5.2 CE & W

All of the reactor internals were placed into one of three categories based on the significance and severity of the potential degradation. These three categories were:

Category A: Component items for which aging degradation significance is minimal orand aging effects are below the screening criteria.
Category B: Component items above screening levels but are not "lead" component items and aging degradation significance is moderate.
Category C: "Lead" component items for which aging degradation significance is high or moderate and aging effects are above screening levels.

5.2.1 Components Placed in Category A Based on FMECA

After review and confirmation by the FMECA panel, all of the components that were not identified in the screening process for potential susceptibility to any of the eight degradation mechanisms were retained as originally placed in Category A.

The FMECA panel also observed that, due to the conservative nature of the screening process, many components that had been identified for potential degradation were known to not be susceptible to degradation. The most obvious example of the conservative nature of the process was that the surveillance capsule components were identified for irradiation embrittlement because the screening process attributed the peak core barrel fluence to all of the potential attachments. However the FMECA panel observed that the surveillance capsules contain dosimetry packages and the fluences were known to be well below the threshold for irradiation embrittlement.

To more accurately reflect the degradation potential for the components and account for the overly conservative nature of the screening process, the FMECA panel recommended that components with low failure likelihood and either low or medium damage likelihood, especially where the potential for any damage was considered to be readily detectable and manageable in attaining a safe operational state, be moved to Category A. <u>Components with low failure likelihood and high damage likelihood were not considered as candidates to be moved to Category A under any conditions</u>. These criteria are illustrated in Figure 2. By definition, all components with potential safety concerns were classified as high damage likelihood. <u>Therefore, no components with identified safety concerns were affected by this re-classification.</u>

Failure Likelihood	Consequence (Damage Likelihood)		
	Low	Medium	High
High	2	3	3
Medium	1	2	3
Low	1	1	2
None	0	0	0

Category A

Figure 2 FMECA Criteria for Aging Significance Table

The 41 Westinghouse components with one or more identified degradation mechanisms that were moved to Category A based on the FMECA results are listed in Table 5. The 48 CE components moved to Category A based on the FMECA are listed in Table 6. The FMECA panel identified 27 Westinghouse and 27 CE components with low failure probability and low damage consequence. There were an additional 14 Westinghouse and 21 CE components with low failure probability and medium damage consequence. Although the FMECA panel identified a potential economic consequence of failure in the components with medium likelihood of damage, the low failure probability resulted in minimal risk to plant operation. Therefore these 14 Westinghouse and 21 CE components were also placed in Category A. Application of the FMECA process to the Lower Core Plate Fuel Alignment Pin Bolts is provided in Example 1.

Example 1: Lower Core Plate Fuel Alignment Pin Bolts Placed In Category A Based on FMECA

Original screening results: MRP-191 Table 5-1

- **IASCC, Wear, Fatigue, Irradiation Embrittlement, Void Swelling, Irradiation Induced Stress Relaxation/Creep**

Functional Description: MRP-191 Section C.2.1

- **The LCP is bolted at the periphery to a ring welded to the ID of the core barrel. The span of the plate is supported by lower support columns that are attached at their lower end to the lower support plate. At the center, a removable plate is provided for access to the vessel lower head region.**

FMECA Conclusion: MRP-191 Table 6-5

- **Low Failure Probability, Low Consequence**
  - **Screening process overestimated fluence because it assumed components attached to LCP saw same peak fluence. These bolts are located on periphery.**
  - **No history of failures**
  - **Bolts are redundant fasteners.**

Table 5. Westinghouse Components Moved to Category A Based on FMECA Process (Data extracted from MRP-191 Table 6-5)

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
						L, M, H	L, M, H
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	Enclosure pins	304 SS	NONE	SCC, Wear	L	M
		Upper guide tube enclosures	304 SS	NONE	SCC, Wear	L	M
		Flanges-intermediate	304 SS	G	SCC, Fatigue	L	M
		Flanges-intermediate	CF8	G	SCC, Fatigue, TE	L	M
		Flanges-lower	304 SS	G	SCC, Fatigue	L	M
		Guide tube support pins	316 SS	NONE	Wear, Fatigue, ISR	L	M
	Mixing Devices	Mixing devices	CF8	NONE	SCC, TE, ISR	L	L
	Upper Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS	NONE	Wear	L	L
		Upper core plate	304 SS	A, G	Wear, Fatigue	L	M
	Upper Plenum	UHI flow column bases	CF8	G	TE, IE	L	L
	Upper Support Column Assemblies	Bolts	316 SS	G	Wear, Fatigue, ISR	L	M
		Column bases	CF8	G	SCC, TE, IE	L	M
		Extension tubes	304 SS	G	SCC	L	M
	Upper Support Plate Assembly	Deep beam ribs	304 SS	G	SCC	L	M
		Deep beam stiffeners	304 SS	G	SCC	L	M

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H
		Inverted top hat (ITH) flange	304 SS	N/A	SCC, Fatigue	L	M
		Inverted top hat (ITH) upper support plate	304 SS	N/A	SCC	L	M
Lower Internals Assembly	Baffle and Former Assembly	Baffle bolting lock bars	304 SS	NONE	IASCC, IE, VS	L	L
	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column bolts	316 SS	NONE	Fatigue	L	L
		BMI column extension bars	304 SS	G	IASCC, IE, VS	L	L
		BMI column nuts	304 SS	NONE	IASCC,Wear, Fatigue, IE, VS, ISR	L	L
	Irradiation Specimen Guides	Irradiation specimen guides	304 SS	NONE	Wear, IE	L	L
		Irradiation specimen guide bolts	316 SS	NONE	IASCC, Wear, Fatigue, IE, ISR	L	L
		Irradiation specimen guide lock caps	304L SS	NONE	IE	L	L
		Specimen plugs	304 SS	NONE	IE	L	L
	Lower Core Plate and Fuel Alignment Pins	Fuel alignment pins	316 SS	NONE	IASCC, Wear,IE, VS	L	L
		LCP-fuel alignment pin bolts	316 SS	NONE	IASCC, Wear, Fatigue, IE, VS, ISR	L	L
		LCP-fuel alignment pin lock caps	304L SS	NONE	IASCC, IE, VS	L	L

MRP-227 Roadmap
October 29, 2010

Assembly	Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
						L, M, H	L, M, H
	Neutron Panels/Thermal Shield	Neutron panel bolts	316 SS	NONE	IASCC, Wear, Fatigue, IE, ISR	L	L
		Neutron panel lock caps	304 SS	NONE	IE	L	L
		Thermal shield bolts	316 SS	NONE	IASCC, Wear, Fatigue, IE, ISR	L	L
		Thermal shield dowels	316 SS	NONE	IE	L	L
		Thermal shield or neutron panels	304 SS	G	IE	L	L
	Radial Support Keys	Radial support key bolts	304 SS	G	Wear	L	L
	Radial Support Keys	Radial support keys	304 SS	G	SCC, Wear	L	L
	Secondary Core Support (SCS) Assembly	SCS base plate	304 SS	NONE	SCC	L	L
Interfacing Components	Interfacing Components	Clevis inserts	Alloy 600	G	Wear	L	L
		Clevis inserts	304 SS	G	Wear	L	L
		Clevis inserts	Stellite	G	Wear	L	L
		Internals hold-down spring	304 SS	G	Wear	L	L
		Internals hold-down spring	403 SS	G	Wear, TE	L	L

IMT Consequence of Failure - G: Causes significant economic impact
A: Precludes a safe shutdown

Table 6. CE Components Moved to Category A Based on FMECA Process (Data extracted from MRP-191 Table 6-6)

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
					L,M,H	L,M,H
Upper Internals Assembly	Upper guide structure support plate	304 SS	G	SCC	L	M
	Upper guide structure support flange-upper	304 SS	G	SCC, Wear	L	M
	Upper guide structure support flange-lower	304 SS	G	SCC	L	M
	Cylindrical skirt	304 SS	G	SCC	L	M
	Grid plate	304 SS	G	SCC	L	M
	Control rod shroud-grid ring	304 SS	N/A	SCC	L	M
	Control rod shroud-grid beams	304 SS	N/A	SCC	L	M
	Control rod shroud-cross braces	304 SS	N/A	SCC	L	M
	GSSS guide structure plate	304 SS	N/A	SCC	L	M
	GSSS support cylinder	304 SS	N/A	SCC	L	M
	Flange blocks	304 SS	N/A	Wear	L	L

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
					L,M,H	L,M,H
	Flange block bolts	410 SS	N/A	TE	L	L
	RVLMS support structure tubes	304 SS	N/A	SCC, Wear, Fatigue	L	L
	Fuel bundle guide pins	316 SS	N/A	Wear, Fatigue, ISR	L	L
	Fuel bundle guide pin nuts	304 SS	N/A	Wear, Fatigue, ISR	L	L
	Hold down ring	403 SS/ F6NM	G	Wear, TE	L	L
	Belleville washer	Alloy 718	N/A	Wear	L	L
Lower Support Structure	Core support plate bolts	316 SS	N/A	IASCC, Wear, Fatigue, IE, ISR	L	L
	Core support plate dowel pins	304 SS	N/A	IE	L	L
	Anchor block bolts	316 SS	N/A	Wear, Fatigue, IE, ISR	L	L
	Anchor block dowel pins	304 SS	N/A	IE	L	L
	Fuel alignment pins	304 SS	NONE	IE	L	M
	Core support beams	304 SS	A, G	SCC, Wear	L	L
	Bottom plate	304 SS	N/A	SCC	L	L
	ICI support columns	304 SS	N/A	SCC	L	L

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
					L,M,H	L,M,H
Control Element Assembly (CEA)–Shroud Assemblies	CEA shrouds	304 SS	G	SCC	L	M
	CEA shrouds	CPF8/CF8	G	SCC, TE	L	M
	CEA shroud bases	304 SS	G	SCC	L	M
	CEA shroud bases	CF8	G	SCC, TE	L	M
	CEA shroud extension shaft guides	304 SS	G	SCC	L	M
	Modified CEA shroud extension shaft guides	CF8	G	SCC, TE	L	M
	Internal/external spanner nuts	304 SS	NONE	SCC	L	M
	CEA shroud bolts	A286 SS	NONE	Wear, Fatigue, ISR	L	M
	CEA shroud tie rods	304 SS	N/A	SCC	L	M
	Snubber blocks	304 SS	N/A	SCC	L	L
	Snubber shims	XM-29	N/A	Wear	L	L
Core Support Barrel Assembly	Core barrel snubber lugs	304, 321 or 348 SS	G	SCC, Wear	L	L
	Alignment keys	A286 SS	NONE	Wear	L	L
	Alignment keys	304 SS	NONE	Wear	L	L
	Core barrel outlet nozzles	304 SS	G	SCC, Wear	L	M

Assembly/ Sub-Assembly	Component	Material	IMT Conseq. of Failure	Screened-in Degradation Mechanisms	Likelihood of Failure	Likelihood of Damage
					L,M,H	L,M,H
	Thermal shield	304 SS	G	SCC	L	L
	Thermal shield support pins	304 SS	NONE	Wear	L	L
Core Shroud Assembly	Guide lugs	304 or 348 SS	NONE	SCC	L	L
	Guide lug inserts	304, 321 or 348 SS	NONE	Wear	L	L
In-Core Instrumentation (ICI)	ICI guide tubes	316 SS	NONE	SCC, IE	L	L
	ICI nozzle support plate	304 SS	G	SCC	L	L
	ICI thimble support plate	304 SS	G	SCC, Wear	L	L
	ICI thimble tubes-upper	304 SS	NONE	SCC, Wear	L	L

5.2.2 Components Placed in Categories B and C

The remaining 31 Westinghouse and 23 CE "non-Category A" components were evaluated and placed in Category B or Category C based on the FMECA results and analysis using the Category definitions. Each component was assigned a FMECA aging significance grouping based on the FMECA categories as indicated in Figure 2.

Two exceptions were noted to the components identified by the screening and FMECA process. First, it was observed that the X-750 flexures in Westinghouse plants were obsolete due to plant modifications to resolve the aging concerns. These flexures were removed from subsequent consideration. Second, it was noted that the Zr-4 thimble tubes in the CE In-Core Instrumentation system were known to be subject to an irradiation growth phenomenon that was not addressed as one of the eight degradation modes. These thimble tubes were automatically placed in Category C.

Of the remaining components, 12 Westinghouse and 13 CE components ranked as medium failure likelihood and low failure consequence were automatically placed in Category B. Evaluations of the impact of each of the identified degradation mechanisms were used to rank the significance of the remaining 19 Westinghouse and 9 CE components. Based on that ranking, 12 Westinghouse components were identified as Category C and an additional 6 Westinghouse components were added to the Category B list. A total of 6 CE components (including the Zr-4 thimble tubes mentioned above) were identified as Category C, with the remaining 4 components added to Category B.

There were two additional exceptions to this categorization process discussed in Section 7.2 of MRP-191:

1. *The Westinghouse lower support casting, had been identified as a FMECA Group 2 component based on the consequences of an assumed failure. However, consistent with the MRP-134 definitions, this component was placed into Category A after consideration of the very low probability of degradation and consequence due to the identified thermal embrittlement degradation mechanism.*

2. *The otherOne exception is the internals hold down spring fabricated from 304 SS. Thermal "ratcheting", leading to permanent deformation, is not one of the explicitly characterized degradation mechanisms from MRP-175 but may occur in this component and reduce the spring hold-down force over time. This particular phenomenon was assessed to have a moderate likelihood of occurrence; hence, it was assigned to Category B to warrant attention during the development of Inspection and Evaluation (I&E) guidelines.*

MRP-227 Roadmap
October 29, 2010

The final list of 31 Westinghouse and 23 CE Category B and Category C items is provided in MRP-191 Tables 7-2 and 7-3. This information is summarized here in Tables 6 and 7. This list of Category B and C Components is carried forward into MRP-227 Tables 3-2 and 3-3. The aging management strategy for the reactor internals is built around examination of these items.

Table 76 Summary of Westinghouse Category B and Category C Components

Assembly	Sub-Assembly	Component	Material	Degradation Mechanism	IMT Conseq. of Failure	Likelihood of Failure	Likelihood of Damage	MRP-191 Category
						L, M, H	L, M, H	A, B or C
Upper Internals Assembly	Control Rod Guide Tube Assemblies and Flow Downcomers	C tubes	304 SS	Wear	G	M	M	C
		Flanges-lower	CF8	SCC, Fatigue,TE, IE	G	M	M	B
		Flexures	X-750	SCC, Fatigue,TE, IE	G	H	M	
		Guide plates/cards	304 SS	SCC, Wear, Fatigue	G	H	M	C
		Guide tube support pins	X-750	SCC, Wear, Fatigue, ISR	NONE	H	M	C
		Sheaths	304 SS	Wear	G	M	M	C
	Upper Support Plate Assembly	Upper support ring or skirt	304 SS	SCC, Fatigue,TE, IE	G	M	M	B
Lower Internals Assembly	Baffle and Former Assembly	Baffle-edge bolts	316 SS/347 SS	IASCC, Wear, Fatigue, IE, VS, ISR	NONE	H	M	C
		Baffle plates	304 SS	IASCC, IE, VS	G	M	L	B
		Baffle-former bolts	316 SS/347 SS	IASCC, Wear, Fatigue, IE, VS, ISR	G	H	L	C

Assembly	Sub-Assembly	Component	Material	Degradation Mechanism	IMT Conseq. of Failure	Likelihood of Failure	Likelihood of Damage	MRP-191 Category
						L, M, H	L, M, H	A, B or C
		Barrel-former bolts	316 SS/347 SS	IASCC, Wear, Fatigue, IE, VS, ISR	N/A	H	L	C
		Former plates	304 SS	IASCC, IE, VS	G	M	L	B
	Bottom Mounted Instrumentation (BMI) Column Assemblies	BMI column bodies	304 SS	SCC, IASCC, Fatigue, IE, VS	G	M	L	B
		BMI column collars	304 SS	IASCC, IE, VS	G	M	L	B
		BMI column cruciforms	CF8	IASCC, TE, IE, VS	G	M	L	B
		BMI column extension tubes	304 SS	SCC, IASCC, Fatigue, IE, VS	G	M	L	B
	Core Barrel	Core barrel flange	304 SS	SCC, Wear	A, G	L	H	B
		Core barrel outlet nozzles	304 SS	SCC, Fatigue	G	M	M	B
		Lower core barrel	304 SS	SCC, IASCC, IE	A, G	M	H	C
		Upper core barrel	304 SS	SCC, IASCC, IE	A, G	M	H	C
	Flux Thimbles (Tubes)	Flux thimble tube plugs	304 SS	SCC, IASCC, IE. VS	G	M	L	B
		Flux thimbles (tubes)	316 SS	SCC, IASCC, Wear, IE. VS	G	H	L	C

MRP-227 Roadmap
October 29, 2010

Assembly	Sub-Assembly	Component	Material	Degradation Mechanism	IMT Conseq. of Failure	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	MRP-191 Category A, B or C
	Lower Core Plate and Fuel Alignment Pins	Lower core plate	304 SS	SCC, IASCC, Wear, Fatigue, IE. VS	A, F, G	M	M	C
		XL lower core plate	304 SS	SCC, IASCC, Wear, Fatigue, IE	N/A	M	M	C
	Lower Support Column Assemblies	Lower support column bodies	CF8	IASCC, TE, IE, VS	G	M	L	B
		Lower support column bodies	304 SS	IASCC, IE, VS	G	M	L	B
		Lower support column bolts	304 SS	IASCC, Wear, Fatigue, IE, VS, ISR	G	M	L	B
	Lower Support Casting or Forging	Lower support casting	CF8	TE	A, G	L	H	A
	Neutron Panels/Thermal Shield	Thermal shield flexures	304 SS	IASCC, Wear, Fatigue, IE, ISR	N/A	M	L	B
Interfacing Components	Interfacing Components	Clevis insert bolts	X-750	SCC, Wear	G	M	L	B
		Internals hold-down spring	304 SS	SCC, Wear	G	L	L	B
		Upper core plate alignment pins	304 SS	Wear	NONE	M	L	B

Table 87 Summary of CE Category B and Category C Components

Assembly/ Sub-Assembly	Component	Material	Degradation Mechanism	IMT Conseq. of Failure	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	MRP-191 Category A, B or C
Upper Internals Assembly	Fuel alignment plate	304 SS	SCC, Wear, Fatigue	A, G	M	M	B
Lower Support Structure	Core support plate	304/304L SS	SCC, IASCC, Wear, Fatigue, IE	A, G	M	M	C
	Fuel alignment pins	A286 SS	IASCC, Wear, Fatigue, IE, ISR	NONE	M	M	C
	Core support columns	304 SS	SCC, IASCC, Fatigue, IE	A, G	M	L	B
	Core support columns	CF8	SCC, IASCC, Fatigue, TE, IE	A, G	M	L	B
	Core support deep beams	304 SS	SCC, IASCC, Fatigue, IE	A, G	M	M	C
	Core support column bolts	316 SS	IASCC, Wear, Fatigue, IE, ISR	NONE	M	L	B
Control Element Assembly (CEA)–Shroud Assemblies	Instrument tubes	304 SS	SCC, Fatigue	NONE	M	L	B
Core Support Barrel Assembly	Upper cylinder	304 SS	SCC	A, G	L	H	B
	Lower cylinder	304 SS	SCC, IASCC, IE	A, G	M	H	C
	Upper core barrel flange	304 SS	SCC, Wear	A, G	L	H	B
	Lower core barrel flange	304 SS	SCC, Fatigue	A, G	L	H	B
	Thermal shield positioning pins	UNS S21800	Wear, Fatigue, ISR	NONE	M	L	B

MRP-227 Roadmap
October 29, 2010

Assembly/ Sub-Assembly	Component	Material	Degradation Mechanism	IMT Conseq. of Failure	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	MRP-191 Category A, B or C
Core Shroud Assembly	Shroud plates	304 SS	SCC, IASCC, IE, VS	G	M	M	C
	Former plates	304 SS	SCC, IASCC, IE, VS	G	M	L	B
	Ribs	304 SS	SCC, IASCC, IE, VS	G	M	L	B
	Rings	304 SS	SCC, IASCC, IE, VS	G	M	L	B
	Core shroud bolts	316 SS	IASCC, Wear, Fatigue, IE, VS, ISR	G	M	L	B
	Barrel-core shroud bolts	316 SS	IASCC, Wear, Fatigue, IE, ISR	G	M	L	B
	Core shroud tie rods	348 SS	Wear, Fatigue, IE, ISR	N/A	M	L	B
	Core shroud tie rod nuts	316 SS	Wear, Fatigue, IE, ISR	N/A	M	L	B
	Guide lug insert bolts	A286 SS	Wear, Fatigue, ISR	N/A	M	L	B
In-Core Instrumentation (ICI)	ICI thimble tubes-lower	Zircaloy-4	Wear	NONE	M	L	C

6.0 Step 6. Engineering Evaluation and Assessment

The sixth step of the process was to perform an assessment of the PWR internals components and items that would most be affected by the aging degradation mechanisms (preliminary Category B and C items from the previous steps). Step 6 has previously been identified as a "Functionality Evaluation" or "Functionality Assessment" in each of the reference documents,. It was determined that these terms may have been somewhat misleading. It has been renamed herein as Engineering Evaluation and Assessment to more closely describe the work that has actually been performed
Step 6 has been identified as a "Functionality Evaluation" or "Functionality Assessment" in each of the reference documents, for which the chosen words unfortunately are now felt to have been somewhat misleading. It has been renamed herein for clarification of the work that has actually been performed. [Or, some wording similar to this!]

