Attachment 1: Responses to Supplemental Questions in Support of the Nuclear Regulatory Commission Safety Evaluation of MRP-227, Revision 1

Question 1: Response to Request for Additional Information (RAI) 5 in part discussed functionality considerations in support of the reduced sample size for the core barrel welds. Specifically, a core barrel weld could completely fracture allowing the core to drop, and the reactor could still be safely shut down. However, MRP-191 failure modes, effects and criticality analysis (FMECA) categorizes the core barrel welds as high consequence of failure, because failure of these welds could preclude safe shutdown. Please discuss the apparent inconsistency between the RAI 5 response and the MRP-191 FMECA results. [1]

Response:

This apparent inconsistency stems from two aspects of how the MRP-191, Revision 0 [2] and Revision 1 [3] FMECA assessed the core barrel welds:

1. The FMECA combined the economic and safety consequences into one category labeled "Likelihood of Damage". This can be seen in Table 6-3 of either revision of the document.
2. The expert panel at the time assigned a relatively conservative level to the "Likelihood of Failure" category. There were multiple degradation mechanisms and little experience with conducting detailed examinations of the core barrel welds at the time, so it was conservatively assigned a medium likelihood.

The results for the Westinghouse core barrel welds FMECA ranking and categorization from MRP-191, Revision 1 are provided in Table 1. Similar results for the Combustion Engineering (CE) core support barrel welds are provided in Table 2.

The industry is currently in the process of revising MRP-191 for subsequent license renewal (SLR). The same basic expert panel review and FMECA approach as used for earlier revisions of MRP-191 is being used for Revision 2. However, one key change that has been made is to separate the consequences evaluation into economic consequences and safety consequences. Past evaluations documented in the issue management table, MRP-156 [4], defined significant economic impact events for a component as "those for which we do not have a proven fix and would result in significant regulatory and/or public scrutiny, such as first-of-a-kind consideration." The SLR expert panel considered the economic impact of developing solutions and addressing scrutiny, as well, and distilled the economic impact into a ranking based on the expected order of magnitude cost of degradation in a particular component.

Another key difference is the amount of experience accumulated for the core barrel welds since the original publication of MRP-191, Revision 0. This experience ranged from the detailed MRP-227-A [5] enhanced visual examinations already detailed in the response to RAI 5 [6] to the multiple acceptance criteria calculations developed on a plant-specific basis under the methodologies of WCAP-17096-NP-A [7] to the evaluations of the potential for cold work in austenitic stainless steel components in the reactor vessel internals [8].

Publication of MRP-191, Revision 2 is planned for later in 2018, but the expert panel FMECA ranking and categorization review supporting the revised document has already been conducted [9]. Note that for MRP-191, Revision 2 the core barrel welds were separated into girth welds and axial welds because the ranking and categorization is different due to the lower stresses and reduced consequences from postulated degradation (both safety and economic) in the axial welds.

The MRP-191, Revision 2 expert panel conducted extended discussions of the Westinghouse and CE-designed core barrels and the potential for degradation of the welds in the barrels. These discussions included past operating experience and evaluations and consideration of the function of the core barrel. The key points of those discussions are listed here and are separated between lower and upper core barrel and girth and axial welds [9]:

Lower Core Barrel (Westinghouse) or Lower Cylinder (CE) Girth Welds:

- Likelihood of degradation: Low – Welds not expected to fail
    - Thick welds with low active stresses during operation
    - If degradation were to occur it would be expected in the base metal along the weld (heat-affected zone)
    - Full penetration welds – slight concern for fatigue crack growth
    - Small pressure differential across barrel (it is not a pressure vessel)
    - Residual stresses not expected to cause a complete failure
    - Multiple lower core barrel girth welds have been inspected at an EVT-1 level and no relevant indications have been observed to date (see details provided in response to RAI 5 in [6])
    - CE weld operating experience happened early in plant life and was due to a design issue with the attached thermal shield rather than a material aging issue
- Safety consequence of degradation: Medium
    - If core barrel drops, it would be caught by the secondary core support and clevis inserts and radial keys (addressed in more detail in the response to supplemental question 3)
    - Safety consequence assigned to Medium because there is a concern for core damage but the plant could be shut down safely
- Economic consequence of degradation: High
    - Removal of the core barrel after this failure would be a severe challenge
    - Repair may not be possible and replacement of the core barrel would be prohibitively expensive
- Safety category would be A based on the FMECA group, but the panel conservatively elevated it to B
- Economic category is B based on the FMECA group

Lower Core Barrel (Westinghouse) or Lower Cylinder (CE) Axial Welds:

- Likelihood of degradation: Low
    - o Considered lower than the likelihood of degradation in girth welds
- Safety consequence of degradation: Low
    - o Axial welds do not directly support the core like girth welds
    - o Large potential cracks have been justified in acceptance criteria calculations due to low stresses

- Economic consequence of degradation: Medium
    - o Ability to justify a large potential crack reduces the economic consequences
    - o Economic consequence is medium because actions would likely be required but they would not necessarily lead to extreme cost options

Lower Core Barrel (Westinghouse) or Lower Cylinder (CE) Axial Welds (cont.):

- Safety category is A based on the FMECA group
- Economic category is A based on the FMECA group

Upper Core Barrel (Westinghouse) or Upper Cylinder (CE) Girth Welds:

- Likelihood of degradation: Low – Welds not expected to fail
    - Thick welds with low active stresses during operation
    - If degradation were to occur it would be expected in the base metal along the weld (heat-affected zone)
    - Full penetration welds – slight concern for fatigue crack growth
    - Small pressure differential across barrel (it is not a pressure vessel)
    - Residual stresses not expected to cause a complete failure
    - At least 10-15 upper flange welds (combining both WEC and CE inspections) and several upper girth welds have been inspected with no relevant findings.
        - EVT-1 inspection achieved close to 100% coverage for most of these inspections.
        - See details provided in response to RAI 5 in [6]
    - Little stress corrosion cracking (SCC) has been observed in internals components fabricated from austenitic stainless steel and its weld metals, in general
    - CE weld operating experience happened early in life and was due to a design issue rather than a material aging issue
- Safety consequence of degradation: Medium
    - If core barrel drops, it would be caught by the secondary core support and clevis inserts and radial keys (addressed in more detail in the response to supplemental question 3)
    - Safety consequence assigned to Medium because there is a concern for core damage but the plant could be shut down safely
    - Panel believed that these might be the most fail-safe reactor internals component
- Economic consequence of degradation: High
    - Removal of the core barrel after this failure would be a severe challenge
    - Repair may not be possible and replacement of the core barrel would be prohibitively expensive
- Safety category would be A based on the FMECA group, but the panel conservatively elevated it to B
- Economic category is B based on the FMECA group

