Core Barrel Weld Cracking Issue Safety Significance Evaluation

Prepared by:

Bryan M. Wilson
Joshua K. McKinley
Melanie R. Fici
Westinghouse Electric Co., LLC
Cranberry, PA

Ref. Westinghouse letter LTR-AMLR-19-45, Rev. 0, dated 6/25/2019

Westinghouse

To: James P. Molkenthin
cc: Douglas F. Holderbaum

Date: June 25, 2019

From: Aging Management & License Renewal
Ext: (412) 374-3281
Fax: (724) 940-8564

Your ref: N/A
Our ref: LTR-AMLR-19-45, Rev. 0

Subject: **Core Barrel Weld Cracking Issue Safety Significance Evaluation**

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1.0 Background / Issue Description

In the Spring of 2018, during a MRP-227 visual inspection (EVT-1) of the core support barrel (CSB) circumferential girth welds, approximately 45 crack-like indications, the majority of which were off-axis (circumferentially oriented), were found along the core barrel middle axial weld (MAW) (in the beltline) at St. Lucie Unit 1 [9] The indications were found while using the MAW as an orientation marker when moving from one girth weld to the next. This inspection also revealed one crack perpendicular to the middle girth weld (MGW) (also in the beltline); however the size of this crack was insufficient to trigger the expansion inspection. The MGW and MAW indications were dispositioned in a plant specific evaluation permitting the plant to operate for one additional 18-month cycle. This 18-month limit is based on the requirement in [19] to re-inspect the flaws to validate the crack growth rate used in the disposition calculations. St. Lucie Unit 1 is a Combustion Engineering (CE)-designed pressurized water reactor (PWR); however, the design of the CE CSB is similar to that of the core barrel in Westinghouse-designed PWRs, and the evaluation and conclusions of this report apply to both designs. For simplicity, the terminology "core barrel" will be used throughout this report to describe both designs unless the CE CSB is specifically discussed.

MRP-227-A [1] assigned the core barrel girth welds to the primary component inspection category and made the core barrel axial welds expansion components to the core barrel girth welds. Expansion components are not inspected until specific degradation thresholds, defined in MRP-227, are observed in the primary component. This ranking of circumferential welds as a leading component was based on a failure modes, effects, and criticality analysis (FMECA) conducted using an expert panel process as part of MRP-191 [2]. This FMECA process considers the likelihood of degradation and the significance of

the degradation in terms of potential for causing core damage. The inputs supporting this FMECA were the components' function, potential degradation mechanisms, past operating experience (OE), and operating conditions (stress, fluence, water chemistry, temperature, etc). Based on these inputs, the expert panel screened the core barrel welds in for potential cracking degradation mechanisms driven by the neutron exposure of the barrel and the residual stresses and material changes caused by the welding process. Since the welding residual stress is expected to be oriented perpendicular to the weld and the heat affected zone follows the edge of the weld, the panel expected that any cracking, if present, would be oriented parallel to the weld and in the heat-affected zone. This assumption then entered into the consideration of the safety significance of an axially-oriented crack versus a circumferentially-oriented crack. The potential circumferentially-oriented crack was assigned a higher significance, because it would be more likely to result in a condition with potential for core damage, such as a core barrel separation. This ultimately led to the ranking of the girth welds as primary components and axial welds as expansion components.

Operation with a separated core barrel has not specifically been confirmed to result in core damage; however, for the purposes of the MRP-191 [2] FMECA process this assumption was applied. The assumption was conservatively based on the lack of design and qualification analysis supporting operation in this condition. The core barrel is designed and qualified to sustain all design basis conditions without impact on function. Operation with a separated core barrel is beyond the plant's design basis; therefore, no component qualification or safety analysis considered the impacts of operating in this condition. The only evaluations conducted for such an event were the design and structural analysis of the secondary core support system, the design of the reactor vessel, and the driveline alignment. This was done to ensure that a complete separation event could be accommodated without failure of the vessel and without preventing control rod insertion for safe shutdown following the event.