As was the case with the FMECA and the severity categorization, the engineering evaluation processes used by AREVA and Westinghouse varied in their specific steps but accomplished the intended goal. A summary of each approach is described below. Finite element analyses of the core barrel regions for the three designs were performed as described in MRP-229 for the B&W units and MRP-230 for the CE and W units. The details of the approaches and results are described in MRP-229 231 for the B&W units and MRP-230 232 for the CE and W units. The results were carried into the aging management strategies documented in MRP-231 for B&W units and MRP-232 for CE and W units.

6.1 B&W

The engineering evaluation and assessment (aka, functionality assessment) work performed included structural evaluation with finite element analysis (FEA), engineering analysis, operating experience, and review of inservice inspection results. (Note: the functionalityengineering evaluation and analysis assessment effort is not a requalification of the design basis considering the potential age-related degradation).

6.1.1 FEA Analyses

Two finite element analyses (FEA) (also call "functionality analyses" in Sections 2.1 and 2.2 of MRP-231) were performed for the B&W units within the MRP effort:

- A genericn analysis of the core barrel assembly, which includes the core barrel cylinder, baffle plates, former plates, baffle-to-former (FB) bolts, baffle-to-baffle (BB) bolts, and core barrel-to-former (CB) bolts. The thermal shield and bolt locking devices are not modeled and analyzed in this evaluation.
- A genericn analysis of the currently installed upper core barrel (UCB) bolts, lower core barrel (LCB) bolts, and flow distributor (FD) bolts on a generic basis.

6.1.1.1 Core Barrel Assembly

FEA is performed for the core barrel assembly due to the large number of Category "C" and "B" items in the assembly and potential interactions between the aging degradation mechanisms. The modeling was based on a representative B&W PWR internals unit design, using irradiated and aged material properties, and was performed to model several irradiation-induced aging degradation mechanisms and their interactions (see details in MRP-229).

Included in this analysis was the evaluation of selected austenitic stainless steel components that were judged to be susceptible to irradiation-induced degradation of mechanical and/or physical properties using an ANSYS-based subroutine developed by ANATECH Corporation for EPRI. The stainless steel material models employed in the calculations account for the effects of plasticity, irradiation-enhanced creep, stress relaxation , irradiation-assisted stress corrosion cracking, void swelling, and irradiation embrittlement as a function of temperature and dose. The project team focused on finding the integrated effects of material aging combined with steady-state operational characteristics of the reactor internals.

These analyses subjected representative internals components/assemblies to core heating and dose for 40 fuel cycles or 60 years. Conservative core loading, heat transfer, and mechanical preloads were applied. The aging degradation modeling provided insight for the locations and progression of aging degradation. However, it is not considered capable of predicting the precise timing or location of various aging degradation effects. Therefore, the MRP-227 inspection schedule for the core barrel assembly is primarily based on the industry operating experience and inspection results to date. The core barrel assembly FEA aging modeling results provided additional assurance that the inspection schedule will detect the aging degradation and their interactions before functionality is affected.

The FEA modeling of aging degradation for the core barrel assembly was based on representative configurations and operational histories, which were generally conservative, but not necessarily bounding in every parameter. These assumptions were a conservative representation of U.S. PWR operating units, all of which implemented low-leakage core-loading patterns early in operating life.

Certain items were found to exhibit possible susceptibility to age-related degradation due to prolonged radiation, stress, and temperature (for example, baffle-to-former bolts). Other items are not likely to exhibit susceptibility to age-related degradation that could affect functionality from long-term reactor operation. Results are summarized in Section 4 of the MRP-229 report.

None of the Core Barrel Assembly components were downgraded to "No Additional Measures" as a result of the FEA analysis. In addition, some aging degradation effects such as baffle-to-former bolt overload were identified based on the FEA analysis.

However, some of the Core Barrel Assembly components were downgraded from "C" to "B". For example, the baffle plates were downgraded from "C" to "B", which were eventually placed in the "Primary" group. In addition, some components had an individual aging degradation mechanism downgraded from "C" to "B" or to "A", but could not be downgraded to "No Additional Measures" due to the remaining aging degradation mechanisms. For example, void swelling was downgraded from "C" to "A" for baffle-to-former bolts, which remained as a "C" item and eventually was placed in the "Primary" group.

6.1.1.2 High-Strength Bolt Rings

The UCB and LCB bolt locations have a core support function and are categorized as "C". Detailed FEA is performed in accordance with the current ASME Section 3 design criteria under normal operating and upset conditions.

Variations in bolt replacement patterns or non-functional bolts were not considered in the analysis. The loads considered were:

- Preload
- Thermal stresses for the case of High-Leakage End-Of-Cycle (HL-EOC)
- Mechanical load including hydraulic forces and flow-induced vibration (FIV)
- Deadweight loads

None of these bolts/components were downgraded to "No Additional Measures" as a result of the FEA analysis. The UCB and LCB bolts remained as Category C and were eventually placed in the Primary group. The FD bolts were downgraded from Category C to Category B and were eventually placed in the Expansion group.

The results are used in the final two steps of the MRP work for assessing the previous B&WOG Materials Committee conclusions and recommendations regarding these bolts. Due to the considerable differences among the units, additional analysis on a unit-specific basis is underway withinwas performed by the PWROG.

6.1.2 Other Evaluations

Evaluations for the remaining preliminary Category B and C items (i.e., engineering assessment, operating experience, and review of inservice inspection results) were performed as necessary and are documented in Section 2 of MRP-231.

The results from the engineering evaluation and assessment efforts functionality assessments provide the basis for updating the Category A, Category B, and Category C PWR internals items for the B&W-design. The final Category A, B, and C results are provided in Table 98.
A brief discussion of these two steps is provided below.

6.1.2.1 Engineering Evaluations

Several B&W RV internals components were "resolved" (downgraded to "No additional measures") by engineering evaluations as documented in MRP-231, Sections 2.3 and 2.4, and are listed below.

- CRGT Guide Tubes and Sectors

- Thermal Shield Upper Restraint Cap Screws
- Lower Grid Rib-to-Shell Forging Cap Screws
- Lower Grid Support Post Pipe Cap Screws

The CRGT guide tubes (C-tubes) and guide sectors (split-tubes) in B&W units were initially categorized as "Not-A" for wear, and were placed in Category "B" after FMECA. Subsequently, AREVA reviewed past wear investigations of control rods within the guide path as documented in MRP-231 Section 2.3. It was concluded that there was no evidence of wear on the control rod, and thus there should not be any wear on the CRGT guide tubes and guide sectors. Therefore, the CRGT guide tubes and sectors were downgraded to "A" from "B" and were eventually placed in the "No Additional Measure" group.

The thermal shield upper restraint cap screws, lower grid rib-to-shell forging cap screws, and lower grid support post pipe cap screws were initially categorized as "Not-A" for irradiation-induced stress relaxation and creep, and the resulting mechanisms of fatigue and wear. These three items were placed in Category "B" after FMECA. Subsequently, AREVA determined the maximum 60-year fluence of these locations. Based on the irradiation stress relaxation data from similar material and temperature, the 60-year stress relaxation was estimated to be insignificant. Therefore, irradiation-enhanced stress relaxation and creep, and the resulting mechanisms of fatigue and wear are downgraded to "A" from "B" and the three cap screw items were eventually placed in the "No Additional Measure" group.

6.1.2.2 Engineering Assessment

Several B&W RV internals weld locations were "resolved" (downgraded to "No additional measures") by assessing the functionality as documented in MRP-231, Sections 2, and are listed below.

- Alloy X-750 dowels-to-plenum cover bottom flange welds
- Alloy X-750 dowel-to-upper grid rib section bottom flange welds
- Alloy X-750 core barrel-to-former plate dowels and the locking welds
- Alloy X-750 dowel-to-lower grid shell forging welds
- Alloy X-750 dowel-to-lower grid rib section welds
- Alloy X-750 dowel-to-flow distributor flange welds

The above welds used nickel-based Alloy 69 (INCO 69) and Alloy 82 (INCO 82) materials, which are susceptible to PWSCC. However, these particular locking welds are for Alloy X-750 alignment dowels that were used only to facilitate the internals assembly process. These dowels do not have any function after the internals items were assembled. Therefore, these welds were downgraded to "A" and were eventually placed in the "No Additional Measure" group.

Table 98
Final Categorization (A, B and C) and Aging Management Strategy (P, E, N and A) Results for Selected B&W Components

Component	ABC Before MRP-231 Evaluation and Assessment (MRP-231 Rev. 1 Table 1-1)	Final ABC After MRP-231 Evaluation and Analysis Assessment (MRP-231 Rev. 1 Table 2-8) (Note 1)	Final P, E, N, A List (MRP-231 Rev. 1 Table 3-8)
CRGT Spacer Castings	B	B	E
CRGT Control Rod Guide Tubes	B	A	N
CRGT Control Rod Guide Sectors	B	A	N
CSS Vent Valve Top and Bottom Retaining Rings	B	B	P
CSS Vent Valve Disc	B	B	P
CSS Vent Valve Disc Shaft or Hinge Pin	B	B	P
Core Barrel Cylinder	B	B	E
Baffle Plates	C	B	P
Former Plates	C	B	E
Core Barrel-to-Former Plate Dowels	B	A	N
Lower Grid Support Post Cap Screw	B	A	N
Flow Distributor (FD) Bolts	C	B	E

Note 1: MRP-231 Table 2-8 only contains “non-A” items; hence the “A” items listed in this column do not appear in Table 2-8.

6.2 CE & W

The functionality analysis provides an opportunity to understand each degradation mode in more detail and to analyze how they interact. The results of the functionality analysis were used to determine that there were a number of potential degradation modes in the Category B and Category C components that were of low failure probability and low failure consequence. These potential degradation modes had little or no potential impact on the function of the component. The three basic types of functionality analysis were: 1) Irradiation Aging Analysis; 2) Extension of Irradiation Analysis to Other Components; and 3) Functionality of Remaining Components. These are discussed below.

6.2.1 Irradiation Aging Analysis

The functionality assessment began with a detailed irradiation aging analysis to understand the complex interactions between active degradation mechanisms in highly irradiated components. These detailed modeling efforts were applied to the Westinghouse baffle-former-barrel structure, the Westinghouse lower core plate, and a welded CE core shroud assembly. The intent of the irradiation aging analysis was to identify trends and limits in the component behavior. The analysis was used to identify factors that could potentially cause component failure. Representative plant designs with relatively severe irradiation conditions were selected for the irradiated aging analysis. These conditions were chosen to test the capability of the structure and identify points of potential concern.

The most severe assumption in the irradiation aging analysis was that the reactor had operated for an extended period of time with "out-in" fuel loading patterns. As the "out-in" pattern is known to produce high neutron fluences in the reactor internals structures and all W and CE NSSS plants in the U.S. fleet are known to have moved away from this core loading strategy relatively early in plant life, the peak baffle-former fluences in the representative plant will significantly exceed the peak 30 EFPY fluences in any currently operating plant. Although the power distributions assumed for the remainder of the 60 EFPY analysis were more realistic, the average power density chosen for this portion of the analysis corresponds to the upper end of the current practice for power uprates. The resulting peak 60 year fluences are expected to be limiting for the current fleet.

Because the irradiation aging analysis applies a multi-parameter model to a complex structure, it is not possible nor is it appropriate to identify bounding conditions. Although the analysis as performed is expected to predict peak neutron fluences in the baffle formers that exceed any realistic evaluation of the operating structures, alternative power distributions may produce higher fluences at off-peak locations. The analysis clearly demonstrates that there are competing effects of irradiation induced void swelling and irradiation induced stress relaxation on the aging behavior of bolts and other key components in the reactor internals structure. Although the highest irradiation doses may provide conservative estimates of the stress increase caused by differential swelling, they may mask the effects of stress relaxation on the bolt pre-load. Therefore, it is not possible to accurately define any set of conditions that bounds the range of potential responses. However, due to the size and complexity of the baffle-former structure it is possible to find locations in the structure that represent a wide range of potential conditions. The interpretation of the irradiation aging analysis described in MRP-232 is based on evaluation of this range of conditions and extrapolation to similar internals structures. However, it does not purport to be a bounding analysis.

The irradiation aging analysis of the representative Westinghouse and CE plants incorporated the most highly irradiated components in the reactor internals. These results were used to provide guidance that was used in the evaluation of the remaining irradiated components.

6.2.1.1 Results from Irradiation Aging Analysis of Westinghouse Lower Core Plate

The analysis of the lower core plate was based on the assumption that the plant had operated for 13 cycles of "out-in" core loading followed by 27 cycles of operation with power distributions representative of current practice in plant uprates. The peak reported 60-year neutron dose in the lower core plate was 19 dpa. The potential for IASCC cracking was evaluated in terms of the ratio of the effective stress to a dose dependent threshold stress for cracking. Over the entire 60-year analysis, there was no location in the lower core plate where the calculated stress exceeded the IASCC threshold stress. The results of this analysis are presented in Section 4 of MRP-230 and summarized in Section 4.2.3 of MRP-232.

6.2.1.2 Results from Irradiation Analysis of Westinghouse Baffle-Former-Barrel Structure

The most highly irradiated components in the Westinghouse reactor internals are the flux thimbles, which are inserted in the core and the core baffle structure that immediately surround the core. This analysis was based on the assumption that the plant had operated for twenty 18 month cycles of "out-in" core loading followed by twenty 18 month cycles of operation with power distributions representative of current practice in plant uprates. The peak reported 60-year neutron dose of 147 dpa in this assembly occurred in the baffle plates. There is a large variation in neutron fluence over the volume of this assembly, with a peak fluence in the core barrel of only 13 dpa. The highest peak damage rates occurred during the period of "out-in" operation. The detailed analysis of the baffle-former barrel structure included the baffle plates, former plates, core barrel and associated bolting. The results of this analysis are presented in Section 3.1 of MRP-230. Results of detailed local modeling of selected baffle-former bolts areis presented in Section 5 of MRP-230.

Void swelling rates in localized regions near the baffle-former interface imposed significant stresses on the surrounding bolts. During the first thirty years of operation, a significant fraction of the baffle-former bolts exceeded the IASCC threshold stress. Conditions were found to be significantly less damaging during the period of operation with low leakage cores. Although significant localized deformation was noted in sections of the baffle-former structure, the resultant stresses are relatively low. No IASCC concerns were identified in the baffle plates or the former plates. There were tTwo barrel-former bolt locations were identified where the 60-year stress could potentially exceed the IASCC threshold. However, the vast majority of baffle-former bolts indicated a slowing falling preload. Complete loss of load in the system is not expected. A summary of the baffle-former-barrel conclusions and recommendations is provided in MRP-232 Sections 4.2.1 and 4.2.2.

6.2.1.3 Results from Irradiation Analysis of CE Welded Core Shroud

The most highly irradiated components in the CE reactor internals are located in the core shroud assembly that immediately surrounds the core. There are sSeveral different core shroud designs are present included in the CE fleet. The core shroud design selected for the detailed irradiation analysis consists of stacked upper and lower welded structures, held together by tie rods. This design was selected for study because it was believed to have features that would demonstrate the most sensitivity to void swelling. Where the two welded structures meet, there are matching 1.5 inch thick horizontal plates producing a 3 inch thick section near the core midplane with no internal cooling. Gamma heating was expected to produce relatively high internal temperatures, which may result in void swelling.

The detailed aging analysis used for the CE core shroud, which is described in MRP-230 Section 3.2 used the same basic neutron loading assumptions as the Westinghouse baffle-former-barrel assembly analysis. The peak neutron dose in the CE core shroud at 60 years of operation was 132 dpa. Despite the large amount of void related distortion near the peak temperature locations, swelling induced increases in stress were limited to a relatively small volume of surrounding welds. Analysis and recommendations based on these results are provided in MRP-232 Section 4.1.1.

The tie rods in the CE core shroud are located near the outside of the shroud structure and operate near the fluid temperature. The peak 60 year neutron fluence in the tie rod is 19 dpa. Under these conditions, minimal void swelling is expected. However, the neutron dose at the tie rod location is sufficient to cause irradiation induced stress relaxation. The analysis indicates a gentle drift of tie rod loads over the 60 year period, but sufficient load appears to be maintained.

6.2.2 Extension of Irradiation Analysis to Other Components

There were a number of lessons learned from the analysis of the lower core plate, core shroud and baffle-former-barrel structure that were directly applicable to other irradiated components in the system. Most notably, a number of components had been identified for potential susceptibility for irradiation-related degradation mechanisms based primarily on their proximity to the lower core plate or core barrel. The detailed fluence maps developed to support the analysis of the highly irradiated components were used to provide more realistic fluence estimates for many of these components. The results from the irradiation aging analysis clearly demonstrated that the conditions at these locations were not severe enough to cause significant degradation.