Upper Core Barrel (Westinghouse) or Upper Cylinder (CE) Axial Welds:

- (Same discussion as Lower core barrel axial welds)
- Likelihood of degradation: Low
    - Considered lower than the likelihood of degradation in girth welds due to lower active stress during normal operation

Upper Core Barrel (Westinghouse) or Upper Cylinder (CE) Axial Welds (cont.):

- Safety consequence of degradation: Low
    - Axial welds do not directly support the core like girth welds
    - Large potential cracks have been justified in acceptance criteria calculations due to low stresses
- Economic consequence of degradation: Medium
    - Ability to justify a large potential crack reduces the economic consequences
    - Economic consequence is medium because actions would likely be required but they would not necessarily lead to extreme cost options
- Safety category is A based on the FMECA group
- Economic category is A based on the FMECA group

These evaluations by the expert panel resulted in the preliminary results for the Westinghouse core barrel welds provided in Table 3 and the preliminary results for the CE core support barrel welds provided in Table 4 [9]. MRP-191, Revision 2 has not yet been published, so the evaluation results presented above must be called "preliminary." It should be noted that this expert panel evaluation was conducted prior to the core barrel operating experience gained during the spring 2018 outage season. This recent operating experience at a Combustion Engineering-designed plant may have an impact on the likelihood of degradation rankings provided here, but the technical discussions supporting the safety consequence rankings should not be affected.

Table 3 and Table 4 show two key points that resolve the inconsistency noted between the RAI 5 response and the MRP-191 FMECA results:

1. When economic consequence and safety consequence are separated, the safety consequence is medium, while the economic consequence is high. The medium safety consequence is due to the same reasoning about core shutdown still being possible, which has been detailed in the response to RAI 5 [6]. The high economic consequence is due to the possibility of core barrel degradation requiring a replacement or costly repairs.
2. The likelihood of degradation was reduced to medium based on the extensive operating experience from MRP-227-A inspections and the significantly improved understanding of stress in the core barrel and its welds.

The FMECA ranking and categorization results from the MRP-191, Revision 2 expert panel are consistent with the technical basis provided in the response to RAI 5 [6] and are due to the increased experience and better technical understanding of the welds and core barrel.

Table 1: MRP-191, Revision 1 FMECA Ranking and Categorization Table for the Westinghouse Core Barrel Welds

Assembly	Subassembly	Component	Material	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group	Category
Lower internals assembly	Core barrel	Lower core barrel (includes LGW, LFW, MAW, and LAW)	304 SS	Weld, IASCC, IE	M	H	3	C
		Upper core barrel (includes UFW, UGW, and UAW)	304 SS	Weld, IE	M	H	3	C

Table 2: MRP-191, Revision 1 FMECA Ranking and Categorization Table for the Combustion Engineering Core Support Barrel Welds

Assembly/ Subassembly	Component	Material	Screened-in Degradation Mechanisms	Likelihood of Failure L, M, H	Likelihood of Damage L, M, H	FMECA Group	Category
Core Support Barrel Assembly	Upper cylinder (includes UFW, UGW, and UAW)	304 SS	Weld	L	H	2	B
	Lower cylinder (includes MGW, LGW/LFW, MAW, LAW, and CSBFW)	304 SS	Weld, IASCC, IE	M	H	3	C

Table 3: MRP-191, Revision 2 (SLR) FMECA Ranking and Categorization Table for the Westinghouse Core Barrel Welds

Assembly	Subassembly	Component	Material	Screened-in Degradation Mechanisms	Likelihood of Failure	Safety Consequence	Economic Consequence	Safety FMECA Group	Economic FMECA Group	Safety Category	Economic Category
Lower Internals Assembly	Core barrel	Lower core barrel axial welds (includes middle axial weld (MAW) and lower axial weld (LAW))	304 SS	Weld, IASCC, Fatigue, IE, VS	L	L	M	1	1	A	A
		Lower core barrel girth welds (includes lower girth weld (LGW) and lower flange weld (LFW))	304 SS	Weld, IASCC, Fatigue, IE, VS	L	M	H	1	2	B	B
		Upper core barrel axial welds (includes upper axial weld (UAW))	304 SS	Weld, Fatigue	L	L	M	1	1	A	A
		Upper core barrel girth welds (includes upper flange weld (UFW) and upper girth weld (UGW))	304 SS	Weld, Fatigue	L	M	H	1	2	B	B

Table 4: MRP-191, Revision 2 (SLR) FMECA Ranking and Categorization Table for the Combustion Engineering Core Support Barrel Welds

Assembly	Component	Material	Screened-in Degradation Mechanisms	Likelihood of Failure	Safety Consequence	Economic Consequence	Safety FMECA Group	Economic FMECA Group	Safety Category	Economic Category
Core Support Barrel assembly	Upper cylinder girth welds (Upper Flange Weld, Upper Girth Weld)	304 SS	Weld, IASCC, Fatigue, IE	L	M	H	1	2	B	B
	Upper cylinder axial welds (Upper Axial Weld)	304 SS	Weld, IASCC, Fatigue, IE	L	L	M	1	1	A	A
	Lower cylinder girth welds (Middle Girth Weld, Lower Girth Weld /Lower Flange Weld)	304 SS	Weld, IASCC, Fatigue, IE	L	M	H	1	2	B	B
	Lower cylinder axial welds (Middle Axial Weld, Lower Axial Weld)	304 SS	Weld, IASCC, Fatigue, IE	L	L	M	1	1	A	A

Question 2: RAI 5, Tables 1-3, what is the probability of detection for each crack size? How do you get >25% probability when inspecting a 25% sample with only one crack present? (Maybe provide an example of one of these calculations). [1]

Response:
"Probability of Detection" refers to two aspects of this particular question and response. The first aspect is the direct subject of RAI 5 [6], which provides information on the probability of detecting one or more cracks of a specific size in a typical core barrel weld length during an EVT-1 inspection with 25% coverage. The tables apply to either a single inspection with one or more cracks present of that particular size or to multiple welds with one or more cracks of that size.