Westinghouse provided input directly to the Electric Power Research Institute (EPRI) in the creation of the reactor internals inspection and evaluation guidance contained in MRP-227. This input included the multiple basis documents that MRP-227 references and the final guidance document itself. MRP-227 is used by utilities to manage plant reactor internals aging degradation to maintain safe and reliable operation during the period of extended operation beyond the 40-year design life of the plant. As a result, this document (MRP-227) and the input provided by Westinghouse used to create the guidance contained within can be conservatively treated as a delivered basic component for the purpose of this evaluation. The St. Lucie Unit 1 OE from 2018 potentially invalidates the assumptions about crack orientation and the assignment of Primary and Expansion components in MRP-227. As a result, there is a potential that following the current guidance in MRP-227 could result in not identifying and properly managing circumferential cracking, should it exist at an axial weld location before the expansion criteria from the girth welds are triggered. Therefore, the purpose of this letter is to evaluate if having guidance that could potentially result in missing circumferential cracks in the core barrel during MRP-227 inspections constitutes a deviation involving a basic component in that it increases the risk to plant safety from the standpoint of shutdown capability and core damage.

2.0 Issue Evaluation

2.1 Evaluation of Shutdown Capability

An evaluation of the ability to insert the control rods in the event of a hypothetical core barrel separation was evaluated in [3] for both Westinghouse and Combustion Engineering (CE) design plants. This report discusses design features that limit the drop associated with a core barrel separation, the impact of this drop on critical interfaces intended to maintain alignment of the control rod path, and historical testing performed to confirm rod insertion capability with the misalignments that would be expected in a separated condition. This report concluded that, following a complete separation of the core barrel at any elevation, the control rod path would remain aligned to allow for complete insertion of the rods should a reactor trip be initiated either automatically or manually.

2.2 Likelihood of Operating with a Separated Core Barrel (detectability)

This same report [3] evaluates the likelihood that a separated core barrel condition during normal or upset operation would be detected and result in an automatic or immediate manual reactor trip. For the postulated case where a core barrel separation results from faulted conditions, such as a Safe Shutdown Earthquake (SSE) or Loss-of-Coolant Accident (LOCA), it was determined that the event itself would result in an automatic reactor trip. Since the control rod path would remain aligned, as previously discussed, detection of the separation itself is not a concern.

On the other hand, for postulated cases where a core barrel separation occurs during normal/upset conditions, there would not be a significant initiating event, such as a LOCA or SSE, that would cause the reactor to trip automatically. Therefore, an evaluation of the likelihood of detecting the condition as it occurs or once it has occurred was required. This evaluation was conducted in [3] by first outlining each reactor trip setpoint and identifying which plant measurement signals (temperature, pressure, and neutron flux) are inputs to each setpoint. From there, various separation locations were evaluated to determine how each of the different measurement signals might be impacted. It was concluded that, while there may be some indications that a separation has occurred, there is uncertainty regarding how long it would take to diagnose the condition and shut down the reactor following the separation. Although the potential may exist for a separated core barrel to go undetected if it should occur during normal/upset operation, the following sections discuss the likelihood of forming or growing cracks to sufficient size to cause a complete separation under these conditions.

2.3 Likelihood of Cracks Existing

Although the root cause of the degradation discovered at St. Lucie Unit 1 is unknown, there is a possibility that these flaws are not due to the aging-related degradation that MRP-227 inspections are intended to manage. Early in the operation of St. Lucie Unit 1, the thermal shield worked loose and caused significant fatigue cracking of the CSB [4]. The thermal shield was removed and the barrel flaws mitigated; however, it is possible that the visual inspections of that day did not detect the small flaws by the MAW that were discovered in 2018.

From a relative susceptibility standpoint, there is no clear reason to expect axial welds to be more susceptible to age-related cracking, such as stress corrosion cracking (SCC) or irradiation-assisted stress corrosion cracking (IASCC), than girth welds. Both locations are full penetration welds. While records are not available to definitively confirm the axial weld geometry, it is anticipated that both the axial and girth welds would be similar in size, with the girth welds potentially being slightly larger due to the local core barrel geometry. Furthermore, in comparing the relative internal constraints associated with pulling the weld joint together, it is expected that bending along the weak axis of the plate and flexing of the shell would result in lower internal constraint forces in the axial welds as compared to closing a radial offset or correcting for ovality differences between different barrel shells with the girth welds. Based on this, it is expected that the axial welds could have lower levels of residual stress; though for the purposes of this evaluation, both locations are assumed to have similar levels of residual stress approaching the yield strength of the material. Sensitization in the heat affected zone from the weld fabrication is expected in both the axial and girth welds. Some core barrel design documents have required annealing post-weld heat treatment of the axial welds, which would tend to make these less susceptible as compared to a weld which has not received such post-weld heat treatment. Similar requirements have not been found for girth welds.