6.2.3 Functionality Analysis of Remaining Components

Functionality analysis is based on evaluation of the relevance of the degradation mode to the design basis requirements for Category B and Category C components. In some cases, the identified degradation mode was either found to be irrelevant to the function of the component, or it was found that existing analysis could be used to demonstrate that the potential change in component condition was not a challenge to the design basis.

It should be noted that the design justification for the reactor internals is based primarily on elastic analysis. The irradiation-induced increase in yield stress only increases the limits for elastic analysis. Notch sensitivity or flaw tolerance is not normally considered as part of the design basis for reactor internals. Therefore, in analyzing the components that have reduced toughness due to irradiation embrittlement, it is important to consider the potential for flaws and other stress risers. The combination of a potential cracking mechanism (SCC, IASCC or fatigue) with irradiation embrittlement may be a particular concern.

6.2.4 Functionality Analysis to Demonstrate No Additional Aging Management Requirements

The FMECA process was completed by considering the combined effects of all identified aging degradation mechanisms on the component. While it is important to consider the potential interactions between the degradation modes, in most cases the FMECA conclusions are controlled by one or two limiting degradation modes. The functionality analysis provides an opportunity to understand each degradation mode in more detail and to analyze how they interact. The results of the functionality analysis were used to determine that there were a number of potential degradation modes in the Category B and Category C components that were of low failure probability and low failure consequence. These potential degradation modes had little or no potential impact on the function of the component.

The Category B and Category C component degradation modes that were determined to have little or no impact on the component function are listed as "Resolved by Analysis" in MRP-232 Tables 2-1 through 2-16. It is important to note that the original categorization of these components was based on the combined effects of all degradation mechanisms. In general, this categorization is based on consideration of the most severe effects and it is possible that some identified mechanisms in the same component with less severe impacts may be considered to be "Resolved by Analysis." Descriptions of the individual degradation mechanisms and functionality concerns are contained in Section 2 of MRP-232. Evaluation of the implications of the functionality analysis for each component is contained in Section 4 of MRP-232. These determinations are reflected in MRP-227 Tables 3-2 and 3-3.

The determination that one or more mechanism was resolved by analysis had no impact on the classification of any component as Category B or Category C. However, determination in any component that the mechanism was "Resolved by Analysis" did imply that further aging management for that mechanism was not required. These components were identified in MRP-227 Tables 3-2 and 3-3 as "No Aadditional mMeasures". Aging management requirements were eventually defined for all of the identified degradation mechanisms in the Category B and Category C components that were not determined to be "Resolved by Analysis". In a limited number of cases, all of the identified degradation mechanisms in a component were determined to be "Resolved by Analysis" and the final aging management recommendation for the component was "No Additional Measures". The remaining Category B and C components were placed into the Primary, Expansion or Existing aging management recommendation tables.

Many of the functionality analysis conclusions were derived by comparing specific degradation modes and their impact on a specific component. Application of this process to the Bottom

Mounted Instrument Column Cruciforms is provided in Example 2a and the application of the process to the Lower Core Plate is in example 2b.

Example 2: Bottom Mounted Instrumentation Cruciforms Degradation Mechanisms Moved to "No Additional Measures"

Original screening results: MRP-191 Table 5-1
- **IASCC, Irradiation Embrittlement, Thermal Embrittlement, Void Swelling**

Functional Description:
- **MRP-232 Section 4.2.6: BMI column assemblies provide a path for the flux thimbles into the core from the bottom of the vessel and protect the flux thimbles during operation of the reactor.**
- **MRP-156 Section 4.2.10: The cruciform columns extend through the flow holes of the lower support forging and attach to the bottom of the LCP.**

FMECA Conclusion: MRP-191 Table 6-5
- **Medium Failure Probability, Low Consequence**

Analysis of Degradation Mechanisms: MRP-232 Section 4.2.6
- **No additional measures required**
  - **Analysis of lower core plate indicated irradiation effects are overestimated.**
  - **The flux thimbles are inserted and withdrawn during refueling outages. It is anticipated that any failure in these columns would be noted during refueling outages and would have minimal impact on normal operation.**
  - **Inspection of BMI columns triggered by difficulty inserting (or withdrawing) flux thimbles.**
  - **BMI system has no structural function.**

Example 2a: Bottom Mounted Instrumentation Cruciforms Degradation Mechanisms Moved to "No Additional Measures"

Original screening results: MRP-191 Table 5-1
- IASCC, Irradiation Embrittlement, Thermal Embrittlement, Void Swelling

Functional Description:
- MRP-232 Section 4.2.6: BMI column assemblies provide a path for the flux thimbles into the core from the bottom of the vessel and protect the flux thimbles during operation of the reactor.
- MRP-156 Section 4.2.10: The cruciform columns extend through the flow holes of the lower support forging and attach to the bottom of the LCP.

FMECA Conclusion: MRP-191 Table 6-5
- Medium Failure Probability, Low Consequence

Analysis of Degradation Mechanisms: MRP-232 Section 4.2.6
- No additional measures required
  - Analysis of lower core plate indicated irradiation effects are overestimated.

– The flux thimbles are inserted and withdrawn during refueling outages. It is anticipated that any failure in these columns would be noted during refueling outages and would have minimal impact on normal operation.
– Inspection of BMI columns triggered by difficulty inserting (or withdrawing) flux thimbles.
– BMI system has no structural function.

Example 2b: Lower Core Plate

The analysis of the lower core plate was based on the assumption that the plant had operated for 13 cycles of "out-in" core loading followed by 27 cycles of operation with power distributions representative of current practice in plant uprates. The results of this analysis are presented in Section 4 of MRP-230 and summarized in Section 4.2.3 of MRP-232. The peak reported 60-year neutron dose in the lower core plate was 19 dpa. The "low leakage" power distributions used in the uprated core designs minimize radial leakage of neutrons, but can result in higher levels of axial leakage. Therefore, the peak reported lower core plate temperature of 635°F occurred during the later period of operation when uprated core power distributions were assumed. The peak volumetric swelling in the lower core plate was 0.18% and occurred in a very small region near the mid-thickness of the plate. The potential for IASCC cracking was evaluated in terms of the ratio of the effective stress to a dose dependent threshold stress for cracking. Over the entire 60 year analysis, there was no location in the lower core plate where the calculated stress exceeded the IASCC threshold stress.

The lower core plate was originally placed in Category C based on the observation that it was a critical core support structure and the fact that there were multiple identified degradation modes.

Following the FMECA process, there were six potential degradation modes were identified.

1. SCC – No additional measures (IASCC predominate)
2. Void Swelling – No additional measures (Calculated 0.18% maximum)
3. IASCC – Existing Inspections Adequate
4. Wear – Existing Inspections Adequate
5. Fatigue – Existing Inspections Adequate
6. Irradiation Embrittlement - Existing (Included in evaluation of IASCC and fatigue)

Based on this analysis, the lower core plate is listed in Table 4-9 as an existing component recommendation.

7.0 Step 7. Categorize for Inspection (Primary, Expansion, Existing, No Additional Measures) and Aging Management Strategy

Thise final step in the process is to take all the remaining Category B and C components and re-classify them based on the need for inspection. The ultimate result of the process was to assign the components into Primary, Expansion, Existing Programs, and No Additional Measures groups, with appropriate recommendations to support unit-specific aging management program development efforts. The four functional groups are summarized below and are defined in Section 3.3.1 of MRP-227:

- **Primary:** those PWR internals items that are highly susceptible to the effects of at least one of the eight aging mechanisms were placed in the Primary group. The aging management requirements that are needed to ensure functionality of Primary components are described in these I&E guidelines. The Primary group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

- **Expansion:** those PWR internals items that are highly or moderately susceptible to the effects of at least one of the eight aging mechanisms, but for which functionality assessment has shown a degree of tolerance to those effects, were placed in the Expansion group. The schedule for implementation of aging management requirements for Expansion components will depend on the findings from the examinations of the Primary components at individual units.

- **Existing Programs:** those PWR internals items that are susceptible to the effects of at least one of the eight aging mechanisms and for which generic and unit-specific existing AMP elements are capable of managing those effects, were placed in the Existing Programs group.

- **No Additional Measures:** those PWR internals items for which the effects of all eight aging mechanisms are below the screening criteria were placed in the No Additional Measures group. Additional components were placed in the No Additional Measures group as a result of FMECA and the functionality assessment. No further action is required by these guidelines for managing the aging of the No Additional Measures components.

It should be noted that the categorization and analysis processes described herein are not intended to supersede any ASME B&PV Code Section XI requirements. Any components that are classified as core support structures as defined in ASME B&PV Code Section XI IWB 2500 Category B-N-3 have requirements that remain in effect and may only be altered as allowed by 10CFR50.55a.

7.1 B&W

The aging management strategy development described in MRP-231 combined the results of Step 6 (functionality assessment, component accessibility, operating experience, existing evaluations, and prior examination results) to determine the appropriate methodologies for maintaining the long-term functions of PWR internals safely and economically. This process permitted further categorization of PWR internals into the functional groups listed above. Figure 1-2 in MRP-231 shows the process used by AREVA to meet this goal, while Figure 2-2 (MRP-227) shows the links between the categorization based on screening criteria, the functionality analysis, the aging management strategy development, and the I&E guidelines.

MRP-227 Roadmap
October 29, 2010

The aging management strategy for each of the B&W-design PWR internals items is developed in MRP-231. Section 3.3 (MRP-231) summarizes the recommended inspection method, inspection frequency, and inspection coverage for the Primary and Expansion items. Each of these is summarized in Tables 3-9 and 3-10 (MRP-231) or Tables 4-1 and 4-4 (MRP-227).

Note: There are no Existing Programs component items for the B&W-designed PWR internals, so there is no Table 4-7 in MRP-227.

The following examples and flow charts provide an illustration of how the process worked for 2 various components in the B&W-design RV internals. Figure 3 below is a flow chart that shows the overall seven step processExample 3 is for the CRGT control rod guide tubes and Example 4 is for CSS vent valve top and bottom retaining rings. Figure 3 below is a flow chart that shows the overall 7 step process. Figure 4 is flow chart for the CRGT control rod guide tubes and Figure 5 is flow chart for CSS vent valve top and bottom retaining rings. The eighth step included in this roadmap refers to the final MRP efforts involved in preparation of the I&E Guidelines in MRP-227.

MRP-227 Roadmap
October 29, 2010

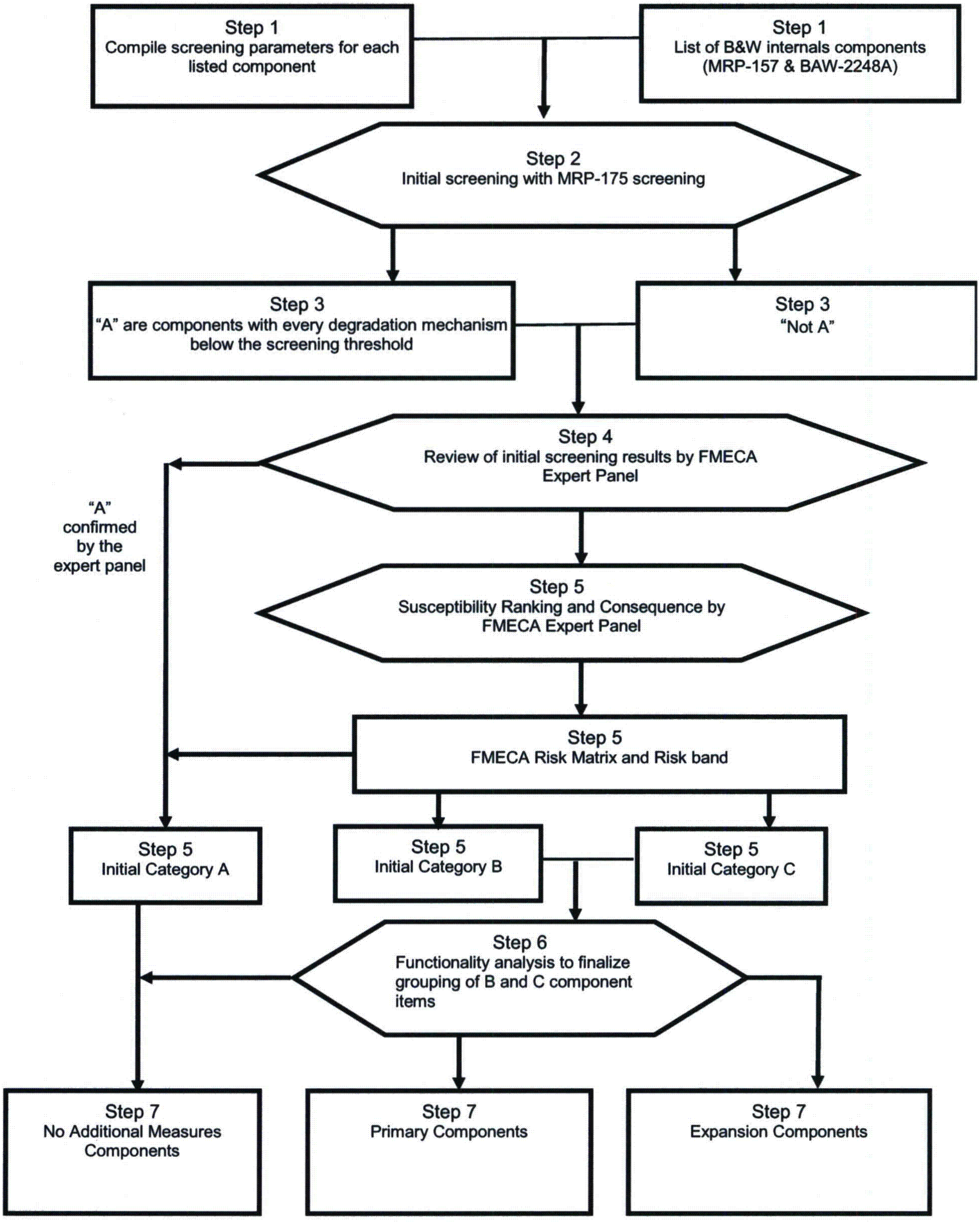

Figure 3, Step 1 through Step 7 for MRP-189 Figure 1-3 flowchart

Example 3: CRGT spacer castings

The function of the spacer castings is to provide structural support to the 12 perforated vertical rod guide tubes and 4 pairs of vertical rod guide sectors within each CRGT assembly. Ten spacer castings keep the 16 guide tubes/sectors in each CRGT assembly aligned with the 16 guide tubes in the fuel assembly below. The control rod spider, which in turn supports the control rods, is guided by the brazement assembly over the entire range of the withdrawal path. In addition, the brazement envelope limits reactor coolant cross flow on the control rods to limit flow induced vibration. The spacer castings do not have a core support function; however, they do have a safety function relative to control rod alignment, insertion and reactivity issues. Degradation of the spacer castings could result in degradation in the unit shutdown capability by hindering the insertion of the control rods into the core in the normal anticipated time.

Initially screened in as Non-A and ultimately grouped as Expansion

- Screened in as Non-A for thermal aging embrittlement in Step 3 (cast austenitic stainless steel Type CF-3M, and information available on chemical composition indicates that ferrite ranges from 6.2% to 27.7%), all other mechanisms screened out

- FMECA results identified susceptibility as "C" and safety consequences as "1," which preliminarily categorizes this item as "Category B" (see Figure 1 in Step 5)

- As shown in Table 3-6 (MRP-231), the incore monitoring instrumentation (IMI) guide tube spiders and the attachment welds, the CSS outlet nozzles at ONS-3 and DB, and the CSS vent valve discs are categorized as Primary items

- The CSS outlet nozzles, the CSS vent valve discs, and the CRGT spacer castings are located above the core and their operating conditions are similar, i.e., at hot leg temperature with an irradiation dose too low to cause irradiation embrittlement. Hence, their extent of thermal embrittlement is expected to be similar. Since the CSS outlet nozzles and the CSS vent valve discs are readily accessible, they are grouped as Primary items and the CRGT spacer castings are grouped as Expansion items.

- However, Type CF-3M material contains 2% to 3% percent molybdenum, which may potentially contribute to a higher thermal embrittlement for the CRGT spacer castings than the other Type CF-8 casting items, depending on the casting method and ferrite content. Thus, in considering any potential synergistic effect of dose on thermal aging embrittlement, the Type CF-8 IMI spiders would be expected to bound the Type CF-3M CRGT spacer castings. Therefore, the CRGT spacer castings are also categorized as Expansion items for the IMI spiders.

The accompanying flow chart is provided as Figure 4 below.

MRP-227 Roadmap
October 29, 2010

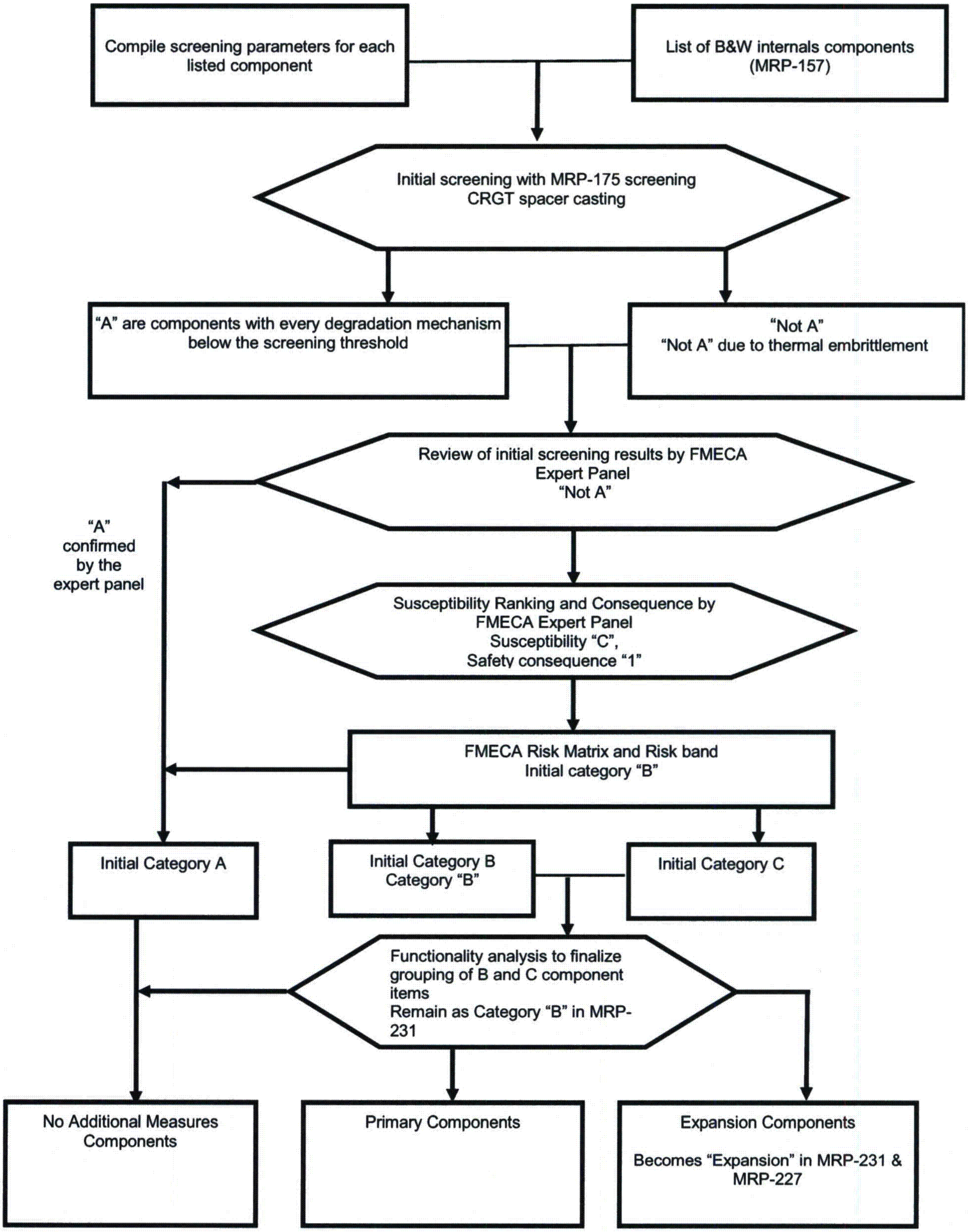

Figure 4, Flowchart for CRGT spacer castings (based on MRP-189 Figure 1-3 flowchart)

Example 4: CRGT control rod guide tubes and sectors

The control rod guide tube assemblies each consist of a pipe (the guide housing), a flange, spacer castings, guide tubes, and rod guide sectors. The assemblies are welded to the plenum cover plate and bolted to the upper grid assembly. Their function is to provide control rod assembly guidance, protect the control rod assembly from the effects of potential coolant cross-flow, and structurally connect the upper grid assembly to the plenum cover.

