Table of Proposed Changes
NRC Draft Safety Evaluation of MRP-227
May 10, 2011

Item Number	Location in Proposed Revision	MRP Change or Issue	Discussion
Editorial Changes			
1	Various	Change "RVI" to "reactor internals."	"RVI" is a standard Industry acronym for reactor vessel integrity. To avoid potential for confusion, this acronym should not be used for reactor vessel internals. For clarity, this change was not tracked in the mark up.
2	Various		Minor editorial suggestions.
3	2.0, first paragraph	Delete IGSCC from degradation modes list.	IGSCC is considered a subset of SCC and may therefore be considered redundant.
4	2.2, fifth paragraph	Revise definition of Category A components.	Revise for consistency with definition in MRP-227.
5	2.2, eighth paragraph, bullet 4	Clarify definition of "No Additional Measures" components.	Binning a component in the "No Additional Measures" category under MRP-227 does not necessarily mean that no inspections will be performed to monitor the component. It simply means that no action is required from an aging management viewpoint. Inspections of core support structures are performed in accordance with ASME Section XI Code requirements.
6	2.3, fifth paragraph	Change "barrel" to "core barrel."	Revise for consistency with nomenclature of B&W plants.
7a	2.5, second paragraph, bullet 1	Change to indicate the *Mandatory* requirement is to develop a reactor internals program, rather than to implement MRP-227.	Revise for consistency with MRP-227.
7b	2.5, second paragraph, bullet 2	Change to indicate that the *Needed* requirement to implement the Tables in MRP-227 are tied to issuance of the Revision A version of MRP-227, rather than Revision 0.	Revise for consistency with MRP-227.

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15a, b, c	3.2.6.2 first and second paragraphs 4.2.2 first paragraph	Delete reference to Table 4-2 of MRP-189 and Table 4-4 of MRP-190; add reference to Table 4-4 of MRP-191.	Revised for correctness. B&W design plant components covered in MRP-227 are listed in Table 4-1; Table 4-2 covers associated weldments. There is no Table 4-4 in MRP-190. Rather, Tables 4-4 and 4-5 of MRP-191 list components covered in MRP-227 for Westinghouse and CE designed plants.
Technical Changes			
8	2.6	Change to indicate that Appendix A of MRP-227 merely summarizes GALL requirements for AMP submittals.	Appendix A of MRP-227 was not intended to require specific information for LRAs, but to provide guidance.
9a, 9c, 9e, 9f	3.2.2, heading and first, second and third paragraphs 4.1.1 heading and first paragraph	Change "High Consequence" to "Core Support"	The term "High Consequence" may be somewhat ambiguous with respect to whether the viewpoint is strictly overall aging management strategy, which considers both failure probability AND failure consequence, or simply failure consequence without consideration of probability. For clarity, the specific components in question (i.e., Core Support) should be cited to avoid potential ambiguity.
9b	3.2.2, first paragraph	Change generic description "components whose failure could cause significant safety consequences" to more specific description "components that are part of the core support structure".	As discussed above, more specific description of the components of concern is desirable to avoid potential ambiguity.
9d	3.2.2, second paragraph	Changed to clarify the results of the FMECA process for core support structures.	The FMECA panel concluded that aging related degradation of the applicable core support structure components would not necessarily result in loss of core support.

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9g	3.2.2 third paragraph 4.1.1 first paragraph	Changed to require designation of the applicable core support structure components as "Existing" Category rather than "Primary" Category.	Inspection of core support structures is already required by ASME Code Section XI. Although the FMECA analysis for these specific components did not indicate that inspections were required for aging management, MRP-227 can recognize that inspections are indeed performed under Section XI and therefore designate inspection of the components in the "Existing" Category.
10a, 10b	3.2.2, first and second paragraphs 4.4.1 first paragraph and table	Deleted reference to lower grid-to-core barrel bolts in B&W design plants and associated discussion of binning process used for B&W components.	The lower grid-to-core-barrel bolts in B&W-designed reactors are abbreviated as LCB bolts in MRP-227-Rev. 0, and are listed as "Primary" in MRP-227-Rev. 0 Table 4-1 (page 4-17). See Explanatory Note 1.
11a,b,c,d	3.2.3 first paragraph 4.1.2 second paragraph	Add text to clarify the results of the FMECA analysis for the Westinghouse upper flange upper and lower core cylinders and the CE core support barrel.	Revised for clarity to support follow-on discussion.
11e	3.2.3 first paragraph	Added discussion that designation of the core barrel welds as "Expansion" category components was based on reduction of IASCC susceptibility due to stress relaxation effects.	For accuracy, it should be noted that stress relaxation effects were considered in the original technical basis for the inspection requirements for these welds and other IASCC-susceptible components.
11f	3.2.3 second paragraph	Added discussion clarifying that only the core barrel/core support barrel welds in the beltline region potentially see sufficient neutron exposure to exceed the IASCC screening criteria.	This added text clarifies that only a limited region of the subject welds are in the high fluence regions and thus potentially subject to IASCC.
11g	3.2.3 second paragraph	Revised to read "... may not truly represent *a sufficiently robust link to* the extent ..."	Revised for technical clarity.

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11h	3.2.3 third paragraph	Replaced "high consequence components" with "Westinghouse core barrel and CE core support barrels"	See discussion in 9a above; more specific description of the components of concern is desirable to avoid potential ambiguity.
11i,j	3.2.3 third paragraph 4.1.2 second paragraph	Revise text to require susceptible circumferential beltline core support/core barrel welds to be inspected as "Primary" components, with an alternative to demonstrate acceptability through analysis.	This focuses inspection on the susceptible portions of the welds.
11k	3.2.3 third paragraph 4.1.2 second paragraph	Establish 75% inspection criteria for susceptible portions of the circumferential beltline core support/core barrel welds and acknowledge that accessibility of core barrel welds may be limited for some plant designs and provide an alternative to evaluate the welds as inaccessible components consistent with Applicant/Licensee Action Item 6.	Dependent on plant design, the areas of greatest interest (that is, highest fluence areas) may be inaccessible. It is thus desirable to acknowledge these potential limitations and provide a technically acceptable alternative in lieu of inspection.
12a	3.2.4 heading 4.1.3 heading	Change "High Consequence" to "Core Support."	See discussion in 9a above; more specific description of the components of concern is desirable to avoid potential ambiguity.
12b,c	3.2.4 second paragraph 4.1.3 first and second paragraphs	Revise second sentence to read "In addition to these degradation mechanisms, *in the absence of plant specific material specifications*, the casting component is *assumed to be* susceptible to thermal embrittlement."	Revised for correctness to reflect that thermal embrittlement susceptibility is a function of material composition.

Item Number	Location in Proposed Revision	MRP Change or Issue	Discussion
12d,e,f	3.2.4 second and third paragraphs 4.1.3 first and second paragraphs	Expanded discussion on the original basis for categorizing the CE lower support structure core support column welds in the "Expansion" Category.	MRP evaluation concluded that since only a fraction of the core support column welds are subject to sufficient fluence to potentially cause IASCC, and the stresses in the welds are generally compressive, these welds would not be a leading indicator of degradation. This was the basis for categorization of the welds as "Expansion" components.
12g	3.2.4 fourth paragraph 4.1.3 second paragraph	Added the words "joining the core support columns to the core support plate" after CE support column welds	Clarify the location of the welds to be inspected.
12h	3.2.4 fourth paragraph 4.1.3 second paragraph	Changed "components to be inspected" to "core support column welds"	Revised for clarity.
12i	3.2.4 fourth paragraph 4.1.3 second paragraph	Revised to require inspection in a 25% sample of the core support column welds, with a different 25% sample inspected at each 10 year interval.	Inspection of a 25% sample is considered sufficient to demonstrate the structural integrity of the core support column assembly considering the high degree of redundancy in this assembly (50 or more support columns).
13	3.2.4 first paragraph 4.1.3 first paragraph	Deleted discussion of B&W flow distributor-to-shell forging bolts.	These bolts are subject to other degradation mechanisms only after SCC initiates. Thus, they should not be considered multi-degradation mode components. Additionally, these bolts do not provide a core support function. See also Explanatory Note 2.
14	3.2.5 4.1.4	Delete this Topical Report Condition entirely.	The MRP screening process did not identify the Control Element Assembly Shroud Bolts in CE-designed plants as susceptible to irradiation related degradation mechanisms because the neutron fluence is well below the MRP-175 screening limit. It is possible that U.S. NRC staff has confused CEA shroud bolts with core shroud bolts, which are Primary components.

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16a, c	3.2.6.3 first and second paragraphs	Added reference to the specific tables and sections in MRP-227 that provide guidance on Industry and plant-specific aging management.	Added for completeness and clarity.
16b	3.2.6.3 first paragraph	Added reference to Westinghouse flux thimble tube inspection program.	The flux thimble tube program is recognized as an acceptable program for management of degradation in these components.
16d	3.2.6.3 third paragraph	Added reference to removal of thermal shields in CE-designed plants and programs in place for aging management of in-core instrument thimble tubes in some plants where thermal shields were not removed.	Added for completeness and clarity.
16e	3.2.6.3 fifth paragraph	Deleted specific reference to CE fuel alignment pins.	CE fuel alignment pins are listed in Section 4.4.2 and Table 4-8 of MRP-227 (CE Existing Program components) as ASME Section XI components and thus are incorrectly included as plant-specific Existing Program items in the draft SE.
16f	3.2.6.3 fifth paragraph	Added wording to clarify expectation for evaluation of the adequacy of Existing programs for aging management of affected components.	Added for clarity.
17a,b	3.2.6.3 fourth paragraph	Revised to read "There have been issues with cracking *of the original* X750 pins and *many* licensees have replaced …"	Revised for correctness.

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18	3.2.6.4	Deleted this Applicant/Licensee Action Item entirely.	The staff appears to agree with the conclusion that the applied stress on this component is low and weld residual stress have been alleviated by a stress relief treatment during the original fabrication. Documentation of the fabrication records that justifies this conclusion has recently been prepared by AREVA for the B&W-design licensees and is available for staff review if necessary (either submitted on a generic basis by the MRP for all of the B&W units or the documentation could be submitted by each licensee).
19a,b	3.3.2 first paragraph	Added the sentence "Coverage requirements for all expansion items should be determined and reviewed as part of the extent of condition evaluation for the "Primary" component." With this change, noted that the 75% minimum coverage requirement should be used as guidance when determining coverage requirement for "Expansion" components.	The coverage required for "Expansion" components should be based on the type of degradation, significance and affected area(s) in the associated "Primary" component. See also Explanatory Note 3.
19c,e	3.3.2 second paragraph 4.1.5 first paragraph	Added words to indicate that 75% coverage of "Expansion" components would be the default minimum in the absence of a valid technical justification for an alternative minimum.	This wording is intended to indicate that 1) it is necessary to establish a minimum coverage requirement for "Expansion" components, and 2) the default value is 75%.
19d,f,i	3.3.2 second and fourth paragraphs 4.1.5 first paragraph	Added clarifying wording to indicate that the coverage requirement is based on the *relevant* area/volume of the accessible component would be inspected, rather than 100%.	The examination area for "Expansion" components should be based on the results of an extent of condition evaluation, and may not be "100% of the accessible like component(s)" in all cases.

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19g	3.3.2 third paragraph	Changed to read "*The intention of the* minimum examination ..." rather than "Application of this minimum examination ...".	Revised wording indicates that the minimum examination coverage for "Expansion" components should be based on inspection findings in the associated "Primary" component.
19h	3.3.2 third paragraph 4.1.5 first paragraph	Added the sentences: "The extent of condition report for any Primary inspection finding should address the relevance of the observed degradation to the conditions in the expansion component. Any technical justification for a minimum examination coverage requirement below the 75% guideline for the Expansion components must provide reasonable assurance that degradation will be detected in a timely manner."	Wording for consistency with other changes in 3.3.2, as discussed above.
20a	3.3.3 first paragraph 4.1.6 first paragraph	Inserted "(EPFY)" after "10 to 15 years"	For correctness since examination frequency for Westinghouse/CE bolts are specified on an EFPY basis in Tables 4.2 and 4.3 of MRP-227, Rev. 0
20b	3.3.3 second paragraph 4.1.6 first paragraph	Added the clarifying wording "following the initial or baseline examination" as the starting point for the 10-year examination interval.	Added for clarity.

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21a,b	3.3.5 first paragraph	Added discussion of the applicable measurement parameters for the B&W plant core clamping, CE plant core shroud segment gap and Westinghouse plant hold down spring dimensional acceptance criteria, and noted that specific acceptance criteria were not provided in part due to plant-specific as-built tolerances.	Added for clarity; see also discussion in 19c,d below. Additionally, a generic acceptance criterion has been established for each of the operating B&W units relative to the core clamping items (plenum cover weldment rib pads, plenum cover support flange, and CSS top flange). It is provided in Table 5-1 of MRP-227 Rev. 0. Therefore, this applicant/licensee action item does not apply to the B&W applicants/licensees. The one time measurement at each of the B&W units has already been performed for five of the seven B&W units. To date, each of the five units has met the acceptance criterion with no discernable wear from the original measurements. Therefore, measurements at the remaining units (still pending) are also expected to be acceptable.
21c	3.3.5 second paragraph	Added the word "methodology" after "acceptance criteria."	Acceptance criteria for physical measurements would generally be condition-specific (core design, time in life, current operational conditions and loads, etc.) and therefore cannot generically be determined more than one or two years beforehand. However, the technique for determination of the acceptance criteria may be specified.
22	3.3.5 first paragraph	Deleted discussion of B&W plant baffle-to-baffle bolts and core barrel-to-former bolts.	No physical measurement of these components is required in MRP-227. Thus, reference to the B&W bolting appears to be in error.
23a, b, c	3.3.6 first paragraph	Revised designation of the inaccessible components in B&W designed plants.	Revised for correctness to identify those components that are inaccessible. See Explanatory Note 5.
24a, b, c, d, h	3.3.7 first paragraph	Revised discussion of RAI 4-15 response.	Discussion revised for consistency with the RAI response. The response was not intended to convey that specific fracture mechanics/flaw tolerance analyses had been or were planned to be performed generically for noted components.
24e	3.3.7 second paragraph	Added reference to WCAP 17096-NP.	This change acknowledges that determination of acceptance criteria will be based on the guidance provided in the WCAP.

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24f,g	3.3.7 third paragraph	Revised to specifically require demonstration of redundancy for the noted CASS components.	Revised for clarity to indicate the specific type of analysis required at the time of license submittal.
25a,b,c	3.3.7 first, second and fifth paragraphs 4.2.7 first paragraph	Changed "B&W IMI guide tube assembly(ies)" to "B&W IMI guide tube spider assembly(ies)" and to note these components are "Primary" category.	Revised for correctness. See Explanatory Note 6.
26	3.3.7 third paragraph	Deleted reference to a plant-specific reactivity analysis as part of an acceptance criteria analysis for B&W CRGT assembly spacer castings.	A reactivity analysis requires identification of condition-specific parameters, including core loading. Thus, this type of analysis can only be performed once these conditions are established and it is not generally possible to provide these analysis results at the time of license submittal. See also Explanatory Note 6.
27a-g	3.5.1 first paragraph including fifth bullet and last paragraph	Suggested wording changes to clarify requirements on a licensee-specific basis based on their current licensing status.	See Explanatory Note 7.
28 a,b,c,d	3.5.1 second bullets	Changed "nonconformance" to "deviation"	Changed for consistency with Industry nomenclature.
29	3.5.1 fourth bullet	Changed last sentence to read "...TS requirements take precedence over the MRP requirements and shall be complied with *unless a TS change is submitted.*"	Revised to reflect that a licensee may elect to change their Technical Specifications for consistency with MRP-227 requirements.

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30	3.6 third paragraph 4.1.8 first paragraph	Delete this Topical Report Condition entirely.	As noted in Appendix A of the response to RAI set 4, the existing Appendix A in MRP-227-Rev. 0 will be deleted in its entirety and replaced with a new Appendix A entitled Operating Experience Summary, with the contents provided in MRP-letter MRP 2009-091 sent to the NRC in December 2009.
31	4.1.7	Added "*consistent with ASME Code Section XI requirements*" after "baseline 10-year inspection re-examination interval ..."	Added to reflect similar scheduling flexibility to that allowed for Code inspections.
32a,b	4.2.1	Changed to read "...each applicant/licensee is responsible for ~~performing an evaluation of its plant's design and operating history~~ *assuring* and demonstrating the applicability of *its plant design and operating history to* the approved version of MRP-227 to the facility."	Revised to clarify intent (i.e., assurance versus analysis).
33	4.2.3	Change "analysis" to "assessment"	Revised to clarify intent.

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2	Various		Minor editorial suggestions.
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4	2.2, fifth paragraph	Revise definition of Category A components.	Revise for consistency with definition in MRP-227.
5	2.2, eighth paragraph, bullet 4	Clarify definition of "No Additional Measures" components.	Binning a component in the "No Additional Measures" category under MRP-227 does not necessarily mean that no inspections will be performed to monitor the component. It simply means that no action is required from an aging management viewpoint. Inspections of core support structures are performed in accordance with ASME Section XI Code requirements.
6	2.3, fifth paragraph	Change "barrel" to "core barrel."	Revise for consistency with nomenclature of B&W plants.
7a	2.5, second paragraph, bullet 1	Change to indicate the *Mandatory* requirement is to develop a reactor internals program, rather than to implement MRP-227.	Revise for consistency with MRP-227.
7b	2.5, second paragraph, bullet 2	Change to indicate that the *Needed* requirement to implement the Tables in MRP-227 are tied to issuance of the Revision A version of MRP-227, rather than Revision 0.	Revise for consistency with MRP-227.

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Technical Changes			
8	2.6	Change to indicate that Appendix A of MRP-227 merely summarizes GALL requirements for AMP submittals.	Appendix A of MRP-227 was not intended to require specific information for LRAs, but to provide guidance.
9a, 9c, 9e, 9f	3.2.2, heading and first, second and third paragraphs 4.1.1 heading and first paragraph	Change "High Consequence" to "Core Support"	The term "High Consequence" may be somewhat ambiguous with respect to whether the viewpoint is strictly overall aging management strategy, which considers both failure probability AND failure consequence, or simply failure consequence without consideration of probability. For clarity, the specific components in question (i.e., Core Support) should be cited to avoid potential ambiguity.
9b	3.2.2, first paragraph	Change generic description "components whose failure could cause significant safety consequences" to more specific description "components that are part of the core support structure".	As discussed above, more specific description of the components of concern is desirable to avoid potential ambiguity.
9d	3.2.2, second paragraph	Changed to clarify the results of the FMECA process for core support structures.	The FMECA panel concluded that aging related degradation of the applicable core support structure components would not necessarily result in loss of core support.