Since not all documentation has yet been recovered on all Westinghouse and CE core barrels, annealing alone is not conclusive related to relative susceptibility of axial versus girth welds. Another aspect to consider is the operating stresses on the barrel. In general, normal operating stresses in the barrel are quite low. As discussed in response to RAI 5 on MRP-227, Revision 1 (sent to the NRC via MRP letter MRP 2018-003 [16] dated January 30, 2018) the most significant operating stress results from the thermal gradient through the wall of the barrel. While these stresses are much larger than the primary stresses on the barrel, they remain much lower than the anticipated residual stress. Even when considering that some of the residual stress will relax over the course of operation due to irradiation stress relaxation, both welds are anticipated to be at a higher stress state as compared to the base metal. While the St. Lucie OE shows circumferential growth for a short distance into the base metal, this is not unprecedented. BWR experience [17] [18] has shown that circumferential growth from axial welds has occurred in a similar component and was attributed to welded attachments and alignment features. It is possible that similar localized residual stress region and heat affected zone could be present along axial welds of the core barrel if similar welded attachments and alignment features were used during fabrication of these components. Without a significant driving stress, any cracks extending circumferentially from an axial weld would be expected to arrest fairly quickly once they begin to extend outside of the heat affected zone. Conversely, circumferentially oriented age-related cracking in or adjacent to a girth weld would have a more likely driving stress (weld residual stress) and more likely environment (heat affected zone) to continue the growth along the circumference of the barrel. Additionally, for these same reasons the girth welds would have a greater potential for multiple SCC or IASCC cracks to form and potentially connect.

Since 2011, the initial baseline inspections of the Primary component CB and CSB assembly flange and girth welds per the requirements of MRP-227 have been performed at 20 PWR units ([5], [6], [7], [8], and Attachment 1). From a review of the inspection results, only one recordable indication was reported (observed adjacent to the MGW at St. Lucie Unit 1 in spring 2018 [8, 9]). Therefore, based on the

inspection experience, significant indications of age-related cracking have not been observed for the CB or CSB flange and girth welds.

More recently, three U.S. PWRs (ANO Unit 2, St. Lucie Unit 1, and Calvert Cliffs Unit 2) have performed visual (EVT-1) examinations of the expansion component axial welds in the CSB assembly. At St. Lucie Unit 1, multiple indications were detected in the heat-affected zone and base metal adjacent to the MAW [8]. No axial weld indications were reported for the other two units. Of the three units which have undergone core barrel axial weld inspection, Calvert Cliffs Unit 2 and St. Lucie Unit 1 share the same CSB design and have a very similar lower internals designs. One of the primary differences in the lower internals design is that the thermal shield was eliminated from the Calvert Cliffs plant designs prior to startup, whereas St. Lucie Unit 1 operated with a thermal shield in place for a number of years prior to it being removed due to result of excess vibration, as previously discussed. While this difference is notable when considering the significant difference in axial weld inspection results, the impact of the thermal shield has not been confirmed as the cause for the cracking along the axial weld detected at St. Lucie Unit 1 in 2018. The ANO Unit 2 design is a newer, larger capacity vintage of CE plants, which was redesigned to provide general strengthening of internals structures in response to increasing reactor core sizes and load changes. A couple notable features of the ANO Unit 2 lower internals are the elimination of the thermal shield from the design, the smaller diameter of the CSB (ANO Unit 2 core barrel inner diameter is about 8% smaller as compared to the St. Lucie Unit 1 core barrel), and the redesign of the CSB to have a thicker wall in the upper segments of the barrel as compared to the lower segments in the region of the core shroud and the lower support structure.

Figure 1 summarizes the effective full power year (EFPY) and unit age data from [5], [6], [7], and [8] corresponding to the date of the baseline barrel weld inspections (not that this figure does not include the most recent results from Calvert Cliffs Unit 2). From these data, operating time alone may not be a reliable indicator for barrel flange and girth weld cracking potential, because units similar in age and EFPY to the unit with confirmed cracking showed no indication of cracking in the girth welds inspected. With only one data point for confirmed detection of cracking for either the girth welds or axial welds, meaningful assessments cannot be made. Considering the measured crack lengths discovered and the inconsistency of these crack lengths with expected slow growth rates for cracks in PWR core barrels [15, 19], it is possible that the initiating mechanism for the observed condition at St. Lucie Unit 1 could be related to the early-life thermal shield failure that occurred at that plant and caused earlier core barrel cracking.