Initially screened in as <u>Non-A</u> and ultimately grouped as <u>No Additional Measures</u>

- Screened in as <u>Non-A</u> for wear in Step 3 (due to the relative motion between these and the control rods), all other mechanisms screened out

- FMECA results identified susceptibility as "B" and safety consequences as "2," which preliminarily categorizes this item as "Category B" (see Figure 1 in Step 5)

- As shown in Section 2.3 (MRP-231), the control rod guide tubes and sectors are re-categorized to "Category A" by an evaluation of control rod wear performed by AREVA and an engineering judgment that wear between these two items would be similar and therefore negligible

- Therefore, the CRGT control rod guide tubes and sectors are categorized as <u>No Additional Measures</u> required

The accompanying flow chart is provided as Figure 5 below.

MRP-227 Roadmap
October 29, 2010

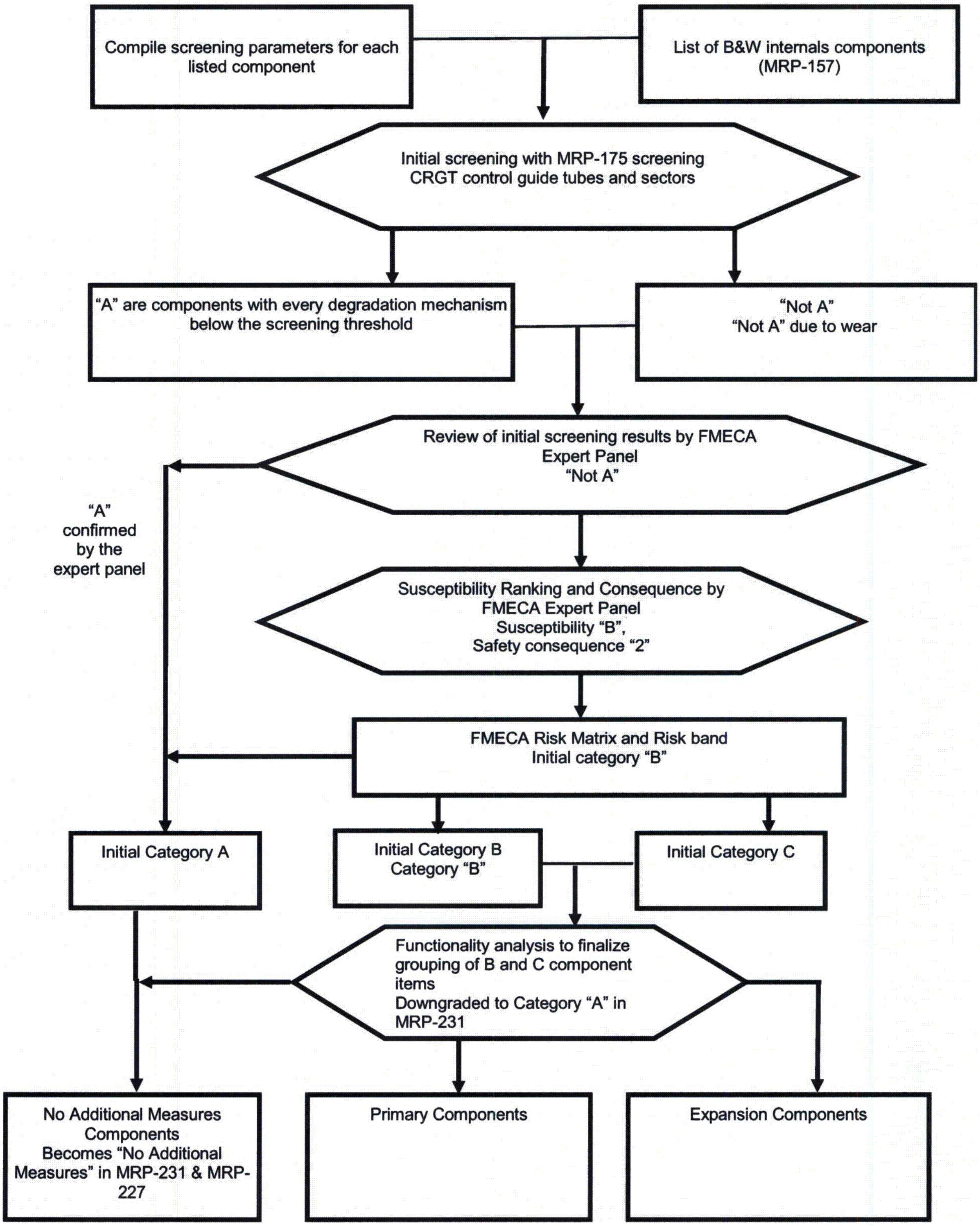

Figure 5, Flowchart for CRGT control rod guide tubes (based on MRP-189 Figure 1-3 flowchart)

Example 5: CSS vent valve top and bottom retaining rings

Vent valves are passive devices and for all normal operating conditions, the vent valve is closed. The pressure on the reactor vessel annulus side is greater than the interior of the core support shield and the pressure differential holds the valve closed to prevent bypass flow. The vent valve top and bottom retaining rings do not have a core support safety function; however, they do have a safety function in that degradation of the vent valve top and bottom retaining rings, which would prevent the vent valve from opening, could result in loss of the vent valve function during a large break loss-of-coolant-accident (LOCA).

Initially screened in as Non-A and ultimately grouped as Primary

- Screened in as Non-A for thermal aging embrittlement in Step 3 (martensitic PH stainless steel, Type 15-5 PH), all other mechanisms screened out

- FMECA results identified susceptibility as "C" and safety consequences as "1," which preliminarily categorizes this item as "Category B" (see Figure 1 in Step 5)

- As shown in Section 3.2.5 (MRP-231) and as noted in the BAW-2248A report, a program is in place at each of the B&W units that requires testing and inspection of the vent valve assemblies each refueling outage. The aging management measures provided in these requirements include a provision primarily to visually inspect the valve body and disc seating surfaces. However, the existing program does not specify the visual inspection technique and the surface coverage. Therefore, to augment the existing vent valve program, these vent valve items are categorized as Primary items for TE with a VT-3 visual inspection of 100% of the accessible surface at every 10-year ISI.

The accompanying flow chart is provided as Figure 6 below.

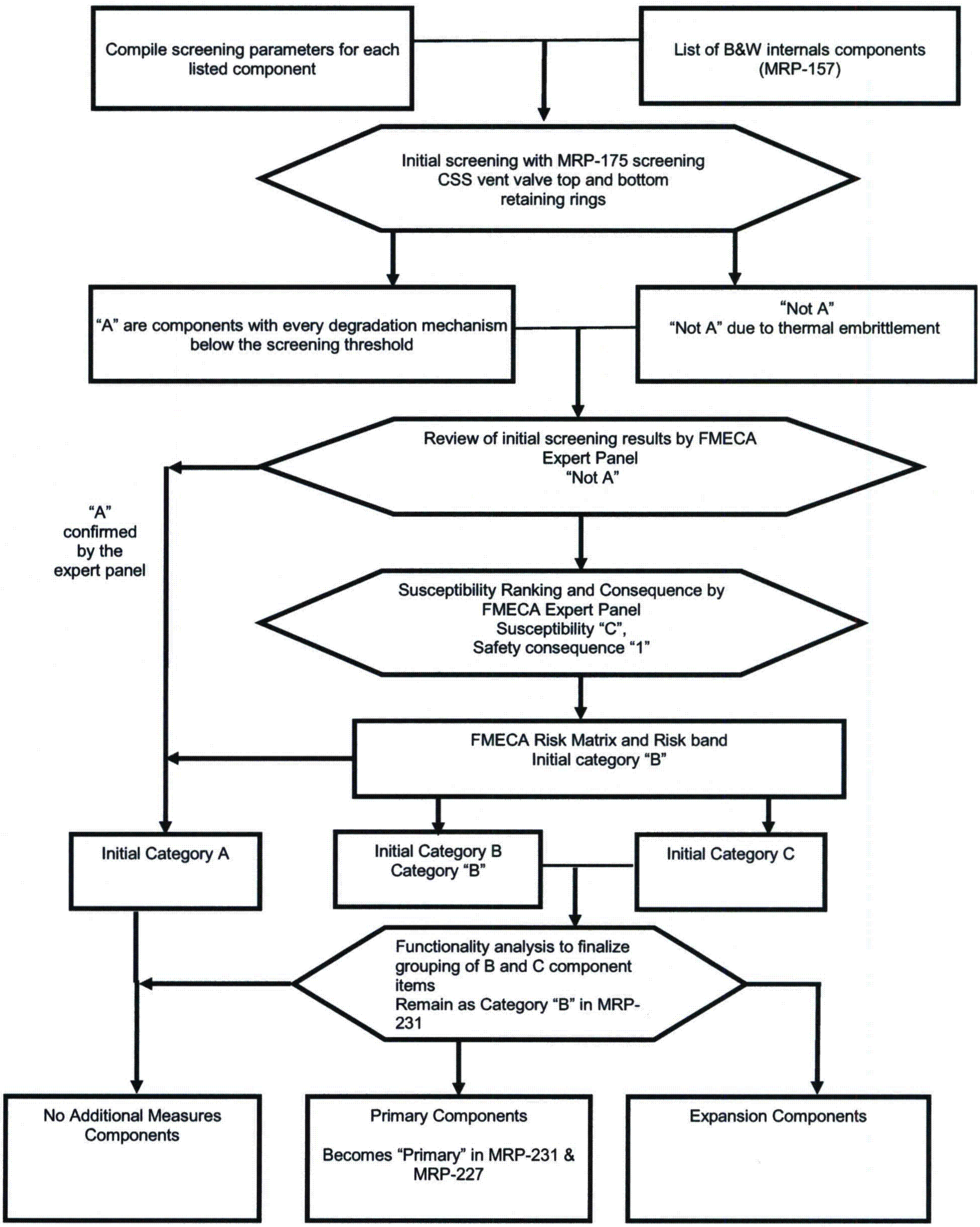

Figure 6, Flowchart for CSS vent valve top and bottom retaining rings (based on MRP-189 Figure 1-3 flowchart)

Example 6: CSS vent valve disc

Vent valves are passive devices that and for all normal operating conditions, the vent valve is closed. The pressure on the reactor vessel annulus side is greater than the interior of the core support shield and the pressure differential holds the valve closed to prevent bypass flow. The vent valve discs do not have a core support safety function; however, they do have a safety function in that degradation of the vent valve discs, which would prevent the vent valve from opening, could result in loss of the vent valve function during a large break loss-of-coolant-accident (LOCA).

Initially screened in as Non-A and ultimately grouped as Primary

- Screened in as Non-A for thermal aging embrittlement in Step 3 (CASS material and CMTR results were not readily available), all other mechanisms screened out

- FMECA results identified susceptibility as "C" and safety consequences as "1," which preliminarily categorizes this item as "Category B" (see Figure 1 in Step 5)

- As shown in Section 3.2.5 (MRP-231) and as noted in the BAW-2248A report, a program is in place at each of the B&W units that requires testing and inspection of the vent valve assemblies each refueling outage. The aging management measures provided in these requirements include a provision primarily to visually inspect the valve body and disc seating surfaces. However, the existing program does not specify the visual inspection technique and the surface coverage. Therefore, to augment the existing vent valve program, this vent valve item is categorized as a Primary item for TE with a VT-3 visual inspection of 100% of the accessible surface at every 10-year ISI.

The accompanying flow chart is provided as Figure 7 below.

MRP-227 Roadmap
October 29, 2010

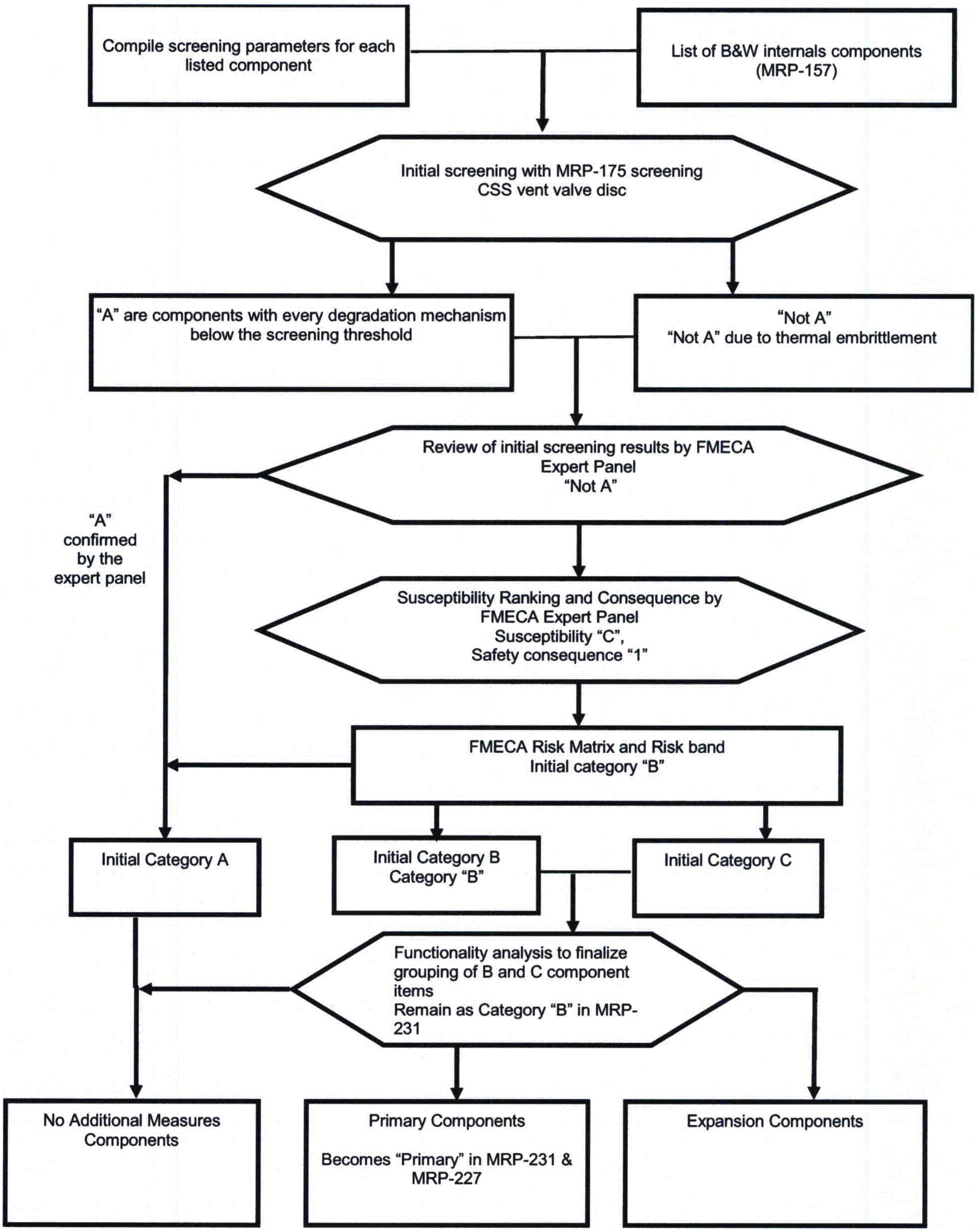

Figure 7 Flowchart for CSS vent valve disc (based on MRP-189 Figure 1-3 flowchart)

Example 7: CSS vent valve disc shaft or hinge pin

Vent valves are passive devices that and for all normal operating conditions, the vent valve is closed. The pressure on the reactor vessel annulus side is greater than the interior of the core support shield and the pressure differential holds the valve closed to prevent bypass flow. The vent valve disc shaft (or, hinge pin) does not have a core support safety function; however, it does have a safety function in that degradation of the disc shaft (or, hinge pin), which would prevent the vent valve from opening, could result in loss of the vent valve function during a large break loss-of-coolant-accident (LOCA).

Initially screened in as Non-A and ultimately grouped as Primary

- Screened in as Non-A for thermal aging embrittlement in Step 3 (martensitic stainless steel, Type 431), all other mechanisms screened out

- FMECA results identified susceptibility as "C" and safety consequences as "1," which preliminarily categorizes this item as "Category B" (see Figure 1 in Step 5)

- As shown in Section 3.2.5 (MRP-231) and as noted in the BAW-2248A report, a program is in place at each of the B&W units that requires testing and inspection of the vent valve assemblies each refueling outage. The aging management measures provided in these requirements include a provision primarily to visually inspect the valve body and disc seating surfaces. However, the existing program does not specify the visual inspection technique and the surface coverage. Therefore, to augment the existing vent valve program, this vent valve item is categorized as a Primary item for TE with a VT-3 visual inspection of 100% of the accessible surface at every 10-year ISI.

The accompanying flow chart is provided as Figure 8 below.