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10a, 10b	3.2.2, first and second paragraphs 4.4.1 first paragraph and table	Deleted reference to lower grid-to-core barrel bolts in B&W design plants and associated discussion of binning process used for B&W components.	The lower grid-to-core-barrel bolts in B&W-designed reactors are abbreviated as LCB bolts in MRP-227-Rev. 0, and are listed as "Primary" in MRP-227-Rev. 0 Table 4-1 (page 4-17). See Explanatory Note 1.
11a,b,c,d	3.2.3 first paragraph 4.1.2 second paragraph	Add text to clarify the results of the FMECA analysis for the Westinghouse upper flange upper and lower core cylinders and the CE core support barrel.	Revised for clarity to support follow-on discussion.
11e	3.2.3 first paragraph	Added discussion that designation of the core barrel welds as "Expansion" category components was based on reduction of IASCC susceptibility due to stress relaxation effects.	For accuracy, it should be noted that stress relaxation effects were considered in the original technical basis for the inspection requirements for these welds and other IASCC-susceptible components.
11f	3.2.3 second paragraph	Added discussion clarifying that only the core barrel/core support barrel welds in the beltline region potentially see sufficient neutron exposure to exceed the IASCC screening criteria.	This added text clarifies that only a limited region of the subject welds are in the high fluence regions and thus potentially subject to IASCC.
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11i,j	3.2.3 third paragraph 4.1.2 second paragraph	Revise text to require susceptible circumferential beltline core support/core barrel welds to be inspected as "Primary" components, with an alternative to demonstrate acceptability through analysis.	This focuses inspection on the susceptible portions of the welds.
11k	3.2.3 third paragraph 4.1.2 second paragraph	Establish 75% inspection criteria for susceptible portions of the circumferential beltline core support/core barrel welds and acknowledge that accessibility of core barrel welds may be limited for some plant designs and provide an alternative to evaluate the welds as inaccessible components consistent with Applicant/Licensee Action Item 6.	Dependent on plant design, the areas of greatest interest (that is, highest fluence areas) may be inaccessible. It is thus desirable to acknowledge these potential limitations and provide a technically acceptable alternative in lieu of inspection.
12a	3.2.4 heading 4.1.3 heading	Change "High Consequence" to "Core Support."	See discussion in 9a above; more specific description of the components of concern is desirable to avoid potential ambiguity.
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12g	3.2.4 fourth paragraph 4.1.3 second paragraph	Added the words "joining the core support columns to the core support plate" after CE support column welds	Clarify the location of the welds to be inspected.
12h	3.2.4 fourth paragraph 4.1.3 second paragraph	Changed "components to be inspected" to "core support column welds"	Revised for clarity.
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20a	3.3.3 first paragraph 4.1.6 first paragraph	Inserted "(EPFY)" after "10 to 15 years"	For correctness since examination frequency for Westinghouse/CE bolts are specified on an EFPY basis in Tables 4.2 and 4.3 of MRP-227, Rev. 0
20b	3.3.3 second paragraph 4.1.6 first paragraph	Added the clarifying wording "following the initial or baseline examination" as the starting point for the 10-year examination interval.	Added for clarity.

Item Number	Location in Proposed Revision	MRP Change or Issue	Discussion
21a,b	3.3.5 first paragraph	Added discussion of the applicable measurement parameters for the B&W plant core clamping, CE plant core shroud segment gap and Westinghouse plant hold down spring dimensional acceptance criteria, and noted that specific acceptance criteria were not provided in part due to plant-specific as-built tolerances.	Added for clarity; see also discussion in 19c,d below. Additionally, a generic acceptance criterion has been established for each of the operating B&W units relative to the core clamping items (plenum cover weldment rib pads, plenum cover support flange, and CSS top flange). It is provided in Table 5-1 of MRP-227 Rev. 0. Therefore, this applicant/licensee action item does not apply to the B&W applicants/licensees. The one time measurement at each of the B&W units has already been performed for five of the seven B&W units. To date, each of the five units has met the acceptance criterion with no discernable wear from the original measurements. Therefore, measurements at the remaining units (still pending) are also expected to be acceptable.
21c	3.3.5 second paragraph	Added the word "methodology" after "acceptance criteria."	Acceptance criteria for physical measurements would generally be condition-specific (core design, time in life, current operational conditions and loads, etc.) and therefore cannot generically be determined more than one or two years beforehand. However, the technique for determination of the acceptance criteria may be specified.
22	3.3.5 first paragraph	Deleted discussion of B&W plant baffle-to-baffle bolts and core barrel-to-former bolts.	No physical measurement of these components is required in MRP-227. Thus, reference to the B&W bolting appears to be in error.
23a, b, c	3.3.6 first paragraph	Revised designation of the inaccessible components in B&W designed plants.	Revised for correctness to identify those components that are inaccessible. See Explanatory Note 5.
24a, b, c, d, h	3.3.7 first paragraph	Revised discussion of RAI 4-15 response.	Discussion revised for consistency with the RAI response. The response was not intended to convey that specific fracture mechanics/flaw tolerance analyses had been or were planned to be performed generically for noted components.
24e	3.3.7 second paragraph	Added reference to WCAP 17096-NP.	This change acknowledges that determination of acceptance criteria will be based on the guidance provided in the WCAP.

Item Number	Location in Proposed Revision	MRP Change or Issue	Discussion
24f,g	3.3.7 third paragraph	Revised to specifically require demonstration of redundancy for the noted CASS components.	Revised for clarity to indicate the specific type of analysis required at the time of license submittal.
25a,b,c	3.3.7 first, second and fifth paragraphs 4.2.7 first paragraph	Changed "B&W IMI guide tube assembly(ies)" to "B&W IMI guide tube spider assembly(ies)" and to note these components are "Primary" category.	Revised for correctness. See Explanatory Note 6.
26	3.3.7 third paragraph	Deleted reference to a plant-specific reactivity analysis as part of an acceptance criteria analysis for B&W CRGT assembly spacer castings.	A reactivity analysis requires identification of condition-specific parameters, including core loading. Thus, this type of analysis can only be performed once these conditions are established and it is not generally possible to provide these analysis results at the time of license submittal. See also Explanatory Note 6.
27a-g	3.5.1 first paragraph including fifth bullet and last paragraph	Suggested wording changes to clarify requirements on a licensee-specific basis based on their current licensing status.	See Explanatory Note 7.
28 a,b,c,d	3.5.1 second bullets	Changed "nonconformance" to "deviation"	Changed for consistency with Industry nomenclature.
29	3.5.1 fourth bullet	Changed last sentence to read "...TS requirements take precedence over the MRP requirements and shall be complied with *unless a TS change is submitted.*"	Revised to reflect that a licensee may elect to change their Technical Specifications for consistency with MRP-227 requirements.

Item Number	Location in Proposed Revision	MRP Change or Issue	Discussion
30	3.6 third paragraph 4.1.8 first paragraph	Delete this Topical Report Condition entirely.	As noted in Appendix A of the response to RAI set 4, the existing Appendix A in MRP-227-Rev. 0 will be deleted in its entirety and replaced with a new Appendix A entitled Operating Experience Summary, with the contents provided in MRP-letter MRP 2009-091 sent to the NRC in December 2009.
31	4.1.7	Added "*consistent with ASME Code Section XI requirements*" after "baseline 10-year inspection re-examination interval …"	Added to reflect similar scheduling flexibility to that allowed for Code inspections.
32a,b	4.2.1	Changed to read "…each applicant/licensee is responsible for ~~performing an evaluation of its plant's design and operating history~~ *assuring* and demonstrating the applicability of *its plant design and operating history to* the approved version of MRP-227 to the facility."	Revised to clarify intent (i.e., assurance versus analysis).
33	4.2.3	Change "analysis" to "assessment"	Revised to clarify intent.

Attachment 2
Explanatory Notes for MRP Comments
NRC Draft Safety Evaluation of MRP-227
May 10, 2011

Note 1: TRC 1

Lower Grid Assembly-To-Core Barrel Bolts In B&W-Designed Reactors

The lower grid-to-core-barrel bolts in B&W-designed reactors are abbreviated as LCB bolts in MRP-227-Rev. 0, and are listed as "Primary" in MRP-227-Rev. 0 Table 4-1 (page 4-17). Confusion on behalf of the NRC reviewers most likely resulted from the fact that the FMECA identifier B.4 in Table A-3 of the MRP-190 report for the 108 Alloy A-286 or Alloy X-750 lower grid assembly-to-core barrel (a.k.a. lower core barrel or LCB) bolts does not define LCB. However, the abbreviated acronym LCB is clearly defined in MRP-189 Rev. 1 in Table 3-2 (Pages 3-9 and 3-10), Table 4-1 (pages 4-9 and 4-10), 5-1 (page 5-10), and Table 5-2 (page 5-17) as these same bolts, which are indeed the high-strength lower core barrel bolts that exist at each of the operating B&W units. Therefore, this item should be removed from TRC 1.

Note 2: TRC 3

B&W Flow Distributor-to-Shell Forging Bolts

The flow distributor-to-shell forging bolts (a.k.a. flow distributor or FD 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.

The potential age-related degradation mechanism for flow distributor bolts in B&W units was determined to be SCC only (see MRP-189 Rev. 1 Table 3-2, Table 4-1, Table 5-1, and Table 5-2; MRP-231 Table 3-4, Table 3-8, and Table 3-10; and MRP-227 Table 4-4) . Initially, thermal stress relaxation (TSR) was considered a potential age-related degradation mechanism for these bolts, but as discussed in MRP-189 Rev. 1 (page 3-21, Section 3.4, item c), TSR was subsequently removed as a separate mechanism for bolting applications remote from the high radiation flux area near the core. The discussion of fatigue or wear (as a result of thermal stress relaxation) for FMECA identifier L.2.3 in Table A-4 of the MRP-190 report was deemed to be non-relevant and/or applicable only as cascading consequences of failure by SCC and therefore not carried forward in the other MRP documents (i.e., MRP-231 and MRP-227); however, MRP-190 was not revised to reflect this change.

The SCC susceptibility of the FD bolts was also reevaluated in MRP-231 (Section 2.2, pages 2-12 and 2-13) based on results from stress analysis work. Because of the lower stress level and temperature, the FD bolts were re-categorized from Category C to Category B. In addition, as discussed in MRP-231 (Section 3.2.4, page 3-10), only the

UCB and LCB bolts have potentially high SCC susceptibility and safety consequences, and the FD bolts were ultimately placed into the MRP-227 Expansion Table (Table 4-4).

It should also be noted that the FD bolts, which connect the flow distributor and shell forging, are installed in a vertically upward orientation, receive a VT-3 examination during ASME Code 10-year ISI examinations and failures would be obvious (i.e., the bolt head would drop, but remain captured by the locking clip) were they to fail.

Note 3: TRC 5

The following categories would apply to B&W Expansion Items:

1. Selecting 100% of accessible surfaces of a set of components or component items

   Applies to: Alloy X-750 dowel-to-upper fuel assembly support pad welds; and the lower fuel assembly support pad items for IE and/or TE (which includes the pad, pad-to-rib Section welds, Alloy X-750 dowel, cap screw, and their locking welds), which also includes the Alloy X-750 dowel-to-lower fuel assembly support pad welds for SCC

   The strategy employed for these component items is to specify that the accessible surfaces of all of the components in the population to be examined to the extent that it would not require disassembly. All of the components are accessible, although some portions of the visible surfaces will be obstructed. In all of these cases, the required VT-3 visual examination is looking for gross degradation, such as separation of material, broken or missing locking welds, etc. Thus, no minimum coverage requirement is deemed necessary.

2. Selecting 100% of accessible bolting locking devices in a bolted assembly

   Applies to: locking devices for UTS, SSHT, LTS, and FD bolts; and the lower grid shock pad bolt locking devices

   For this situation, the required VT-3 visual examination is looking for gross degradation, such as separation of material, broken or missing locking welds/devices, etc. for the applicable bolted assemblies. Camera access to the bolt head and locking devices are, with minimal exceptions, without limitation. Thus, where visual VT-3 examination is required there is virtually no access limitation anticipated. The potential significant limitation on accessibility for this situation only occurs when volumetric examination (UT) is specified and a bolt head design or as-built condition limits access or effectiveness for the UT transducer. Since this is a potential severe limitation, considering the industry's limited examination experience across the entire variety of design variations, the MRP will include minimum coverage requirements in Tables 4-4 through 4-6 to further assure that potential limitations on access and examination coverage will be adequately addressed (this note will be included in the '-A' version of MRP-227):

"A minimum of 75% of the total population (examined + unexamined) must be examined for inspection credit."

The examination of 75% of the accessible surfaces or items will provide the necessary evidence to show whether the degradation is localized or widespread directly, without the need for coverage expansion.

If a minimum coverage is not satisfied, the intent of MRP-227 is not met and, as discussed in RAI response 4-7, a deviation must be prepared and the staff and the MRP notified of the inability to meet the "needed" requirement for coverage. This requirement already applies where specific recommendations are made for the Primary components.

3. Inaccessible items requiring an evaluation or replacement

   Applies to: core barrel cylinder (including vertical and circumferential seam welds); former plates; external baffle-to-baffle bolts; core barrel-to-former bolts; and the locking devices (including locking welds) for external baffle-to-baffle bolts and core barrel-to-former bolts

   No minimum examination coverage is required for this situation. Justification for acceptability will be determined through an evaluation or replacement of the component or component item(s).

4. Accessible items requiring an evaluation or replacement

   Applies to: internal baffle-to-baffle bolts

   No minimum examination coverage is required for this situation. Justification for acceptability will be determined through an evaluation or replacement of the component or component item(s).

<u>Note 4: A/LAI 4</u>

Note: the terminology used for this weld in MRP-189-Rev 1 is "Core support shield upper flange weld".

As stated in the MRP response to RAI 4-4, the core support shield (CSS) upper flange weld in B&W units was screened out of all aging degradation mechanisms, including SCC, and therefore was placed in Category A as documented in MRP-189 Rev. 1 (see "Weld ID WC-43" Tables 3-3 on page 3-17 and 4-2 on page 4-14).

This particular weld is a double U-groove weld made using an automatic submerged arc (ASA) welding process with Type 308L weld metal. The MRP-175 SCC screening criteria for austenitic stainless steel welds are $\geq$ 30 ksi total stress (applied and residual) ***and*** < 5 percent ferrite. The MRP-175 screening criteria are listed in MRP-189-Rev. 1, Table 3-1 on page 3-2. In accordance with ASME Code, the original B&W RV internals

equipment specification specified the ferrite content for austenitic stainless steel welds to be a minimum of 5 percent. Therefore, this weld (WC-43) was screened out of SCC (i.e., Category A for SCC) based on the weld ferrite requirement.

In the process of preparing MRP-189 Rev. 01, the internals welds were reviewed by AREVA and put through the MRP-175 screening process, as noted Section 3.2 (page 3-15). This was completed by reviewing fabrication records of all internals welds for the Davis-Besse internals. In addition, the post-weld stress relief for all seven B&W units was also reviewed. The results are documented in AREVA proprietary document 51-9027395-000 (Reference 7 of MRP-189 Rev. 1). Since the weld stress relief was not needed to screen out SCC, the stress relief information during the original fabrication was not listed in MPR-189-Rev. 1. The stress relief information is summarized below.

During the review, it was found that many internals welds including all double-groove ASA welds such as the CSS upper flange weld for all seven operating B&W units was required to have a post-weld stress relief treatment either by a mechanical or thermal means or both during the original construction by the original RV internals equipment specification. This practice has been confirmed for four B&W units by the actual signed and dated assembly process sheets. Specifically, there was no exception or deviation from the stress relief requirement for any double-groove ASA welds including the CSS upper flange weld for these four units.

The approved RV internals assembly process to be used for the remaining three units was also found. The approved assembly process also listed the same stress relief steps as for the other four units. However, the actual signed and dated RV internals assembly process sheets for these three B&W units could not be found. These three units were built chronologically between the other four units whose signed and dated process sheets were found. During original construction period for the seven B&W units, several revisions to the RV internals equipment specification were made. In each revision, the stress relief requirement was always listed. Therefore, it was concluded that all double-groove ASA welds including the CSS upper flange weld in all seven operating B&W units received a post-weld stress relief treatment.

Note 5: A/LAI 6

Sections 3.3.6 and 4.2.6 of the draft SE incorrectly identifies and discusses "internal" baffle-to-baffle bolts and their locking devices, and "baffle-to-former bolts and their locking devices" as inaccessible items for B&W units. These should be removed from this applicant/licensee action item.

The "internal baffle-to-baffle bolts and their locking devices" are accessible. MRP-231 Rev. 1 Section 3.2.2 contains a discussion of why the internal baffle-to-baffle bolts are not considered primary items and that complete failure of all of these bolts along with a significant number of baffle-to-former bolts could occur before a functionality concern would exist with the baffle plates or core barrel assembly. In addition, these bolts and their locking devices are VT-3 visually examined during the ASME Code 10-year ISI

examinations, which is another reason why they are considered Expansion items. These bolts are therefore included in the proposed justification by evaluation or replacement effort for the other bolt locations that are inaccessible (i.e., external baffle-to-baffle bolts and core barrel-to-former bolts).

The "baffle-to-former bolts and their locking devices" are accessible items and identified as primary items to be examined, as given in Table 4-1 of MRP-227 Rev. 0. Rather, Section 4.2.6 of the draft SE should be rewritten to state "core barrel-to-former bolts and their locking devices".

Note 6: A/LAI 7

Sections 3.3.7 and 4.2.7 of the draft SE incorrectly identifies "B&W in-core monitoring instrumentation (IMI) guide tube assemblies" as a CASS component. The correct term is "B&W in-core monitoring instrumentation (IMI) guide tube assembly spiders." The spider is the only CASS item in the IMI guide tube assemblies.

The recommended methodology for acceptance of the IMI guide tube assembly spiders (given in WCAP-17096) is to perform a generic analysis to show that one or more missing spider arms or a completely missing spider will not result in loss of function of the IMI guide tube. Thus, no fracture mechanics evaluations are needed. Since there are 52 IMI guide tubes in each B&W unit, a redundancy argument may also be adequate.

For the spacer castings inside the CRGT assembly, a generic analysis is being proposed within the PWROG to quantify the stress distribution in the spacer castings as a function of failed screw locations. The results of this evaluation would be used to justify the assumption that a failed threaded screw location would not lead to rapid successive failures. Since there are 69 CRDMs in each B&W unit, an alternative methodology for acceptance of degradation or failure of the CRGT assembly spacer castings (given in WCAP-17096) is to perform a unit-specific reactivity analysis to determine the number of CRDMs that are required for shut down of the reactor. However, this would be specific to each fuel cycle and would not be performed until it was known which particular CRDM would potentially not be operable. Thus again, no fracture mechanics evaluations are needed.

Note 7: A/LAI 8

Section 3.5.1 could be interpreted to require items 1 thru 5 for all licensees when they are only appropriate for those submitting new license renewal applications. Items 3 thru 5 have already been provided by licensees during their original license renewal application submittals. Changes are provided for additional clarity on which of the listed actions in 3.5.1 are required for licensees with a renewed license and which are for new applicants for license renewal.

In Item 5 the third paragraph discusses addressing TLAA CUF analyses for vessel internals and requires that the effects of reactor water environment must be included to

satisfy NRC Code requirements. Currently license renewal guidance in NUREG-1800 (Section 4.3.2.1.3) and NUREG-1801 (X.M1 program scope) only require the consideration of environmental effects for reactor coolant pressure boundary components. They do not include this requirement for reactor vessel internal components since they are not reactor coolant pressure boundary. A review of the 2007 version of these sections of ASME Code Section III did not reveal any requirement to include environmental effects.

Attachment 2
Explanatory Notes for MRP Comments
NRC Draft Safety Evaluation of MRP-227
May 10, 2011

Note 1: TRC 1

Lower Grid Assembly-To-Core Barrel Bolts In B&W-Designed Reactors

The lower grid-to-core-barrel bolts in B&W-designed reactors are abbreviated as LCB bolts in MRP-227-Rev. 0, and are listed as "Primary" in MRP-227-Rev. 0 Table 4-1 (page 4-17). Confusion on behalf of the NRC reviewers most likely resulted from the fact that the FMECA identifier B.4 in Table A-3 of the MRP-190 report for the 108 Alloy A-286 or Alloy X-750 lower grid assembly-to-core barrel (a.k.a. lower core barrel or LCB) bolts does not define LCB. However, the abbreviated acronym LCB is clearly defined in MRP-189 Rev. 1 in Table 3-2 (Pages 3-9 and 3-10), Table 4-1 (pages 4-9 and 4-10), 5-1 (page 5-10), and Table 5-2 (page 5-17) as these same bolts, which are indeed the high-strength lower core barrel bolts that exist at each of the operating B&W units. Therefore, this item should be removed from TRC 1.

