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United States Nuclear Regulatory Commission
Washington, DC 20555-0001

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/RENEWED LICENSE NO. DPR-23

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE PRESSURIZED WATER REACTOR INTERNALS PROGRAM PLAN FOR AGING MANAGEMENT OF REACTOR INTERNALS (TAC NO. ME9633)

Ladies and Gentlemen:

The NRC requested that Duke Energy Progress, Inc., respond to two requests for additional information (RAI) regarding the Aging Management Program for the Reactor Vessel Internals at Robinson Nuclear Power Plant (RNP). The Duke Energy Progress response to these RAIs (RAI-3-1 and RAI-3-3)(NRC ADAMS Accession Numbers.: ML14014A097, ML13240A499 and ML14167A345) is provided in the enclosures to this letter. A copy of Westinghouse Technical Bulletin (TB), Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation (TB-14-5), is also provided in the enclosure to this letter. Based upon communications from the NRC, dated September 3, 2014, the Staff did not require inclusion of a response to Item 2 of RAI-3-1, for this submittal.

On a clarification call with the NRC on August 28, 2014, related to RAI-3-3, the Staff concurred with the conclusion of the above Westinghouse Technical Bulletin that degradation of the Clevis Insert Bolts poses no nuclear safety or operational concern. The Staff further stated that the following items are deemed to constitute a complete response to RAI-3-3:

1. Summary of the latest inspection results on Clevis Insert Bolts

2. Revision of the MRP-227-A inspection requirements to integrate an enhanced visual inspection to be utilized in the next 10-year ASME Section XI, ISI examinations of the Reactor Vessel Internals

In addition to the existing ASME Section XI, ISI required Visual Test (VT-3), RNP will perform an Enhanced Visual Test (EVT-1), which meets and exceeds VT-3 requirements, in the next 10-year ASME Section XI, ISI examinations (i.e., Fifth 10-year ISI examinations). As it was stated above, although degradation of the Clevis Insert Bolts do not pose a nuclear safety or operational concern, RNP has begun

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NRR

performing an impact assessment review arising from the above Technical Bulletin. This effort will give rise to development of an asset management program that would mitigate degradation of the Clevis Insert Bolts and associated reactor internals (i.e., Clevis Insert accessible welds, bolts, locking bars, and dowel pins) reflected in the subject Technical Bulletin.

There are no regulatory commitments made in this submittal. If you have any questions regarding this submittal, please contact Mr. R. Hightower at (843) 857-1329.

I declare under penalty of perjury that the foregoing is true and correct.

Executed On: 9/5/14

Sincerely,

William. R. Gideon
Site Vice President

WRG/am

Enclosure 1: RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE PRESSURIZED WATER REACTOR INTERNALS PROGRAM PLAN FOR AGING MANAGEMENT OF REACTOR INTERNALS

H. B. Robinson Unit 2 Summary Report for the Cold Work Assessment

Enclosure 2: RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE PRESSURIZED WATER REACTOR INTERNALS PROGRAM PLAN FOR AGING MANAGEMENT OF REACTOR INTERNALS

H. B. Robinson Unit 2 Summary Of Inspection Results for the Clevis Insert Bolts

Enclosure 3: RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE PRESSURIZED WATER REACTOR INTERNALS PROGRAM PLAN FOR AGING MANAGEMENT OF REACTOR INTERNALS

Westinghouse Technical Bulletin (TB), Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation (TB-14-5)

cc: Mr. V. M. McCree, NRC, Region II
Ms. Martha Barillas, NRC Project Manager, NRR
NRC Resident Inspector, HBRSEP Unit No. 2

ENCLOSURE 1

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE PRESSURIZED WATER REACTOR INTERNALS PROGRAM PLAN FOR AGING MANAGEMENT OF REACTOR INTERNALS (TAC NO. ME9633) DOCKET NO. 50-261

H. B. Robinson Unit 2 Summary Report for the Cold Work Assessment

In Request for Additional Information RAI-3-1 [1 and 5], the U.S. Nuclear Regulatory Commission (NRC) advised Duke Energy that resolution of Applicant/Licensee Action Item (A/LAI) 1 of MRP-227-A would need to be resolved as part of the staff's review of the H. B. Robinson License Renewal Amendment and Pressurized Water Reactor (PWR) Vessel Internals Program.