MRP-227 Roadmap
October 29, 2010

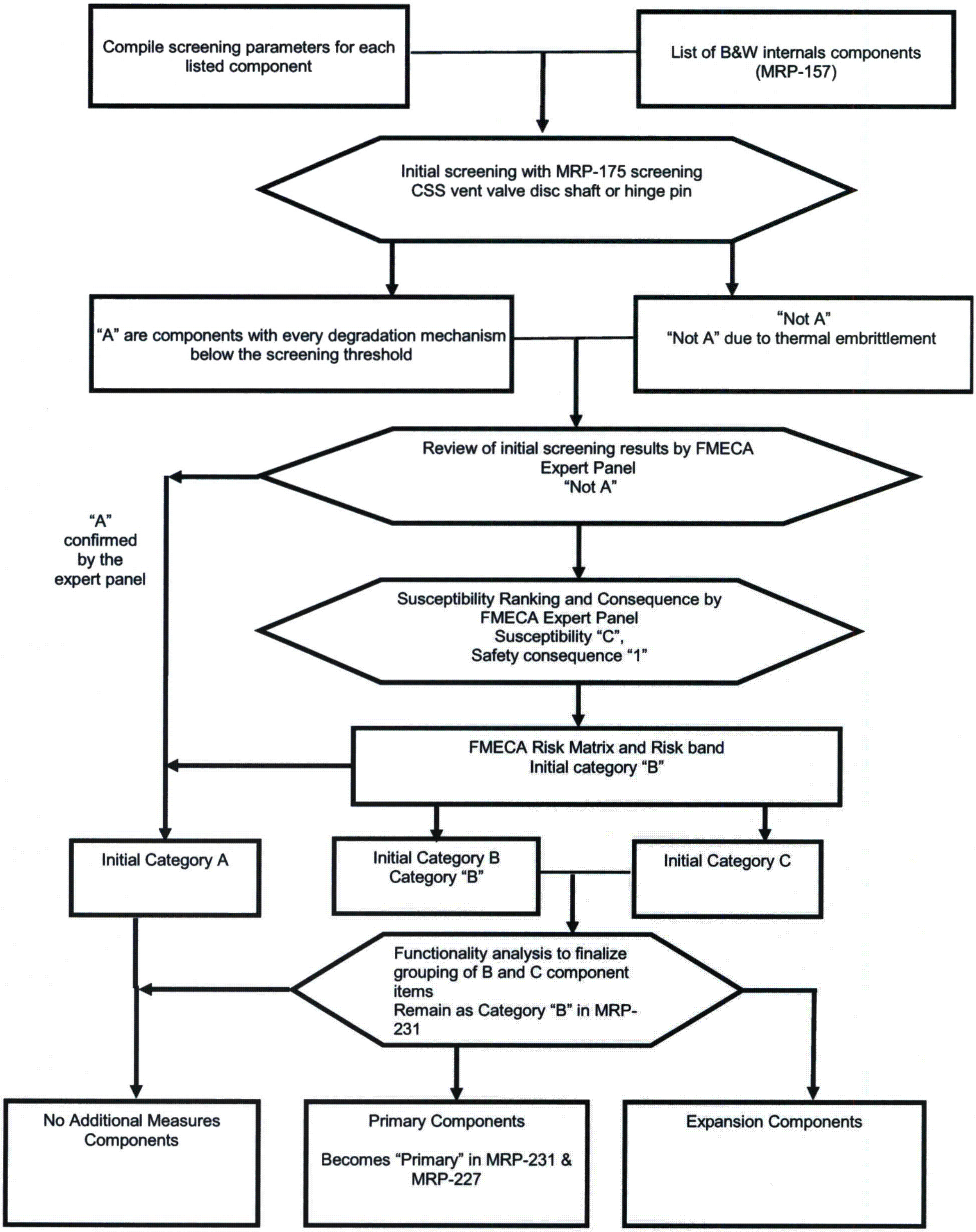

Figure 8, Flowchart for CSS vent valve hinge pin (based on MRP-189 Figure 1-3 flowchart)

Example 8: Core barrel cylinder

The core barrel supports the fuel assemblies, lower grid, flow distributor, and in-core instrument guide tubes. The primary function of the core barrel cylinders and welds during normal power operation is to provide a flow envelope for the core and, thereby limit core bypass flow.

The core barrel cylinders and welds therefore do not have a direct core support safety function; however, they do have a safety function to control bypass around the core during a loss-of-coolant-accident (LOCA).

Initially screened in as Non-A and ultimately grouped as Expansion

- Screened in as Non-A for SCC, fatigue, and irradiation embrittlement in Step 3 (austenitic stainless steel, Type 304 with welds), all other mechanisms screened out

- FMECA expert panel determined that fatigue as an aging mechanism to have a low susceptibility that is supported by no known operating experience of fatigue, and the design criteria containing a significant amount of margin

- FMECA results identified SCC susceptibility as "B" and safety consequences as "1," which preliminarily categorizes this item as "Category A"

- FMECA results identified IE susceptibility as "C" and safety consequences as "1," which preliminarily categorizes this item as "Category B" (see Figure 1 in Step 5)

- As shown in Section 3.2.3 (MRP-231) the core barrel cylinder is considered inaccessible and is not part of the standard 10-year ISI inspection. However, limited access to the former plates, core barrel cylinder, and otherwise inaccessible bolt locking devices is available through the flow bypass holes should a limited examination become necessary

- The baffle plates are the primary item for inspection from IE while the core barrel cylinder is considered to be Expansion item due to its low safety consequences and lower dose

The accompanying flow chart is provided as Figure 9 below.

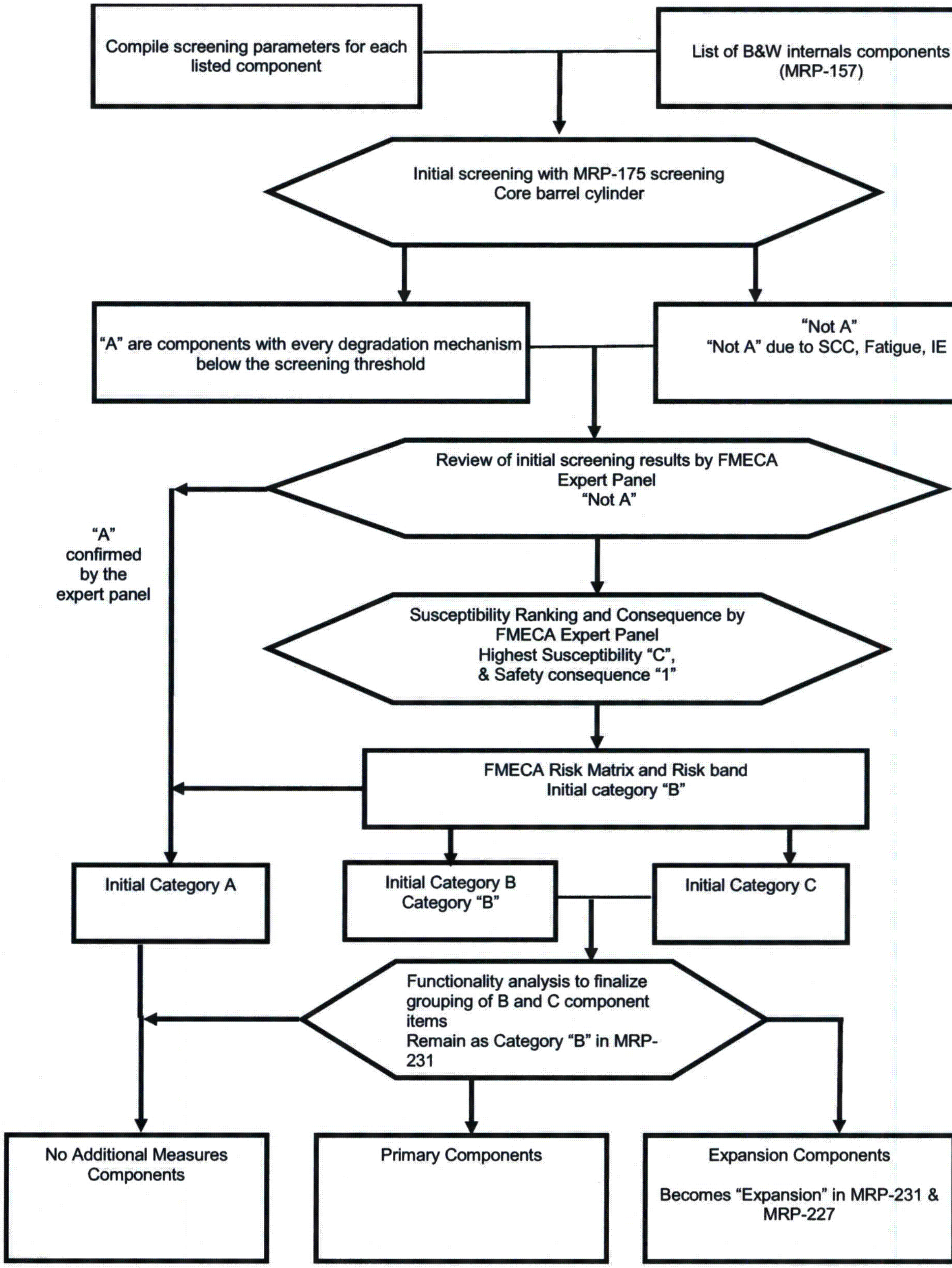

Figure 9, Flowchart for core barrel cylinder (based on MRP-189 Figure 1-3 flowchart)

Example 9: Baffle plates

Degradation of the baffle plates could result in increased core bypass flow and a reduction in margin to departure from nucleate boiling (DNB), but would probably have a negligible effect on unit operations and would not be observed except by direct examination. The core barrel assembly supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. However, the baffle plates do not support any dead weight load. The primary function of the baffle plates during normal power operation is to provide a flow envelope for the core and, thereby limit core bypass flow. There is a differential pressure across the baffle plates during operation and there are thermal stresses induced by both thermal radial gradients and axial gradients primarily resulting from gamma heating. The differential pressure across the plates is amplified during the postulated loss of coolant accident and the plates must be restrained by the baffle plate to former bolts to prevent fuel damage. The baffle plates also provide a horizontal support for the fuel assemblies during a seismic event.

The baffle plates therefore do not have a direct core support function; however, they do have a safety function to control bypass around the core during a loss-of-coolant-accident (LOCA) and maintain the design core geometry during a seismic event.

Initially screened in as <u>Non-A</u> and ultimately grouped as <u>Primary</u>

- Screened in as <u>Non-A</u> for IASCC, IE, and VS in Step 3 (austenitic stainless steel, Type 304), all other mechanisms screened out

- FMECA results identified IASCC susceptibility as “C” and safety consequences as “2,” which preliminarily categorizes this item as “Category C” (see Figure 1 in Step 5)

- FMECA results identified IE susceptibility as “D” and safety consequences as “1,” which preliminarily categorizes this item as “Category B” (see Figure 1 in Step 5)

- FMECA results identified VS susceptibility as “B” and safety consequences as “2,” which preliminarily categorizes this item as “Category B” (see Figure 1 in Step 5)

- As shown in Section 2.1.3.1 (MRP-231), IASCC for the baffle plates was re-categorized to “Category A” as a result of the structural analysis performed

- As shown in Section 2.1.4 (MRP-231), VS for the baffle plates was re-categorized to “Category A” as a result of the structural analysis performed

- As shown in Section 2.5 (MRP-231), as a result of the structural analysis and evaluations performed, the final category for this item is “Category B”

- As shown in Section 3.2.3 (MRP-231) the baffle plates are readily accessible (at least the surface located next to the fuel), while the former plates and the core barrel cylinder are for the most part inaccessible. All three of these items are categorized as "Category B" for IE.

- Therefore, the baffle plates are identified as the Primary item for inspection from IE while the former plates and the core barrel cylinder are considered to be Expansion items due accessibility issues and to their relatively low safety consequences.

The accompanying flow chart is provided as Figure 10 below.

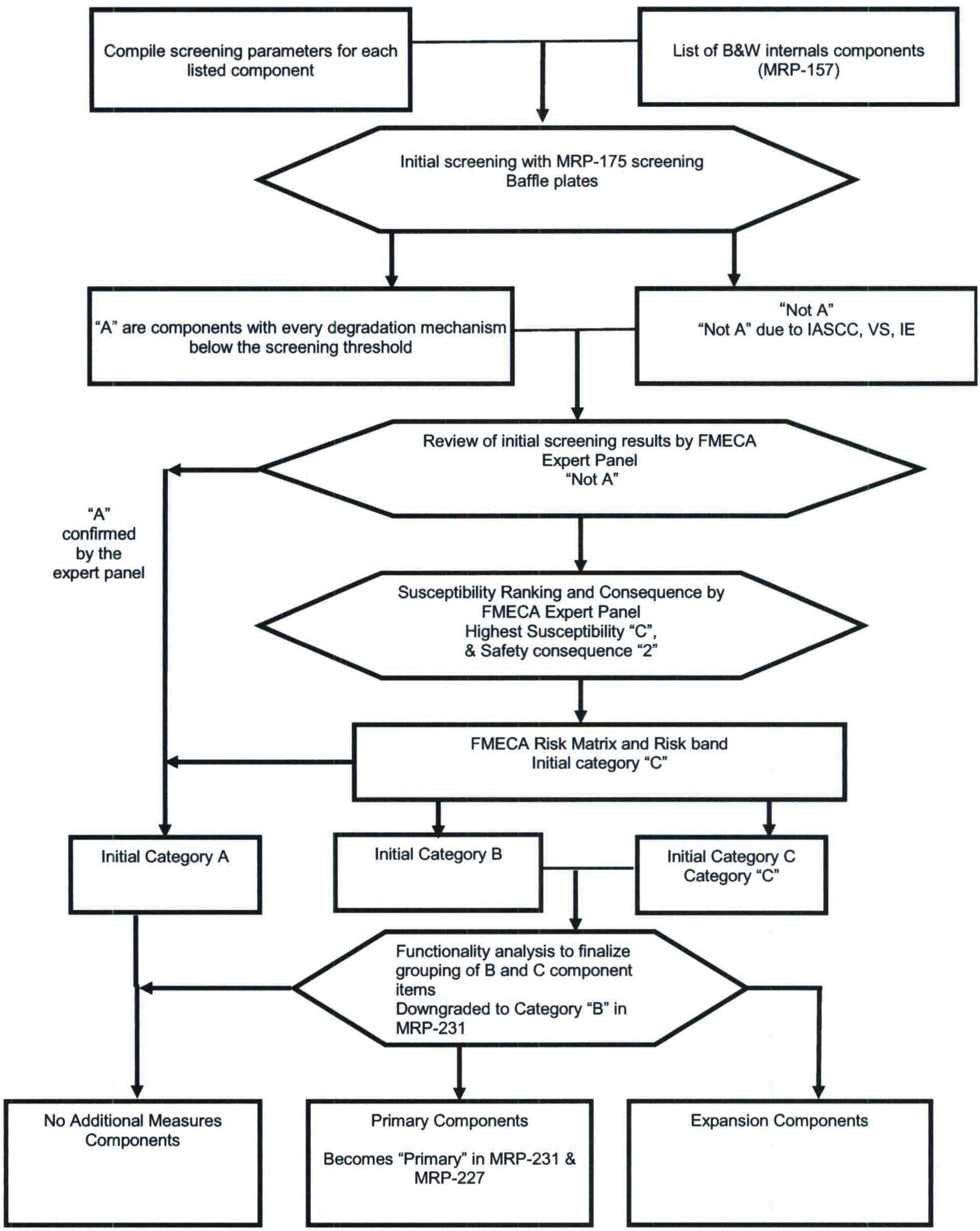

Figure 10, Flowchart for baffle plates (based on MRP-189 Figure 1-3 flowchart)

Example 10: Former plates

The former plates do not have a direct core support safety function; however, they do have a safety function to help maintain the structural integrity of the core barrel assembly during operating conditions.

Initially screened in as Non-A and ultimately grouped as Expansion

- Screened in as Non-A for IASCC, IE, and VS in Step 3 (austenitic stainless steel, Type 304), all other mechanisms screened out
- FMECA results identified IASCC susceptibility as "C" and safety consequences as "2," which preliminarily categorizes this item as "Category C" (see Figure 1 in Step 5)
- FMECA results identified IE susceptibility as "D" and safety consequences as "1," which preliminarily categorizes this item as "Category B" (see Figure 1 in Step 5)
- FMECA results identified VS susceptibility as "C" and safety consequences as "2," which preliminarily categorizes this item as "Category C" (see Figure 1 in Step 5)
- As shown in Section 2.1.3.1 (MRP-231), IASCC for the former plates was re-categorized to "Category A" as a result of the structural analysis performed
- As shown in Section 2.1.4 (MRP-231), VS for the former plates was re-categorized to "Category A" as a result of the structural analysis performed
- As shown in Section 2.5 (MRP-231), as a result of the structural analysis and evaluations performed, the final category for this item is "Category B"
- As shown in Section 3.2.3 (MRP-231) the baffle plates are readily accessible (at least the surface located next to the fuel), while the former plates and the core barrel cylinder are for the most part inaccessible. All three of these items are categorized as "Category B" for IE.
- Therefore, the baffle plates are identified as the Primary item for inspection from IE while the former plates and the core barrel cylinder are considered to be Expansion items due accessibility issues and to their relatively low safety consequences.

The accompanying flow chart is provided as Figure 11 below.

MRP-227 Roadmap
October 29, 2010

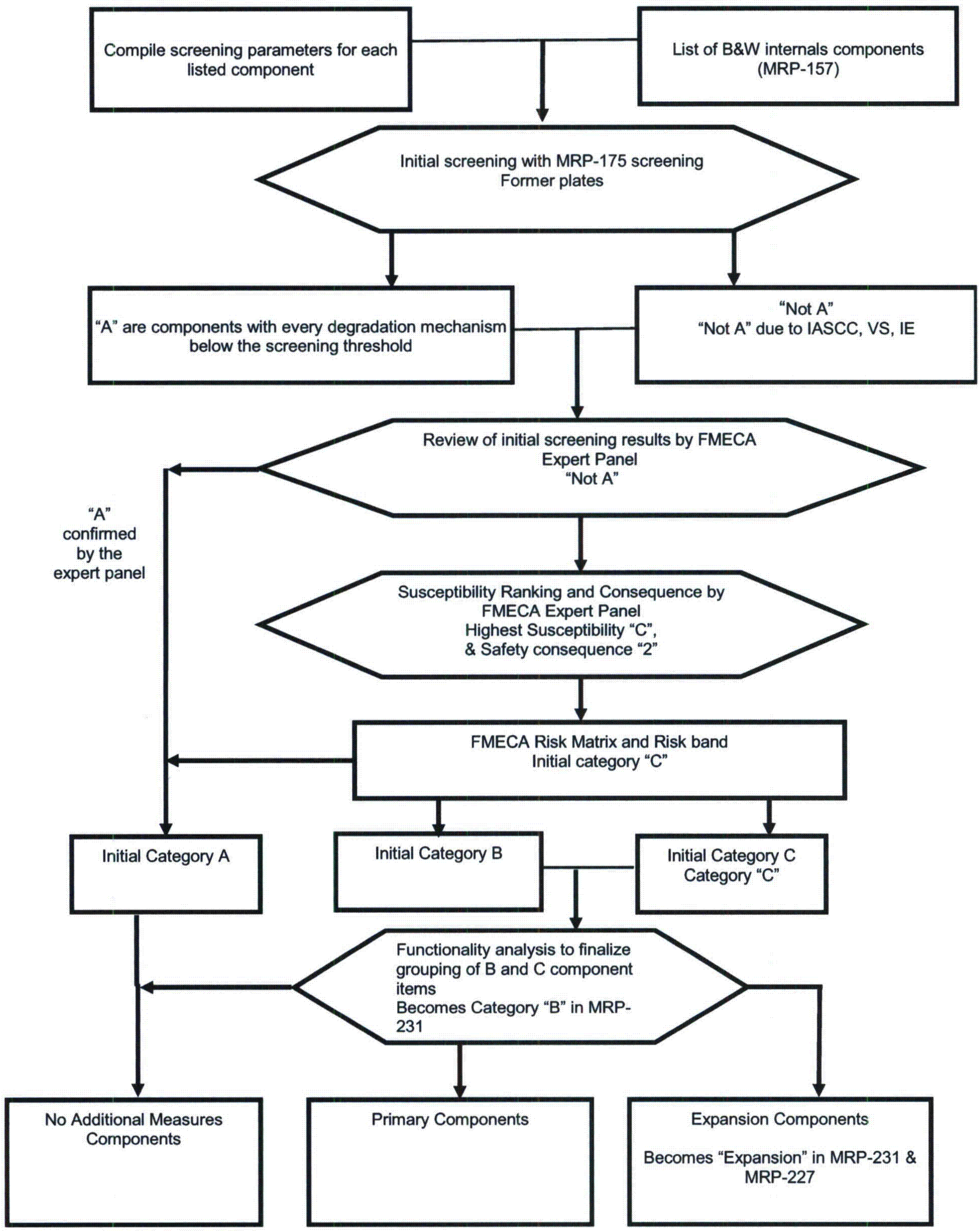

Figure 11 Flowchart for former plates (based on MRP-189 Figure 1-3 flowchart)

Example 11: Core barrel-to-former plate dowels and welds

Welds are used for locking the 32 Alloy X-750 dowels, which were used to align the former plates with the core barrel cylinder at the top and bottom former plate level (16 dowels at each level). After the former plates are bolted to the core barrel cylinder with the core barrel-to-former plate bolts, these Alloy X-750 dowels and their locking welds no longer have any function.