The second aspect of probability of detection is the probability that a particular EVT-1 inspection will detect a crack of a certain size. This is implicitly included in the response to RAI 5 [6] in the assumptions about minimum detectable crack size (0.25 inch in the response) and how it is treated in the calculations. Cracks below 0.25 inch are assumed to have zero probability of detection, which is likely conservative. Cracks 0.25 inch and longer are assumed to have 100% probability of detection. For longer cracks, this assumption is expected to be close to the real probability of detection. Smaller cracks may be more difficult to detect, but the EVT-1 requirements of the inspection standard, MRP-228 [10], are intended to maximize the probability of detection of the inspection technique. These include but are not limited to the demonstration, cleanliness, travel speed, angle, distance, and lighting requirements. A round robin including several inspection vendors was conducted to evaluate the effectiveness of the required EVT-1 inspections in detecting cracks of various sizes and locations [11]. The round robin results showed that with a crack length of 6-10 mm (0.24-0.39 inches) the average detection rate was 62%. This rose to 88% for cracks in the 16-20 mm (0.63 – 0.79 inches) range. The round robin demonstrated that probability of detection is vendor and inspection system specific, so these detection rates were not included in the calculations supporting response to RAI 5 beyond the assumption of a minimum detectable crack size.

The probability of detecting a crack within a length of weld in these calculations is dependent on the size of the crack. As noted in the response to RAI 5:

> *Assumed crack size has a slight effect on the probability of detection, with larger cracks having a higher probability than smaller cracks. In this calculation, the increase in probability comes from the potential to detect the end of a crack that extends into the uninspected portion of the weld.*

This is due to the assumption that 0.25 inch of crack length intersecting the inspection length will be detected, while any length less than 0.25 inch will not be detected. For a 0.25 inch crack, this means that if the crack is fully present in the inspection length, it will be detected, but if even 0.000000001 inch of the crack is outside of the assumed inspection length, it will not be detected. This is demonstrated in Figure 1 for an assumed 0.25 inch flaw. If the 0.25 inch flaw is fully within the inspected length, then it is detected. If even a small portion of it is outside of the inspection length, it is assumed to be missed. This is conservative, because there are practical limits to what would be missed.

Thus, for a 0.25 inch crack:

Probability = inspection coverage (25%) – 0.25"/weld length = 24.9%.

The same logic applies to a longer crack. Figure 2 demonstrates a similar case for a longer flaw. If at least 0.25 inch of the flaw intersects the inspected length, the flaw will be detected. However, if less than 0.25 inch of the flaw intersects; it is assumed to be undetected. Thus, for larger cracks, the inspected length is effectively increased by the assumed crack length minus the minimum detectable size to account for a crack that just intersects the end of the inspected length. For a single 2 inch crack in a weld, this increases the inspected length effectively by 1.75 inch.

For a 2 inch crack:

Probability = inspection coverage (25%)+(crack length-0.25")/weld length-0.25"/weld length = 25.3%

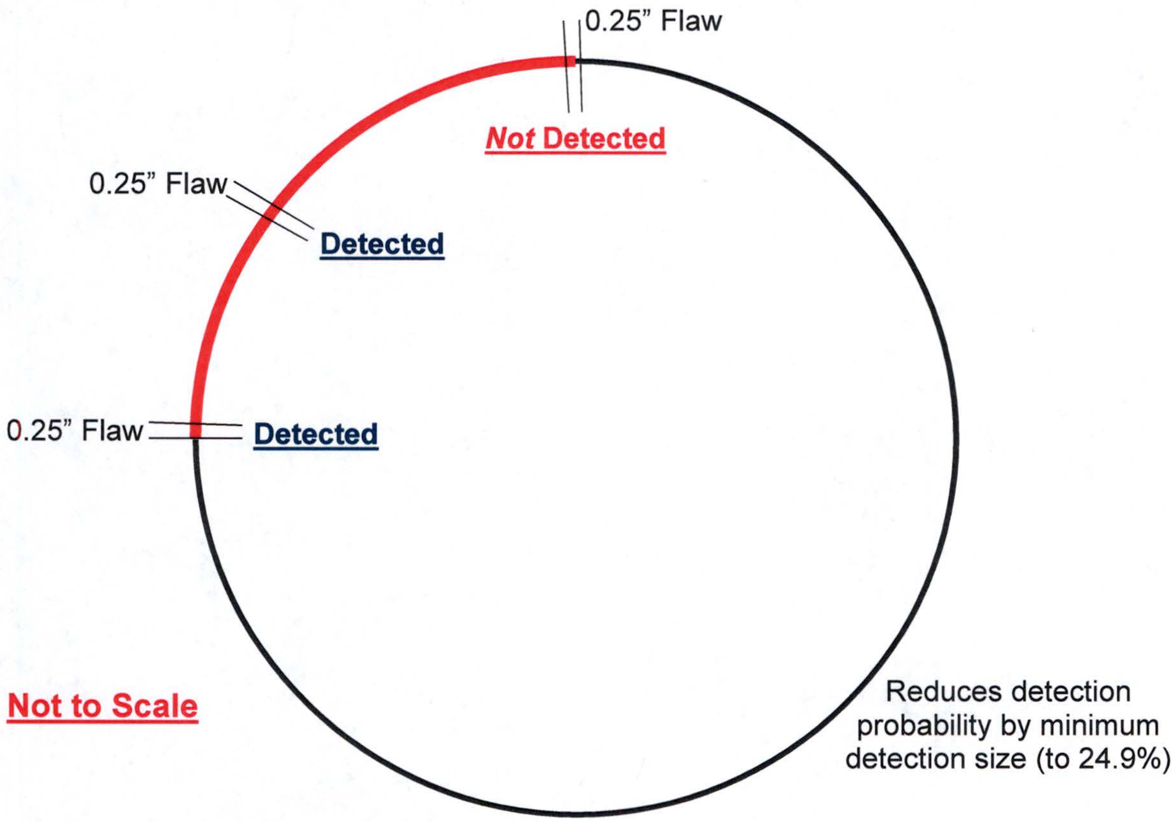

Figure 1: Schematic of core barrel weld inspection coverage and flaw intersection for detection by a 25% EVT-1 sampling inspection (0.25 inch flaw)