> "RAI-3-1 (Action Item 1 of the NRC staff's SE for MRP-227-A):
>
> *Do the RNP RVI have non-weld or bolting austenitic stainless steel components with 20% cold work or greater, and if so do the affected components have operating stresses greater than 30 ksi? If so, perform a plant-specific evaluation to determine the aging management requirements for the affected components."*

Westinghouse has evaluated the H. B. Robinson reactor internals components according to industry guideline MRP 2013-025 [2], as well as the MRP-191 [3] industry generic component listings and screening criteria (including consideration of cold work as defined in MRP-175 [4], noting the requirements of Section 3.2.3). In addition to consideration of the material fabrication, forming, and finishing process, a general screening definition of a resulting reduction in wall thickness of 20% was applied as an evaluation limit. It was confirmed that all H. B. Robinson Unit 2 components, as applicable for the design, are included directly in the MRP-191 component lists.

The evaluation included a review of all plant modifications affecting reactor internals and the plant operating history. The components were procured according to ASTM International or ASME material specifications through applicable quality-controlled protocols. H. B. Robinson Unit 2 components were binned according to the following categories for material fabrication and cold work potential:

1. Cold work categories include the following:
    - cast austenitic stainless steel (CASS) (Category 1)
    - hot-formed austenitic stainless steel (Category 2)
    - annealed austenitic stainless steel (Category 3)
    - fasteners austenitic stainless steel (Category 4)
    - cold-formed austenitic stainless steel without subsequent solution annealing (Category 5)
2. Cold work potential was based on MRP-227-A generic criteria:
    - No (N) typically applies to cold work Categories 1, 2, and 3.
    - Yes (Y) typically applies to cold work Categories 4 and 5.

Where multiple options existed for a component or assembly, the bounding condition, taken as including cold work, was selected for the purpose of the assessment. In some instances cascading fabrication would appear to mitigate any potential for cold work; however, since the historical record was not detailed, the potential is noted, but a conservative approach was selected for this assessment.

The evaluation, consistent with industry guidelines, concluded that the reactor internals Categories 1, 2, and 3 (non-bolting) components at H. B. Robinson Unit 2 contain no cold work greater than 20% as a result of material specification and controlled fabrication construction. Category 4 components were already assumed to have the potential for cold work in the MRP-191 generic assessments. Material fabrication specifications used for H. B. Robinson Unit 2 would suggest that processes were limiting and precluded the introduction of significant cold work in some of the Categories 4 and 5 components. Category 4 components with greater than 20% cold work have been previously dispositioned in MRP-191 [3]. This disposition is accounted for in the aging management program; therefore, H. B. Robinson Unit 2 requires no additional stress analyses for Category 4 components. Following MRP 2013-025 guidelines [2], Category 5 components with greater than 20% cold work are to be analyzed for stresses. H. B. Robinson Unit 2 contains no Category 5 components with greater than 20% cold work; therefore, no stress analyses were performed. The detailed evaluation for the A/LAI for H. B. Robinson Unit 2 cold work assessments concluded both that the plant-specific material fabrication and design were consistent with the MRP-191 basis and that the MRP-227-A sampling inspection aging management requirements, as related to cold work, are directly applicable to H. B. Robinson Unit 2. Therefore, a plant-specific evaluation to determine the aging management for 20% or greater cold work components with operating stresses greater than 30 ksi is not needed for H. B. Robinson Unit 2.

References

1. Email from Siva Lingam (NRC) to Richard Hightower (PGN), "Robinson Unit 2 PWR Vessel Internal Program Plan for Aging Management – Requests for Additional Information (RAIs) (TAC No. ME9633) – Correction to RAI-3-3," September 5, 2013. (NRC ADAMS Accession No.: ML13249A000)

2. Materials Reliability Program Letter, MRP 2013-025, "MRP-227-A Applicability Template Guideline," October 14, 2013.

3. *Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191).* EPRI, Palo Alto, CA: 2006. 1013234.