Note 2: TRC 3

B&W Flow Distributor-to-Shell Forging Bolts

The flow distributor-to-shell forging bolts (a.k.a. flow distributor or FD 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.

The potential age-related degradation mechanism for flow distributor bolts in B&W units was determined to be SCC only (see MRP-189 Rev. 1 Table 3-2, Table 4-1, Table 5-1, and Table 5-2; MRP-231 Table 3-4, Table 3-8, and Table 3-10; and MRP-227 Table 4-4) . Initially, thermal stress relaxation (TSR) was considered a potential age-related degradation mechanism for these bolts, but as discussed in MRP-189 Rev. 1 (page 3-21, Section 3.4, item c), TSR was subsequently removed as a separate mechanism for bolting applications remote from the high radiation flux area near the core. The discussion of fatigue or wear (as a result of thermal stress relaxation) for FMECA identifier L.2.3 in Table A-4 of the MRP-190 report was deemed to be non-relevant and/or applicable only as cascading consequences of failure by SCC and therefore not carried forward in the other MRP documents (i.e., MRP-231 and MRP-227); however, MRP-190 was not revised to reflect this change.

The SCC susceptibility of the FD bolts was also reevaluated in MRP-231 (Section 2.2, pages 2-12 and 2-13) based on results from stress analysis work. Because of the lower stress level and temperature, the FD bolts were re-categorized from Category C to Category B. In addition, as discussed in MRP-231 (Section 3.2.4, page 3-10), only the

UCB and LCB bolts have potentially high SCC susceptibility and safety consequences, and the FD bolts were ultimately placed into the MRP-227 Expansion Table (Table 4-4).

It should also be noted that the FD bolts, which connect the flow distributor and shell forging, are installed in a vertically upward orientation, receive a VT-3 examination during ASME Code 10-year ISI examinations and failures would be obvious (i.e., the bolt head would drop, but remain captured by the locking clip) were they to fail.

Note 3: TRC 5

The following categories would apply to B&W Expansion Items:

1. Selecting 100% of accessible surfaces of a set of components or component items

   Applies to: Alloy X-750 dowel-to-upper fuel assembly support pad welds; and the lower fuel assembly support pad items for IE and/or TE (which includes the pad, pad-to-rib Section welds, Alloy X-750 dowel, cap screw, and their locking welds), which also includes the Alloy X-750 dowel-to-lower fuel assembly support pad welds for SCC

   The strategy employed for these component items is to specify that the accessible surfaces of all of the components in the population to be examined to the extent that it would not require disassembly. All of the components are accessible, although some portions of the visible surfaces will be obstructed. In all of these cases, the required VT-3 visual examination is looking for gross degradation, such as separation of material, broken or missing locking welds, etc. Thus, no minimum coverage requirement is deemed necessary.

2. Selecting 100% of accessible bolting locking devices in a bolted assembly

   Applies to: locking devices for UTS, SSHT, LTS, and FD bolts; and the lower grid shock pad bolt locking devices

   For this situation, the required VT-3 visual examination is looking for gross degradation, such as separation of material, broken or missing locking welds/devices, etc. for the applicable bolted assemblies. Camera access to the bolt head and locking devices are, with minimal exceptions, without limitation. Thus, where visual VT-3 examination is required there is virtually no access limitation anticipated. The potential significant limitation on accessibility for this situation only occurs when volumetric examination (UT) is specified and a bolt head design or as-built condition limits access or effectiveness for the UT transducer. Since this is a potential severe limitation, considering the industry's limited examination experience across the entire variety of design variations, the MRP will include minimum coverage requirements in Tables 4-4 through 4-6 to further assure that potential limitations on access and examination coverage will be adequately addressed (this note will be included in the '-A' version of MRP-227):

"A minimum of 75% of the total population (examined + unexamined) must be examined for inspection credit."

The examination of 75% of the accessible surfaces or items will provide the necessary evidence to show whether the degradation is localized or widespread directly, without the need for coverage expansion.

If a minimum coverage is not satisfied, the intent of MRP-227 is not met and, as discussed in RAI response 4-7, a deviation must be prepared and the staff and the MRP notified of the inability to meet the "needed" requirement for coverage. This requirement already applies where specific recommendations are made for the Primary components.

3. Inaccessible items requiring an evaluation or replacement

   Applies to: core barrel cylinder (including vertical and circumferential seam welds); former plates; external baffle-to-baffle bolts; core barrel-to-former bolts; and the locking devices (including locking welds) for external baffle-to-baffle bolts and core barrel-to-former bolts

   No minimum examination coverage is required for this situation. Justification for acceptability will be determined through an evaluation or replacement of the component or component item(s).

4. Accessible items requiring an evaluation or replacement

   Applies to: internal baffle-to-baffle bolts

   No minimum examination coverage is required for this situation. Justification for acceptability will be determined through an evaluation or replacement of the component or component item(s).

<u>Note 4: A/LAI 4</u>

Note: the terminology used for this weld in MRP-189-Rev 1 is "Core support shield upper flange weld".

As stated in the MRP response to RAI 4-4, the core support shield (CSS) upper flange weld in B&W units was screened out of all aging degradation mechanisms, including SCC, and therefore was placed in Category A as documented in MRP-189 Rev. 1 (see "Weld ID WC-43" Tables 3-3 on page 3-17 and 4-2 on page 4-14).

This particular weld is a double U-groove weld made using an automatic submerged arc (ASA) welding process with Type 308L weld metal. The MRP-175 SCC screening criteria for austenitic stainless steel welds are $\geq$ 30 ksi total stress (applied and residual) ***and*** < 5 percent ferrite. The MRP-175 screening criteria are listed in MRP-189-Rev. 1, Table 3-1 on page 3-2. In accordance with ASME Code, the original B&W RV internals

equipment specification specified the ferrite content for austenitic stainless steel welds to be a minimum of 5 percent. Therefore, this weld (WC-43) was screened out of SCC (i.e., Category A for SCC) based on the weld ferrite requirement.

In the process of preparing MRP-189 Rev. 01, the internals welds were reviewed by AREVA and put through the MRP-175 screening process, as noted Section 3.2 (page 3-15). This was completed by reviewing fabrication records of all internals welds for the Davis-Besse internals. In addition, the post-weld stress relief for all seven B&W units was also reviewed. The results are documented in AREVA proprietary document 51-9027395-000 (Reference 7 of MRP-189 Rev. 1). Since the weld stress relief was not needed to screen out SCC, the stress relief information during the original fabrication was not listed in MPR-189-Rev. 1. The stress relief information is summarized below.

During the review, it was found that many internals welds including all double-groove ASA welds such as the CSS upper flange weld for all seven operating B&W units was required to have a post-weld stress relief treatment either by a mechanical or thermal means or both during the original construction by the original RV internals equipment specification. This practice has been confirmed for four B&W units by the actual signed and dated assembly process sheets. Specifically, there was no exception or deviation from the stress relief requirement for any double-groove ASA welds including the CSS upper flange weld for these four units.

The approved RV internals assembly process to be used for the remaining three units was also found. The approved assembly process also listed the same stress relief steps as for the other four units. However, the actual signed and dated RV internals assembly process sheets for these three B&W units could not be found. These three units were built chronologically between the other four units whose signed and dated process sheets were found. During original construction period for the seven B&W units, several revisions to the RV internals equipment specification were made. In each revision, the stress relief requirement was always listed. Therefore, it was concluded that all double-groove ASA welds including the CSS upper flange weld in all seven operating B&W units received a post-weld stress relief treatment.

Note 5: A/LAI 6

Sections 3.3.6 and 4.2.6 of the draft SE incorrectly identifies and discusses "internal" baffle-to-baffle bolts and their locking devices, and "baffle-to-former bolts and their locking devices" as inaccessible items for B&W units. These should be removed from this applicant/licensee action item.

The "internal baffle-to-baffle bolts and their locking devices" are accessible. MRP-231 Rev. 1 Section 3.2.2 contains a discussion of why the internal baffle-to-baffle bolts are not considered primary items and that complete failure of all of these bolts along with a significant number of baffle-to-former bolts could occur before a functionality concern would exist with the baffle plates or core barrel assembly. In addition, these bolts and their locking devices are VT-3 visually examined during the ASME Code 10-year ISI

examinations, which is another reason why they are considered Expansion items. These bolts are therefore included in the proposed justification by evaluation or replacement effort for the other bolt locations that are inaccessible (i.e., external baffle-to-baffle bolts and core barrel-to-former bolts).

The "baffle-to-former bolts and their locking devices" are accessible items and identified as primary items to be examined, as given in Table 4-1 of MRP-227 Rev. 0. Rather, Section 4.2.6 of the draft SE should be rewritten to state "core barrel-to-former bolts and their locking devices".

Note 6: A/LAI 7

Sections 3.3.7 and 4.2.7 of the draft SE incorrectly identifies "B&W in-core monitoring instrumentation (IMI) guide tube assemblies" as a CASS component. The correct term is "B&W in-core monitoring instrumentation (IMI) guide tube assembly spiders." The spider is the only CASS item in the IMI guide tube assemblies.

The recommended methodology for acceptance of the IMI guide tube assembly spiders (given in WCAP-17096) is to perform a generic analysis to show that one or more missing spider arms or a completely missing spider will not result in loss of function of the IMI guide tube. Thus, no fracture mechanics evaluations are needed. Since there are 52 IMI guide tubes in each B&W unit, a redundancy argument may also be adequate.

For the spacer castings inside the CRGT assembly, a generic analysis is being proposed within the PWROG to quantify the stress distribution in the spacer castings as a function of failed screw locations. The results of this evaluation would be used to justify the assumption that a failed threaded screw location would not lead to rapid successive failures. Since there are 69 CRDMs in each B&W unit, an alternative methodology for acceptance of degradation or failure of the CRGT assembly spacer castings (given in WCAP-17096) is to perform a unit-specific reactivity analysis to determine the number of CRDMs that are required for shut down of the reactor. However, this would be specific to each fuel cycle and would not be performed until it was known which particular CRDM would potentially not be operable. Thus again, no fracture mechanics evaluations are needed.

Note 7: A/LAI 8

Section 3.5.1 could be interpreted to require items 1 thru 5 for all licensees when they are only appropriate for those submitting new license renewal applications. Items 3 thru 5 have already been provided by licensees during their original license renewal application submittals. Changes are provided for additional clarity on which of the listed actions in 3.5.1 are required for licensees with a renewed license and which are for new applicants for license renewal.

In Item 5 the third paragraph discusses addressing TLAA CUF analyses for vessel internals and requires that the effects of reactor water environment must be included to

satisfy NRC Code requirements. Currently license renewal guidance in NUREG-1800 (Section 4.3.2.1.3) and NUREG-1801 (X.M1 program scope) only require the consideration of environmental effects for reactor coolant pressure boundary components. They do not include this requirement for reactor vessel internal components since they are not reactor coolant pressure boundary. A review of the 2007 version of these sections of ASME Code Section III did not reveal any requirement to include environmental effects.

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Note 1: TRC 1

Lower Grid Assembly-To-Core Barrel Bolts In B&W-Designed Reactors

The lower grid-to-core-barrel bolts in B&W-designed reactors are abbreviated as LCB bolts in MRP-227-Rev. 0, and are listed as "Primary" in MRP-227-Rev. 0 Table 4-1 (page 4-17). Confusion on behalf of the NRC reviewers most likely resulted from the fact that the FMECA identifier B.4 in Table A-3 of the MRP-190 report for the 108 Alloy A-286 or Alloy X-750 lower grid assembly-to-core barrel (a.k.a. lower core barrel or LCB) bolts does not define LCB. However, the abbreviated acronym LCB is clearly defined in MRP-189 Rev. 1 in Table 3-2 (Pages 3-9 and 3-10), Table 4-1 (pages 4-9 and 4-10), 5-1 (page 5-10), and Table 5-2 (page 5-17) as these same bolts, which are indeed the high-strength lower core barrel bolts that exist at each of the operating B&W units. Therefore, this item should be removed from TRC 1.

Note 2: TRC 3

B&W Flow Distributor-to-Shell Forging Bolts

The flow distributor-to-shell forging bolts (a.k.a. flow distributor or FD 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.

The potential age-related degradation mechanism for flow distributor bolts in B&W units was determined to be SCC only (see MRP-189 Rev. 1 Table 3-2, Table 4-1, Table 5-1, and Table 5-2; MRP-231 Table 3-4, Table 3-8, and Table 3-10; and MRP-227 Table 4-4) . Initially, thermal stress relaxation (TSR) was considered a potential age-related degradation mechanism for these bolts, but as discussed in MRP-189 Rev. 1 (page 3-21, Section 3.4, item c), TSR was subsequently removed as a separate mechanism for bolting applications remote from the high radiation flux area near the core. The discussion of fatigue or wear (as a result of thermal stress relaxation) for FMECA identifier L.2.3 in Table A-4 of the MRP-190 report was deemed to be non-relevant and/or applicable only as cascading consequences of failure by SCC and therefore not carried forward in the other MRP documents (i.e., MRP-231 and MRP-227); however, MRP-190 was not revised to reflect this change.

The SCC susceptibility of the FD bolts was also reevaluated in MRP-231 (Section 2.2, pages 2-12 and 2-13) based on results from stress analysis work. Because of the lower stress level and temperature, the FD bolts were re-categorized from Category C to Category B. In addition, as discussed in MRP-231 (Section 3.2.4, page 3-10), only the

UCB and LCB bolts have potentially high SCC susceptibility and safety consequences, and the FD bolts were ultimately placed into the MRP-227 Expansion Table (Table 4-4).

It should also be noted that the FD bolts, which connect the flow distributor and shell forging, are installed in a vertically upward orientation, receive a VT-3 examination during ASME Code 10-year ISI examinations and failures would be obvious (i.e., the bolt head would drop, but remain captured by the locking clip) were they to fail.

Note 3: TRC 5

The following categories would apply to B&W Expansion Items:

1. Selecting 100% of accessible surfaces of a set of components or component items

   Applies to: Alloy X-750 dowel-to-upper fuel assembly support pad welds; and the lower fuel assembly support pad items for IE and/or TE (which includes the pad, pad-to-rib Section welds, Alloy X-750 dowel, cap screw, and their locking welds), which also includes the Alloy X-750 dowel-to-lower fuel assembly support pad welds for SCC

   The strategy employed for these component items is to specify that the accessible surfaces of all of the components in the population to be examined to the extent that it would not require disassembly. All of the components are accessible, although some portions of the visible surfaces will be obstructed. In all of these cases, the required VT-3 visual examination is looking for gross degradation, such as separation of material, broken or missing locking welds, etc. Thus, no minimum coverage requirement is deemed necessary.

2. Selecting 100% of accessible bolting locking devices in a bolted assembly

   Applies to: locking devices for UTS, SSHT, LTS, and FD bolts; and the lower grid shock pad bolt locking devices

   For this situation, the required VT-3 visual examination is looking for gross degradation, such as separation of material, broken or missing locking welds/devices, etc. for the applicable bolted assemblies. Camera access to the bolt head and locking devices are, with minimal exceptions, without limitation. Thus, where visual VT-3 examination is required there is virtually no access limitation anticipated. The potential significant limitation on accessibility for this situation only occurs when volumetric examination (UT) is specified and a bolt head design or as-built condition limits access or effectiveness for the UT transducer. Since this is a potential severe limitation, considering the industry's limited examination experience across the entire variety of design variations, the MRP will include minimum coverage requirements in Tables 4-4 through 4-6 to further assure that potential limitations on access and examination coverage will be adequately addressed (this note will be included in the '-A' version of MRP-227):

"A minimum of 75% of the total population (examined + unexamined) must be examined for inspection credit."

The examination of 75% of the accessible surfaces or items will provide the necessary evidence to show whether the degradation is localized or widespread directly, without the need for coverage expansion.

If a minimum coverage is not satisfied, the intent of MRP-227 is not met and, as discussed in RAI response 4-7, a deviation must be prepared and the staff and the MRP notified of the inability to meet the "needed" requirement for coverage. This requirement already applies where specific recommendations are made for the Primary components.

3. Inaccessible items requiring an evaluation or replacement

   Applies to: core barrel cylinder (including vertical and circumferential seam welds); former plates; external baffle-to-baffle bolts; core barrel-to-former bolts; and the locking devices (including locking welds) for external baffle-to-baffle bolts and core barrel-to-former bolts

   No minimum examination coverage is required for this situation. Justification for acceptability will be determined through an evaluation or replacement of the component or component item(s).

4. Accessible items requiring an evaluation or replacement

   Applies to: internal baffle-to-baffle bolts

   No minimum examination coverage is required for this situation. Justification for acceptability will be determined through an evaluation or replacement of the component or component item(s).

Note 4: A/LAI 4

Note: the terminology used for this weld in MRP-189-Rev 1 is "Core support shield upper flange weld".

As stated in the MRP response to RAI 4-4, the core support shield (CSS) upper flange weld in B&W units was screened out of all aging degradation mechanisms, including SCC, and therefore was placed in Category A as documented in MRP-189 Rev. 1 (see "Weld ID WC-43" Tables 3-3 on page 3-17 and 4-2 on page 4-14).

This particular weld is a double U-groove weld made using an automatic submerged arc (ASA) welding process with Type 308L weld metal. The MRP-175 SCC screening criteria for austenitic stainless steel welds are $\geq$ 30 ksi total stress (applied and residual) ***and*** < 5 percent ferrite. The MRP-175 screening criteria are listed in MRP-189-Rev. 1, Table 3-1 on page 3-2. In accordance with ASME Code, the original B&W RV internals

equipment specification specified the ferrite content for austenitic stainless steel welds to be a minimum of 5 percent. Therefore, this weld (WC-43) was screened out of SCC (i.e., Category A for SCC) based on the weld ferrite requirement.

In the process of preparing MRP-189 Rev. 01, the internals welds were reviewed by AREVA and put through the MRP-175 screening process, as noted Section 3.2 (page 3-15). This was completed by reviewing fabrication records of all internals welds for the Davis-Besse internals. In addition, the post-weld stress relief for all seven B&W units was also reviewed. The results are documented in AREVA proprietary document 51-9027395-000 (Reference 7 of MRP-189 Rev. 1). Since the weld stress relief was not needed to screen out SCC, the stress relief information during the original fabrication was not listed in MPR-189-Rev. 1. The stress relief information is summarized below.

During the review, it was found that many internals welds including all double-groove ASA welds such as the CSS upper flange weld for all seven operating B&W units was required to have a post-weld stress relief treatment either by a mechanical or thermal means or both during the original construction by the original RV internals equipment specification. This practice has been confirmed for four B&W units by the actual signed and dated assembly process sheets. Specifically, there was no exception or deviation from the stress relief requirement for any double-groove ASA welds including the CSS upper flange weld for these four units.

The approved RV internals assembly process to be used for the remaining three units was also found. The approved assembly process also listed the same stress relief steps as for the other four units. However, the actual signed and dated RV internals assembly process sheets for these three B&W units could not be found. These three units were built chronologically between the other four units whose signed and dated process sheets were found. During original construction period for the seven B&W units, several revisions to the RV internals equipment specification were made. In each revision, the stress relief requirement was always listed. Therefore, it was concluded that all double-groove ASA welds including the CSS upper flange weld in all seven operating B&W units received a post-weld stress relief treatment.

Note 5: A/LAI 6

Sections 3.3.6 and 4.2.6 of the draft SE incorrectly identifies and discusses "internal" baffle-to-baffle bolts and their locking devices, and "baffle-to-former bolts and their locking devices" as inaccessible items for B&W units. These should be removed from this applicant/licensee action item.