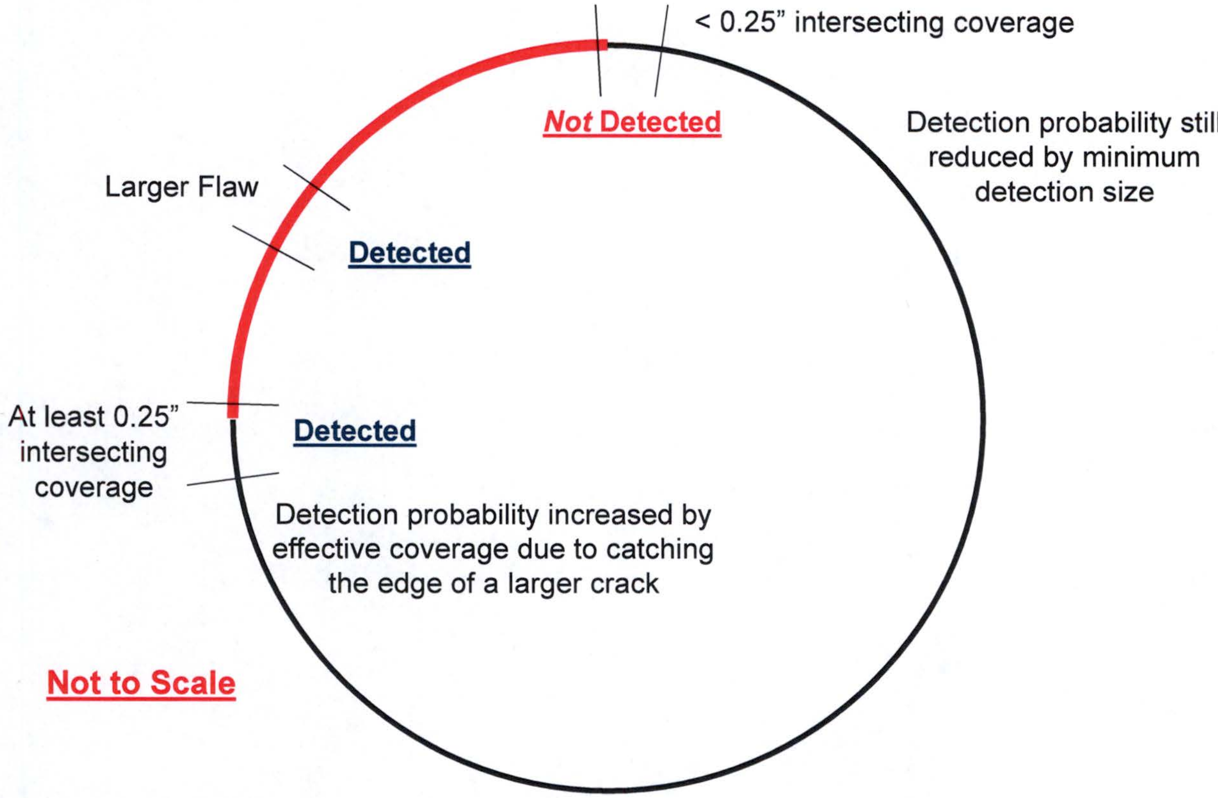

Figure 2: Schematic of core barrel weld inspection coverage and flaw intersection for detection by a 25% EVT-1 sampling inspection (flaw sizes greater than 0.25 inch)

Question 3: RAI-5—Response from the Material Reliability Program (related to the functionality of the core barrel under a faulted condition) states in part the following—Page 11 of LTR-AMLR-17-9, Rev.2:

"Testing was conducted to measure the effect of various abnormal conditions on the ability to insert the control rods and the time to scram. One of these tests investigated the effect of a full core drop type accident.

The testing performed ... also tested the effect of significant fuel deflections (i.e., the center of the fuel assembly was deflected laterally while the top and bottom were pinned) and determined that effect on scram time was acceptable. This provides evidence that the small 'bend' in the control rod insertion path that could be caused by a tilted core barrel would not have an impact on the ability to insert the control rods for core shutdown."

The staff requests industry discuss the following:

- During the scenario addressed above, how many control rod assemblies are allowed to encounter the small "bend" in the control rod insertion path due to a tilted core barrel?
- Is the deflection due to small "bend" observed (in the testing) in the control rod insertion path bounded by the safety margin established in a plant loss-of-coolant-accident accident analyses for each Combustion Engineering and Westinghouse unit?
- Provide a brief summary on how the full core drop test was conducted. [1]

Response:

Overview of Core Girth Weld Failure

Some additional background information is required prior to describing these test conditions and results. One of the purposes of the control rod insertion tests was to simulate the effects of a core barrel failure on rod insertion times and the capability for full rod insertion. This requires an understanding of what will occur during a hypothetical core barrel failure. Figure 3 contains Figure 3-5 from MRP-227, Revision 1 [12], which shows a Westinghouse-designed reactor vessel with the reactor vessel internals inside. Key components that are relevant to understanding what occurs if a core barrel fails include:

- the core barrel itself
- the core barrel outlet nozzles and interfacing vessel outlet nozzles
- the radial keys and interfacing clevis inserts
- the secondary core support (SCS) structure
- the upper core plate alignment pins

The components attached directly to the core barrel are shown in more detail in Figure 4.

Several scenarios for failure of a core barrel girth weld can be considered. Each of these will result in some interaction between the alignment and interfacing components listed above.

- Girth weld above the baffle-former assembly (core shroud assembly in CE plants) fails
  - Complete 360° failure
  - Partial failure
- Girth weld within or below the baffle-former assembly (core shroud assembly in CE plants) fails
  - Complete 360° failure
  - Partial failure

Complete, 360° failure of a girth weld above the baffle-former assembly would likely result in the entire lower internals dropping down. The SCS structure is specifically designed to limit the vertical displacement of the lower internals after a postulated core barrel girth weld failure at the core barrel to flange weld. Should this postulated failure occur, the drop of the core barrel would cause the bottom plate of the SCS to displace vertically by a small amount and come into contact with the bottom head of the reactor vessel. The energy absorbers are designed to absorb the impact from this drop and prevent damage. The maximum possible drop from a complete girth weld failure is less than the distance the control rods are inserted into the fuel during normal operation, so they will stay aligned with the fuel assemblies and ready for insertion to achieve safe shutdown. The maximum possible drop is also less than the length of the fuel alignment pins, which prevents the top of the fuel assembly from shifting out of alignment with the control rod guide tube assembly and the control rods. Lateral support to the barrel as a whole during this hypothetical complete girth weld failure would be provided by the core barrel outlet nozzles, the radial keys, and the fuel alignment pins. The outlet nozzles are nearly in contact with the vessel outlet nozzles during normal operation, and would not allow significant tilting or shifting of the barrel near the top. The radial keys have a small gap interface with the clevises on the vessel and provide lateral and rotational support to the barrel, also limiting the potential for offset or twist in the barrel.