4. *Materials Reliability Program: PWR Internals Material Aging Degradation Mechanism Screening and Threshold Values (MRP-175).* EPRI, Palo Alto, CA: 2005. 1012081.

5. Duke Energy Letter, "Response to NRC Request for Additional Information Related to the Pressurized Water Reactor Internals Program Plan for Aging Management of Reactor Internals (TAC No. ME9633)," January 9, 2014. (NRC ADAMS Accession No.: ML14014A097)

ENCLOSURE 2

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE PRESSURIZED WATER REACTOR INTERNALS PROGRAM PLAN FOR AGING MANAGEMENT OF REACTOR INTERNALS (TAC NO. ME9633) DOCKET NO. 50-261

H. B. Robinson Unit 2 Summary Of Inspection Results for the Clevis Insert Bolts

AREVA

HB Robinson - RO-27

Progress Energy

Component / Description	Azimuth	Exam Type	Cal #	Cleaning Assmt. Req'd.	Cleaning By:	Video File	Examiner	Level	Exam Results	Date/Time	Comments/Remarks
Reactor Vessel Lower Head (Group 4A)											
Clevis Insert Keyways (101/CIK)	N/A	VT-3		**No**	N/A		Dan Langenfeld (DJL)	**III**	**NRI**	2012-02-13 09:56:23	Performed a VT-3 of the Lower Clevis Insert Keyways and Lower Radial Support Lugs looking for debris, wear, deformation and abnormal conditions. (0, 90, 180 & 270 degrees). (SUM #533200) SECT-XI Non-Relevant indications such as minor scoring, scratches, gouges from typical lower internals assembly/re-assembly operations were noted. Previously noted indications were observed - no change. No Relevant Indications. Total Coverage 100%
		VT-3	70			Vid 0218 avi	Dennis Bryant (DNB)	**II**	**NRI**	2012-02-13 09:32:27	All four Lower Clevis Inserts/Radial Supports examined. Normal wear/scratches noted associated with lifting operations. Previously noted indications were observed - no change.
								Customer Review	NRI	2012-02-16 04:01:57	No abnormal conditions noted other than previously identified inconsequential wear on all four Keyways. MPD

Examined By: Victor Morton (VM) Level: III Date: 2012-02-15
Examined By: Dennis Bryant (DNB) Level: II Date: 2012-02-15
Level III Review: Andrew Clay (AWC) Date: 2012-02-16
Level III Review: Dan Langenfeld (DJL) Date: 2012-02-13

ENCLOSURE 3

RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION RELATED TO THE PRESSURIZED WATER REACTOR INTERNALS PROGRAM PLAN FOR AGING MANAGEMENT OF REACTOR INTERNALS (TAC NO. ME9633) DOCKET NO. 50-261

Westinghouse Technical Bulletin (TB), Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation (TB-14-5)

Westinghouse

Technical Bulletin

An advisory of a recent technical development pertaining to the installation or operation of Westinghouse-supplied nuclear plant equipment. Recipients should evaluate the information and recommendation, and initiate action where appropriate.

1000 Westinghouse Drive, Cranberry Township, PA 16066

Subject: **Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation**	Number: **TB-14-5**
System(s): Reactor Coolant System, Reactor Internals, Lower Radial Support System	Date: 08/25/2014
Affects Safety-Related Equipment Yes ☐ No ☒	S.O.: NA

This Technical Bulletin supersedes InfoGram IG-10-1 (Reference 1).

BACKGROUND

Westinghouse IG-10-1 was issued to communicate an operating experience (OE) related to clevis insert cap screw (bolt) degradation. Since the issuance of IG-10-1, additional information has become available related to the root cause of the OE that was previously communicated. This Technical Bulletin (TB) provides a summary of the OE as well as root cause findings and the applicability of these findings on Westinghouse and Combustion Engineering (CE) pressurized water reactor designs. This TB also reviews the safety implications of the OE and root cause analysis results as well as inspection recommendations for licensees to consider including as part of their aging management program to address this OE.