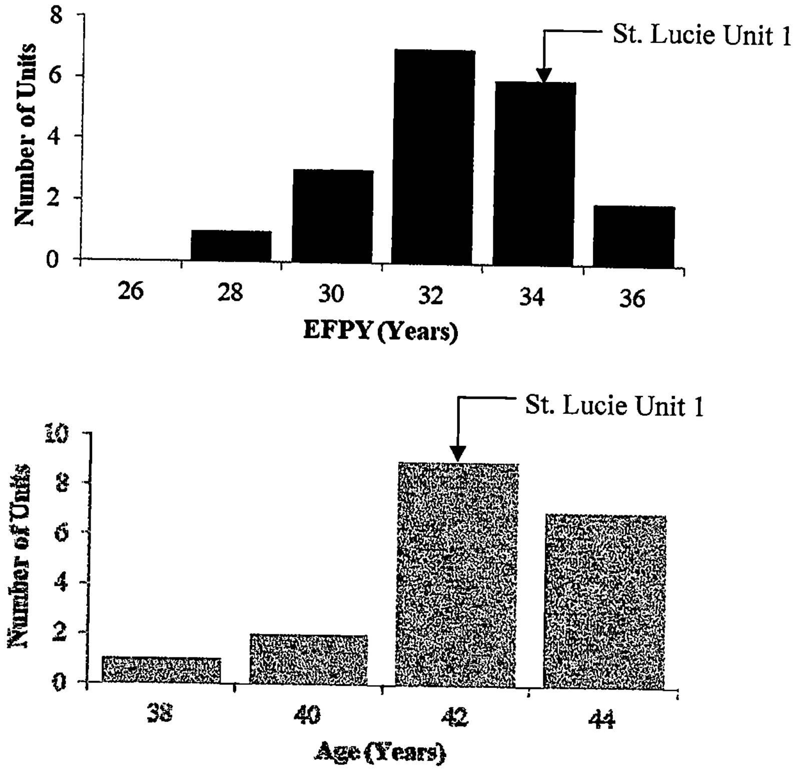

Figure 1: Operating Time at Baseline Barrel Weld Inspections

In addition to the OE at St. Lucie Unit 1, boiling water reactors (BWRs) have experienced similar circumferential cracking from axial welds; these are referred to in BWR documentation as "off-axis" flaws. Off-axis flaws associated with BWR OE have been linked to a combination of construction or field fabrication issues and age-related degradation issues (see BWRVIP-302 [10] as an example). Intergranular SCC is a commonly observed phenomenon in BWR core shrouds. In 1990, cracks were observed in the heat-affected zone of a circumferential weld in the beltline region of a foreign BWR core shroud [11]. In 1993, in response to recommendations made after the 1990 BWR core shroud OE, a visual examination of a domestic BWR core shroud identified cracks in the heat-affected zone [11]. Subsequent examinations of the core shroud at other BWR plants identified additional instances of cracking in the heat-affected zones [11]. The BWR experience has shown that portions of the core shrouds most susceptible to SCC are associated with the base metal in areas immediately adjacent to the shroud welds (i.e., the heat-affected zone) [11]. While the service experience between BWR and PWR plants differs [12], the observation of SCC in the heat-affected zone of BWR core shrouds supports the assessment that SCC in PWR barrels is most likely to be observed in the heat-affected zone.

2.4 Likelihood of Cracks Growing to a Critical Length

The overall likelihood of a circumferential flaw growing long enough under normal/upset operating conditions to cause a core barrel failure is extremely low. Although emergency and faulted conditions

could cause limited flaw growth, the probability of these events is sufficiently low that they are not considered. Furthermore, as previously discussed, separation due to faulted conditions would initiate an automatic reactor trip and therefore, this is not a limiting condition in the consideration of core barrel separation. The normal operating stresses are relatively small and are not expected to drive a flaw to the length necessary to cause a failure during the period of license extension. As a flaw grows around the core barrel, it is assumed that it extends through wall. This relieves the stresses caused by thermal gradients in the core barrel. Although the dead weight of the core barrel components results in tensile stress that contributes to circumferential flaw growth, it is counteracted by the coolant flow under normal operating conditions. The net result is a relatively small tensile axial stress. This is evidenced by larger critical crack sizes for normal/upset conditions as compared to critical crack sizes based on faulted conditions. Any fatigue flaw growth resulting from transient stresses is minimized under base load operation.