A partial failure of a girth weld above the baffle-former assembly could result in one side of the barrel being tilted downwards while the other side is still partially attached to the upper portion of the barrel. The vertical drop distance on the fractured side is still limited by the SCS, and any tilt would be limited by the radial keys. The same discussion of the potential vertical and lateral offsets of the fuel assemblies and control rods discussed for a complete failure still apply. The interfaces of the outlet nozzles and the radial keys will also limit the potential offsets and bends that can occur in this case.

This first group could be further divided into failures above and below the core barrel outlet nozzles. However, the SCS, the radial keys, and the engagement of the fuel alignment pins will limit potential lateral or circumferential movement in all cases.

Girth weld failures below the top of the baffle-former assembly present even less of a concern due to the presence of the other lower internals components attached to the core barrel. All of the discussion about the constraints provided by the SCS and radial keys for failures above the baffle-former assembly still apply, but the baffle-former assembly structure and other components like the lower support columns will act as a secondary support to reduce the potential lateral and vertical offsets. Thus, any test or argument that addresses a failure in the upper part of the core barrel will address failures in the lower part.

Note that CE-designed core support barrels have components that perform the same functions as described above for the Westinghouse-designed case. These components typically have different names, often look a bit different, and in some cases were placed in a different location, but the reasoning presented above is equally applicable to CE-designed internals.

Description of Control Rod Insertion Test (Response to Question Part 3)

Since a brief summary of how the test was conducted (question in bullet 3) is pertinent in responding to the other two questions, this is addressed next. The overall purpose of the test conducted was to evaluate how deflections at various points in the drive line could impact control rod insertion times. Various tests were conducted during the test program; however, only the two tests relevant to the questions asked are discussed.

The test setup was comprised of a simulated control rod drive mechanism (CRDM) and related driveline components, a full size prototypic 17x17 standard guide tube assembly, and a full size prototypic 17x17 fuel assembly [13]. The fuel assembly was submerged in water and the guide tube was wetted in an attempt to achieve a prototypic friction between the control rods and guide tube and fuel assembly. The simulated CRDM contained features which allowed for tuning of drop times to match drop times achieve during full-flow loop drop tests. The drop times were also confirmed prior to performing any tests to be repeatable within less than ± 1-percent.

The first test was conducted considering a vertical and lateral offset of the fuel assembly top nozzle [13]. The purpose of this test was to assess the impact that a core drop would have on rod insertion times. The vertical offset considered in the test is on the order of and slightly greater than the vertical offset expected for the core drop condition. Similarly, the lateral offset simulated was representative of the lateral offset that would occur between the fuel assembly top nozzle and the fuel alignment pin when the fuel top nozzle was vertically offset due to the core drop. The results of this test show less than an 8-percent impact on rod insertion times in the configuration described and that the rods were able to fully insert.

The second test was conducted considering a lateral offset of the fuel assembly at mid span with the top and bottom of the fuel assembly pinned [13]. The purpose of this test was to assess the impact that a fuel assembly deflections would have on rod insertion times. The lateral offset used for this test was representative of approximately one-half of the cumulative gap between fuel assemblies at the widest width of the core for a plant that uses 17x17 fuel assemblies. In other words, it assumed a case where half of the fuel assemblies in the core were deflected to one side such that the grids nearest to the mid-span of the fuel assembly were in contact across the core and the grid on the peripheral fuel assembly is in contact with the baffle plate. The results of this test show less than a 2-percent impact on rod insertion times in the configuration described and that the rods were able to fully insert.

Response to Question Part 1

With regards to the first part of this question (bullet 1), only one path was simulated by the test. In the case of a core drop, it would be expected that all fuel assemblies would be laterally offset by a similar amount and have some slight variation in vertical offset associate with the slight tilt of the barrel prior to contact at the outlet nozzle gaps. Therefore, the results of this test are expected to be representative of all control rod locations.

Response to Question Part 2

With regards to the second part of this question (bullet 2), Westinghouse has not performed a review of deflections calculated for all Westinghouse and CE loss-of-coolant accident analyses to confirm that the deflections simulated in the test are bounding of these analyses. However, it is expected that the 17x17 test adequately represents all of the other applicable fuel designs. This is based on the testing parameters and results, specifically, the significant deflections applied relative to the total available gap across the core, the nearly negligible impact on rod insertion times, and the ability to fully insert the rods. It is expected that a more expansive search would result in showing that the test results are representative or bounding.