During a 10-year in-service inspection (ISI) in March 2010, an anomaly was observed in the lower radial support clevis inserts. The 10-year ISI program includes a remotely operated visual inspection of the vessel after the core barrel is removed. This visual inspection examines the condition of the six clevis inserts, including general integrity of bolted and welded connections. Figure 1 provides an overview of the radial support clevis insert configuration. Each clevis insert at the plant inspected features 8 cap screws and 2 dowel pins which provide for retention of the insert. At a total of 7 (out of 48) bolt locations, there was visual evidence that the bolt head had detached from the shank. On one of the clevis dowel pins, the lock welds on the pin had broken and the pin was slightly rotated and displaced deeper into the insert. Reviews of video images from the previous 10-year ISI vessel inspection showed no indications of wear, fractures, or other anomalies with the clevis insert cap screws or dowel pins at any location.

Additional information, if required, may be obtained from Bryan Wilson, (412) 374-3281

Author:	Reviewer	Manager:
William J. Smoody Regulatory Compliance	John T. Crane Regulatory Compliance	James A. Gresham Regulatory Compliance
Verifier: Bryan M. Wilson Reactor Internals Design and Analysis I	Verifier: Michael A. Burke, PhD Primary Systems Design and Repair	

BACKGROUND (Continued)

During the spring 2013 refueling outage, 29 bolts were removed and replaced. The primary concern was that the clevis insert bolt would continue to degrade eventually leading to a condition where the clevis insert would be able to completely dislodge once the lower internals were removed. When the internals are assembled into the reactor vessel the clevis insert is effectively trapped by the interfacing lower internals components. Once the lower internals are removed, and if there is significant degradation of the designed fits and hardware, there is a potential for the insert to be dislodged from the mating vessel lug. Once dislodged from the vessel lug, reinstallation was viewed as being of high commercial consequence due to the customization needed to reestablish the required design gaps for this component. The number of replacement bolts was determined based on structural evaluations of the bolts assuming that none of the original bolts were functional. Evaluations (discussed later) showed that no bolts were required to maintain the intended safety function. The evaluations, to determine the number of required replacement bolts, were primarily focused on ensuring that the bolts would remain functional for the remaining life of the plant.