Critical weld crack lengths for faulted conditions have been calculated generically for Westinghouse and CE plants in WCAP-17684-P [13]. While the critical flaw sizes are relatively small and potentially smaller than some inaccessible locations around the core barrel (such as behind neutron panels), this analysis contains a significant amount of conservative assumptions and uses bounding inputs due to the generic nature of the report. Some of the key conservative assumptions are:

- Critical crack lengths are based on a thru-wall crack and linear-elastic fracture mechanics
- The crack growth rate is based on the highest fluence at any location around the barrel for a given weld elevation
- Maximum stresses at any locations around the barrel for a given weld elevation are used. No consideration is given for stress variations around the circumference.

Therefore, it is anticipated that if further work were to be done to refine the flaw tolerance analysis, it would show that critical cracks sizes are much greater than those conservatively calculated in this report.

The next limiting condition is associated with normal operation and is the crack growth due to high-cycle fatigue (HCF). Although the stresses associated with this condition are much lower, the concern is the large number of cycles that can be accumulated over a short period of time. Some scoping studies for HCF show the critical crack sizes to be on the order of one foot or greater considering many of the same conservative assumptions outlined above. These evaluations also considered a large factor applied to the stress amplitude to account for uncertainties in the forcing functions. Therefore, it is anticipated that if further work were to be done to refine the flaw tolerance analysis in these scoping studies, it would show that critical crack sizes addressing HCF are much greater than those conservatively calculated.

Currently, there is little OE regarding crack growth rates for SCC or IASCC in barrels of operating PWRs. As a result, crack growth rates developed for SCC in BWRs have generally been used in the evaluations of flaws in PWR barrels (e.g., see MRP-227-A [1] and WCAP-17684-P [13]). Interim guidance to WCAP-17096-NP-A [19] in PWROG-17071-NP [14] references an EPRI Report on IASCC crack growth rates [15] for an updated crack growth rate model that can be used in weld locations subjected to fluence levels greater than 3×10^{21} n/cm^2 (E > 1 MeV). Assumed crack growth rates in WCAP-17684-P are generally low for normal operating conditions as evidenced by the small difference

between 18-month and 10-year allowable flaw sizes. Based on these rates, crack growth at the end of the inspection interval is anticipated to be small and catastrophic crack growth would not be expected.

3.0 Identification of Affected Plants

The issue applies to any Combustion Engineering and Westinghouse plant that uses MRP-227 guidance for CSB (CE) or Core Barrel (Westinghouse) weld inspections.

4.0 Conclusion

Based on the evaluation conducted herein, it is concluded that age-related cracking at axial welds is not likely to be any more prevalent or safety significant than cracking at circumferential welds. While the St. Lucie Unit 1 OE and BWR experience highlights the potential for circumferential cracking to be present in axial welds and the conclusions of [3] highlight the potential for a fully separated core barrel to go undetected should it occur during normal/upset operation, it is not credible that circumferential cracks extending from axial welds will grow substantially into the base metal during normal/upset operation due to few known driving mechanisms and relatively low crack growth rates. Therefore, full separation during normal/upset operation due to this cracking is not expected. Furthermore, as discussed herein the automatic reactor trip associated with a faulted condition as well as the alignment provided by design features of the reactor internals will ensure safe shutdown can be achieved if a separation were to occur during a faulted event. As a result, while the MRP-227 reactor internals inspection and evaluation guidance as written may allow for an inspection to overlook circumferential cracking in an axial weld, the monitoring of girth welds remains a reasonable surrogate for identification of age-related cracking of the core barrel prior to any significant degradation which could impact plant safety from the standpoint of shutdown capability and core damage. However, as discussed within, there remains a great deal of uncertainty associated with the cause of the cracking discovered at St. Lucie. Therefore, it is reasonable that the industry take a conservative position relative to future inspections of the core barrel axial welds until further information becomes available.

If you have any questions or desire further information, please contact the undersigned.

Author: ELECTRONICALLY APPROVED*
Bryan M. Wilson
Reactor Internals Design & Analysis I

Reviewer: ELECTRONICALLY APPROVED*
Joshua K. McKinley
Aging Management & License Renewal

Approved: ELECTRONICALLY APPROVED*
Melanie R. Fici, Manager
Aging Management & License Renewal

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