Initially screened in as Non-A and ultimately grouped as No Additional Measures

- Screened in as Non-A for IE and VS in Step 3 (Alloy X-750 material and nickel-base weld), all other mechanisms screened out

- FMECA results identified IE susceptibility as "D" and safety consequences as "1," which preliminarily categorizes this item as "Category B" (see Figure 1 in Step 5)

- FMECA results identified VS susceptibility as "B" and safety consequences as "1," which preliminarily categorizes this item as "Category A" (see Figure 1 in Step 5)

- As shown in Section 2.6 (MRP-231), the core barrel-to-former plate dowels and welds are re-categorized to "Category A" by engineering judgment that the welds are used for locking the Alloy X-750 alignment dowels in place, which facilitated the internals assembly process. These dowels and welds do not have any function after the internals items were joined by bolting. Thus, they are ultimately grouped as No Additional Measures.

The accompanying flow chart is provided as Figure 12 below.

MRP-227 Roadmap
October 29, 2010

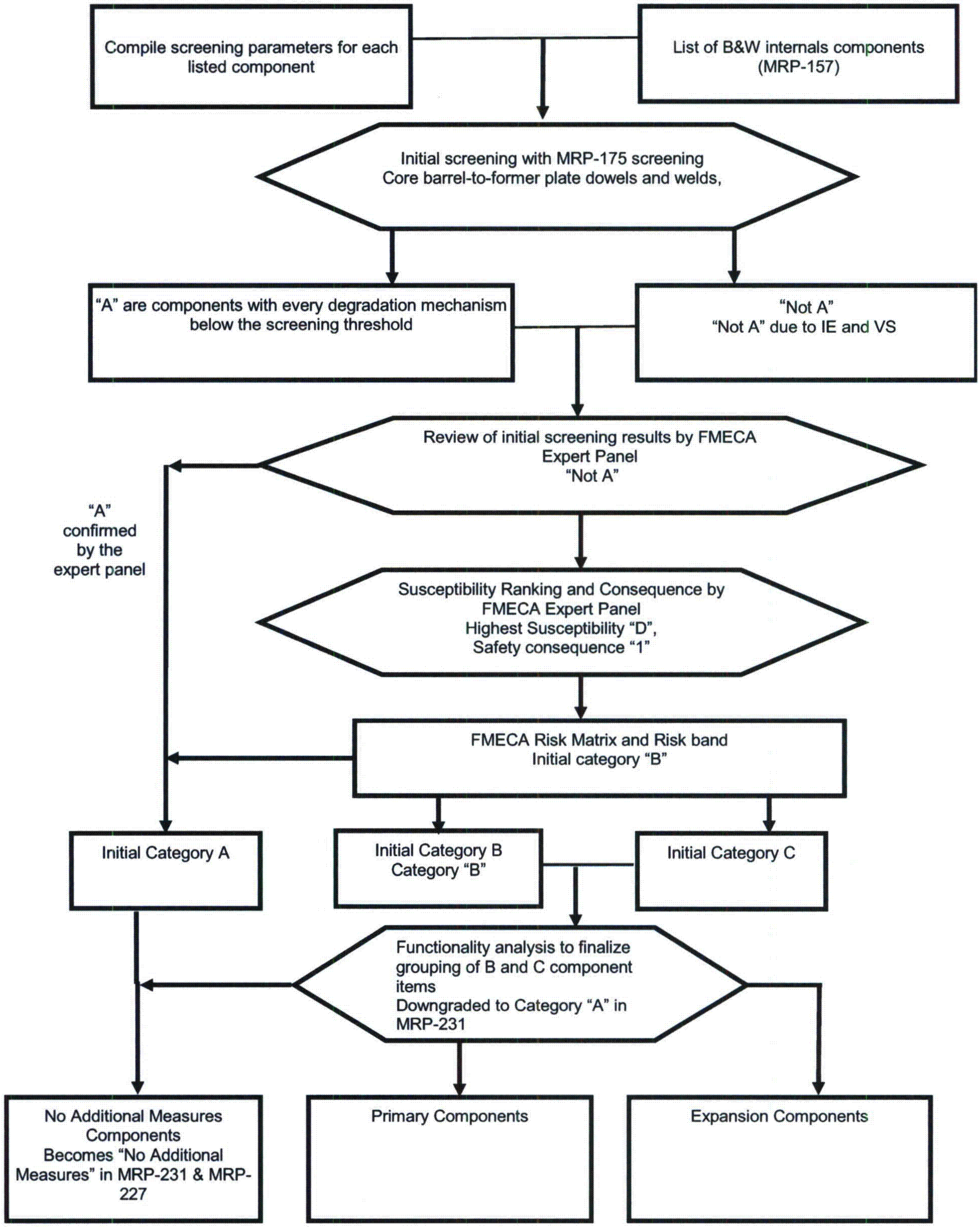

Figure 12, Flowchart for core barrel-to-former plate dowels and welds (based on MRP-189 Figure 1-3 flowchart)

Example 12: Lower grid support post cap screw

The lower grid assembly provides alignment and support for the fuel assemblies, supports the core barrel assembly and flow distributor, and aligns the IMI guide tubes with the fuel assembly instrument tubes. The lower grid consists of three grid structures or flow plates. From top to bottom, they are the lower grid rib section, the flow distributor plate, and the lower grid forging. Each of these flow plates has holes or flow-ports to direct reactor coolant flow upward towards the fuel assemblies. The lower grid assembly is surrounded by the lower grid shell forging. The lower grid shell forging is a flanged cylinder ("ring"), which supports the various horizontal grid structures and flow plates.

The support posts are 48 cylinders placed between the lower grid forging and the lower grid rib section to provide support. The support post assemblies consist of the support pipes and the associated bolting plugs. The support pipes are made from 10½ inch high sections of 4 inch schedule 160 pipe. There are four equally spaced notches at the bottom of the cylinders, where they are welded to the top of the lower grid forging that allow coolant flow upward from below. The bolting plugs are 1¾ inch high disks welded to the top of the support pipes. The bolting plugs have four scallop-shaped holes machined out of the edges so that the tops have a cruciform shape through which coolant can flow. The top of each bolting plug is drilled and tapped to accept the cap screw used to hold it to the lower grid rib section.

Initially screened in as Non-A and ultimately grouped as No Additional Measures

- Screened in as Non-A for irradiation-enhanced stress relaxation, wear, fatigue, and irradiation embrittlement in Step 3 (austenitic stainless steel, Type 304), all other mechanisms screened out

- FMECA results identified ISR susceptibility, with subsequent concerns for wear and fatigue, as "C" and safety consequences as "1," which preliminarily categorizes this item as "Category B" (see Figure 1 in Step 5)

- FMECA results identified IE susceptibility as "B" and safety consequences as "1," which preliminarily categorizes this item as "Category A" (see Figure 1 in Step 5)

- As shown in Section 2.4 (MRP-231), the lower grid support post cap screws are re-categorized to "Category A" by a calculation of potential stress relaxation and engineering judgment that these cap screws will have an estimated 60-year stress relaxation of about 18.7%, which would not be a significant concern. Thus, they are ultimately grouped as No Additional Measures.

The accompanying flow chart is provided as Figure 13 below.

MRP-227 Roadmap
October 29, 2010

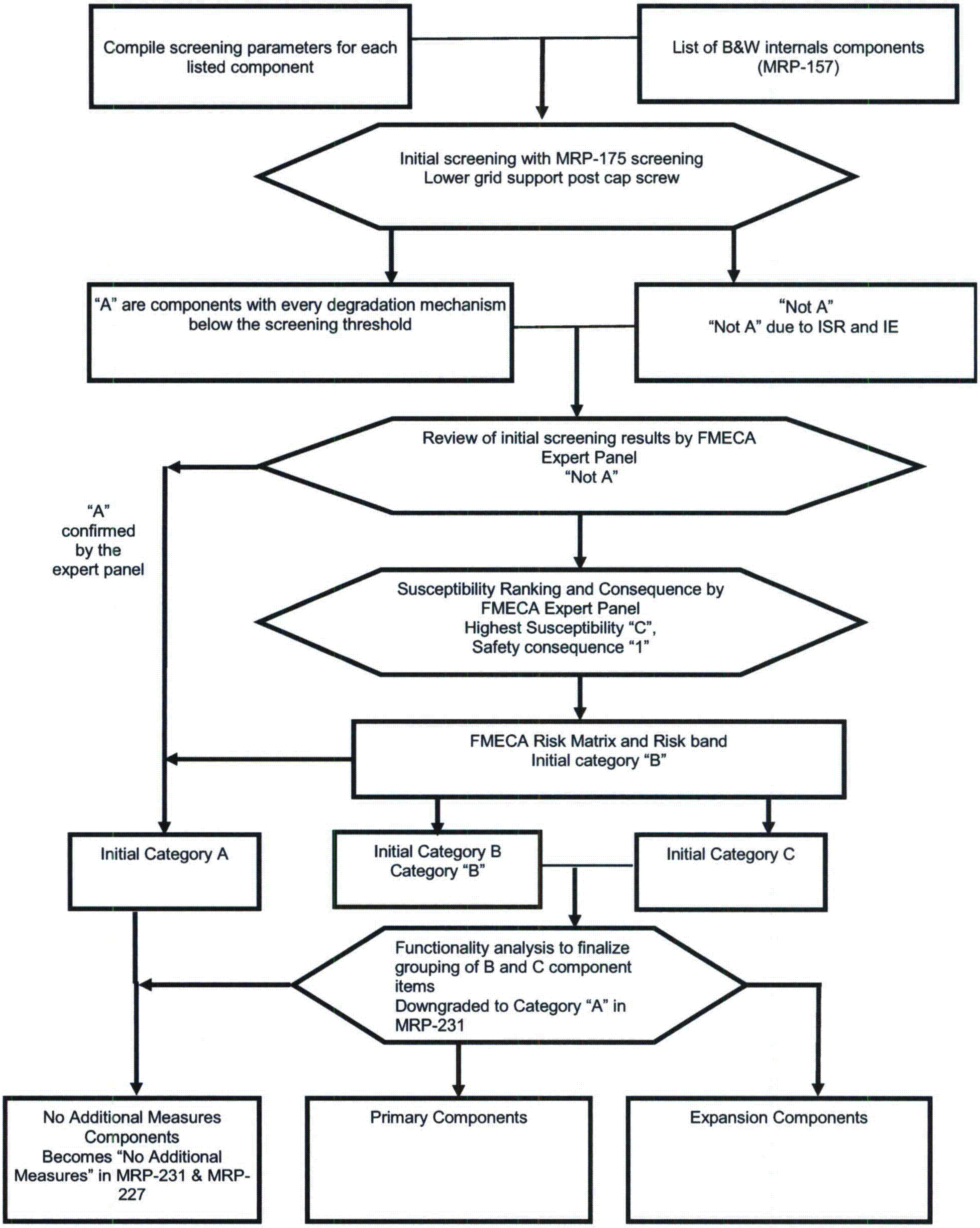

Figure 13, Flowchart for lower grid support post cap screw (based on MRP-189 Figure 1-3 flowchart)

Example 13: Flow distributor bolts

As defined, the purpose of the flow distributor bolts is to secure the flow distributor assembly to the reactor vessel lower internals. The flow distributor assembly is used to direct flow into the RV core and to provide support and alignment for the in-core monitoring instrumentation guide tubes. The flow distributor bolts support the deadweight of the flow distributor head and flange, IMI guide tubes, IMI guide tube support plate and the clamping ring. The flow distributor bolts do not provide a core support function. Therefore, failure of a single or even multiple flow distributor bolts would not necessarily prevent the flow distributor assembly from performing its function.

Initially screened in as <u>Non-A</u> and ultimately grouped as <u>Expansion</u>

- Screened in as <u>Non-A</u> for SCC in Step 3 (age-hardenable stainless steel, Alloy A-286, except TMI-1, which is Alloy X-750 material), all other mechanisms screened out

- FMECA results identified SCC susceptibility as "D" and safety consequences as "1," which preliminarily categorizes this item as "Category B" for a few bolts being failed (see Figure 1 in Step 5)

- However, the FMECA team also discussed cascading failures of bolts, and raised the safety consequences to "3," which led to a preliminary categorization of "Category C" for this situation (see Figure 1 in Step 5)

- As shown in Section 2.2 (MRP-231), the flow distributor bolts are predicted to have a lower SCC susceptibility than the UCB and LCB bolts, and thus its SCC category is downgraded to "Category B."

- As shown in Section 3.2.4 (MRP-231), of the six structural bolting rings and the lower grid shock pad bolts, only the UCB and LCB bolting have a core support function. Therefore, the UCB and LCB bolts are ultimately grouped as <u>Primary</u> items and the flow distributor bolts (and other structural bolt locations) are ultimately grouped as an <u>Expansion</u> items.

The accompanying flow chart is provided as Figure 14 below.

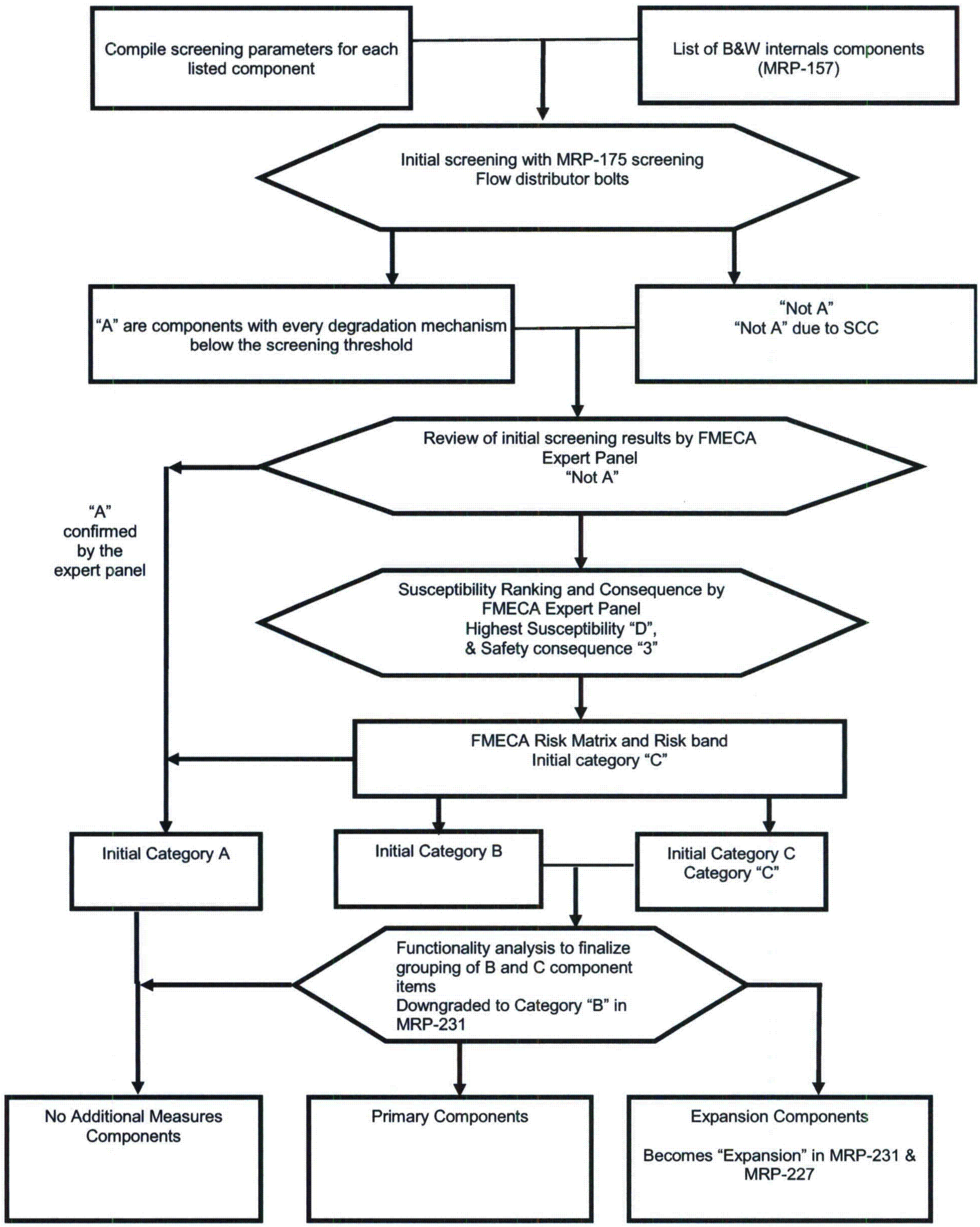

Figure -14, Flowchart for flow distributor bolts (based on MRP-189 Figure 1-3 flowchart)

7.2 W and CE & W

To facilitate the development of the aging management recommendations, the Westinghouse B and C components were grouped by the following list of assemblies:

- Baffle-Former
- Core Barrel
- Lower Core Plate
- Lower Core Support Structure
- Control Rod Guide Tube Assembly
- BMI System
- Flux Thimbles
- Upper Support Plate Assembly
- Alignment and Interfacing Components

Section 4.2 of MRP-232 is organized into subsections by this list of assemblies. The potential degradation mechanisms for the components in each assembly are discussed and recommendations provided. The recommendations are based on multiple factors including data collected in the screening and FMECA processes and the results of the functionality analyseis and data on the degradation mechanisms. The following sequence test describes this effort as a sequential process to clarify the underlying logic. The actual activities were carried out in parallel and involved complex interactions.

7.2.1 Basis for Primary, Expansion, Existing Programs and No Additional Measures Determination

The Category B and C components remaining in the pool following this process of elimination all have at least one identified mechanism that could potentially degrade their function. All of these components were considered in the comprehensive aging management strategy that combines existing inspection and monitoring programs with a set of newly defined programs.