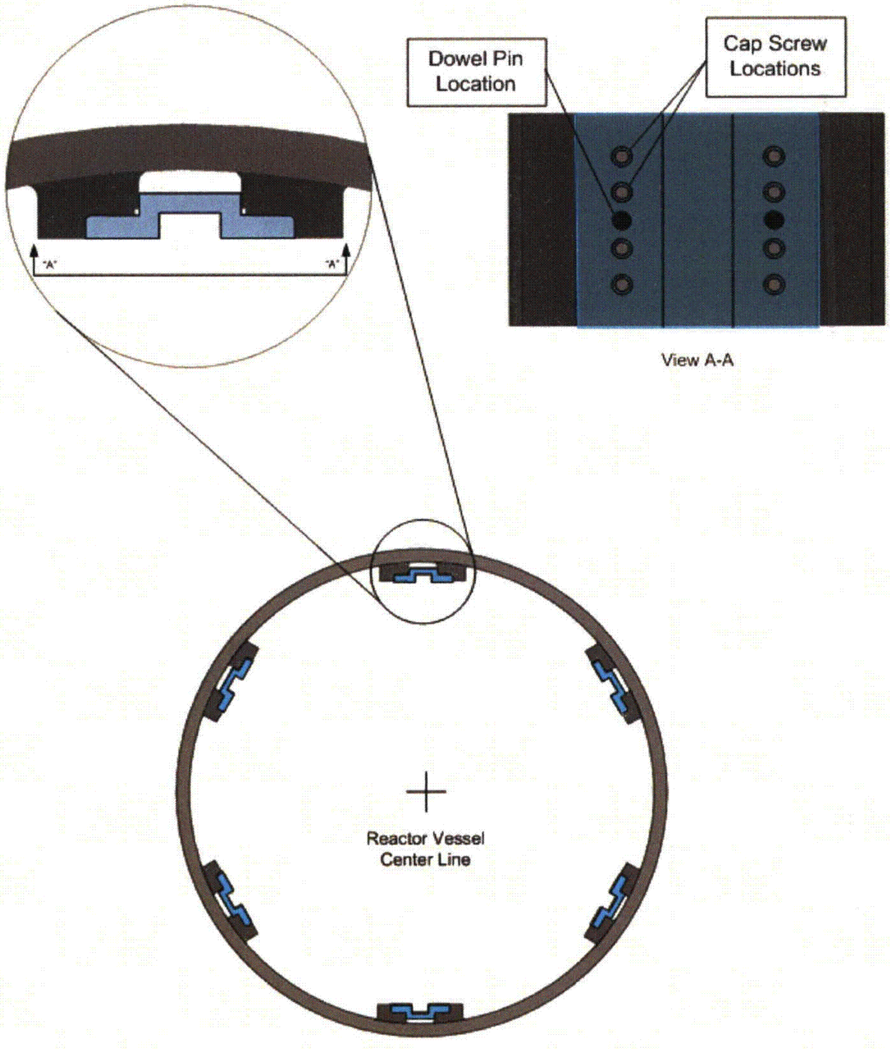

Figure 1 - Example of Lower Radial Support Clevis Insert, Cap Screw and Dowel Pin Locations

ROOT CAUSE ANALYSIS RESULTS SUMMARY

The licensee's root cause analysis identifies the cause of the clevis insert bolt failures to be primary water stress corrosion cracking (PWSCC) due to the use of Alloy X-750 with a susceptible heat treatment. Additional details of particular interest are as follows:

- Of the 29 bolts removed, 13 were originally considered intact (later found to be cracked) and 16 were fractured.
- Of the 16 bolts found to be fractured, only 8 showed signs of wear between the head and the lock bar during visual inspections prior to removal.
- All 29 bolts examined contained cracking in the head-to-shank transition.
- Fractographic scanning electron microscopy (SEM) analysis and cross section metallographic examinations confirmed the fracture mode was essentially 100% intergranular on all of the bolts.
- There was no evidence that the bolts failed due to fatigue cracking or mechanical overload.

While the conclusion of PWSCC as the failure cause is consistent with the supposition made in Reference 1, the new information related to the extent of cracking suggests that clevis insert bolt cracking is potentially a latent issue which has not yet manifested itself as visual indications at other plants. This new information also confirms that visual inspection is less than adequate in detecting clevis insert bolt failures. However, as discussed herein, the lower radial support structure will maintain its intended safety function despite bolt failures.

PLANT APPLICABILITY

This issue potentially impacts all Westinghouse- and CE-designed plants. All Westinghouse and CE plant designs have been confirmed to contain a similar susceptible heat treatment of Alloy X-750 to that used in the subject cap screws. Furthermore, preload stresses in the Alloy X-750 clevis insert bolts (or snubber lug bolts in the case of the CE design) have generally been found to be consistent across the fleet. While there are differences among the designs used across the Westinghouse and CE fleets, these differences are not considered to have a significant impact on susceptibility.

SAFETY IMPLICATIONS

In support of an operability assessment, Westinghouse performed engineering evaluations of the as-found condition, including an evaluation of the potential for loose parts. The loose parts evaluation concluded that the separated cap screw heads will remain captured in the clevis insert counterbores and will not impact operation. However, lock bars at the degraded cap screw locations were observed to have experienced wear-related degradation; therefore, the potential for loose parts from the lock bars to affect other locations in the reactor vessel was evaluated. Westinghouse concluded that no degradation of mechanical components is expected as a result of the potential presence of loose parts from the lock bars in the primary system.

Westinghouse also performed a detailed structural evaluation of the as-found condition of the clevis inserts. The analysis demonstrated that failure of the bolts would not result in a loss of safety function. Furthermore, it was demonstrated that a significant number of redundancies prevent the loss of the intended safety function.