The "internal baffle-to-baffle bolts and their locking devices" are accessible. MRP-231 Rev. 1 Section 3.2.2 contains a discussion of why the internal baffle-to-baffle bolts are not considered primary items and that complete failure of all of these bolts along with a significant number of baffle-to-former bolts could occur before a functionality concern would exist with the baffle plates or core barrel assembly. In addition, these bolts and their locking devices are VT-3 visually examined during the ASME Code 10-year ISI

examinations, which is another reason why they are considered Expansion items. These bolts are therefore included in the proposed justification by evaluation or replacement effort for the other bolt locations that are inaccessible (i.e., external baffle-to-baffle bolts and core barrel-to-former bolts).

The "baffle-to-former bolts and their locking devices" are accessible items and identified as primary items to be examined, as given in Table 4-1 of MRP-227 Rev. 0. Rather, Section 4.2.6 of the draft SE should be rewritten to state "core barrel-to-former bolts and their locking devices".

Note 6: A/LAI 7

Sections 3.3.7 and 4.2.7 of the draft SE incorrectly identifies "B&W in-core monitoring instrumentation (IMI) guide tube assemblies" as a CASS component. The correct term is "B&W in-core monitoring instrumentation (IMI) guide tube assembly spiders." The spider is the only CASS item in the IMI guide tube assemblies.

The recommended methodology for acceptance of the IMI guide tube assembly spiders (given in WCAP-17096) is to perform a generic analysis to show that one or more missing spider arms or a completely missing spider will not result in loss of function of the IMI guide tube. Thus, no fracture mechanics evaluations are needed. Since there are 52 IMI guide tubes in each B&W unit, a redundancy argument may also be adequate.

For the spacer castings inside the CRGT assembly, a generic analysis is being proposed within the PWROG to quantify the stress distribution in the spacer castings as a function of failed screw locations. The results of this evaluation would be used to justify the assumption that a failed threaded screw location would not lead to rapid successive failures. Since there are 69 CRDMs in each B&W unit, an alternative methodology for acceptance of degradation or failure of the CRGT assembly spacer castings (given in WCAP-17096) is to perform a unit-specific reactivity analysis to determine the number of CRDMs that are required for shut down of the reactor. However, this would be specific to each fuel cycle and would not be performed until it was known which particular CRDM would potentially not be operable. Thus again, no fracture mechanics evaluations are needed.

Note 7: A/LAI 8

Section 3.5.1 could be interpreted to require items 1 thru 5 for all licensees when they are only appropriate for those submitting new license renewal applications. Items 3 thru 5 have already been provided by licensees during their original license renewal application submittals. Changes are provided for additional clarity on which of the listed actions in 3.5.1 are required for licensees with a renewed license and which are for new applicants for license renewal.

In Item 5 the third paragraph discusses addressing TLAA CUF analyses for vessel internals and requires that the effects of reactor water environment must be included to

satisfy NRC Code requirements. Currently license renewal guidance in NUREG-1800 (Section 4.3.2.1.3) and NUREG-1801 (X.M1 program scope) only require the consideration of environmental effects for reactor coolant pressure boundary components. They do not include this requirement for reactor vessel internal components since they are not reactor coolant pressure boundary. A review of the 2007 version of these sections of ASME Code Section III did not reveal any requirement to include environmental effects.

Attachment 3
MRP Mark Up
NRC Draft Safety Evaluation of MRP-227
May 10, 2011

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

MATERIALS RELIABILTY PROGRAM: PRESSURIZED WATER REACTOR INTERNALS

INSPECTION AND EVALUATION GUIDELINES (MRP-227, REVISION 0)

PROJECT NO. 669

1.0 INTRODUCTION

1.1 Background

By letter dated January 12, 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090160204), the Electric Power Research Institute (EPRI) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review and approval Materials Reliability Program (MRP) Report 1016596 (MRP-227), Revision 0, "Pressurized Water Reactor (PWR) Internals Inspection and Evaluation Guidelines." A non-proprietary version of MRP-227, Revision 0 is also available in ADAMS at Accession No. ML003691748.

By letter dated March 2, 2010 (ADAMS Accession No. ML100640166), EPRI informed the NRC that MRP-227 Revision 0, was made publicly available and is no longer proprietary.

MRP-227, Revision 0 contains a discussion of the technical basis for the development of an aging management program (AMP) for reactor vessel internal (Item 1) components in PWR vessels supplied by Westinghouse, Babcock and Wilcox (B&W) and Combustion Engineering (CE). MRP-227, Revision 0 provides inspection and evaluation (I&E) guidelines as part of the AMP for use by the applicants/licensees.

1.2 Purpose

The NRC staff reviewed MRP-227, Revision 0 to determine whether its guidance will provide reasonable assurance that the I&E of the subject reactor internals components will maintain their required performance during the period of extended operation. The review also considered compliance with license renewal (LR) requirements in order to allow licensees or applicants the option of incorporating the MRP-227, Revision 0 guidelines by reference in a plant-specific integrated plant assessment (IPA) related to the AMP and associated time-limited aging analyses (TLAAs).

1.3 Organization of the Safety Evaluation

Section 2.0 of this safety evaluation (SE) summarizes MRP-227, Revision 0. Section 3.0 documents the staff's evaluation and findings pertaining to the adequacy of the MRP's AMP recommendations. In particular, Section 3.0 documents staff concerns with MRP-227, Revision

0 and the basis for limitations and conditions being placed on the use of MRP-227 as well as licensee/applicant action items that shall be addressed by applicants/licensees who choose to implement the NRC-approved version of MRP-227. Section 4.0 summarizes the limitations and conditions and the applicant/licensee action items. Section 5.0 provides the conclusions resulting from this SE.

1.4 Regulatory Requirements

Title 10 of the *Code of Federal Regulations* (CFR) Part 54 addresses the requirements for plant license renewal. The regulation at 10 CFR Section 54.21 requires that each application for LR contain an IPA and an evaluation of TLAAs. The IPA shall identify and list those structures and components subject to an aging management review (AMR) and demonstrate that the effects of aging (cracking, loss of material, loss of fracture toughness, dimensional changes, loss of preload) will be adequately managed so that their intended functions will be maintained consistent with the current licensing basis (CLB) for the period of extended operation as required by 10 CFR 54.29(a). In addition, 10 CFR 54.22 requires that a LR application include any technical specification (TS) changes or additions necessary to manage the effects of aging during the period of extended operation as part of the LR application.

Structures and components subject to an AMP shall encompass those structures and components that (1) perform an intended function, as described in 10 CFR 54.4, without moving parts or without a change in configuration or properties and (2) are not subject to replacement based on a qualified life or specified time period. These structures and components are referred to as "passive" and "long-lived" structures and components, respectively. The scope of components considered for inspection under MRP-227, Revision 0 guidance includes core support structures (typically denoted as Examination Category B-N-3 by the American Society of Mechanical Engineers (ASME) Code, Section XI) and those reactor internals components that serve an intended LR safety function pursuant to criteria in 10 CFR 54.4(a)(1). The scope of the program does not include consumable items such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation because these components are not typically within the scope of the components that are required to be subject to an AMP, as defined by the criteria set in (Item 2)
10 CFR 54.21(a)(1).

Some owners of PWR units were granted renewed licenses and each of these licensees made a commitment to conform with the recommendations specified in NUREG-1801, "Generic Aging Lessons Learned (GALL), Revision 1, AMP XI.M16, "PWR Vessel Internals." AMP XI.M16 requires that the applicant provide a commitment in the Final Safety Analysis Review (FSAR) supplement to (a) participate in the industry programs for investigating and managing aging effects on reactor internals components; (b) evaluate and implement the results of the industry programs as applicable to the reactor internals components; and (c) upon completion of these programs, but not less than 24 months before entering the period of extended operation, submit an inspection plan for reactor internals components to the NRC for review and approval. Each applicant/licensee that made a commitment to conform with the recommendation specified in NUREG-1801, Revision 1, AMP XI.M16 also made a commitment in its FSAR that it will implement (Item 2) an industry developed AMP for its reactor internals components.

If a LR applicant confirms that it will implement MRP-227, Revision 0 guidelines, as modified by this SE, at its plant, then no further review of the AMP for the PWR internals components is

necessary, except as specifically identified in Section 4.0 of this SE. With these exceptions, an applicant may rely on the MRP-227, Revision 0 report for the demonstration required by Section 54.21(a)(3) with respect to the reactor internals components and structures within the scope of the report. Under such circumstances, the staff intends to rely on the evaluation in this SE to make the findings required by 10 CFR 54.29 with respect to a particular application.

2.0 SUMMARY OF MRP-227

MRP-227, Revision 0 contains a discussion of the technical basis for implementing inspection requirements for PWR internals components that are subject to any of the applicable degradation mechanisms (e.g., stress corrosion cracking (SCC), (Item 3) ~~intergranular stress corrosion cracking (IGSCC),~~irradiation-assisted stress corrosion cracking (IASCC), wear, fatigue, thermal and/or neutron embrittlement, void swelling, and irradiation-enhanced stress relaxation) during the LR period. This report, in addition, provides a brief, high-level summary of flaw evaluation guidelines for reactor internals components that exhibit active degradation mechanisms, and establishes requirements for inspection of additional components if an active degradation mechanism is discovered (i.e., expansion of the scope of reactor internals component inspections). Extensive information was provided with respect to the effects of the applicable degradation mechanisms on various reactor internals components and the inspection requirements for these components.

The following sections include a brief description of the information contained in MRP-227, Revision 0.

2.1 MRP-227, Revision 0 - Section 1

Section 1 of the MRP-227, Revision 0 report includes an overall synopsis related to aging management of the PWR internals components by identifying the following steps in the MRP's process for developing the AMP: (1) development of screening criteria for the applicable degradation mechanisms; (2) screening of the different reactor internals components designed by Westinghouse, B&W, and CE based on the components' susceptibility to degradation; (3) functionality analyses and failure modes, effects, and criticality analyses (FMECAs) performed for the components which resulted in the binning of components into different inspection categories; and (4) development of the proposed I&E guidelines and flaw evaluation methodology.

Step (1) of this process was not discussed in MRP-227, Revision 0 but was documented in MRP-175, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values." MRP-227 also referenced MRP-211, "Materials Reliability Program: PWR Internals: Age Related Material Properties Degradation Mechanisms, Models and Basis Data," which addresses screening criteria for the degradation mechanisms in PWR internals components. Screening of PWR internals components for susceptibility to the degradation mechanisms was performed by establishing a set of screening criteria for each relevant degradation mechanism. The MRP-175 report provided technical data that was obtained from experiments to provide the basis that the MRP used to develop the screening criteria for different degradation mechanisms. The screening criteria for the degradation mechanisms considered in MRP-227, Revision 0 depend on various factors. For example, the screening factors for SCC depend on type of material and applied stress.

2.2 MRP-227, Revision 0 - Sections 2 and 3

In Sections 2 and 3 of the report, the MRP provided an expanded discussion regarding steps (2) and (3) identified in Section 2.1 of this SE. In this SE, these steps, which lead up to the binning of components into inspection categories, may be referred to as the "categorization" phase of the MRP's process.

As background material, Section 3 of MRP-227, Revision 0 discussed the various design characteristics, and their functions, of the reactor internals components supplied by Westinghouse, CE, and B&W. This section also discussed potential aging effects that may result from the identified degradation mechanisms. These aging effects included: (1) various forms of cracking, (2) loss of material induced by wear; (3) loss of fracture toughness due to either individual or synergistic contributions from thermal aging or neutron irradiation embrittlement; (4) dimensional changes and potential loss of fracture toughness due to void swelling and irradiation growth; and (5) loss of preload due to either individual or synergistic contributions from thermal and irradiation-enhanced stress relaxation or creep.

Initial screening of reactor internals components for all three (B&W, CE, and Westinghouse) designs was based on a consideration of material properties (e.g., chemical composition) and operating conditions (e.g., neutron fluence exposure, temperature history, and representative stress levels) in order to determine the susceptibility of PWR internals components to the applicable aging mechanisms. This resulted in the binning of these reactor internals components as either susceptible or not susceptible to each of the eight degradation mechanisms, based on the degradation screening criteria.

Next, the MRP performed a failure modes, effects and criticality analysis (FMECA) of the reactor internals components. The FMECA process was discussed in detail in MRP-190, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals," and MRP-191, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs." The FMECA was a qualitative process that included expert elicitation by technical experts. Expert elicitation was used for developing the technical basis for categorization of various reactor internals components under different categories based on the combination of the likelihood of component degradation due to one or more of the eight degradation mechanisms, and the severity of safety consequences. Each component was assigned to one of three categories (for each degradation mechanism) ranging from insignificant effects (Category A) to potentially moderately significant effects (Category B) to potentially significant effects (Category C). Category C components were associated with higher risk in that they are more susceptible to aging degradation and the consequences of their failure are more severe. Category C components were also often considered the likely lead components for providing telltale signs of the associated aging degradation. Category B components, on the other hand, can still be susceptible to aging degradation but their consequences of failure are typically less than Category C components. Category A components are (Item 4) a) those for which ~~have been judged to be not susceptible to any of the eight degradation mechanisms. That is, each of the degradation mechanisms is believed to be~~ aging effects are below the ~~associated~~ screening ~~criterion used~~ criteria, so that age-related degradation significance is minimal and in ~~MRP-227, Revision 0~~addition, b) those that are just above the screening criteria but tolerant of aging effects with no loss of functionality as determined through the FMECA or additional evaluations.

The MRP then performed a functionality assessment of the PWR internals components and items that would most be affected by the degradation mechanisms (i.e., preliminary Category B and C items from the FMECA). This assessment was based on representative plant designs using irradiated and aged material properties. The functionality analyses included finite element analyses (FEA) on selected reactor internals components that were deemed to be susceptible to irradiation-induced degradation mechanisms (e.g., IASCC, neutron embrittlement, void swelling, and irradiation-induced stress relaxation) where the effects are dependent on multiple variables and develop with time to assess the evolution of degradation. The functionality analyses were used to demonstrate that although some Category C components were susceptible to one or more degradation mechanisms, the effect of the degradation mechanisms on their performance was not significant.

It should be noted that the FMECA and functionality analyses were based on the assumption of thirty years of operation with high leakage core loading patterns followed by thirty years of low leakage core loading patterns. In the U.S. PWR fleet, low leakage core loading patterns were implemented early in the unit's operating lives. Hence, MRP considered this assumption conservative. The MRP also assumed a base load operation such that the modeled plants operate at fixed power levels and do not vary power on a calendar or load demand schedule.

Industry considered the results from the FMECA and functionality analysis along with operating experience, component accessibility, and existing inspection programs to develop the recommended inspection categories for maintaining the long-term functionality of PWR internals components. In Section 3, the MRP, based on this assessment, developed four inspection categories:

1. Primary – reactor internals components that are either highly susceptible to effects of aging due to any active degradation mechanism, or components that have a degree of tolerance for a specific degradation mechanism but for which no leading highly susceptible or accessible component exists. These components are to be periodically inspected as part of a reactor internals component AMP.

2. Expansion – reactor internals components that are moderately or highly susceptible to the effects of aging due to one or more active degradation mechanisms, but for which the functionality analyses indicated that these components have a degree of tolerance to the aging effects associated with these degradation mechanisms. These components will be inspected as part of a reactor internals component AMP if unacceptable degradation is identified during inspections of relevant "Primary" inspection category components.

3. Existing (Programs) – reactor internals components that are susceptible to the effects of aging due to one or more active degradation mechanisms, but that are managed under an existing generic or plant-specific AMP currently implemented by the PWR fleet, which adequately manages the aging effect. MRP-227, Revision 0 consistently calls this category the "Existing" inspection category, but for clarity it will be referred to as the "Existing (Programs)" inspection category in this SE.

4. No Additional Measures – reactor internals components that are below the screening criteria for the applicable degradation mechanisms, or were classified under this category due to FMECA and functionality analysis findings. (Item 5) ~~These components~~

~~are not to be inspected as part of a RVI component AMP~~No further action is required by the MRP-227 Rev. 0 Guidelines for managing the aging of these components.

Tables 3-1 through 3-3 in Section 3 of MRP-227, Revision 0 summarize the proposed inspection categories for each B&W, CE, and Westinghouse reactor internals component that was initially placed into Categories B and C as a result of the initial screening and FMECA analyses. These tables identify the proposed inspection categories associated with each of the individual degradation mechanisms as well as the final grouping. The final I&E guidelines were based on the summary classifications contained in these tables.

2.3 MRP-227, Revision 0 - Sections 4 and 5

In Sections 4 and 5 of MRP-227, Revision 0, a detailed discussion regarding: (1) the examination method to be applied for a particular component based on its final categorization (see Section 2.2 of this SE); (2) qualifications for the examinations; (3) examination frequency; (4) sampling and coverage; (5) expansion scope of examination based on the extent of observed degradation; and (6) evaluation of examination results. In this SE, the staff will refer to this information as the MRP's proposed I&E guidelines for components subject to MRP-227. Tables 4-1, 4-2, and 4-3 of MRP-227, Revision 0 address the identification of "Primary" inspection category components, their relevant aging effects, and the type of examination methods to be used for plants designed by B&W, CE, and Westinghouse, respectively. Similar information is provided in Tables 4-4, 4-5, and 4-6 for the "Expansion" inspection category components designed by B&W, CE, and Westinghouse, respectively. Tables 4-8 and 4-9 include similar information for some components in the "Existing (Programs)" inspection category for plants designed by CE and Westinghouse, respectively. No existing generic industry programs were considered sufficient to monitor the aging effects in reactor internals components designed by B&W and, hence, no Table 4-7 was included. Although categorized under the "Existing (Programs)" inspection category, CE thermal shield positioning pins, CE in-core instrumentation (ICI) thimble tubes, and Westinghouse guide tube support pins (split pins) were not included in Tables 4-8 and 4-9 because the adequacy of the plant-specific existing programs to manage degradation of these components for the period of extended operation could not be verified in the development of MRP-227, Revision 0.

The examination methods endorsed by MRP-227, Revision 0 include: (1) ASME Code, Section XI, visual (VT-3 and VT-1) examinations; (2) enhanced visual (EVT-1) and VT-1 examinations; (3) surface examination [eddy current testing (ET)], (4) volumetric examination using ultrasonic techniques (UT), and (5) physical measurements. Selection of an examination method was based on the characterization of a particular degradation mechanism. It was also based on the examination method that is capable of identifying the aging effect associated with the degradation mechanism. MRP's proposed examinations are to be implemented by well-established standard procedures and these procedures are to be qualified per industry inspection standards addressed in MRP-228, "Materials Reliability Program: Inspection Standard for Reactor Internals." Some examination methods require additional qualifications per ASME Code, Section V, "Non-Destructive Examinations."

In general, the "Primary" and "Existing (Programs)" inspection category components are to be examined once during every 10-year ISI interval. Tables 4-1, 4-2, 4-3, 4-8, and 4-9 address the frequency of examinations to be used for these components in plants designed by B&W, CE, and Westinghouse. For some components (e.g., baffle bolts), MRP-227, Revision 0 specifically

notes that the frequency of examination may be increased based on inspection results. In general, operating experience gathered from inspections conducted in accordance with the NRC-approved version of MRP-227 will be reviewed and used to update inspection requirements.

Tables 4-1, 4-2, 4-3, 4-4, 4-5, 4-6, 4-8, and 4-9 address the requirements for the examination coverage for reactor internals components in plants designed by B&W, CE, and Westinghouse. In addressing the coverage to be obtained when examinations are performed, MRP-227, Revision 0 states that for all "Primary" and "Expansion" inspection category components, one hundred percent of accessible surfaces/volumes are required to be examined, with the exception of some components for which limited accessibility is known to exist. In this case, known limited accessibility was related to the need to disassemble the reactor internals components in order to achieve full accessibility to all of a set of like components for examination. Types of like components with known limited accessibility included, for example, Westinghouse guide cards in control rod guide tube (CRGT) assemblies. For these sets of components, MRP-227, Revision 0 required an inspection sample, ranging from 10 percent to 20 percent of each subject set of like components. For the 10 percent to 20 percent sample of each set of components to be inspected, MRP-227, Revision 0 required that one hundred percent of the accessible surfaces/volumes be examined.