The Category B/C classification indicates the severity of the potential degradation mechanism, however, it provides little guidance about the timing of the degradation or the relation to similar degradation mechanisms in other components. To provide the basis for the development of reactor internals inspection guidelines, the remaining degradation mechanisms were sorted into the following four functional groups described above; Primary, Expansion, Existing Programs and No Additional Measures.

An effective aging management strategy requires a coordinated set of recommendations. Within the Westinghouse reactor internals design, there are 29 Category B and C items as listed in MRP-227 Table 3-3. There are multiple identified degradation mechanisms for each of these components, bringing the total number of identified degradation mechanism/component pairings in the Westinghouse design to 62. Within this set of identified degradation issues there remains significant variation in both the predicted timing of and the impact of the effect. The development of the inspection strategy for the Westinghouse reactor internals is described in Section 4 of MRP-232.

The key to developing an efficient aging management strategy is to utilize appropriate groupings of components and degradation mechanisms that will allow a common strategy to be applied to multiple components. These groupings allow the aging management strategy to take advantage of the "waterfall" effect, where inspection of a Primary component can be shown to provide a leading indicator or reasonable sample for degradation in related Eexpansion components. The relationships between the Primary and Expansion components must be defined in terms of the relationships between the identified degradation mechanisms. Tables 12 through 19 summarize the final sorting of the screened-in components into inspection categories for each degradation mechanism.

The determination that a potential damage mechanism could be placed in the No Additional Measures Category was based on the Functionality Analysis, as described in Section 6.3. The determination that a mechanism was resolved by analysis did not change the Category B/C classification for the component, which is based on the consideration of the most severe degradation concerns. In some cases, a degradation mode in a Category C component may be identified as "No Aadditional mMeasures" because it had no impact on the potential component function. This would generally imply that the degradation mechanism was not the limiting concern that resulted in the Category C classification.

In the course of the evaluation, it was determined that there were several potential degradation concerns that were already adequately managed either through the existing ASME Section XI examinations or through other repair or replacement programs that had been implemented across the industry. These items were all placed in the Existing Programs category.

Application of this process to the Bottom Mounted Instrument Column Bodies is provided in Example 5.

Example 514: Bottom Mounted Instrumentation Column Bodies listed as Expansion Item

Original screening results: MRP-191 Table 5-1

- SCC, IASCC, Irradiation Embrittlement, Fatigue, Void Swelling

Functional Description:

- MRP-232 Section 4.2.6: BMI column assemblies provide a path for the flux thimbles into the core from the bottom of the vessel and protect the flux thimbles during operation of the reactor.

FMECA Conclusion: MRP-191 Table 6-5

- Medium Failure Probability, Low Consequence

Analysis of Degradation Mechanisms: MRP-232 Section 4.2.6.1

- Expansion based on cracking in CRGT lower flanges
  - The primary function of the BMI columns is to allow insertion and withdrawal of the flux thimbles, and as was noted several times, failures within the columns would be indicated by difficulty with the insertion of the flux

thimbles during a refueling outage. Thus, detailed inspections are not required, and this component is classified as being an Expansion inspection component, required only when the regular withdrawal and insertion of the flux thimble indicates malfunction.

– Analysis of lower core plate indicated irradiation effects are overestimated.
– BMI system has no structural function.

7.2.2 Development of Inspection Recommendations

Inspection strategies were designated for all of the Primary and Expansion components. These strategies were developed by Westinghouse engineering staff and subjected to a common internal peer review committee. To facilitate the process, the Category B and Category C components were regrouped into the following assemblies:

- Westinghouse
  - Baffle-Former
  - Bottom Mounted Instrumentation
  - Control Rod Guide Tube and Upper Internals
  - Core Barrel and Thermal Shield
  - Lower Support Plate and Support Columns
  - Interfacing Components
- CE
  - Control Element Assemblies Upper Internals
  - Core Shroud
  - Core Support Barrel
  - Lower Support Structure

Section 4.2 of MRP-232 contains subsections for each assembly grouping with detailed documentation supporting the aging management recommendations.

7.2.3 Basis for Inspection Method

The instructions given for the determination of an appropriate inspection method are defined in Section 2.5 of MRP-232. Although Westinghouse recommended VT-1 examinations for the detection of surface-breaking cracks, the MRP concluded that the use of the EVT-1 standard would be more appropriate and consistent with current practice for detecting stress corrosion cracking in BWR internals. This change is incorporated in MRP-227. Further discussion of the inspection methods is provided in MRP-228.

7.2.4 Degradation Mechanisms with No Direct Inspection Requirements

The proposed inspection methods are appropriate for degradation when cracking is the primary effect. The cracking-related mechanisms would include SCC, IASCC and fatigue. The VT-3 examination can also be used to detect visible signs of wear. Gross deformation due to swelling may also be detectable in a visual exam, but effects of swelling (i.e. stress) may occur before the deformation is observable. However, there is no non-destructive inspection technique capable of detecting thermal or irradiation embrittlement. At this time there is no practical way to monitor stress relaxation by measuring loads in reactor internal bolting. Although MRP-227 has identified irradiation embrittlement, thermal embrittlement, void swelling and irradiation induced stress relaxation as Pprimary or eExpansion degradation mechanisms for multiple components, there are no effective inspections techniques for these mechanisms. Although there are no inspection requirements for these components' aging management strategies for the degradation are required.

The aging management strategies for void swelling and stress relaxation must rely on detection of the secondary consequences of these mechanisms. The irradiation aging analysis conducted on the baffle-former structure provides the basis for determining these consequences. The aging analysis does suggest relative displacement along seams in the baffle structure that may be directly observable. The only other observable consequence of void swelling in the baffle-former-barrel assembly is IASCC failure of baffle-former bolts and baffle-edge bolts caused by swelling in the former plates. The timing of the failure is affected by compensating loss of load due to stress relaxation. Therefore, inspections of the bolting systems for IASCC failure provide an indicator of these related degradation mechanisms. Void swelling and stress relaxation are not listed in MRP-227 as aging effects monitored in the bolt examinations because they are not directly observed in the examination.

The aging management strategies for thermal embrittlement and irradiation embrittlement rely largely on trend curves compiled from laboratory data. Embrittlement can lead to loss of toughness that reduces the flaw tolerance of the materials. This loss of toughness can have a drastic effect on the acceptable flaw size in the component. Section 6.2.2 of MRP-227 provides guidance on fracture mechanics analysis of irradiated components. Because the irradiated components and thermally embrittled components have a reduced flaw tolerance, it is particularly important that any active cracking mechanism in these components be actively managed. In the inspection strategy, every component with an identified embrittlement concern has a corresponding requirement for inspection related to one or more potential cracking mechanism.

7.2.5 Basis for Inspection Time and Interval

The objective of the screening evaluation process was to identify components and locations where aging-related degradation could impair plant function. Operating experience with reactor internals has been generally positive. Therefore, there is no basis for establishing a risk-based inspection program. The irradiation aging study and other functionality analyseis can provide some general insights into the process and rate of component degradation. However, given the lack of established failure rates, the selection of inspection times and intervals is based largely on engineering judgment. These recommendations are included in the general inspection guidelines suggested in Section 4.2 of MRP-227.

7.2.6 Influence of Irradiation and Thermal Embrittlement on Inspection Timing

The MRP-227 recommendations do not include any inspections to detect the presence of irradiation or thermal embrittlement. There is ample experimental data to demonstrate that irradiation embrittlement will occur in all of the wrought stainless steel components that exceed MRP-175 screening fluence. In the most highly irradiated sections of the baffle structure, embrittlement will occur in the first few years of reactor operation. The region subject to irradiation embrittlement will grow over time. This behavior is evident in the irradiation aging analysis. Similarly, there is sufficient data on thermal embrittlement to suggest that ferritic steels with high ferrite contents will gradually lose toughness over the life of the internals. The MRP-227 recommendations reasonably assume that these changes in material properties will occur under the described conditions. The timing of the inspection strategy is not determined by the need to detect embrittlement.

Loss of fracture toughness due to irradiation or thermal embrittlement does result in increased emphasis on the detection of cracks and other flaws in the component. The inspection recommendations do recognize the need to inspect for potential cracking in embrittled components. In this case, the time of the inspection is determined by the onset of the cracking mechanism.

7.2.7 Influence of Void Swelling and Irradiation Induced Stress Relaxation/Creep on Inspection Time and Interval

Concerns about void swelling and stress relaxation/creep are effectively limited to the baffle-former-barrel assembly. The MRP-227 inspections do include some visual inspections of this assembly to identify gross distortion caused by void swelling. The intention of this inspection is to encourage general monitoring for the effects of void swelling later in life. Although the recommendation provides a broad window based on the number of effective full power years of operation for the initial inspection, the 10 EFPY inspection interval provides regular monitoring during the period of license renewal.

Differential swelling can have a significant effect on the stress distributions in the Westinghouse baffle-former structure. The effects of void swelling and irradiation-induced stress relaxation on the stresses and strains in the baffle-former assembly are calculated in the irradiation aging analysis. The relatively complex stress histories are the basis for the evaluation of IASCC susceptibility in the baffle-former bolts. However, there are no requirements for detection of local swelling or stress relaxation effects because they are not directly observable. Therefore, these calculations do not directly impact the timing of the proposed inspections.

When stress relaxation of bolted structures is a potential degradation mechanism, there are associated concerns about fatigue and wear. The impact of stress relaxation in the core barrel bolts was a factor in the timing consideration for these bolts. Although it is possible that some bolts in the core barrel will experience significant load loss during the first forty years of operation, the overall system of bolts is expected to maintain load carrying capability.

7.2.8 Influence of SCC, IASCC and Fatigue on Inspection Time and Interval

The majority of the MRP-227 inspection recommendations are intended to detect cracking due to one or more of the three cracking-related mechanisms: SCC, IASCC and fatigue. Therefore, the timing of the required inspections is controlled by the cracking mechanisms. Where multiple cracking mechanisms are concerned, the most limiting recommendation was controlling.

Although the regulatory and Ccode requirements for fatigue qualification have evolved over time, all plants currently operating in the US were designed and licensed for forty years of operation. The design requirements include the ability to maintain function through the normally expected fatigue cycles. Problems with vibration and high cycle fatigue were encountered and resolved early in plant life. There is no existing operating experience or analysis that suggests that the reactor internals are subject to fatigue cracking in the first forty years of operation. The Westinghouse recommendations to inspect for fatigue cracking within two refueling cycles of entering license renewal are meant to provide a basis for the period of license extension. Fatigue-related issues during the period of license renewal may also be addressed by time-limited aging analysis (TLAA). Should inspections of the operating fleet indicate fatigue related failures in the reactor internals components, the MRP would consider more frequent inspections.

Type 304 and Type 316 stainless steels are used extensively in the primary system of a Westinghouse plant. Stress corrosion cracking failures of these alloys in primary systems is highly unusual and generally associated with specialized local conditions. There is no reason to believe that the reactor internals are more susceptible to primary water SCC than other stainless steel components in the reactor primary system. The upper core barrel flange weld was selected as a region of potentially high stress that would provide an accessible inspection sample suitable for monitoring SCC of stainless steel in the Westinghouse internals. The Westinghouse recommendations to inspect for SCC of stainless steel within two refueling cycles of entering license renewal are meant to provide a basis for the period of license extension. The interval for subsequent inspections was chosen to be consistent with the ASME Section XI inspection cycle. The MRP and the PWROG have undertaken additional studies of primary water SCC in stainless steels. Should the industry studies or the MRP-227 inspections indicate SCC-related concerns in the reactor internals, the MRP would consider more frequent inspections.

Stress corrosion cracking of high strength nickel-based alloys has led to replacement of flexures in the control rod guide tube assemblies and guide tube support pins. The flexures are no longer a concern because they have been universally replaced with flexureless inserts. The guide tube support pins have either been replaced with Alloy X-750 pins with an improved heat treatment or with Type 316 stainless steel pins with a modified design. The utilities are responsible to establish, or are working with their equipment vendors to establish appropriate monitoring of the replacement items. Similar failures have been recently reported in Alloy X-750 bolts used to secure clevis inserts to the guide lugs. These failures were discovered in the course of a normal ASME Section XI examination. No safety issues were identified, and the plant returned to operation for another cycle without removing or replacing the broken bolts. The MRP-227 recommendations list inspection of the clevis insert for wear resulting from failure of the Alloy X-750 bolts as an Existing Programs component. The MRP has established training procedures to make inspectors aware of this type of operating history. Should additional failures occur, the MRP would consider more frequent inspections.

The irradiation aging analysis described in MRP-232 provided an estimate of the number and locations of bolts exceeding the IASCC threshold stress as a function of plant operating history. These bolts are the reactor internals components subjected to the most severe combinations of irradiation exposure and stress. The irradiation aging analysis indicated that the period of time when the plant operated with "out-in" core loading patterns caused the highest rates of irradiation-induced bolt loading and potential IASCC. The power history assumed for the aging analysis included 30 years of operation at full power with these high leakage core loading patterns. Beyond thirty years of operation, when low-leakage core loading patterns were assumed the bolt loads were observed to faill. Therefore, in the irradiation aging analysis, most of the IASCC failures occurred beyond 30 effective full-power years (EFPY) of operation.

Westinghouse worked with the Owners' Group to conduct several major studies of IASCC failures in baffle-former bolts during the 1990's. These studies, which were conducted in response to reports of failed bolting in several French plants, included both inspections of operating plants and assessments of the effect of bolt failures on plant operation. Inspections conducted after approximately 20 EFPY at Point Beach, Farley, and Ginna reported relatively low bolt failure rates. The plant assessments indicated that there was not an immediate safety issue related to IASCC failures in baffle-former bolting. In the Safety Evaluation of WCAP-15029, the NRC concluded that:

> *Finally, in consideration of the WOG assessment and conclusion that the baffle bolt issue is not an immediate safety concern and that it is appropriate to treat baffle former bolt degradation as an aging management issue, subsequent to replacement of baffle bolts, licensees are expected to develop an appropriate inspection monitoring and aging management program for baffle bolting.*

MRP-227 recommends inspection of the baffle-former bolts for cracking between 25-35 EFPY. The intention of this inspection is to establish a basis for aging management of the baffle-former bolts during the period of license extension. The lower exposure limit was selected based on the previous inspection experience, which indicated acceptable rates of bolt failure at 20 EFPY. The upper exposure limit was selected to provide a baseline consistent with the peak damage in the irradiation aging analysis. The irradiation aging analysis indicated diminishing rates of bolt failure in the later stages of plant life. Therefore the recommendation is to provide a subsequent inspection after 10-15 additional EFPY to demonstrate the stability of the bolting pattern.

7.2.9 Influence of Wear on Inspection Time and Interval

Many of the wear related examinations are addressed by the ASME Section XI. The schedule for the remaining wear mechanisms follows a similar requirement.

Although the current MRP-227 recommendations for wear in the control rod guide tube assembly follow the ASME Section XI examination schedule, inspection requirements for wear in the control rod guide tube assembly are being actively reviewed by the PWROG. Should changes in this recommendation occur, it is anticipated that they would be implemented through the NEI-03-08 protocol.

8.0 Step 8: Preparation of MRP-227 I&E Guidelines

The final step involved taking the results of the NSSS vendor's work and recommendations and developing the final approach for managing aging of reactor internals. The NSSS recommendations are discussed in Section 7.0 above and can be found in MRP-231 and MRP-232. The MRP Core Writers Group, composed of utility representatives, including early license renewal applicants, and other technical consultants, reviewed the recommendations for adequacy and to assure that the proposed recommendations could be accomplished. The NSSS recommendations were then placed into MRP-227 as appropriate. For example Table 3-8 from MRP-231 translates into Table 3-1 in MRP-227, Table 3-9 from MRP-231 translates into Table 4-1 of MRP-227, and Table 3-10 of MRP-231 translates into Table 4-4 of MRP-227. A similar process was used to move information from MRP-232 into MRP-227. The final industry positions were documented in MRP-227 and approved through the MRP process. MRP-227 was approved with "needed" requirements as defined in NEI 03-08 and will be implemented by all domestic PWR utilities consistent with those requirements.

9.0 References

The following is a list of the documents discussed in this roadmap, including the revision that is applicable.

1. *Pressurized Water Reactor Issue Management Table, PWR-IMT,* Consequence of Failure (MRP-156), EPRI, Palo Alto, CA: 2005, 1012110.
2. *Updated B&W Design Information for the Issue Management Tables* (MRP-157). EPRI, Palo Alto, CA: 2005. 1012132
3. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values* (MRP-175). EPRI, Palo Alto, CA: 2005. 1012081.
4. *Materials Reliability Program: Screening, Categorization, and Ranking of B&W-Designed PWR Internals* (MRP-189-Rev. 1). EPRI, Palo Alto, CA: 2009. 1018292.
5. *Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals* (MRP-190). EPRI, Palo Alto, CA: 2006. 1013233.
6. *Screening, Categorization, and Ranking of reactor Internals Components for Westinghouse and Combustion Engineering PWR Designs* (MRP-191). EPRI, Palo Alto, CA: 2006. 1013234.
7. *Pressurized Water Reactor Issue Management Tables* (MRP-205, rev 1). EPRI, Palo Alto, CA: 2006. 1014446.
8. *Inspection Standard for Reactor Internals Components* (MRP-228). EPRI, Palo Alto, CA: 2009. 1016609
9. *Materials Reliability Program: Functionality Analysis for B&W Representative PWR Internals* (MRP-229). EPRI, Palo Alto, CA: 2008. 1016598.

10. *Functionality Analysis for Westinghouse and Combustion Engineering Representatives PWR Internals* (MRP-230, rev 0). EPRI, Palo Alto, CA: 2008. 1016597.
11. *Materials Reliability Program: Aging Management Strategies for B&W PWR Internals* (MRP-231). EPRI, Palo Alto, CA: 2008. 1016592.
12. *Reactor Internals Aging Management Strategy Reports – Westinghouse/Combustion Engineering Designs* (MRP-232 rev 0). EPRI, Palo Alto, CA: 2008. 1016593
13. *Pressurized Water Reactor Internals Inspection and Evaluation Guidelines* (MRP-227 rev 0). EPRI, Palo Alto, CA: 2008. 1016596.
14. *Demonstration of the Management of Aging Effects for the Reactor Vessel Internals*, BAW-2248, July 1997.
15. CE NPSD-1216
16. WCAP-14577-R1-A
17. WCAP-15029
18. ASME Boiler & Pressure Vessel Code, Section XI, Division 1, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, NY, 2001 Edition, Plus 2003 Addenda, or later.
19. 10CFR 50.54– Codes and Standards, Title 10 (Energy), Part 50 (Domestic Licensing of Production and Utilization Facilities) of the Code of Federal Regulations
20. 10CFR 50.55a – Codes and Standards, Title 10 (Energy), Part 50 (Domestic Licensing of Production and Utilization Facilities) of the Code of Federal Regulations
21. *EPRI Materials Degradation Matrix – Revision 1*. EPRI, Palo Alto, CA: 2008. 1016486.

C MRP-227-A CHANGES

The changes from MRP-227 Revision 0 incorporated in this revision are summarized below.