Similar evaluations to those performed for the subject clevis insert design have been performed for the other three basic Westinghouse clevis insert designs. These evaluations also considered the results of the root cause analysis by assuming all bolts on the same insert were non-functional. The conclusions resulting from these evaluations were consistent with the conclusions reached for the evaluations performed for the plant, which are:

- Failure of the bolts would not result in a loss of safety function
- Significant number of redundancies prevent the loss of the intended safety function
- Concerns related to bolt failures are generally non-safety in nature

No formal evaluations have been conducted for the CE snubber lug designs, which are in a similar location in the reactor as the Westinghouse lower radial support, have the same intended safety function, and also use Alloy X-750 attachment bolts. However, based on a comparison of the design features, installation, and loading, it is concluded that the snubber lug will be able to perform its intended safety function in the event of bolt failures. Furthermore, the snubber lug bolt heads would remain trapped and would not become loose parts.

MAINTENANCE IMPLICATIONS

While clevis insert bolt failures do not present a safety concern, continued operation with degraded clevis insert hardware increases the likelihood and consequence of maintenance risks due to the complexity and cost of repair, and the required level of contingency planning. The primary risks are related to significant loosening and disengagement of the clevis insert, which could result in difficulty or inability to remove the core barrel or need to replace the insert to reestablish customized design gaps. Either of these would have a significant outage impact. As previously mentioned, maintenance considerations were one of the primary considerations which led to replacement of bolts. Other potential implications of operation with failed clevis insert bolts include the following:

- Bolt remnants wearing on clevis insert bolt counterbores, seating surfaces, or mating threads could lead to additional scope for reconstituting these surfaces prior to bolt replacement.
- Bolt head remnants wearing into adjacent surfaces of the radial key could lead to wedging of the bolt head during barrel removal.
- Worn lock bars releasing from the insert and becoming loose parts has been evaluated and shown to not be a safety concern. However, as the length of operation with failed clevis insert bolts increases, the probability of lock bars releasing from the insert increases. Furthermore, the potential for these lock bars to tumble, break, or continue to wear into smaller fragments within the internals also increases. If they become small enough in size, they can potentially enter the fuel and lead to cladding wear, potentially including wear-through (leaking fuel) or foreign object search and retrieval (FOSAR) activities.
- Increased time for bolt replacement if a significant number of bolts are broken. It generally requires much less effort to remove intact bolts versus separated bolts.

Maintenance risks should be considered in evaluating the inspection methods used for managing the aging of clevis insert bolts.

INSPECTION RECOMMENDATIONS

The U.S. nuclear industry–through the Materials Reliability Program (MRP)–has developed Inspection and Evaluation (I&E) guidelines (MRP-227-A; Reference 2) for the management of reactor internals age-related degradation issues in the U.S. pressurized water reactor (PWR) fleet. MRP-227-A focuses on those internals components identified as susceptible to aging effects and provides information to support effective aging management, while simultaneously maintaining safety and reliability. The inspection and evaluation recommendations in MRP-227-A were issued under the Nuclear Energy Institute (NEI) NEI 03-08 (Reference 3) protocol for industry wide implementation.

Westinghouse-Designed Plants:

In MRP-227-A, the clevis insert bolts for the Westinghouse-designed plants are classified as an Existing Programs component managed for loss of material (wear). To manage this mechanism, MRP-227-A recommends the use of the existing ASME Section XI visual inspection (VT3) of all accessible surfaces. While this broadly covers the inspection scope and remains the recommended approach to aging management of the clevis insert bolts, there are some clarifications needed on scope of inspection to ensure that aging management inspection programs are focused on monitoring conditions that are directly related to the safety function of the component.