MRP-227, Revision 0 addressed the examination of "Expansion" inspection category components, which is based on the extent of aging degradation observed in a related "Primary" inspection category component, in Tables 4-4, 4-5, 4-6, 5-1, 5-2, and 5-3. The criteria for initiating the examination of the "Expansion" inspection category components is based on the column on the linkage between the "Primary" and "Expansion" inspection category components established in these tables. In general, a single "Primary" inspection category component that is being inspected to monitor for a particular degradation mechanism may be linked to more than one "Expansion" inspection category component. The observation of degradation in the "Primary" inspection category component could trigger the need to examine the associated "Expansion" inspection category components, depending on the licensee's evaluation of the significance of observed degradation in the "Primary" inspection category component. Certain "Expansion" inspection category reactor internals components were determined to be completely inaccessible for examination, including the B&W core barrel cylinder (including vertical and circumferential seam welds), former plates, external baffle-to-baffle bolts and their locking devices, (Item 6) ~~baffle~~core barrel-to-former bolts and their locking devices, and core support shield vent valve disc shafts or hinge pins. For these inaccessible "Expansion" category components, MRP-227, Revision 0 stated that, when their inspection is called for based upon the observation of a degradation mechanism in the associated "Primary" inspection category component, the applicant/licensee must evaluate the continued operability of the inaccessible "Expansion" inspection category component or, alternatively, replace the component.

With regard to the evaluation of examination results, Tables 5-1, 5-2, and 5-3 and the text of Section 5 provide: (1) relevant conditions for each specified examination method and (2) general guidance on the evaluation of relevant conditions for plants designed by B&W, CE, and Westinghouse, respectively. For example, for EVT-1 examinations, the specific relevant condition identified in MRP-227, Revision 0 is a detectable crack on the surface of a reactor internals component. The acceptance criteria then provided for the relevant conditions associated with this examination method was that only the absence of a relevant condition would require no further evaluation. An acceptable process to disposition relevant conditions

may include supplemental examinations, accepting the condition until the next examination, or replacement of the component. The outcome of the evaluation of the relevant condition may also affect the implementation of the examination of associated "Expansion" inspection category components.

2.4 MRP-227, Revision 0 - Section 6

Section 6 of the MRP-227, Revision 0 provided guidance on the application of flaw evaluation methodologies to be implemented when an examination reveals the presence of a relevant condition. Various subsections in Section 6 provided details on:

1. The loading conditions to be considered when evaluating core support structures, including deadweight loads, mechanical loads, hydraulic loads, thermal loads, and loads from operating basis and safe shutdown earthquakes.

2. The requirements and limitations (based on accumulated neutron fluence) for the application of limit load evaluation methodologies for flawed reactor internals components. The requirements include application of limit load procedures similar to those given in ASME Code, Section XI.

3. The application of linear elastic fracture mechanics (LEFM) and elastic-plastic fracture mechanics (EPFM) for reactor internals components with an accumulated neutron fluence that exceeds the limit load application threshold limit.

4. The application of existing crack growth rate values for the evaluation of SCC in stainless steel components and IASCC in irradiated stainless steel components.

5. The evaluation of flaws in bolts and bolted assemblies. This includes the assessment of the functionality of bolted assemblies that may contain one or more non-functional bolts. This evaluation is to be based on the minimum number required to maintain the functionality of the assembly until the next examination.

While this evaluation guidance is included in MRP-227, it is important to note that the industry has submitted WCAP-17096-NP for staff review. This WCAP report supersedes the guidance contained in Section 6 of MRP-227. The guidance in the WCAP will be used to evaluate component degradation that exceeds the acceptance criteria in Section 5 when it is observed during required inspections.

2.5 MRP-227, Revision 0 - Section 7

Section 7 of MRP-227, Revision 0 provided a summary of the implementation requirements for the guidelines described in the MRP-227, Revision 0 report. The implementation requirements are defined by the latest edition of Nuclear Energy Institute (NEI) Implementation Protocol NEI 03-08, "Guidelines for the Management of Materials Issues," which includes implementation categories used in MRP-227, Revision 0 including: (a) "Mandatory," which requires implementation of the guidelines at all plants; (b) "Needed," which provides an option for implementing the guidelines wherever possible or implementing alternative approaches, or (c) "Good Practice," which recommends implementation of the guidelines as an option whereby significant operational and reliability benefits can be achieved at a given plant. Failure to meet a

"Needed" or a "Mandatory" requirement is a deviation from the guidelines and a written justification for deviation must be prepared and approved as described in Addendum D to NEI-03-08. A copy of the deviation is sent to the MRP so that, if needed, improvements to the guidelines can be developed. A copy of the deviation is also sent, for information, to the NRC.

Section 7 of MRP-227, Revision 0 specified the following with respect to the implementation of specific MRP-227, Revision 0 guidelines:

1. Each PWR unit shall (Item 7a) ~~implement MRP-227, Revision 0~~develop and document a PWR reactor internals aging management program (AMP) within thirty-six months following ~~the~~ issuance of ~~this report.~~MRP-227 Rev. 0. This is a "Mandatory" requirement.

2. Each PWR unit shall implement Tables 4-1 through 4-9 and Tables 5-1 through 5-3 of the MRP-227, Revision (Item 7b) ~~0~~A report for the applicable design within twenty-four months following the issuance of MRP-227, Revision ~~0.~~A. This is a "Needed" requirement.

3. Examination of the reactor internals components shall comply with the MRP-228 Revision 0, "Materials Reliability Program: Inspection Standard for PWR Internals." This is a "Needed" requirement.

4. Examination results that do not meet the examination acceptance criteria defined in Section 5 of the MRP-227, Revision 0 guidelines shall be recorded and entered in the plant corrective action program and dispositioned.

5. A summary report of all inspections and monitoring, evaluation, and new repairs shall be provided within one hundred and twenty days of the completion of an outage during which the reactor internals components were examined. The summary of the examination results shall be included in an industry report that is updated every six months. This report will monitor the industry progress on the AMP related to PWR reactor internals components and it will also list the emerging operating experience. This is a "Good Practice" requirement.

2.6 MRP-227, Revision 0 - Appendix A

Appendix A addresses how the AMP defined in MRP-227, Revision 0 meets specific AMP attributes as defined by NUREG-1801, the License Renewal Generic Aging Lessons Learned report. Specifically, Appendix A discusses how the MRP-227, Revision 0 program meets the "Scope of Program" (Attribute 1 from NUREG-1801), "Parameters Monitored" (Attribute 3 from NUREG-1801), and "Detection of Aging Effects" (Attribute 4 from NUREG-1801). Appendix A also stated that supplementary information (Item 8) ~~shall~~must be ~~provided~~assembled by the applicants/licensees when submitting their AMPs to satisfy all of the NUREG-1801 AMP requirements for the remaining program elements~~ when implementing MRP-227, Revision 0~~.

3.0 STAFF EVALUATION

The staff reviewed the MRP-227, Revision 0 report to determine if it demonstrated that the effects of aging on the components covered by the report would be adequately managed so that

the components' intended functions would be maintained consistent with the CLB for the period of extended operation, in accordance with 10 CFR 54.21(a)(3). Besides the IPA, Part 54 requires an evaluation of TLAAs, in accordance with 10 CFR 54.21(c). The staff reviewed the MRP-227, Revision 0 report to determine if the TLAAs covered by the report were evaluated for LR in accordance with 10 CFR 54.21(c).

During its review of MRP-227, Revision 0, the staff issued four sets of requests for additional information (RAIs) that addressed technical issues. The details of the staff's RAIs and the corresponding responses are available in ADAMS (proprietary version). However, the staff did not include all the RAIs and the MRP's responses in this SE; it included only those salient RAIs and MRP responses that address specific points of emphasis. References 15 through 17 contain all of the staff's technical RAIs and the MRP's responses.

3.1 Evaluation of MRP-227, Revision 0 - Section 1

The staff reviewed Section 1 of the MRP-227, Revision 0 and accepts the approach used by the MRP to develop the screening criteria for initially binning the reactor internals components into Category A, B, and C. In this section, the MRP provided technical data that was used as the basis for the screening criteria for different degradation mechanisms. The screening criteria were based on: (1) type of material used in reactor internals components, (2) operating stress levels, and in some cases, (3) neutron fluence values. For example, IASCC screening criteria were established by (1) type of material, (2) threshold limit of neutron fluence value and (3) stress values. The threshold limits for neutron fluence and stress levels were developed by valid research data that is widely used by the industry. Similar criteria were developed for the other degradation mechanisms. The NRC staff has not officially reviewed the technical basis for the screening criteria that is contained in MRP-175 and MRP-211. Therefore, the NRC staff does not specifically endorse the screening criteria used in MRP 227. However, the MRP-227, Revision 0 strategy of identifying "Primary" inspection components based on the relative likelihood of degradation compared to other components diminishes the importance of the specific screening criteria values used in MRP-227, Revision 0.

3.2 Evaluation of MRP-227, Revision 0 - Sections 2 and 3

The staff's review of Sections 2 and 3 of MRP-227, Revision 0 resulted in the staff, in principle, accepting the MRP's categorization process for the development of (Item 2) an AMP for the reactor internals components. The MRP considered susceptibility of reactor internals components to one or more degradation mechanisms and the safety consequences as a result of the failure of the reactor internals components. However, the staff identified some concerns with the MRP's categorization process and/or its application. The staff's evaluation of the MRP's process is provided below, focusing on the staff's concerns (Item 2), which led to the imposition of conditions and limitations on the use of MRP-227, Revision 0 and plant-specific action items associated with the use of
MRP-227, Revision 0 (as discussed in Section 4 of this SE).

3.2.1 General Evaluation of MRP's Categorization Process - Initial Screening, FMECA, Functionality Analyses, and the Assigning of Components to Inspection Categories

In Sections 2 and 3 of MRP-227, Revision 0, the MRP discussed at length, their categorization process for various reactor internals components. The categorization process (i.e., initial

screening, FMECA, and functionality analyses) described in MRP-227, Revision 0 provides an adequate approach for identifying the degradation mechanisms for reactor internals components within the scope of LR. Those components that were assessed to be most affected by one or more of the degradation mechanisms addressed in Section 2.0 of this SE were binned under Category C, those components that were expected to be moderately affected by the degradation mechanisms were binned under Category B, and components that were expected to be (Item 2) ~~un~~insignificantly affected by the degradation mechanisms were binned under Category A. The initial screening process entailed evaluation of material properties, corrosion resistance of materials, the effect of neutron fluence on some components, and loading conditions. The staff concluded that the MRP had adopted a systematic approach in the initial screening of the reactor internals components into various categories, and the staff accepts this approach.

The staff, in principle, also agrees with the technical basis used in the development of the recommended component inspection groupings identified in Section 2.2 of this SE based, in part, on using FMECA and functionality analysis. However, in its review of the FMECA process described in MRP-190 and MRP-191 and the functionality analyses described in MRP-229 and 230, the staff identified concerns with the MRP's approach. Some of the staff's concerns were resolved via MRP responses to staff RAIs, while concerns that were not adequately resolved are reflected in plant-specific action items and/or conditions and limitations on the use of MRP-227, Revision 0. Examples of significant staff concerns that were resolved are given in the following paragraphs, and those that were not adequately resolved are addressed in Sections 3.2.2, 3.2.3, and 3.2.4 of this SE.

The staff requested that the MRP address the impact of the potential aging effects on the reactor internals components and reactor system performance in transient and accident conditions. In its response, the MRP provided information to demonstrate that component loadings assumed in the FMECA process included normal operating loads and, in some cases, both normal operating loads and transient loadings. The MRP stated that the expert elicitation process also assessed the safety implications of potentially failed components, and that it could be inferred that the non-escalation of consequences was considered during the FMECA process. The MRP also stated that, as discussed in MRP-190, the expert elicitation process explicitly considered whether the aging effects considered in the FMECA process would result in more severe consequences if a design basis transient occurred. Further, the MRP indicated that if degradation is found during inspections, the subsequent evaluation of the degraded component's integrity is performed using the guidance in WCAP-17096-NP, Revision 2 (Item 2) , which is currently under staff review. The WCAP evaluation requires that acceptable component performance be demonstrated under all design basis conditions such that the licensing basis is maintained. Component repair or replacement is required if this evaluation demonstrates that the licensing basis cannot be maintained. The staff accepts this response and this issue is resolved pending the review of WCAP-17096-NP, Revision 2.

The staff also had concerns associated with some of the FMECA results and the outcome of some of the functionality analyses. Some reactor internals components that were originally identified for potential aging degradation due to single or multiple aging degradation mechanisms (Categories B and C) were placed under the "No Additional Measures" inspection category as a result of the FMECA or functionality analyses. The staff was concerned that these components could be subject to damage and possible deterioration of the original mechanical properties due to aging degradation. Hence, the structural integrity of these reactor

internals components could be challenged under licensing basis loading conditions. The MRP provided a few examples and included acceptable technical justification for categorizing some reactor internals components from Category B and C to the "No Additional Measures" Inspection category. The examples include: (1) Westinghouse bottom mounted instrumentation cruciforms, and (2) Westinghouse lower core plate fuel alignment bolts. The staff accepts their response and considers this issue resolved.

3.2.2 (Item 9a) ~~High Consequence~~Core Support Components in the "No Additional Measures" Inspection Category

During the review of the FMECA process, the staff identified a concern regarding the categorization of some of the (Item 9b) ~~RVI components whose failure could cause significant safety consequences.~~reactor internals components that are part of the core support structure. In some cases, the MRP placed these components under the "No Additional Measures" inspection category. The following paragraphs discuss ~~the categorization of~~ these (Item 9c) ~~high consequence RVI~~ core support components. The relevant ~~high consequence~~ components are: (1) the upper core plate and lower support ~~forging~~forgings or casting in Westinghouse-designed reactors~~,~~
(Item 2) and (2) the lower core support beams, core support barrel assembly (CSBA) upper cylinder and CSBA upper core barrel flange in CE-designed reactors ~~and (3) the lower grid-to-core barrel bolts in B&W-designed reactors~~. (Item 10a)

CE and Westinghouse reactor internals components were grouped in risk categories as part of the FMECA based on the combination of (1) their likelihood of failure and (2) (Item 9d) ~~a qualitative assessment of the potential for core~~ their likelihood of damage ~~associated with their failure.~~ to the reactor. Although it is recognized that loss of core support would have significant safety consequences, the FMECA panel concluded that aging related degradation would not necessarily result in loss of core support and determined that the likelihood of damage in these components ranged from low to high. The staff's concern is related to ~~those components that were qualitatively assessed as having a "high" potential for~~ core (Item 9e) ~~damage associated with their failure (i.e., high consequence~~support components~~)~~ that are not already identified for inspection within the "Primary" or "Expansion" categories. ~~An RVI component was considered to have a "high" potential for core damage when it was believed that some core damage could result from failure of the component, for example, related to the inability to safely shutdown the reactor.~~ The likelihood of degradation in these components was typically assessed in MRP-227, Revision 0, as being "low." A component was identified as having a "low" likelihood of failure when there were no known failures of this component based on operating experience, and it is believed that the failure is unlikely to occur during extended period of operation. (Item 10b)~~A similar approach was used for the B&W components, although different terminology was used. For B&W components, those in "Risk Band III" were understood to be similar to the combination of "high" potential for core damage associated with their failure and a "low" likelihood of failure from the Westinghouse/CE characterization.~~

The staff determined that the MRP did not provide an adequate justification regarding how these (Item 9f) ~~high consequence/~~core support components with low likelihood of failure ~~RVI components~~ were assigned to the "No Additional Measures" inspection category. The staff is concerned that these components could be subject to loss of structural integrity due to one or more degradation mechanisms. To ensure that the structural integrity and functionality of these reactor internals components are maintained under all licensing basis conditions during the

period of extended operation, the staff ~~has determined~~requires that these components shall be included in the (Item 9g) ~~"Expansion" inspection category in the NRC approved version of MRP-227. The staff proposed "Primary" inspection category links for the upper core plate and lower support forging or casting in Westinghouse-designed reactors, the lower core support beams, upper cylinder and upper core barrel flange in the core support barrel assembly in CE-designed reactors and the lower grid-to-core barrel bolts in B&W-designed reactors in~~ASME Section ~~4.1.1 of this SE.~~XI examination of core support structures. To assure that all plants consider these components part of the scope of the ASME Section XI examination, the listed core support components will be added to the list of "Existing" inspections in MRP-227 Revision A. The examination method and coverage requirements for to be used for these additional "~~Expansion~~Existing" inspection category components shall be consistent with the (Item 9h) ~~examination method for the "Primary" inspection category component to which they are linked. The expectations regarding the examination coverage and re-examination frequency are addressed in Sections 4.1.5 and 4.1.7 of this SE.~~ ASME Section XI requirements. **This is addressed as Topical Report Condition 1 in Section 4.1.1.**

3.2.3 Inspection of Components Subject to Irradiation-Assisted Stress Corrosion Cracking

MRP-227, Revision 0 grouped the following components under the "Expansion" inspection category: (1) the upper and lower core barrel welds and lower core barrel flange weld in Westinghouse-designed reactors; and (2) the lower cylinder welds in the core support barrel assembly (CSBA) in CE-designed reactors. (Item 11a) ~~These components were qualitatively assessed as having a "high" potential for core damage associated with their failure (i.e., they are high consequence components) and a "medium" likelihood of failure.~~The upper flange and upper and lower cylinders of the Westinghouse core barrel, and the CE core support barrel are identified in the FMECA process as having a high damage likelihood, recognizing the fact that failure of these components could result in core damage. These components were determined to be susceptible to aging effects due to SCC~~,~~,(Item 11b) with highly irradiated sections also being potentially susceptible to IASCC and neutron embrittlement.- (Item 11c) Due to the presence of multiple potential failure mechanisms, these components were assigned a "medium" likelihood of failure. In MRP-227~~,~~ Revision 0, the corresponding "Primary" inspection category components were the upper core barrel flange weld in Westinghouse-designed reactors and the upper core support barrel flange weld in CE-designed reactors. These "Primary" inspection category components (Item 11d) ~~are prone~~were judged most susceptible to SCC, but were judged to be not susceptible to aging effects due to IASCC and neutron embrittlement. (Item 11e) Their linked "Expansion" components, (1) and (2), above were likewise judged to have low susceptibility to IASCC based on the effect of radiation induced stress relaxation of the otherwise sensitized welds.

Unlike SCC, the onset of degradation due to IASCC and neutron embrittlement depends on neutron fluence (Item 2) ~~and~~ in addition to the stress levels. (Item 11f) ~~The incubation period for initiating cracks due to SCC is different from IASCC.~~ The MRP employed a fluence-dependent threshold stress to evaluate the likelihood of IASCC based on these two factors. The ratio of the operating stress to the threshold stress (designated IASCC ratio) was used to identify locations susceptible to IASCC. The staff recognizes that many of the remaining core barrel welds do not receive sufficient fluence to make them susceptible to IASCC. However, sections of the welds in the beltline region of the core barrels and support barrels may exceed the required neutron fluence for IASCC susceptibility. For these welds MRP credited irradiation

induced stress relaxation to offset the high residual stresses resulting from fabrication. While the staff recognizes that there may be offsetting benefits from irradiation stress relaxation, there are insufficient data at this time to precisely quantify the effect for large structural welds. Since these aging mechanisms are ~~so~~ different, ~~with respect to crack initiation and crack propagation, any identifiable aging effects associated with~~ SCC in the "Primary" inspection category components may not truly represent (Item 11g) a sufficiently robust link to the extent of actual aging degradation due to IASCC and neutron embrittlement in the associated "Expansion" inspection category components. Lack of any evidence of cracking due to SCC in the "Primary" inspection category components does not mean that the "Expansion" inspection category components are free of cracks due to IASCC. Therefore, the staff is concerned that the aging effects associated with IASCC and neutron embrittlement in the "Expansion" inspection category components may not be identified in a timely manner during the period of extended operation.