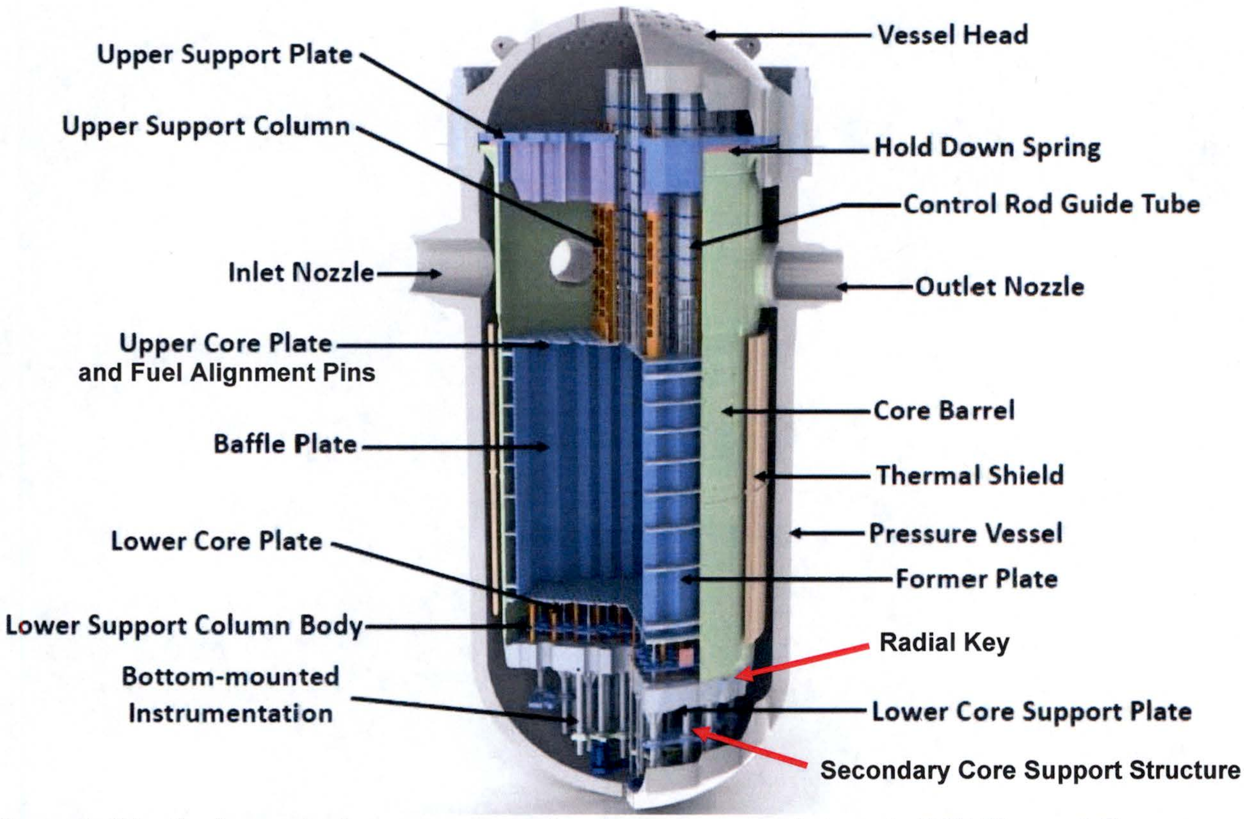

Figure 3: Westinghouse-design reactor vessel and reactor vessel internals ([12] Figure 3-5)

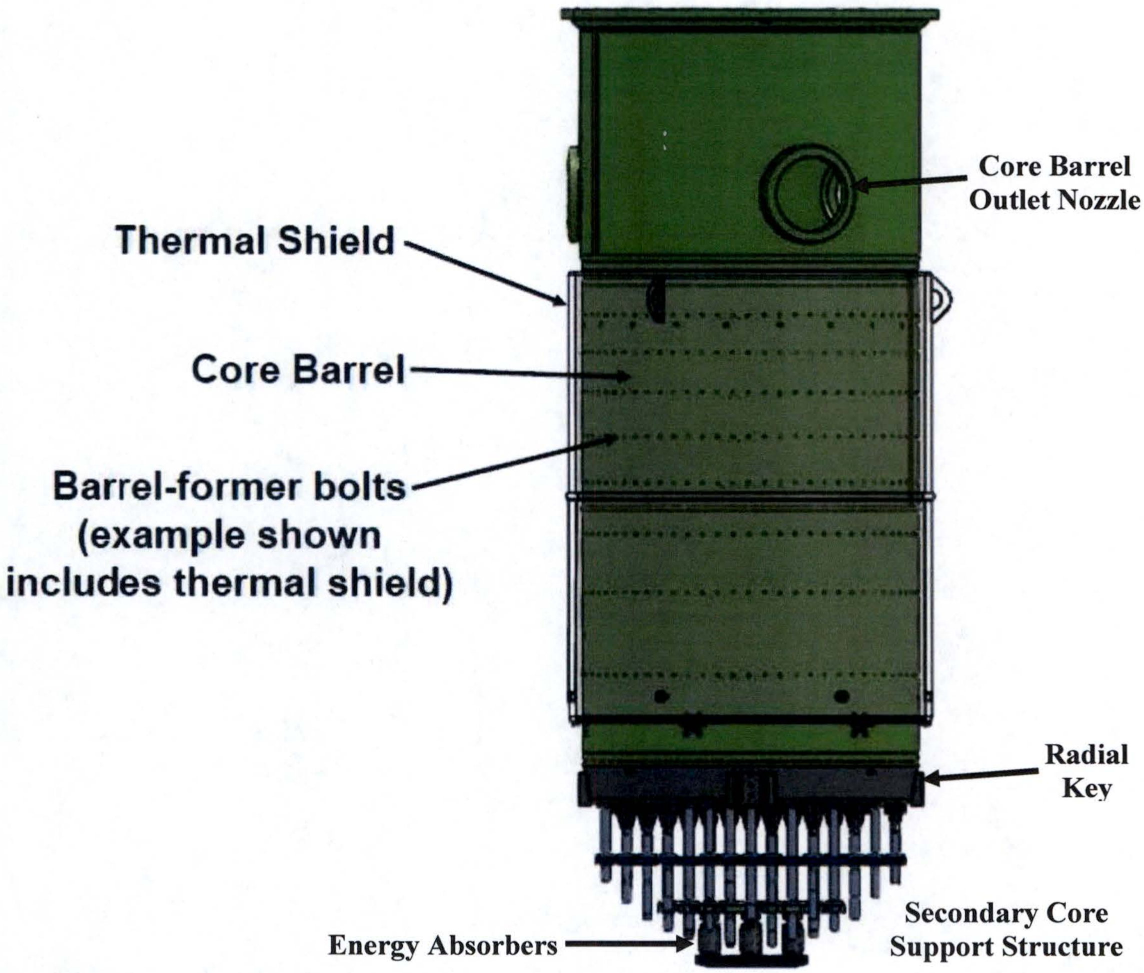

Figure 4: Westinghouse-design core barrel and secondary core support structure ([12] Figure 4-21)

Question 4: The response to RAI 10, which concerns the adequacy of a 25% sample inspection of the deep beam welds, provides a markup of Table 5-2 showing the expansion to the remaining deep beam welds if cracking is found in the initial sample. The markup shows this expansion inspection must be completed by the end of the next refueling outage.