Page	Section	Description of Change
iii	Front Matter	Added U. S. NRC Safety Evaluation
vii, viii, 2-1, 2-4	Report Summary, 2.1, 2.4	Revised first paragraph to clarify that I&E guidance applies both to development of reactor internals aging management engineering programs to satisfy industry commitments under NEI 03-08 and to development of Aging Management Programs (AMP) to support License Renewal commitments.
1-1	1.0	Deleted reference to Good Practice requirement since the only Good practice requirement from MRP-227 Revision 0 has been changed to a Needed requirement (reporting of results) per RAI set 4 response.
2-1	2.1	Added reference to NUREG 1801, Chapter XI.M16A to satisfy Topical Report Condition 7 in the Safety Evaluation (SE).
2-3	2.2, Figure 2-2	Changed “functionality analysis” to “functionality assessment” for accuracy.
3-16	3.3.2	Updated the number of Primary and Expansion category components to reflect changes resulting from SE Topical Report Conditions 1, 2 and 3.
3-17	Table 3-1	Changed B&W Core Support Shield Cast Outlet Nozzles to “No Additional Measures” since all components in the fleet screen out from thermal embrittlement effect due to low delta ferrite content.
3-17	Table 3-1	Deleted B&W Core Support Shield vent valve discs and vent valve disc shaft/hinge pins from table since these are active components not subject to aging management
3-20	Table 3-1	Changed designation of B&W Flow Distributor (FD) Bolts from Expansion to Primary in accordance with SE Topical Report Condition 3. Added note 5 to reflect this change for reconciliation with functionality assessment.
3-21	Table 3-2	Changed designation of CE Core Support Columns (wrought and cast) from Existing to Primary in accordance with SE Topical Report Condition 3. Added note 4 to reflect this change for reconciliation with functionality assessment.
3-21	Table 3-2	Added CE Lower Core Support Beams as an Expansion component in accordance with SE Topical Report Condition 1. Added note 5 to reflect this change for reconciliation with functionality assessment.
3-21	Table 3-2	Added CE Core Support Barrel Assembly Upper Cylinder as an Expansion component in accordance with SE Topical Report Condition 1. Added note 5 to reflect this change for reconciliation with functionality assessment.
3-22	Table 3-2	Changed designation of CE Lower Cylinder Welds from Existing to Primary in accordance with SE Topical Report Condition 2. Added note 4 to reflect this change for reconciliation with functionality assessment.
3-22	Table 3-2	Added CE Core Support Barrel Assembly Upper Core Barrel Flange as

Page	Section	Description of Change
		an Expansion component in accordance with SE Topical Report Condition 1. Added note 5 to reflect this change for reconciliation with functionality assessment.
3-24	Table 3-3	Added Upper Core Plate as an Expansion component in accordance with SE Topical Report Condition 1. Added note 5 to reflect this change for reconciliation with functionality assessment.
3-25	Table 3-3	Changed designation of W Core Barrel Welds from Existing to Primary in accordance with SE Topical Report Condition 2. Added note 6 to reflect this change for reconciliation with functionality assessment.
3-25	Table 3-3	Added Lower Support Casting and Forgings as Expansion components in accordance with SE Topical Report Condition 1. Added note 5 to reflect this change for reconciliation with functionality assessment.
4-1	4.0	Deleted reference to Appendix A in accordance with RAI Set 4 response that Appendix A will contain Operating Experience and not aging management program contents guidance.
4-3	4.1.3	Revised text on inspection qualification requirements in accordance with response to RAI set 3.
4-7	4.3.1	Added clarifying note that bolts associated with locking devices are examined by volumetric (UT) inspection.
4-8	4.3.1	Elevated B&W Control Rod Guide Tube spacer castings from Expansion to Primary since previous Primary links have been deleted (see below).
4-9	4.3.1	Deleted B&W Core Support Shield Cast Outlet Nozzles since all components in the fleet screen out from thermal embrittlement effect due to low delta ferrite content.
4-9	4.3.1	Deleted B&W Core Support Shield vent valve discs and vent valve disc shaft/hinge pins since these are active components not subject to aging management.
4-10	4.3.1	Revised note on B&W FD bolt locking devices to reflect that these locking devices may be inspected during the FD bolt Primary component inspection in accordance with SE Topical Report Condition 3.
4-10	4.3.1	Added B& W FD bolt locking devices as a Primary component in accordance with SE Topical Report Condition 3.
4-11	4.3.1	Added B&W FD bolts as a Primary component in accordance with SE Topical Report Condition 3.
4-13	4.3.2	Added CE core support column welds as a Primary component in accordance with SE Topical Report Condition 3.
4-14	4.3.2	Under CE upper core support barrel flange weld, added lower core support beams, core support barrel assembly upper cylinder and core support barrel assembly upper core barrel flange as Expansion components in accordance with SE Topical Condition 1. Deleted Expansion to core support barrel assembly lower cylinder welds and core support column welds in accordance with SE Topical Report Conditions 2 and 3 (respectively).
4-14	4.3.2	Added CE core support barrel assembly lower cylinder girth welds as a Primary component in accordance with SE Topical Report Condition 2. Added lower cylinder axial welds as an Expansion component linked to the girth welds.
4-16	4.3.3	Under W upper core barrel flange weld Primary inspection entry,

Page	Section	Description of Change
		deleted Expansion to remaining core barrel welds in accordance with SE Topical Report Condition 2.
4-16	4.3.3	Added as Primary components W lower core barrel flange weld and upper and lower core barrel girth welds in accordance with SE Topical Report Conditions 2. Added upper and lower core barrel axial welds as an Expansion component linked to the girth welds.
4-16	4.3.3	Under W CRGT lower flange weld Primary inspection entry, added upper core plate and lower support forging/casting in accordance with SE Topical Report Condition 1.
4-18	Table 4-1	Moved B&W **Control Rod Guide Tube** <u>CRGT spacer castings</u> from Expansion to Primary since previous Primary links have been deleted (see below). Revised description of coverage requirement for clarity.
4-18	Table 4-1	Deleted B&W Core Support Shield Cast Outlet Nozzles since all components in the fleet screen out from thermal embrittlement effect due to low delta ferrite content.
4-19	Table 4-1	Deleted B&W Core Support Shield vent valve discs and vent valve disc shaft/hinge pins since these are active components not subject to aging management.
4-19	Table 4-1	Per response to RAI set 4, added locking devices to table entries for bolts.
4-20	Table 4-1	In accordance with response to RAI set 4, added clarifying note under the *Effect (Mechanism)* column for the B&W **Core Barrel Assembly** <u>Baffle-to-former bolts</u> entry.
4-21	Table 4-1	Added B&W <u>FD bolts</u> as a Primary inspection component in accordance with SE Topical Report Condition 3.
4-21	Table 4-1	For clarity and completeness, added Expansion to <u>Alloy X-750 dowel-to-plenum cover weldment rib pad welds for ONS-1 only</u> under B&W **Lower Grid Assembly** <u>Alloy X-750 dowel-to-guide lock welds</u> entry, per response to RAI set 4.
4-21	Table 4-1	Revised coverage requirement for B&W **Lower Grid Assembly:** <u>Alloy X-750 dowel-to-guide block welds</u> entry for clarity.
4-21	Table 4-1	Revised coverage requirement for B&W **IMI Guide Tube Assembly:** <u>IMI guide tube spiders</u> entry for clarity and deleted Expansion link to CRGT spacer castings since these components have been elevated to Primary.
4-22	Table 4-1	In accordance with response to RAI set 4, added 75% minimum coverage requirement for specific Primary component inspections. Requirement added as a note to the table.
4-23 to 27	Table 4-2	In accordance with response to RAI set 4, added Aging Management information under the *Effect (Mechanism)* column for specific entries.
4-23	Table 4-2	Under the *Examination Method/Frequency* column for the CE **Core Shroud Assembly (Bolted)** <u>core shroud bolts</u> entry, revised the re-examination frequency to a ten year interval in accordance with SE Topical Report Condition 5.

Page	Section	Description of Change
4-25	Table 4-2	Under the *Expansion Link* column for the CE **Core Support Barrel Assembly** Upper (core support barrel) flange weld entry, added lower core support beams, core support barrel assembly upper cylinder and core support barrel assembly upper core barrel flange as Expansion components in accordance with SE Topical Condition 1. Deleted Expansion to remaining core barrel assembly welds and core support column welds in accordance with SE Topical Report Conditions 2 and 3 (respectively)
4-25	Table 4-2	Added entry for CE **Core Support Barrel Assembly** lower cylinder girth welds as a Primary component in accordance with SE Topical Report Condition 2. Added lower cylinder axial welds as an Expansion component linked to the girth welds.
4-25	Table 4-2	Added entry for CE **Lower Support Structure** core support column welds as a Primary component in accordance with SE Topical Report Condition 3.
4-26	Table 4-2	Under *Examination Coverage* column for the lower flange weld, core support plate and fuel alignment plate entries, deleted reference to a plant-specific fatigue analysis, as discussed in response to RAI set 4.
4-27	Table 4-2	In accordance with response to RAI set 4, added 75% minimum coverage requirement for specific Primary component inspections. Requirement added as notes to the table.
4-28 to 31	Table 4-3	In accordance with response to RAI set 4, added Aging Management information under the *Effect (Mechanism)* column for specific entries.
4-28	Table 4-3	Under the *Expansion Link* column for the W **Control Rod Guide Tube Assembly** Lower flange weld entry, added Upper core plate and lower support forging/casting as Expansion components in accordance with SE Topical Condition 1.
4-28	Table 4-3	Under the *Expansion Link* column for the W **Core Barrel Assembly** Upper core barrel flange weld entry, deleted Remaining core barrel welds as Expansion components in accordance with SE Topical Condition 2.
4-28	Table 4-3	Added entry for W **Core Barrel Assembly** upper and lower core barrel cylinder girth welds as Primary components in accordance with SE Topical Report Condition 2. Added upper and lower core barrel cylinder axial welds as an Expansion component linked to the girth welds.
4-29	Table 4-3	Added entry for W **Core Barrel Assembly** lower core barrel flange weld as a Primary component in accordance with SE Topical Report Condition 2.
4-29	Table 4-3	Deleted "or as supported by plant specific justification" from the *Examination Coverage* column for the **Baffle-Former Assembly** Baffle-former bolts entry in accordance with the response to RAI set 4.
4-30	Table 4-3	Adding clarifying text under the *Item* column for the **Baffle-Former Assembly** Assembly entry in accordance with the response to RAI set 4.
4-30	Table 4-3	Deleted the text "Replacement of 304 springs by 403 springs is required when the spring stiffness is determined to relax beyond design tolerance" under the *Examination Coverage* column for the **Alignment and Interfacing Components** Internals hold down spring entry in accordance with the response to RAI set 4.

Page	Section	Description of Change
4-31	Table 4-3	In accordance with response to RAI set 4, added 75% minimum coverage requirement for specific Primary component inspections. Requirement added as notes to the table.
4-32 to 34	Table 4-4	Under the *Examination Method* column, added the requirement for a 10-year re-inspection interval for specific Expansion components in accordance with SE Topical Report Condition 6.
4-32 to 35	Table 4-4	Under the *Examination Coverage* column, added the requirement for 75% minimum coverage for specific Expansion components in accordance with SE Topical Report Condition 5. Requirement added as note 2 to the Table.
4-32 to 34	Table 4-4	Per response to RAI set 4, added locking devices to table entries for bolts.
4-33 and 35	Table 4-4	Added FD bolts as a *Primary Link* for the B&W **Core Barrel Assembly** UTS bolts , SSHT bolts, Lower grid shock pad bolts and LTS bolts entries in accordance with SE Topical Report Condition 3.
4-32	Table 4-4	Revised description of coverage requirement for B&W **Upper Grid Assembly:** Alloy X-750 dowel-to-upper fuel assembly support pad welds entry for clarity.
4-32	Table 4-4	Moved B&W Control Rod Guide Tube spacer castings from Expansion to Primary since previous Primary links have been deleted.
4-32	Table 4-4	Added clarifying text "An acceptable examination technique currently not available" under the *Examination Coverage* column for the B&W **Core Barrel Assembly** Baffle-to-baffle bolt and Core barrel-to-former bolts entries.
4-33	Table 4-4	Revised description of coverage requirement for B&W **Lower Grid Assembly:** Lower fuel assembly support pad items entry for clarity.
4-34	Table 4-4	Revised description of coverage requirement for B&W **Lower Grid Assembly:** Alloy X-750 dowel-to-lower grid fuel assembly support pad welds entry for clarity.
4-36 to 41	Table 4-5	Under the *Examination Method* column, added the requirement for a 10-year re-inspection interval for specific Expansion components in accordance with SE Topical Report Condition 6.
4-36 to 41	Table 4-5	Under the *Examination Coverage* column, added the requirement for 75% minimum coverage for specific Expansion components in accordance with SE Topical Report Condition 5. Requirement added as a note to the Table.
4-36 to 41	Table 4-5	In accordance with response to RAI set 4, added Aging Management information under the *Effect (Mechanism)* column for specific entries.
4-36 and 37	Table 4-5	Added entries for CE **Core Support Barrel Assembly** upper cylinder, upper core barrel flange and **Lower Support Structure** lower core support beams as Expansion components in accordance with SE Topical Condition 1.
4-36	Table 4-5	Deleted entry **Core Support Barrel Assembly** Remaining core barrel assembly welds in accordance with SE Topical Report Condition 2 which raised this component from Expansion to Primary. Added entry applicable only to the lower cylinder axial welds.
4-37	Table 4-5	Deleted entry **Lower Support Structure** Core support column welds in accordance with SE Topical Report Condition 3 which raised this component from Expansion to Primary.
4-39 to 41	Table 4-6	Under the *Examination Method* column, added the requirement for a

Page	Section	Description of Change
		10-year re-inspection interval for specific Expansion components in accordance with SE Topical Report Condition 6.
4-39 to 41	Table 4-6	Under the *Examination Coverage* column, added the requirement for 75% minimum coverage for specific Expansion components in accordance with SE Topical Report Condition 5. Requirement added as a note to the Table.
4-39 to 41	Table 4-6	In accordance with response to RAI set 4, added Aging Management information under the *Effect (Mechanism)* column for specific entries.
4-39	Table 4-6	Added entries for W **Upper Internals Assembly** Upper core plate and **Core Barrel Assembly** lower support forging or castings as Expansion components in accordance with SE Topical Condition 1.
4-39	Table 4-6	Deleted entry **Core Barrel Assembly** Lower core barrel flange weld in accordance with SE Topical Report Condition 2 which raised this component from Expansion to Primary. Added entry applicable only to the upper and lower core barrel axial welds.
4-75	Table 4-8	In accordance with response to RAI set 4, added Aging Management information under the *Effect (Mechanism)* column for specific entries.
4-76	Table 4-9	In accordance with response to RAI set 4, added Aging Management information under the *Effect (Mechanism)* column for specific entries.
5-1	5.0	Added reference to WCAP 17096 to second paragraph, in accordance with response to RAI set 4.
5-2	Table 5-1	Added entry for B&W **Control Rod Guide Tube** CRGT spacer castings which have been raised from Expansion to Primary since previous Primary links have been deleted.
5-2	Table 5-1	Deleted B&W Core Support Shield Cast Outlet Nozzles since all components in the fleet screen out from thermal embrittlement effect due to low delta ferrite content.
5-3	Table 5-1	Deleted B&W Core Support Shield vent valve discs and vent valve disc shaft/hinge pins since these are active components not subject to aging management.
5-4 and 5-5	Table 5-1	Per response to RAI set 4, added locking devices to table entries for bolts.
5-8	Table 5-1	Added B&W FD bolts as a Primary inspection component in accordance with SE Topical Report Condition 3.
5-9	Table 5-2	Revised B&W **Incore Monitoring Instrumentation (IMI) Guide Tube Assembly** item description for clarity.
5-13	Table 5-2	Under the *Expansion Link* column for the CE **Core Support Barrel Assembly** Upper (core support barrel) flange weld entry, added lower core support beams, upper cylinder and upper core barrel flange as Expansion components in accordance with SE Topical Condition 1. Deleted Expansion to remaining core barrel assembly welds and core support column welds in accordance with SE Topical Report Conditions 2 and 3 (respectively)
5-13	Table 5-2	Added entry for CE **Core Support Barrel Assembly** lower cylinder girth welds as a Primary component in accordance with SE Topical Report Condition 2. Included expansion link to lower cylinder axial welds.
5-13	Table 5-2	Added entry for CE **Lower Support Structure** core support column welds as a Primary component in accordance with SE Topical Report Condition 3.

Page	Section	Description of Change
5-16	Table 5-3	Under the *Expansion Link* column for the W **Control Rod Guide Tube Assembly** Lower flange weld entry, added Upper core plate and lower support forging/casting as Expansion components in accordance with SE Topical Condition 1.
5-17	Table 5-3	Under the *Expansion Link* column for the W **Core Barrel Assembly** Upper core barrel flange weld entry, deleted Remaining core barrel welds as Expansion components in accordance with SE Topical Condition 2.
5-18	Table 5-3	Added entry for W **Core Barrel Assembly** upper and lower core barrel cylinder girth welds as Primary components in accordance with SE Topical Report Condition 2. Included expansion link to upper and lower core barrel axial welds.
5-17	Table 5-3	Added entry for W **Core Barrel Assembly** lower core barrel flange weld as a Primary component in accordance with SE Topical Report Condition 2.
6-1	6.0	Added reference to WCAP 17096 to third paragraph, in accordance with response to RAI set 4.
7-1	7.2	Revised to clarify that requirement applies to development of reactor internals aging management engineering programs to satisfy industry commitments under NEI 03-08.
7-2	7.3	Added text on NEI 03-08 implementation in accordance with response to RAI set 4.
7-2	7.6	Revised NEI 03-08 reporting requirement from "Good Practice" to "Needed" in accordance with response to RAI set 4.
7-3	7.7	Added new NEI 03-08 "Needed" requirement to use NRC-approved evaluation methodology to disposition examination results via an engineering evaluation, in accordance with response to RAI set 4.
8-2	References	Added WCAP 17096 and the NRC Safety Evaluation as references
A-1 to 7	Appendix A	Replaced existing Appendix A with an Operating Experience summary, in accordance with response to RAI set 4.
B-1	Appendix B	Added RAI and associated responses as Appendix B in accordance with the NRC Safety Evaluation.
C-1 to 7	Appendix C	Added revision summary in accordance with MRP Administrative Procedures (MRP-130, Rev. 2).

The Electric Power Research Institute Inc., (EPRI, www.epri.com) conducts research and development relating to the generation, delivery and use of electricity for the benefit of the public. An independent, nonprofit organization, EPRI brings together its scientists and engineers as well as experts from academia and industry to help address challenges in electricity, including reliability, efficiency, health, safety and the environment. EPRI also provides technology, policy and economic analyses to drive long-range research and development planning, and supports research in emerging technologies. EPRI's members represent more than 90 percent of the electricity generated and delivered in the United States, and international participation extends to 40 countries. EPRI's principal offices and laboratories are located in Palo Alto, Calif.; Charlotte, N.C.; Knoxville, Tenn.; and Lenox, Mass.

Together...Shaping the Future of Electricity

Programs:

Nuclear Power

Pressurized Water Reactor Materials Reliability Program

1022863

Electric Power Research Institute
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com