To ensure that the structural integrity and functionality of (Item 11h) ~~these high consequence of failure RVI components which are subject to IASCC and neutron embrittlement~~the Westinghouse core barrel and CE core support barrels are maintained under all licensing basis conditions of operation during the period of extended operation, the staff has determined that (Item 11i) the utility must either justify the acceptability of the circumferential welds in the core belt-line region for continued operation by performing an analysis or include these beltline welds in the "Primary" inspection category in the NRC-approved version of MRP-227. The list of degradation mechanisms for these welds shall include IASCC and irradiation embrittlement. The examination method shall meet the requirements as identified for visual inspection of core barrel welds or alternatively permit eddy current or ultrasonic examinations. (Item 11j) ~~shall be included in the "Primary" inspection category in the NRC-approved version of MRP-227. The examination methods shall be consistent with the MRP's recommendations addressed in MRP-227, Revision 0 for these components, the examination coverage for these components shall conform to the criteria described in Section 3.3.1 of this SE, and T~~the examination frequency shall be on a 10-year interval consistent with other "Primary" inspection category components. (Item 11k) In consideration of accessibility issues with these welds, plants must demonstrate coverage of at least 75% of the exterior surface of the potentially susceptible circumferential weld material or, as an alternative to inspection, perform an analysis to justify continued operation per the requirements of Applicant/Licensee Action Item 6 (Section 4.2.6). The remaining core barrel welds would remain as "Expansion" components in Tables 4-5 and 4-6 respectively. **This is addressed as Topical Report Condition 2 in Section 4.1.2 of this SE.**

3.2.4 Inspection of (Item 12a) ~~High Consequence~~Core Support Components Subject to Multiple Degradation Mechanisms

The staff evaluated the effect of multiple degradation mechanisms on the (Item 12a) ~~high consequence reactor internals~~core support structure components and identified that the ~~B&W flow distributor-to-shell forging bolts and~~ CE lower support structure core support column (casting or wrought) welds as needing to be included in the "Primary" inspection category.

(Item 13)~~B&W flow distributor-to-shell forging bolts are susceptible to SCC, fatigue, and wear. Section 3.5 of MRP-190 bases the risk band only on the single most likely aging mechanism. In Table A-1 of MRP-190 (pages A-41 and A-42), the MRP stated that SCC in the flow distributor-to-shell forging bolts is very likely to occur, whereas degradation due to fatigue and wear is less likely to occur. The safety consequence of failure of the subject component, on the other hand,~~

~~is classified as "Severe" which could lead to core damage (i.e., multiple damaged fuel assemblies) with reduced margins to adequately cool the core. While SCC is regarded as the most likely degradation mechanism, the staff is concerned that the synergistic effects of SCC, fatigue and wear could potentially cause greater degradation in these bolts than just the consideration of SCC alone. Due to these synergistic effects, degradation in these bolts could then be equivalent to or greater than other components susceptible only to SCC. Therefore, the staff has concluded that B&W flow distributor-to-shell forging bolts shall be inspected as a "Primary" inspection category component.~~

The CE lower support structure core support column (casting or wrought) welds are susceptible to SCC, IASCC, fatigue, and irradiation embrittlement. In addition to these degradation mechanisms, (Item 12b) ~~this~~ in the absence of plant specific material specifications, the casting component is (Item 12c) assumed to be susceptible to thermal embrittlement. (Item 12d) ~~These components were qualitatively assessed as having a "high" potential for core damage associated with their failure (i.e., they are high consequence components) and a "medium" likelihood of failure.~~The MRP-191 FMECA assigned a "Medium" likelihood of failure due to the potential concern for a component with multiple failure mechanisms. However, the MRP-191 assigned a damage likelihood of "Low" based on the observation that the core support columns are a highly redundant item. However, the NRC staff is concerned that failure of the support column welds would lead to a loss of core support, which would be a high consequence event. Therefore the NRC has concluded that the MRP-191 damage likelihood rating understates the consequence of failure in the core support column welds.

(Item 12e) MRP noted (MRP-232 Figure 4-24) that the neutron fluence gradients below the core support plate are very sharp and ~~a "medium" likelihood~~even at 60 years of ~~failure. MRP-232 identified~~exposure, only a limited volume of the core support column welds experience neutron fluences sufficient to cause IASCC ~~and~~ or irradiation embrittlement ~~as potential degradation mechanisms for these welds~~. (Item 12f) In addition, susceptibility to SCC and IASCC is limited because the stresses in these support columns are primarily compressive. Therefore the MRP concluded that the core support column welds are not the leading location for any degradation mechanism and placed them in the "Expansion" category. However, the staff is concerned that the synergistic effects of SCC, fatigue, and thermal embrittlement (casting only) could potentially cause greater degradation in these welds than just the consideration of IASCC and irradiation embrittlement alone. Degradation in these welds could then be equivalent to or greater than other components susceptible only to IASCC and irradiation embrittlement due to the synergistic effects. Therefore, the staff has concluded that CE lower support structure core support column (casting or wrought) welds (Item 12g) joining the core support columns to the core support plate shall be inspected as a "Primary" inspection category component.

The examination methods for the ~~aforementioned components~~(Item 12h) core support column welds shall be consistent with the MRP's recommendations addressed in MRP-227, Revision 0 for these components. The examination coverage for these components shall (Item 12i) include the welds in a minimum sample of 25% of the core support column population. The surface of these welds is observable from the top side of the core support plate. The examination frequency shall be on a 10-year interval, with an additional 25% sampled at each interval. ~~he examination frequency shall be on a 10-year interval consistent with other "Primary" inspection category components.~~ **This is addressed as Topical Report Condition 3 in Section 4.1.3 in this SE.**
(Item 14)

~~3.2.5 Inspection of Alloy A-286 CE Control Element Assembly (CEA) Shroud Bolts~~

~~The staff noted that CE CEA shroud bolts manufactured from Alloy A-286 stainless steel material are not currently identified in either Tables 4-2 or 4-5 of the MRP-227, Revision 0 for CE "Primary" or "Expansion" inspection category components. These bolts exceeded the screening criteria for wear, fatigue, and irradiation-induced stress relaxation and the failure consequences are medium. However, the typeType 316 stainless steel core shroud bolts are currently in the "Primary" inspection category. These bolts are identified as being susceptible to IASCC, wear, fatigue, irradiation embrittlement, void swelling, and irradiation-induced stress relaxation. No basis is provided why the Alloy A-286 stainless steel bolts are not also susceptible to IASCC, irradiation embrittlement, and void swelling, especially since they exceeded the screening criteria for irradiation-induced stress relaxation. The fluence levels and radiation sensitivity of the Alloy A-286 stainless steel bolts are expected to be similar to the typeType 316 stainless steel bolts. Therefore, the staff concludes that the Alloy A-286 stainless steel bolts shall also be included in the "Primary" inspection category. The examination method, coverage, and frequency prescribed for Alloy A-286 bolts shall be consistent with the requirements for typeType 316 stainless steel core shroud bolts. **This is addressed as Topical Report Condition 4 in Section 4.1.4 in this SE.**~~

3.2.6 Plant-Specific Confirmation of the Applicability and Completeness of MRP-227, Revision 0

3.2.6.1 Applicability of FMECA and Functionality Analysis Assumptions

In Section 2.2 of this SE, the staff noted some of the assumptions made in the industry's FMECAs and functionality analyses. The staff questioned how it would be determined whether the operating history of a particular plant (including, for example, the effects of any plant power uprate) was adequately represented by the assumptions made in support of the industry's FMECAs and functionality analyses. In its October 29, 2010, response to RAI 4-6 from the NRC staff's fourth set of RAIs, the MRP indicated that each applicant/licensee was responsible for performing an evaluation of its plant's operating history and demonstrating the applicability of MRP-227, Revision 0 to the facility. Each applicant/licensee shall describe the process used for determining plant-specific differences in the design of their reactor internals components or plant operating conditions, which result in different component inspection categories. **This issue is Applicant/Licensee Action Item 1, and it is addressed in Section 4.2.1 of this SE.**

However, the staff is also concerned that the MRP does not provide adequate guidance to allow an applicant/licensee to assess the applicability of the MRP-227, Revision 0 to its plant. The MRP should consider developing guidance that will allow an applicant/licensee to determine if the plant-specific differences in the design of their reactor internals components or plant operating conditions result in different component inspection categories. This guidance could be issued in a separate MRP report or included in a future revision of MRP-227.

3.2.6.2 PWR Vessel Internal Components (Item 2) ~~Within~~within the Scope of License Renewal

The list of reactor internals components for which the effects of aging will be managed by application of the AMP defined by MRP-227, Revision 0 is defined by (Item 15a) Table~~s~~ 4-1 ~~and 4-2~~ in MRP-189, Revision 1, "Materials Reliability Program: Screening, Categorization, and

Ranking of B&W-Designed PWR Internals," (Item 15b) ~~Table 4-4 in MRP-190,~~ and (Item 15c) Tables 4-4 and 4-5 in MRP-191.

Consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which reactor internals components are within the scope of LR for its facility. Applicants/licensees shall review the information in (Item 15a) Table~~s~~ 4-1 ~~and 4-2~~ in MRP-189, Revision 1 (Item 15b) ~~, Table 4-4 in MRP-190~~ and (Item 15c) Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the reactor internals components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the reactor internals components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP such that the effects of aging on the missing component(s) will be managed for the period of extended operation. **This issue is Applicant/Licensee Action Item 2, and it is addressed in Section 4.2.2 of this SE.**

3.2.6.3 Evaluation of the Adequacy of Plant-Specific Existing Programs

The MRP (Item 2) ~~identified~~placed certain CE and Westinghouse (Item 16a) ~~RVI components which were categorized into~~reactor internals, as listed in MRP-227 Tables 4-8 and 4-9, in the "Existing (Programs)" inspection category ~~as requiring further plant-specific evaluation to verify the acceptability of the existing programs, or to identify changes to the existing programs which should be implemented to manage the aging of these components for the period of extended operation. If the existing programs are not acceptable, it is necessary to identify and implement changes to the programs to manage aging of applicable components over the period of extended operation.~~. Generically, these were components for which existing ~~plant-specific programs other than a plant's~~ ASME Code, Section XI program were being credited for managing aging~~.~~ (Item 16b) related degradation. In addition to these ASME required inspections, NUREG 1801 Rev. 2XI.M37 provides an acceptable program for managing degradation in Westinghouse flux thimbles.

(Item 16c) Beyond the components listed in the "Existing Programs" tables, Sections 4.3.2 and 4.3.3 of MRP-227 credit plant-specific programs for aging management. These components were left for plant-specific evaluation because, although the MRP was able to identify that plant-specific programs already exist for the management of these components, the MRP was unable to evaluate in detail the content of each facility's plant-specific program. The CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227, Revision 0), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227, Revision 0). Considerations that should be included in this evaluation follow for these specific Westinghouse and CE components.

(Item 16d) Thermal shields were removed from many CE plants early in reactor life. Remaining CE plants with thermal shields require plant-specific programs to monitor the performance of the thermal shield positioning pins. Anisotropic growth of zirconium alloys in CE in-core instrumentation thimble tubes can result in axial growth. This issue has been recognized by plants with zirconium alloys in-core instrumentation thimbles and most plants have instituted appropriate monitoring procedures.

Westinghouse guide tube support pins are made from either (Item 2) Type 316 stainless steel or Alloy X750. There have been issues with cracking (Item 17a) ~~in the~~of the original X750 pins and (Item 17b) ~~most~~ many licensees have replaced them with (Item 2) ~~type~~Type 316 stainless steel materials. The applicants/licensees shall evaluate the adequacy of their plant-specific existing program and ensure that the aging degradation is adequately managed during the extended period of operation for both X750 and (Item 2) ~~type~~Type 316 stainless steel guide tube support pins (split pins). Therefore, it is recommended that the evaluation consider the need to replace the X750 support pins (split pins), if applicable, or inspect the replacement (Item 2) ~~type~~Type 316 stainless steel support pins (split pins) to ensure that cracking has been mitigated and that aging degradation is adequately monitored during the extended period of operation.

(Item 16e) ~~CE fuel alignment pins are susceptible to IASCC, wear, fatigue, irradiation embrittlement, and irradiation-enhanced stress relaxation. The applicants/licensees shall evaluate the adequacy of their plant-specific existing program with respect to CE fuel alignment pins and ensure that the synergistic effects of aforementioned degradation mechanisms are adequately monitored during the extended period of operation.~~

(Item 16f) Therefore, the staff determined that these plant-specific programs require an evaluation to verify the acceptability of the existing programs, or to identify changes to the existing programs which should be implemented to manage the aging of these components for the period of extended operation. If the existing programs are not acceptable, it is necessary to identify and implement changes to the programs to manage aging of applicable components over the period of extended operation. ~~Therefore, the staff determined that CE thermal shield positioning pins and in-core instrumentation thimble tubes, and Westinghouse guide tube support pins (split pins) require plant-specific evaluation to verify the acceptability of the existing programs, or to identify changes to the program that should be implemented to manage the aging of these components, for the period of extended operation.~~ **This issue is Applicant/Licensee Action Item 3, and it is addressed in Section 4.2.3 of this SE.**

(Item 18) ~~3.2.6.4 B&W Core Support Structure Upper Flange Stress Relief~~

~~In its October 29, 2010, response to RAI 4-4, the MRP stated that the core support structure upper flange weld was below the screening criteria for all aging degradation mechanisms including SCC because the applied stress on this component is low and weld residual stresses have been alleviated by a stress relief heat treatment during the original fabrication. The staff accepts this technical basis, but has concluded that each applicant/licensee shall confirm the accuracy of this assumption for its facility. Therefore, B&W applicants/licensees shall confirm that the core support structure upper flange at their facilities were stress relieved during original fabrication/construction. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" inspection category component consistent with the upper core support barrel weld in Westinghouse and CE units. These Westinghouse and CE components have a similar function, but have not been stress relieved.~~

~~If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227 for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of this SE.~~ **~~This issue is Applicant/Licensee Action Item 4, and it is addressed in Section 4.2.4 of this SE.~~**

3.3 Evaluation of MRP-227, Revision 0 - Sections 4 and 5

The staff's review of Sections 4 and 5 of the MRP-227, Revision 0 resulted in the staff, in principle, accepting MRP's development of I&E guidelines for the subject reactor internals components. The MRP considered susceptibility of reactor internals components to one or more degradation mechanisms and the safety consequences as a result of the failure of the reactor internals components in developing the I&E guidelines. However, the staff identified concerns with the MRP's proposed I&E guidelines for some components subject to MRP-227, Revision 0. In the following sections, the staff's evaluation of the proposed I&E guidelines for components subject to MRP-227, Revision 0 is provided, focusing on the staff's concerns which led to the imposition of conditions and limitations on the use of MRP-227, Revision 0 and plant-specific action items associated with the use of MRP-227, Revision 0 (as summarized in Section 4 of this SE).

3.3.1 General Evaluation of the MRP-227, Revision 0 I&E Guidelines

The staff's review of Sections 4 and 5 of the MRP-227, Revision 0 indicated that the MRP generally provided an adequate justification regarding the examination criteria imposed for the "Primary" and "Expansion" inspection category components. "Primary" inspection category components were considered the lead components in which a degradation mechanism was expected to occur prior to the expansion components. Therefore, "Primary" inspection category components are inspected periodically. Further, the analyses indicated that "Expansion" inspection category components have a higher degree of tolerance to the aging effects to which they may be subject than their associated "Primary" inspection category components. Therefore, the initiation of inspections of "Expansion" inspection category components begins only when a particular degradation mechanism is identified in the associated "Primary" inspection category components. The staff noted that for "Primary" and "Expansion" inspection category components, the MRP generally provided examination guidelines including examination methods to be used, sampling and coverage of the examinations, expansion scope based on the extent of degradation, and evaluation of examination results for the reactor internals components. The staff reviewed the frequency of examinations of the reactor internals components addressed in tables in Section 4 of MRP-227, Revision 0 and concluded that, typically, the "Primary" inspection category components are to be examined during every 10-year interval.

Therefore, the staff, in principle, agrees with the I&E guidelines developed for components subject to MRP-227, Revision 0. However in its review of the I&E guidelines, the staff identified several concerns with the MRP's proposal. Some of the staff's concerns were resolved via MRP responses to staff RAIs, and those that were not adequately resolved are reflected in plant-specific action items and/or conditions and limitations on the use of MRP-227, Revision 0. An example of a significant staff concern that was resolved is given in the following paragraphs, while those that were not adequately resolved are addressed in **Sections 3.3.2, 3.3.3, 3.3.4, 3.3.5, 3.3.6, and 3.3.7 of this SE.**

One of the staff's concerns was that, for components in the "Primary" and "Expansion" inspection categories, MRP-227, Revision 0 did not provide a minimum examination coverage criterion related to the total surface area/volume of the component in order to define a successful examination. The staff's concern was that, although MRP-227, Revision 0 states

that all accessible surfaces/volumes of a component subject to inspection are to be examined, this may result in a very limited examination if plant-specific conditions limit the accessible surface area/volume.

In its October 29, 2010, response to NRC staff RAI 4-8, the MRP indicated that they will update MRP-227, Revision 0 to require, in addition to the requirement to examine one hundred percent of the accessible inspection area/volume for "Primary" and "Expansion" inspection category components, a minimum of 75% coverage of the entire examination volume (i.e., including both accessible and inaccessible regions) for all "Primary" inspection category components in order to define an inspection meeting the intent of MRP-227, Revision 0. For certain like-components (e.g., CE core shroud bolts) in the "Primary" inspection category, the examination "coverage" requirements are specified in terms of a minimum percentage of like components that must be inspected. In these cases, the MRP stated that the minimum sample size for inspection is 75% of the total population of like components. When considering the inspection of a set of like components, it is understood that essentially one hundred percent of the area/volume of each accessible like component will be examined.

The staff has concluded that, if there are no defects discovered during the inspection, the 75 percent sample size based on inspection area/volume or total population of like components is acceptable. The staff believes that the minimum inspection area/volume or sample size is acceptable because the examined area/volume/population will provide reasonable assurance regarding the presence or absence of an active degradation mechanism in the subject component. Further, the minimum inspection area/volume is acceptable because it is assumed that the component locations that are 1) most susceptible to the degradation mechanism that is the subject of the examination and 2) most critical to component integrity will be adequately covered by the examinations as a result of the large design margins typically associated with these components. Applicants/licensees may be able to use available information to identify those specific component areas/volumes, or the subset of a group of like components, that are most likely to exhibit degradation and most important to component integrity. Using this information to prioritize the examinations will help to ensure their effectiveness.

If defects are discovered in the 75 percent sample size, the licensee shall enter that information into the plant's corrective action program and to evaluate whether the results of the examination ensure that the component (or set of like components) will continue to meet its intended function under all licensing basis conditions of operation until the next scheduled examination. Hence, the staff finds that the MRP has adequately addressed the staff's concern regarding a minimum examination coverage requirement for the "Primary" inspection category components.

3.3.2 Imposition of Minimum Examination Coverage Criteria for "Expansion" Inspection Category Components

In MRP-227, Revision 0, a requirement to examine one hundred percent of the accessible area/volume, or one hundred percent of accessible components when a population of like components (e.g., bolting) is examined, is proposed for "Expansion" inspection category components. The staff's concern is that this criterion may result in a limited examination if only a small part of a given component, or a limited number of a population of like components, is accessible for examination. (Item 19a) Coverage requirements for all expansion items should be determined and reviewed as part of the extent of condition evaluation for the "Primary" component.