Why not require the expansion be completed during the same refueling outage during which the cracking was found in the initial sample, consistent with the approach for the core barrel welds? [1]

Response:

The technical basis for allowing one cycle in which to complete the inspection is the difference in redundancy and function of an individual weld between the deep beams and the core barrel. Each beam is kept in place by more than one weld, and the failure of an entire weld would not result in loss of functionality. This is quite different from the impact of a core barrel girth weld failure. Additionally, the insertion and removal of fuel each outage provides an element of regular monitoring to the deep beams. Both of these elements are described in the response to RAI 10 [6]:

> *The function of the deep beams is to directly support the core, to keep the fuel in place and to maintain alignment for control element assembly insertion. From the standpoint of functionality, the welded array is a redundant structure. If one weld of a cross-beam fails completely, the other end of that particular beam would still be attached to another main beam. The main beams are welded at multiple locations and would require multiple weld failures to compromise function. Assurance of the continued functionality of the deep beams is also aided by the fact that the onset of the loss of structural functionality would be likely to be first detected during fuel loading or unloading conducted during each refueling outage. The fuel loading and unloading operations are expected to detect this loss of functionality as misaligned fuel assemblies or abnormal difficulty with removing or placing fuel assemblies.*

The additional cycle also has the added benefit of providing the utility with additional flexibility in planning and implementing this expansion inspection most efficiently and effectively, allowing it to be completed during the same outage or the next one.

Clarification on the Response to RAI 8 Provided in Letter MRP 2017-027 [14]

The response to RAI 8 included a note on the determination of re-examination periods for the baffle-former bolts. This was Note 12 and stated:

> *"12. Re-examination periods shall be determined by plant-specific evaluation per the MRP-227 Needed Requirement 7.5 as documented and dispositioned in the owner's plant corrective action program. If atypical or aggressive baffle-former bolt degradation is observed, the interim guidance (MRP 2016-021 and MRP 2017-009) provides limitations to the permitted re-inspection interval (not to exceed 6 years maximum) unless further evaluation is performed to justify a longer interval. If evaluation justifies a longer re-inspection interval, it is not permitted to exceed 10 years."* [14]

The definition of "atypical or aggressive baffle-former bolt degradation" was not explicitly defined in this note. The intention was to use the same definition of "atypical or aggressive baffle-former bolt degradation" used in the baffle-former bolt interim guidance in letter MRP 2017-009 [15]. Thus, Note 12 will be modified as follows:

> "12. Re-examination periods shall be determined by plant-specific evaluation per the MRP-227 Needed Requirement 7.5 as documented and dispositioned in the owner's plant corrective action program. If atypical or aggressive baffle-former bolt degradation as defined in MRP 2017-009 (i.e., ≥3% of baffle-former bolts with UT or visual indications or clustering* for downflow plants and ≥5% of baffle-former bolts with UT or visual indications or clustering* for upflow plants) is observed, the interim guidance (MRP 2016-021 and MRP 2017-009) provides limitations to the permitted re-inspection interval (not to exceed 6 years maximum) unless further evaluation is performed to justify a longer interval. If evaluation justifies a longer re-inspection interval, it is not permitted to exceed 10 years.
> *"Clustering" is defined per NSAL-16-1 Rev.1 as three or more adjacent defective BFBs or more than 40% defective BFBs on the same baffle plate. Untestable bolts should be reviewed on a plant-specific basis consistent with WCAP-17096-NP-A for determination if these should be considered when evaluating clustering."

Clarification 1 on the Response to RAI 9 Provided in Letter MRP 2018-003 [6]

The response to RAI 9 in letter MRP 2018-003 [6] clarified the intention of the coverage requirements for the Westinghouse lower support columns (LSC) and the CE core support columns in MRP-227, Revision 1 and included revised entries for Table 4-5 and Table 4-6. However, these tables did not specify the distribution of the 25% coverage requirement for the LSCs or core support columns. The stress and dose in the columns for both designs are expected to vary from the center of the lower core plate or core support plate to the outer edge of the plate. To address the variation in column degradation behavior that may occur due to these variations in stress and dose, a requirement to evenly distribute the inspection across the population of LSCs or core support columns will be added to these two tables. Note that the CE core support columns were a Primary component in MRP-227, Revision 1 and therefore included in Table 4-2. The response to RAI 9 in [6] provides the basis, and corresponding markup, to move the core support columns from a Primary to an Expansion component, and therefore include in Table 4-5.

For the CE core support columns, the revised text for the "Examination Coverage" column of Table 4-5 will read (showing changes from MRP-227, Revision 1 text):

> "Plants with full height bolted core shroud plates: 25% of the total number of column assemblies (both visible and non-visible from above the core support plate) ~~as visible~~ using a VT-3 examination from above the core support plate. The inspection coverage must be evenly distributed across the population of column assemblies.
>
> Plants with core shrouds assembled in two vertical sections: 25% ~~100%~~ of the accessible surfaces of the core support column welds, from the top side of the core support plate ~~(Note 3)~~. The inspection coverage must be evenly distributed across the population of core support column welds.
>
> (Notes 3 and 4)"

The added Note 4 for Table 4-5 will state:

> "4. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. Possible examples include, but are not limited to, inspection of all column assemblies or accessible core support column welds in one quadrant of the core support plate (based on the azimuthal symmetry of the plate) or inspecting every fourth column or weld across the entire plate."

For the Westinghouse LSCs, the revised text for the "Examination Coverage" column of Table 4-6 will read (showing changes from MRP-227, Revision 1 text):

> "25% of the total number of column assemblies (both visible and non-visible from above the lower core plate) using a VT-3 examination from above the lower core plate. The inspection coverage must be evenly distributed across the population of column assemblies.
>
> (Notes 3 and 4)"

The added Note 4 for Table 4-6 will state:

> "4. Justification that adequate distribution of the inspection coverage has been achieved can be based on geometric or layout arguments. Possible examples include, but are not limited to, inspection of all column assemblies in one quadrant of the lower core plate (based on the azimuthal symmetry of the plate) or inspecting every fourth column across the entire plate."