To ensure that the effects of aging are adequately monitored in the "Expansion" inspection category components, when the examination of these components is required, the staff has concluded that the minimum examination coverage requirement proposed by the MRP for "Primary" inspection category components (discussed in Section 3.3.1 above) (Item 19b) ~~shall also be applied to~~ should be used as a guideline for the inspection of components in the "Expansion" inspection category. (Item 19c) ~~That is~~This would imply that, in the absence of a valid technical justification, a minimum of 75 percent coverage of the entire examination (Item 19d) area or volume (i.e., including both accessible and inaccessible regions) for all "Expansion" inspection category components or a minimum sample size for inspection is 75 percent of the total population of like components (Item 19e) ~~will define~~would be used as the default condition for an inspection meeting the intent of MRP-227, as approved by the NRC. For the inspection of a set of like components, it is understood that (Item 19f) ~~essentially 100%~~relevant of the area/volume of each accessible like component will be examined. ~~Application of this~~

(Item 19g) The intention of the minimum examination coverage requirement (Item 19g) ~~will~~ is to ensure that the inspections of "Expansion" ~~inspection category~~ components will be effective at identifying degradation, if present. (Item 2) ~~However, applicants~~Applicants/licensees may also be able to use available information to identify those specific component areas/volumes, or the subset of a group of like components, that are most likely to exhibit degradation and most important to component integrity. Using this information to prioritize the examinations will help to ensure their effectiveness. (Item 19h) The extent of condition report for any Primary inspection finding should address the relevance of the observed degradation to the conditions in the expansion component. Any technical justification for a minimum examination coverage requirement below the 75% guideline for the Expansion components must provide reasonable assurance that degradation will be detected in a timely manner.

If defects are discovered in the (Item 19i) ~~75 percent sample size~~Expansion Inspection, the licensee shall enter that information into the plant's corrective action program and evaluate whether the results of the examination ensure that the component (or set of like components) will continue to meet its intended function under all licensing basis conditions of operation until the next scheduled examination. **This is addressed as Topical Report Condition 5 in Section 4.1.5 of this SE.**

3.3.3 Examination Frequencies for Baffle-Former Bolts and Core Shroud Bolts

For some components, the staff was concerned over their assigned inspection frequency. For baffle-former bolts in B&W and Westinghouse-designed reactors and core shroud bolts in CE-designed reactors, the examination frequency can vary from 10 to 15 years (Item 20a) (or EFPY). In Appendix B to its October 29, 2010, RAI response, the MRP indicated that the rate of radiation-induced degradation of these components may decrease in the later stage of a plant's life. The analysis that describes the reduction in the rate of degradation is described in MRP-230, "Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals." Since the rate of radiation-induced degradation may decrease in the later stage of a plant's life, the inspection interval may be able to be increased. Hence, MRP-227, Revision 0 provided a proposed examination frequency range of every 10 to 15 years (Item 20a) (or EFPY).

Although the staff understands the general argument made in MRP-227, Revision 0, it has concluded that the information for the aforementioned components under the column "Examination Method/Frequency" in Tables 4-1, 4-2, and 4-3 of MRP-227, Revision 0 is not sufficiently prescriptive to address this issue. The entry for these components provides too much latitude with insufficient oversight of an applicant's/licensee's determination of its examination frequency. Hence, the staff determined that the NRC-approved version of MRP-227 shall specify a 10-year inspection frequency for these components (Item 20b) following the initial or baseline examination unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections. **This is addressed as Topical Report Condition 6 in Section 4.1.6 of this SE.**

3.3.4 Periodicity of the Re-Examination of "Expansion" Inspection Category Components

The I&E guidelines for "Expansion" inspection category components are addressed in Tables 4-4, 4-5 and 4-6 in MRP-227, Revision 0. However, Tables 4-4, 4-5, and 4-6 in MRP-227, Revision 0 do not address the periodicity of subsequent re-examination for all of the "Expansion" inspection category components. For those "Expansion" inspection category components for which Tables 4-4, 4-5, and 4-6 do not specify a periodicity of subsequent re-examination, the MRP stated that the periodicity of the subsequent re-examinations depends on the results of the initial examination.
(Item 2)
The staff has concluded that the NRC-approved version of MRP-227 shall specify a baseline periodicity of subsequent re-examination for all "Expansion" inspection category components and that a baseline 10-year interval between examinations of "Expansion" inspection category components once degradation is identified in the associated "Primary" inspection category component and examination of the "Expansion" inspection category component commences unless an applicant/licensee provides an evaluation for NRC staff approval which justifies a longer interval between inspections. This periodicity is consistent with ASME Code, Section XI requirements. Hence, the staff has concluded that MRP-227, Revision 0, Tables 4-4, 4-5, and 4-6 should be modified to apply a baseline 10-year re-examination interval to all "Expansion" inspection category components. **This is Topical Report Condition 7, and it is addressed in Section 4.1.7 of this SE.**

3.3.5 Application of Physical Measurements as Part of the I&E Guidelines for B&W, CE, and Westinghouse Reactor Internals Components

Physical measurements were proposed as part of the I&E guidelines for some reactor internals components. By letter dated April 20, 2010, the MRP responded to NRC RAIs 3-11 and 3-12 and indicated that physical measurements must be utilized to monitor for loss of core clamping pre-load for B&W core clamping items, for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. In its response to the aforementioned RAIs, the MRP further stated that the (Item 2) specific physical measurement techniques are not within the scope of MRP-227, Revision 0, and, therefore,(item 2) ~~it~~ did not provide specific (Item 21a) techniques for completing the required measurements. The core clamping acceptance criteria for the B&W plants was established numerically as a minimum average height differential in Table 5-1 of MRP-227 Rev/ 0. While the initial acceptance for determining the lack of distortion in the CE core shroud segments is a lack of discernable gap, no numerical maximum acceptable was established if a gap were detected. Also, for the

Westinghouse hold down springs no specific numerical acceptance criteria for these examinations (Item 21b) was provided due to the plant specific "as-built" nature of the interfacing components' tolerances.
(Item 22)
~~MRP also identified that B&W baffle-to-baffle bolts and core barrel-to-former bolts are susceptible to irradiation-enhanced stress relaxation, irradiation creep, IASCC, irradiation embrittlement, and overload. Loss of preload can occur due to irradiation-enhanced stress relaxation and irradiation creep. These components are currently in the "Expansion" inspection category; however, there are no examination requirements and the integrity of these components needs to be justified by evaluation or replacement if examination is triggered by degradation in the baffle-to-former bolts (i.e., their associated "Primary" inspection category component). In its response to RAI 4-17, dated October 29, 2010, the MRP indicated that a plant-specific analysis is required for evaluating the effect of loss of preload in these bolts on the closure integrity of the core barrel assembly to demonstrate that functionality is maintained. Therefore, B&W applicants/licensees shall perform a plant-specific analysis on the effect of loss of closure integrity on the functionality of the core barrel assembly and propose physical measurements or examinations, if necessary, to confirm that adequate closure integrity will be maintained over the period of extended operation.~~

Applicants/licensees shall identify the plant-specific acceptance criteria (Item 21c) methodology to be applied for their facilities when these physical examinations are made, and these acceptance criteria (Item 21c) methodology will be consistent with the plant's licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation. **This is Applicant/Licensee Action Item 5, and it is addressed in Section 4.2.5 of this SE.**

3.3.6 Evaluation of Inaccessible B&W Components

MRP-227, Revision 0 indicates that certain B&W core barrel assembly components are known to be inaccessible for inspection. They are the core barrel cylinder (including vertical and circumferential seam welds), the former plates, the (Item 23a) ~~internal and~~ external baffle-to-baffle bolts~~, core barrel-to-former bolts,~~ (Item 23b) and their locking devices, and (Item 23c) ~~the baffle~~core barrel-to-former bolts and their locking devices. Each of these is an "Expansion" inspection category component. In addition, in its October 29, 2010 response to NRC staff RAI 4-8, the MRP indicated that the B&W core support shield vent valve disc shafts or hinge pins are also inaccessible. This component is a "Primary" inspection category component and it does not have an associated "Expansion" inspection category component.

MRP-227, Revision 0 does not propose that applicants/licensees examine these inaccessible components. Applicants/licensees will justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and/or provide their plan for the replacement of the components. **This is Applicant/Licensee Action Item 6, and it is addressed in Section 4.2.6 of this SE.**

3.3.7 Plant-Specific Evaluation of CASS Components

In its response dated October 29, 2010, to the fourth set of RAIs, MRP identified that (Item 24a) acceptance of degradation effects in some cast austenitic stainless steel (CASS) reactor internals components require a plant-specific analysis to demonstrate (Item 24b) the acceptable minimum number of functional subcomponent members required within a component and/or the minimum number of functional components within an assembly. The analysis would demonstrate that ~~their~~ structural integrity and functionality (Item 24c) of the component or its assembly are maintained during the extended period of operation (Item 24d) given a detected level of degradation or number of failures. This analysis would take into account aging effects that degrade fracture toughness to establish a level of redundancy within the component and/or within the component assembly's population such that acceptance of examination results can be established.

In its response to RAI 4-15, dated October 29, 2010, the MRP identified B&W in-core monitoring instrumentation (IMI) guide tube (Item 25a) ~~assemblies ("Expansion~~assembly spiders ("Primary" inspection category) and CRGT assembly spacer castings ("Expansion" inspection category), CE lower support columns ("Primary" inspection category), and Westinghouse lower support column bodies ("Expansion" inspection category) as (Item 2) components and/or component assemblies requiring such (Item 24e) an approach for acceptance and that it would follow methodologies provided in WCAP-17096 currently under review by the staff.

(Item 24f) Because of the staff's concern with the potential challenge to structural integrity due to a ~~plant-specific~~combination of the reduced fracture toughness and difficult to detect service or fabrication related flaws an analysis (Item 24g) using these approaches for acceptance shall be completed to demonstrate acceptable redundancy. An analysis for the B&W IMI guide tube assembly (Item 25b) spiders is necessary to determine the minimum number of spider arms (Item 24h) necessary for individual spider functionality, and the minimum number of functional spiders (out of 52) that are needed for continued (Item 26) ~~operation. For B&W CRGT assembly spacer castings, a plant-specific reactivity analysis is necessary to determine the number of control rod drive mechanisms (CRDMs) that are required for shut down of the reactor and assess how many CRGT spacer castings could fail and still retain sufficient operability of the remaining CRDMs to shut down the reactor.~~ functionality of the IMI guide tubes.

An analysis for the CE lower support columns and Westinghouse lower support castings is necessary to demonstrate that these components maintain functionality during the extended period of operation.

Therefore, applicants/licensees shall develop a plant-specific analysis for the B&W IMI guide tube assembly (Item 25b) spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies to demonstrate that these components will maintain their functions during the period of extended operation. These analyses should consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227. **This is Applicant/Licensee Action Item 7, and it is addressed in Section 4.2.7 of this SE.**

3.4 Evaluation of MRP-227, Revision 0 - Section 6

Section 6 of the MRP-227, Revision 0 includes a description of the flaw evaluation methodology that is to be implemented when an examination reveals indications that do not meet acceptance criteria. Based on its review of this section, the staff concludes that this section adequately addresses, at a high level, the evaluation methodologies that could be used by the licensee or applicant for evaluating flaws detected during the examination of the reactor internals components. However, industry indicated in its response to RAI 4-14 that Section 6 of MRP-227 will not be used by licensees for evaluating examination results that do not meet the acceptance criteria identified in Section 5 of MRP-227. Rather, WCAP-17096-NP, Revision 2 is the document that will be used as the framework to develop those generic and plant-specific evaluations triggered by findings in the reactor internals examinations. The NRC staff is currently reviewing WCAP-17096-NP, Revision 2.

3.5 Evaluation of MRP-227, Revision 0 - Section 7

The staff reviewed Section 7 of MRP-227, Revision 0 and concludes that the implementation of MRP-227, Revision 0 shall comply with the implementation protocol specified in the NEI 03-08. NEI 03-08 requires that when a licensee does not implement a "Mandatory" or "Needed" element (defined in Section 2.5 of this SE) at its facility, it shall notify the NRC staff of the deviation and justification for the deviation no later than 45 days after approval by a licensee executive. Consistent with requirements addressed in Section 7.3 of MRP-227, Revision 0, all PWR licensees shall implement a program that is consistent with the implementation requirements addressed under the "Needed" category in NEI 03-08. Reporting of the inspection results is very essential to document the operating experience of the fleet. However, the reporting of inspection results to the industry is only addressed as a "Good Practice" element in MRP-227, Revision 0. Since this information will be used to update the I&E guidelines and to inform subsequent examinations at nuclear power plants, the staff recommends that reporting of inspection results both be classified under the "Needed" category.

3.5.1 Submittal of Information for Staff Review and Approval

In addition to the implementation of MRP-227, Revision 0 in accordance with NEI 03-08, applicants/licensees (Item 27a) whose licensing basis contains a commitment to submit a PWR RVI AMP shall also make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility (Item 27b) consistent with the requirements of their commitments. An applicant's/licensee's (Item 27c) ~~application to implement~~submittal that implements MRP-227, as amended by this SE, as an AMP for its facility shall include the following (Item 27d) items (1) and (2). Applicants who submit applications for LR after the issuance of this SE shall, in accordance with the NUREG-1801, Revision 2, submit the information identified in items (1) through (5) for staff review and approval with the application:

1. An AMP for the facility that addresses the 10 program elements as defined in NUREG-1801, Revision 2, AMP XI.M16A.

2. To ensure the MRP-227, Revision 0 program and the plant-specific action items will be carried out by applicants/licensees, applicants/licensees are to submit an inspection plan

which addresses the identified plant-specific action items for staff review and approval consistent with the licensing basis for the plant. If an applicant/licensee plans to implement an AMP which (Item 28a) ~~does not conform to~~deviates from the guidance provided in MRP-227, as approved by the NRC, the applicant/licensee shall identify where their program (Item 28b) ~~does not conform to~~deviates from the ~~recommendations~~ requirements of MRP-227, as approved by the NRC, and shall provide a justification for (Item 28c) ~~the nonconformance which includes a~~any deviations. The technical justification shall include consideration of how the (Item 28d) ~~nonconformance~~deviation affects both "Primary" and "Expansion" inspection category components.

3. The regulation at 10 CFR 54.21(d) requires that an FSAR supplement for the facility contain a summary description of the programs and activities for managing the effects of aging and the evaluation of TLAAs for the period of extended operation. Those applicants for LR referencing MRP-227, as approved by the staff, for their RVI component AMP shall ensure that the programs and activities specified as necessary in MRP-227, as approved by the NRC, are summarily described in the FSAR supplement.

4. The regulation at 10 CFR 54.22 requires each applicant for LR to submit any TS changes (and the justification for the changes) that are necessary to manage the effects of aging during the period of extended operation as part of its LR application (LRA). For the plant CLBs that include mandated inspection or analysis requirements for RV internals either in the operating license for the facility or in the facility TS, the applicant/licensee shall compare the mandated requirements with the recommendations in the NRC-approved version of MRP-227. If the mandated requirements differ from the recommended criteria in MRP-227, as approved by the NRC, the conditions in the applicable license conditions or TS requirements take precedence over the MRP recommendations and shall be complied with (Item 29) unless a TS change is submitted.

5. Pursuant to 10 CFR 54.21(c)(1), (Item 2) ~~the~~an applicant is required to identify all analyses in the CLB for their RVI components that conform to the definition of a TLAA in 10 CFR 54.3 and shall identify these analyses as TLAAs for the application in accordance with the TLAA identification requirement in 10 CFR 54.21(c)(1). MRP-227, as approved by the NRC, does not specifically address the resolution of TLAAs that may apply to applicant/licensee RVI components. Hence, applicants/licensees who implement MRP-227, as approved by the NRC, shall still evaluate the CLB for their facilities to determine if they have plant-specific TLAAs that shall be addressed. If so, the applicant's/licensee's TLAA shall be submitted for NRC review along with the applicant's/licensee's application (Item 27e) for license renewal to implement the NRC-approved version of MRP-227.

   For those cumulative usage factor (CUF) analyses that are TLAAs, the applicant may use the PWR Vessel Internals Program as the basis for accepting these CUF analyses in accordance with 10 CFR 54.21(c)(1)(iii) only if the RVI components in the CUF analyses are periodically inspected for fatigue-induced cracking in the components during the period of extended operation. The periodicity of the inspections of these components shall be justified to be adequate to resolve the TLAA. Otherwise, acceptance of these TLAAs shall be done in accordance with either 10 CFR 54.21(c)(1)(i) or (ii), or in accordance with 10 CFR 54.21(c)(1)(iii) using the applicant's program that corresponds to NUREG-1801, Revision 2, AMP X.M1, "Metal

Fatigue of Reactor Coolant Pressure Boundary Program." (Item 27f) (Item 27g)~~To satisfy the evaluation requirements of ASME Code, Section III, Subsections NG-2160 and NG-3121, the fatigue CUF analyses shall include the effects of the reactor coolant system water environment.~~

~~Applicants whose licensing basis contains a commitment to submit a PWR RVI AMP shall make their submittal using MRP-227, as approved by the NRC, consistent with the requirements of their commitments. Applicants who submit applications for LR after the issuance of this SE shall, in accordance with the NUREG-1801, Revision 2, submit the information identified in items (1) through (5) for staff review and approval with the application.~~ **This is Applicant/Licensee Action Item 8, and it is addressed in Section 4.2.8 of this SE.**

(Item 30) ~~3.6 Evaluation of MRP-227, Revision 0 - Appendix A~~

~~The staff reviewed Appendix A of MRP-227, Revision 0 which originally addressed 3 of the 10 program attributes of an AMP. The staff noted that discussion of the three AMP attributes in the MRP-227, Revision 0, Appendix A did not entirely conform to the NRC's recommended program element criteria for AMPs that are given in Section A.1.2.3 of NRC Branch Technical Position RLSB-1.~~

~~It was the staff's intent to use the information provided in MRP-227, Revision 0, Appendix A to develop Revision 2 of NUREG-1801, AMP XI.M16A, "PWR Vessel Internals Program." By letter dated November 12, 2009, the staff requested that the MRP provide additional information in a format that conforms to the recommended program element criteria in Section A.1.2.3 of NRC Branch Technical Position RLSB-1 that could be used to develop NUREG-1801, Revision 2, AMP XI.M16A and that could be adopted for the contents of an applicant's PWR reactor internals AMP. By letter dated December 2, 2009, the MRP provided a revised AMP that the MRP recommended for the development of the NUREG-1801, Revision 2. The staff evaluated and incorporated the MRP's recommended program elements, with some minor adjustments, into NUREG-1801, Revision 2, AMP XI.M16A.~~

~~When the approved version of MRP-227 is published, MRP-227, Appendix A shall be updated to be equivalent with NUREG-1801, Revision 2, AMP XI.M16A.~~ **~~This is addressed as Topical Report Condition 8 in Section 4.1.8 of this SE.~~**

4.0 CONDITIONS AND LIMITATIONS AND APPLICANT/LICENSEE PLANT-SPECIFIC ACTION ITEMS

Based on its review, the staff identified some issues and concerns in Section 3.0 of this SE that were not adequately resolved regarding the implementation of MRP-227. Some of the staff's issues that are not adequately resolved and remaining concerns are related to conditions and limitations on the use of MRP-227. These conditions and limitations address deficiencies in the AMP defined by MRP-227, Revision 0 and are identified in Section 4.1 of this SE. In addition, some of the staff's issues and concerns that were not adequately resolved are related to applicant/licensee action items related to the use of MRP-227. These plant-specific actions items address topics related to the implementation of MRP-227 that could not be effectively addressed on a generic basis in MRP-227, Revision 0 and are identified in Section 4.2 of this SE.

4.1 Limitations and Conditions on the Use of MRP-227, Revision 0

4.1.1 (Item 9a) ~~High Consequence Components~~Core Support Structures in the "No Additional Measures" Inspection Category

As discussed in Section 3.2.2 of this SE, the staff determined that (Item 2) inspections are required for certain (Item 9c) ~~high consequence of failure components~~ core support structures that were binned in the MRP-227, Revision 0 "No Additional Measures" inspection category. ~~The "Primary" inspection category components to which these additional "Expansion" inspection category components shall be linked is shown below~~ (Item 9g) To ensure that the structural integrity and functionality of these reactor internals components are maintained under all licensing basis conditions during the period of extended operation, the staff has determined that each of these components shall be included in the "Existing" inspection category in the NRC-approved version of MRP-227. The examination method to be used for these additional "Existing" inspection category components shall be consistent with the examination method for inspection of removable core support structures in ASME Section XI. The examination coverage and re-examination frequency requirements for these "Existing" inspection category components shall be as described for the ASME Section XI examinations. **This is Topical Report Condition 1.**
(Item 10b)

~~Component~~	~~Link to "Primary" Inspection Category Components~~
~~Upper core plate in Westinghouse-designed reactors~~	~~CRGT lower flange weld~~
~~Lower support forging or casting in Westinghouse-designed reactors~~	~~CRGT lower flange weld~~
~~Lower core support beams in CE-designed reactors~~	~~Upper core support barrel flange weld~~
~~Core support barrel assembly upper core barrel flange in CE-designed reactors~~	~~Upper core support barrel flange weld~~
~~Lower grid-to-core barrel bolts in B&W-designed reactors~~	~~Lower core barrel bolts~~

~~To ensure that the structural integrity and functionality of these RVI components are maintained under all licensing basis conditions during the period of extended operation, the staff has determined that each of these components shall be included in the "Expansion" inspection category in the NRC-approved version of MRP-227. The examination method to be used for these additional "Expansion" inspection category components shall be consistent with the examination method for the "Primary" inspection category component to which they are linked. The examination coverage and re-examination frequency requirements for these "Expansion" inspection category components shall be as addressed in Sections 4.1.5 and 4.1.7 of this SE.~~

~~When publishing the approved version of MRP-227, Revision 0, Tables 4-4, 4-5, and 4-6 in MRP-227, Revision 0 shall be revised accordingly. **This is Topical Report Condition 1.**~~

Component
Upper core plate in Westinghouse-designed reactors
Lower support forging or casting in Westinghouse-designed reactors
Lower core support beams in CE-designed reactors
Core support barrel assembly upper core barrel flange in CE-designed reactors

4.1.2 Inspection of Components Subject to Irradiation-Assisted Stress Corrosion Cracking

As discussed in Section 3.2.3 of this SE, the staff noted that there are inconsistencies between the degradation mechanisms (Item 2) ~~between~~ in some of the "Primary" and associated "Expansion" inspection category components in Westinghouse and CE-designed reactors. The MRP identified IASCC and neutron embrittlement as the degradation mechanisms for ~~the following~~ "Expansion" inspection category components (Item 11a) in the Westinghouse core barrel and CE core support barrel. ~~, whereas~~ SCC was identified as the degradation mechanism for the corresponding "Primary" inspection ~~category components~~ (Item 11b) of the upper flange welds. The beltline circumferential welds in these components may be subject to the irradiation related degradation mechanisms in addition to SCC. ~~The following table identifies the subject "Expansion" inspection category components and their corresponding tables from MRP-227, Revision 0.~~

~~**"Expansion" Inspection Category Components Subject to IASCC**~~	~~**Tables in MRP-227, Revision 0**~~
~~Upper and lower core barrel welds in Westinghouse-designed reactors~~	~~Table 4-6~~
~~Lower core barrel flange weld in Westinghouse-designed reactors~~	~~Table 4-6~~
~~Core support barrel assembly lower cylinder welds in CE-designed reactors~~	~~Table 4-5~~

To ensure that the structural integrity and functionality of (item 11a) ~~these RVI components~~the Westinghouse core barrel and CE core support barrel are maintained under all licensing basis conditions during the period of extended operation, the staff has determined that (Item 11i) the

potential impact of IASCC and irradiation embrittlement in the beltline circumferential welds must be either evaluated or these welds must~~eachsections of these components shall~~ be included in the "Primary" inspection category in the NRC-approved version of MRP-227. The examination methods shall (Item 11i) meet the requirements for visual inspection of core barrel welds or alternatively examined by eddy current or ultrasonic examination.~~ be consistent with the MRP's recommendations for these components, the examination coverage for these components shall conform to the criteria described in Section 3.3.1 of this SE, and the~~ As described in Section 3.3.1 of this SE, the examination coverage shall be determined by plant specific accessibility. The re-examination frequency shall be on a 10-year interval consistent with other "Primary" inspection category components. (Item 11k) The default inspection requirement shall be a 30° arc centered at the peak fluence location in each quadrant. Plant-specific neutron dose calculations may be employed to further refine the required examination zone. Plants must demonstrate coverage of at least 75% of the exterior surface of potentially IASCC susceptible zones of circumferential weld material. As an alternative to inspection, a licensee/applicant may chose to provide an analysis to justify continued operation per the requirements of Applicant/Licensee Action Item 6.

When publishing the approved version of MRP-227, Revision 0 Tables 4-2 and 4-3 shall be revised accordingly. **This is Topical Report Condition 2.**

4.1.3 Inspection of (Item 12a) ~~High Consequence~~Core Support Components Subject to Multiple Degradation Mechanisms

As discussed in Section 3.2.4 of this SE, the staff determined that (Item 12b) ~~two high~~ the consequence of failure (Item 12d) ~~components~~core support column welds in CE designed reactors is high. The staff concern is that these core support column welds are potentially subject to important combinations of multiple degradation mechanisms~~ were binned in the MRP-227, Revision 0 "Expansion" inspection category. The following table includes the identification of these components and their corresponding tables from MRP-190 used in FMECA process~~. (Item 13)

~~Component~~	~~Relevant Table~~
~~Flow distributor-to-shell forging bolts in B&W-designed reactors~~	~~Table 4-4~~
~~Core support column (casting or wrought) welds in lower support structure in CE-designed reactors~~	~~Table 4-5~~

To ensure that the structural integrity and functionality of (Item 12b) ~~these RVI components~~the CE core support columns are maintained under transient loading conditions during the period of extended operation, the staff has determined that the ~~subject components~~core support column welds joining the core support columns to the core support plate shall be included in the "Primary" inspection category in the NRC-approved version of MRP-227. (Item 12i) The inspection will require an EVT-1 quality visual examination of the subject welds. The visual inspection from the top of the core support plate should assess the condition of welded joints

between the core support columns and the core support plate. The inspection shall be based on a sample of a minimum of 25% of the columns in the inspection population. The re-examination frequency shall be on a 10-year interval similar to other "Primary" inspection category components, with an additional 25% sample inspected at each interval.

When publishing the approved version of MRP-227, Revision 0, (Item 13) Table~~s 4-1 and~~ 4-2 in MRP-227, Revision 0 shall be revised accordingly. **This is Topical Report Condition 3.**

(Item 14)~~ 4.1.4Inspection of Alloy A-286 CE Control Element Assembly (CEA) Shroud Bolts~~

~~As discussed in Section 3.2.5 of this SE, typeType 316 stainless steel CEA shroud bolts are currently in the "Primary" inspection category. However, MRP does not provide I&E guidelines for Alloy A-286 stainless steel CEA shroud bolts. To ensure that the structural integrity and/or functionality of these bolts are maintained under all licensing basis loading conditions during the period of extended operation, the staff has determined that Alloy A-286 stainless steel CEA shroud bolts shall be included in the "Primary" inspection category in the NRC-approved version of MRP-227. The examination method, coverage, and frequency prescribed for Alloy A-286 bolts shall be consistent with typeType 316 stainless steel core shroud bolts.~~

~~When publishing the approved version of MRP-227, Revision 0, Table 4-2 in MRP-227, Revision 0 shall be revised accordingly. **This is Topical Report Condition 4.**~~

4.1.5 Imposition of Minimum Examination Coverage Criteria for "Expansion" Inspection Category Components

As discussed in Section 3.3.1 of this SE, for "Primary" inspection category components, the MRP has proposed to revise MRP-227, Revision 0 to require that (Item 19c) minimum coverage requirements for "Expansion" inspections be established as part of the extent of condition evaluation for the degradation identified in the "Primary" inspection. A technical justification will be required for any minimum coverage requirement below
75 percent of a "Primary" inspection category component's total (accessible + inaccessible) inspection area/volume be examined or, when addressing a set of like components (e.g., bolting), that the inspection examine a minimum sample size of 75 percent of the total population of like components. For the inspection of a set of like components, it is understood that (Item 19f) ~~essentially 100 percent of the~~relevant volume/(Item 19d) ~~area~~areas of each accessible like component will be examined. (Item 19h) ~~This defines the minimum inspection required to meet the intent of MRP-227, Revision 0 provided that no defects are discovered during the inspection.~~ Any technical justification for a minimum examination coverage requirement below the 75% guideline for the "Expansion" inspections must provide reasonable assurance that degradation will be detected in a timely manner and ensure that the component (or set of like components) will continue to meet its intended function under all licensing basis conditions of operation until the next scheduled examination.

If defects are discovered in the 75 percent sample size, the licensee shall enter that information into the plant's corrective action program and evaluate whether the results of the examination ensure that the component (or set of like components) will continue to meet its intended function under all licensing basis conditions of operation until the next scheduled examination.

As discussed in Section 3.3.2 of this SE, an equivalent requirement shall be imposed concerning the inspection of components in the MRP-227, Revision 0 "Expansion" inspection category (Item 19e) based on the extent of condition requirement for the related "Primary" inspections. Similarly to the requirements in Tables 4-1 through 4-3, subsequent examinations shall be carried out in the 10-year ISI interval unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between examinations. When the approved version of MRP-227 is published, Tables 4-4, 4-5, and 4-6 shall be updated to include this requirement. **This is Topical Report Condition 5.**

4.1.6 Examination Frequencies for Baffle-Former Bolts and Core Shroud Bolts

As discussed in Section 3.3.3 of this SE, Tables 4-1, 4-2, and 4-3 of MRP-227, Revision 0 indicate that the frequency of examinations for the baffle-former bolts of B&W and Westinghouse-designed reactors and core shroud bolts in CE-designed reactors can vary from 10 to 15 years (Item 20a) (or EFPY). However, the staff notes that the MRP-227, Revision 0 report provides too much latitude with insufficient oversight of an applicant's/licensee's determination of its examination frequency. Hence, the staff has determined that the NRC-approved version of MRP-227 shall specify a 10-year inspection frequency for these components (item 20c) following the initial or baseline examination unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections. MRP-227, Revision 0 Tables 4-1, 4-2, and 4-3 shall be modified when the approved version of MRP-227 is published to reflect this change. **This is Topical Report Condition 6.**

4.1.7 Periodicity of the Re-examination of "Expansion" Inspection Category Components

As discussed in Section 3.3.4 of this SE, MRP-227, Revision 0, Tables 4-4, 4-5, and 4-6 shall be modified when the approved version of MRP-227 is published to apply a baseline 10-year re-examination interval, (item 31) consistent with ASME Code Section XI requirements, to all "Expansion" inspection category components (once degradation is identified in the associated "Primary" inspection category component and examination of the "Expansion" category component commences) unless an applicant/licensee provides an evaluation for NRC staff approval that justifies a longer interval between inspections. **This is Topical Report Condition 7.**

(Item 30) ~~4.1.8Updating of MRP-227, Revision 0, Appendix A~~

~~As discussed in Section 3.6 of this SE, when the approved version of MRP-227 is published, MRP-227, Appendix A shall be updated to be equivalent with NUREG-1801, Revision 2, AMP XI.M16A.~~ ~~**This is Topical Report Condition 8.**~~

4.2 Plant-Specific Action Items

4.2.1 Applicability of FMECA and Functionality Analysis Assumptions

As addressed in Section 3.2.6.1 of this SE, each applicant/licensee is responsible for (Item 32a) ~~performing an evaluation of its plant's design and operating history~~assuring and demonstrating the applicability of (Item 32b) its plant design and operating history to the approved version of MRP-227 to the facility. Each applicant/licensee should refer, in particular, to the assumptions

regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. **This is Applicant/Licensee Action Item 1.**

4.2.2 PWR Vessel Internal Components (Item 2) ~~Within~~within the Scope of License Renewal

As discussed in Section 3.2.6.2 of this SE, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which reactor internals components are within the scope of LR for its facility. Applicants/licensees shall review the information in (Item 15a) Table~~s~~ 4-1 ~~and 4-2~~ in MRP-189, Revision 1 (Item 15b) ~~, Table 4-4 in MRP-190~~ and (Item 15c) Table~~s~~ ~~4-4 and~~ 4-5 in MRP-191 and identify whether these tables contain all of the reactor internals components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the reactor internals components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation. **This issue is Applicant/Licensee Action Item 2.**

4.2.3 Evaluation of the Adequacy of Plant-Specific Existing Programs

As addressed in Section 3.2.6.3 in this SE, applicants/licensees of CE and Westinghouse are required to perform plant-specific (Item 33) ~~analysis~~assessments either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of (Item 2) ~~this~~ these plant-specific (Item 33) ~~analyses~~assessments and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. CE and Westinghouse components identified for this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227, Revision 0), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227, Revision 0). **This is Applicant/Licensee Action item 3.**

(Item 18) ~~4.2.4B&W Core Support Structure Upper Flange Stress Relief~~

~~As discussed in Section 3.2.6.4 of this SE, the B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the RPV in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" inspection category component. If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of this SE. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval. **This is Applicant/Licensee Action Item 4.**~~

4.2.5 Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse Reactor Internals Components

As addressed in Section 3.3.5 in this SE, applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the NRC-approved version of MRP-227 for loss of (Item 22) ~~core clamping pre-load for B&W core clamping items, loss of closure integrity in B&W's core barrel assembly, for loss of~~ compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections.(Item 22) ~~Based on results of the plant-specific evaluation discussed in Section 3.3.5, B&W baffle-to-baffle bolts and core barrel-to-former bolts may also require physical examination.~~ The applicant/licensee shall include its proposed acceptance criteria (Item 21c) methodology and an explanation of how the proposed acceptance criteria (Item 21c) methodologies are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation as part of their submittal to apply the approved version of MRP-227. **This is Applicant/Licensee Action Item 5.**

4.2.6 Evaluation of Inaccessible B&W Components

As addressed in Section 3.3.6 in this SE, the MRP does not propose to inspect the following inaccessible components: the (Item 2) ~~B&W~~ core barrel cylinder~~s~~ (including vertical and circumferential seam welds), (Item 2) ~~B&W~~the former plates, (Item 23b) ~~B&W internal and~~the external baffle-to-baffle bolts and their locking devices, (Item 23c) ~~B&W baffle~~core barrel-to-former bolts and their locking devices, and (Item 2) ~~B&W~~ core support shield vent valve disc shafts or hinge pins.

(Item 11l) As addressed in Sections 3.2.3 and 4.1.2 in this SE, depending upon the accessibility of IASCC-susceptible component locations, the MRP does not propose to inspect the upper and lower core barrel welds and the lower core barrel flange weld in Westinghouse-designed reactors; and the lower cylinder welds in the core support barrel assembly in CE-designed reactors.

Applicants/licensees will justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components and, if necessary, provide their plan for the replacement of the components for NRC review and approval. **This is Applicant/Licensee Action Item 6.**

4.2.7 Plant-Specific Evaluation of CASS Materials

As discussed in Section 3.3.7 of this SE, the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube (Item 25b) ~~assemblies~~ assembly spiders and CRGT assembly spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation. These analyses should also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement. The plant-specific analysis shall be consistent with the

plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicants/licensees shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227. **This is Applicant/Licensee Action Item 7.**

4.2.8 Submittal of Information for Staff Review and Approval

As addressed in Section 3.5.1 in this SE, applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the reactor internals components at their facility. This submittal shall include the information identified in Section 3.5.1 of this SE. **This is Applicant/Licensee Action Item 8.**

5.0 CONCLUSIONS

The staff has reviewed MRP-227, Revision 0 and concludes that MRP-227, as modified by the conditions and limitations and applicant/licensee action items summarized in Section 4.0 of this SE, provides for the development of an AMP for PWR reactor internals components within the scope of the report which will adequately manage their aging effects such that there is reasonable assurance that they will perform their intended functions in accordance with the CLB during the extended period of operation.

Any applicant may reference this MRP-227, Revision 0, as modified by this SE, in a LRA to satisfy the requirements of 10 CFR 54.21(a)(3) for demonstrating that the effects of aging on the reactor internals components within the scope of this topical report will be adequately managed. The staff also concludes that, upon completion of plant-specific action items set forth in Section 4.0, referencing this topical report in a LRA and summarizing the AMP contained in this topical report in a FSAR supplement will provide the staff with sufficient information to make necessary findings required by Section 54.29(a)(1) for reactor internals components within the scope of MRP-227, as approved by the NRC.

6.0 REFERENCES

The following MRP reports and supporting information was submitted for information only, and these supporting documents were used by the staff as part of its review of the MRP-227, Revision 0 report.

1. NUREG-1801 Revision 2, "Generic Aging Lessons Learned (GALL).

2. MRP-175 Revision 0, "Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values," ADAMS Accession Number ML063470637.

3. MRP-189 Revision 1, "Materials Reliability Program: Screening, Categorization, and Ranking of B& W-Designed PWR Internals," ADAMS Accession Number ML092250189.

4. MRP-190 Revision 0, "Materials Reliability Program: Failure Modes, Effects, and Criticality Analysis of B&W-Designed PWR Internals," ADAMS Accession Number ML091910128.

5. MRP-191 Revision 0, "Materials Reliability Program: Screening, Categorization and Ranking of Reactor Internals of Westinghouse and Combustion Engineering PWR Designs," ADAMS Accession Number ML091910130.

6. MRP-210 Revision 0, "Materials Reliability Program: Fracture Toughness Evaluation of Highly Irradiated PWR Stainless Steel Internal Components," ADAMS Accession Number ML092230736.

7. MRP-211 Revision 0, "Materials Reliability Program: PWR Internals: Age Related Material Properties Degradation Mechanisms, Models and Basis Data," ADAMS Accession Number ML093020614.

8. MRP-228 Revision 0, "Materials Reliability Program: Inspection Standard for PWR Internals," ADAMS Accession Number ML092120574.

9. MRP-229 Revision 3, "Materials Reliability Program: Functionality Analysis for B& W Representative PWR Internals," ADAMS Accession Numbers ML110280110, ML110280111, and ML110280112.

10. MRP-230 Revision 1," Materials Reliability Program: Functionality Analysis for Westinghouse and Combustion Engineering Representative PWR Internals," ADAMS Accession Numbers ML093210269, ML093210270, and ML093210271.

11. MRP-231 Revision 2, "Materials Reliability Program: Aging Management Strategies for B& WPWR Internals," ADAMS Accession Number ML110280113.

12. MRP-232 Revision 0, "Materials Reliability Program: Aging Management Strategies for Westinghouse and Combustion Engineering PWR Internals," ADAMS Accession Numbers ML091671780, ML092250192, and ML092230745.

13. MRP-276 Revision 0, "Materials Reliability Program: Thermal Aging and Neutron Embrittlement Assessment of Cast Austenitic Stainless Steel Welds in PWR Internals" ADAMS Accession Number ML102950165.

14. WCAP-17096-NP, Revision 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," ADAMS Accession Number ML101460157.

15. Response to the staff RAIs dated August 24, 2009, ADAMS Accession Number ML092870179.

16. Response to the staff RAIs dated November 12, 2009 ADAMS Accession Number ML101120660.

17. Response to the staff RAIs dated September 30, 2010, ADAMS Accession Number ML103160381.

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