Clarification 2 on the Response to RAI 9 Provided in Letter MRP 2018-003 [6]

The response to RAI 9 in letter MRP 2018-003 [6] states that if degradation is observed in the initial inspection population for either the Westinghouse LSCs or the CE core support columns, then the examination would expand to include the remainder of the population of the column bodies. Note 3 was added to Table 4-5 and Table 4-6:

> "*3. The stated minimum coverage requirement is the minimum if no significant indications are found. However, the Examination Acceptance criteria in Section 5 require that additional coverage must be achieved in the same outage if significant flaws are found. This contingency should be considered for inspection planning purposes.*" [6]

This note clearly states that the expansion must be conducted during the same outage in which significant flaws are found. However, the text in Table 5-2 for CE and Table 5-3 for Westinghouse requires some modification to clearly state this timing requirement.

Additionally, Table 5-2 and Table 5-3 need to clearly state the level of degradation required to trigger the expansion from the 25% sample to the remaining LSCs or core support columns. Descriptions of the level of degradation appropriate for triggering the expansion were already provided in the "Expansion Item Examination Acceptance Criteria" column of Table 5-3 of MRP-227, Revision 1 for the Westinghouse LSCs and in the same column for the revised Table 5-2 entry provided in the response to RAI 9 in letter MRP 2018-003.

To improve the clarity of MRP-227 Tables 5-2 and 5-3, text providing the expansion criteria and timing for the LSCs and core support columns will be added to the "Examination Method/Frequency" column of the tables.

The following paragraphs will be added to the "Expansion Criteria" column of Table 5-2 for the CE core support columns:

> "Plants with full height bolted core shroud plates: The confirmed detection of missing or separated welds in a core support column or fractured, misaligned, or missing core support columns shall require examination of 100% of the accessible uninspected core support column assemblies using a VT-3 examination from above the core support plate (minimum of 75% of the total population of core support column assemblies) during the same outage.
>
> Plants with core shrouds assembled in two vertical sections: The confirmed detection of a relevant disruption of discontinuity in the surface of a core support column weld shall require examination of 100% of the accessible uninspected core support column welds from the top side of the core support plate (minimum of 75% of the total population of core support column welds) during the same outage."

The following paragraph will be added to the "Expansion Criteria" column of Table 5-3 for the Westinghouse LSCs:

> "The confirmed detection of fractured, misaligned, or missing lower support columns shall require examination of 100% of the accessible uninspected lower support column assemblies using a VT-3 examination from above the lower core plate (minimum of 75% of the total population of lower support column assemblies) during the same outage."

Clarification on the Response to RAI 19 Provided in Letter MRP 2017-027 [14]

The response to RAI 19 in letter MRP 2017-027 [14] provides a revised entry for the Control Rod Guide Tube Assembly Guide plates (cards) to be used in Table 4-3 of MRP-227, Revision 1. This revised table entry implements the requirements of WCAP-17451-P [16] under both the "Examination Method/Frequency" and "Examination Coverage" columns and references Note 7 to provide more information on WCAP-17451-P:

> "7. *In WCAP-17451-P Revision 1 the baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. Initial inspection prior to the license renewal period may be required. Refer to the latest revision of WCAP-17451-P including the results of the NEI 03-08 Generic Topical Report screening and/or NRC review for the specific guidance elements."* [14]

The statement in the last sentence of this note implementing the "latest revision of WCAP-17451-P" will be revised to avoid potential issues for the development of a safety evaluation on MRP-227, Revision 1.

The current applicable version of WCAP-17451-P is Revision 1 [16]. This revision is still applicable for many plants. However, recent operating experience has led to the creation of interim guidance on WCAP-17451-P, Revision 1 for certain control rod designs. This interim guidance has been issued as NEI 03-08 "Needed" guidance under Pressurized Water Reactor Owners Group (PWROG) letter OG-18-46 [17] and was provided to the staff for information under Pressurized Water Reactor Owners Group letter OG-18-76 [18]. For MRP-227, Revision 1, Note 7 will be revised to reference WCAP-17451-P, Revision 1 as modified by the interim guidance of letter OG-18-46. The revised note will state:

> "7. In WCAP-17451-P ~~Revision 1~~ the baseline examination schedule has been adjusted for various CRGT designs, the extent of individual CRGT examination modified, and flexible subsequent examination regimens correlating to initial baseline sample size, accuracy of wear estimation and examination results. Initial inspection prior to the license renewal period may be required. ~~Refer to the latest revision of~~ Use WCAP-17451-P, Revision 1, including the modified requirements due to the interim guidance provided in letter OG-18-46. ~~results of the NEI 03-08 Generic Topical Report screening and/or NRC review for the specific guidance elements.~~"

References:

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4. *Materials Reliability Program: Pressurized Water Reactor Issue Management Table, PWR-IMT Consequence of Failure (MRP-156).* EPRI, Palo Alto, CA: 2005. 1012110.
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6. Materials Reliability Program Letter, MRP 2018-003, "Responses to NRC Requests for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline" (CAC No. MF7740)," January 30, 2018. (ML18038A875)
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8. Pressurized Water Reactor Owners Group Document, PWROG-15105-NP, Revision 0, "PA-MSC-1288 PWR RV Internals Cold-Work Assessment," April 2016.
9. Westinghouse Letter, LTR-AMLR-17-28, Revision 0, "MRP-191 Revision 2 Expert Panel Meeting Minutes," January 22, 2018 (Westinghouse Proprietary).
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13. Westinghouse Report, WCAP-9251, "SCRAM Deflection Test Report 17x17 Guide Tubes, 96 Inch and 150 Inch," December 1977 (Westinghouse Proprietary).
14. Materials Reliability Program Letter, MRP 2017-027, "Responses to NRC Request for Additional Information for Electric Power Research Institute Topical Report MRP-227, Revision 1, 'Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluations Guideline' (CAC No. MF7740)," October 16, 2017. (ML17305A056)
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16. Pressurized Water Reactor Owners Group Report, WCAP-17451-P, Revision 1, "Reactor Internals Guide Tube Wear – Westinghouse Domestic Fleet Operational Projections," October 2013.
17. Pressurized Water Reactor Owners Group Letter, OG-18-46, "Transmittal of Approved "Needed" Interim Guidance for Addressing Accelerated Guide Card Wear Issue Described in NSAL-17-1, (LTR-ARIDA-17-270, Revision 0), PA-MSC-1471," February 20, 2018.
18. Pressurized Water Reactor Owners Group Letter, OG-18-76, "Transmittal of Interim Guidance for Addressing Accelerated Guide Card Wear Issue Described in NSAL-17-1 (PA-MSC-1471)," March 23, 2018. (ML18088A199)