As Note 2 of Table 4-5 of MRP-227-A indicates, the bolts were screened-in because of stress relaxation and associated cracking. However, the failures of the bolts alone do not result in a loss of the intended safety function of the lower radial supports. The loss of function of the lower radial supports is linked more closely to increased motion of the lower end of the core barrel caused by wear on the clevis insert or vessel attachment interface. Therefore, although it is one of many barriers to loosening of the insert, management of the bolt failures does not directly address management of the functional performance of the component. To ensure that the functional performance is being managed, it is recommended to ensure that the MRP-227-A inspections (ASME Section XI, VT3) for the clevis inserts include the following scope:

- Radial key/clevis insert interfacing surfaces; look for aggressive or abnormal wear as compared to previous inspection, if available. The radial key does not make contact with the full length of the clevis insert; if wear is significant it would be visible as a step located toward the bottom end of the clevis insert as shown in Figure 2.
- Interface between the clevis insert and vessel lug (see Figure 3); look for signs of looseness or dislocation. Faces of the insert and vessel lug are generally flush; dislocations may be visible by the insert protruding toward the vessel centerline as compared to the vessel lug.

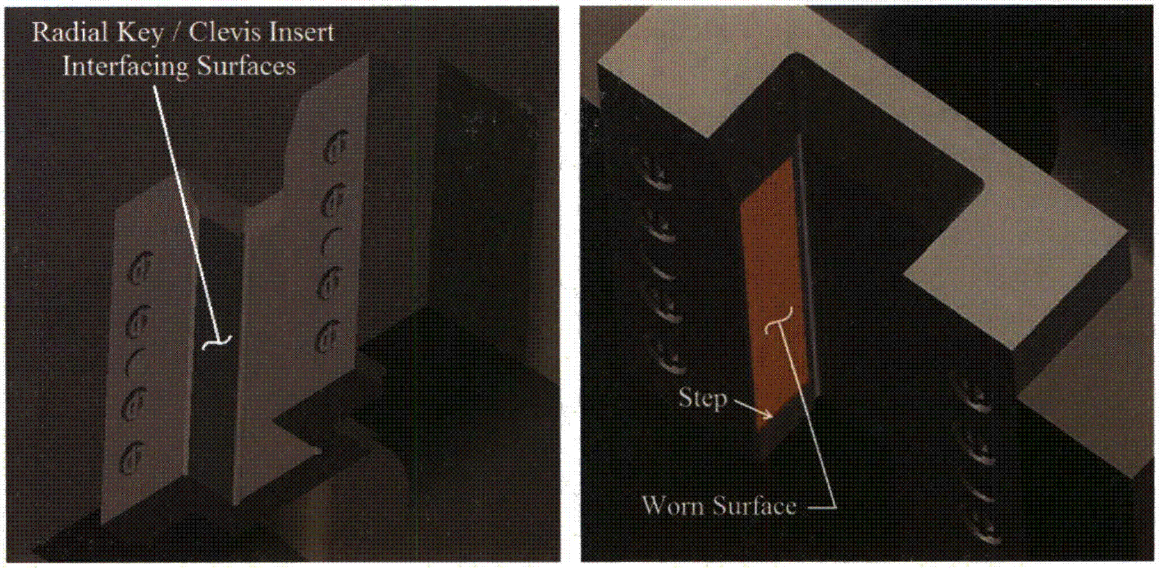

Figure 2 - Example of Wear at the Radial Key/Clevis Insert Interfacing Surface

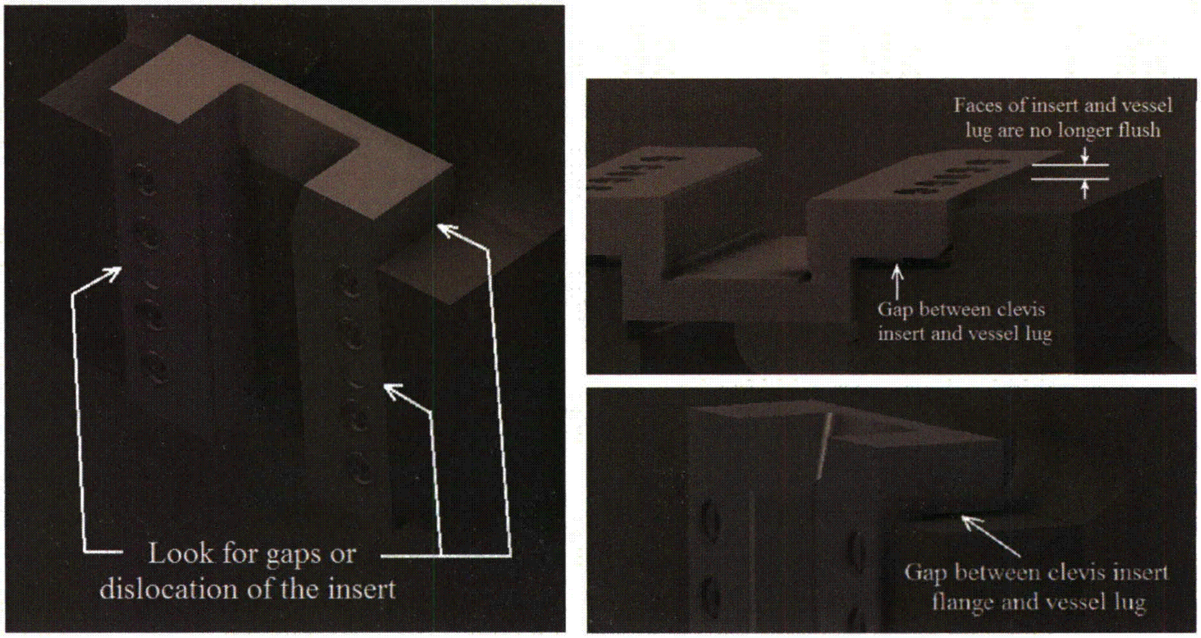

Figure 3 - Example of Dislocation of the Clevis Insert from the Vessel Lug

While managing the functional performance is the primary goal of the recommended inspections, this TB recommends that the scope of the inspection also include the following as a means of reducing maintenance risk associated with clevis insert bolt degradation (Figure 4):

- Look for wear between the bolt head and lock bar and/or bolt head dislocation.
- Look for broken tack welds and dislocation of the dowel pin.

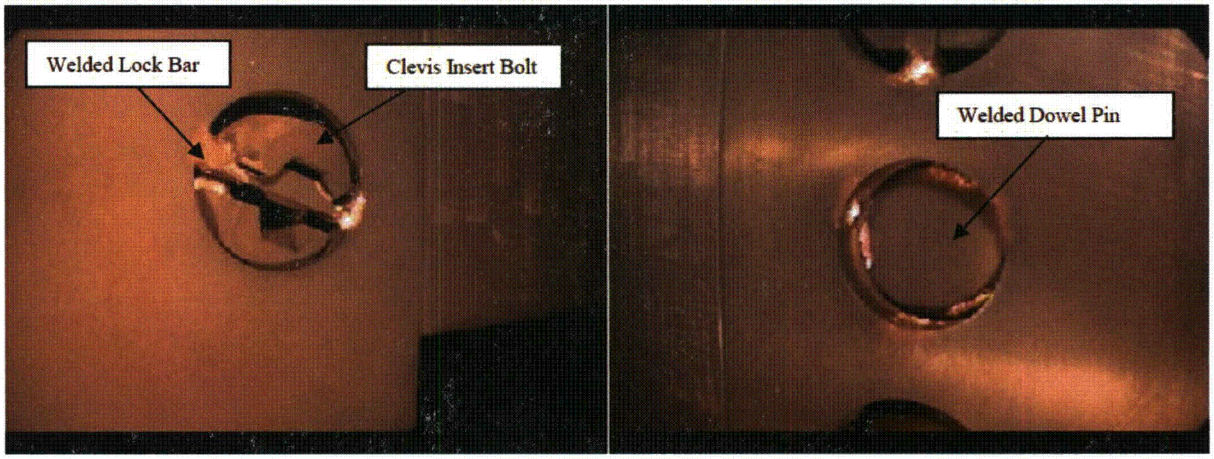

Figure 4 - Examples of Bolt and Dowel Pin Related Degradation

While these methods have been shown to be less than adequate in detecting bolt failure, the indications described have been shown in one case to manifest themselves as a result of bolt failure and prior to any notable loosening of the clevis insert or increased wear between the radial key and clevis insert.

Some optional methods that could be used to improve the reliability of crack detection and enhance the management of maintenance risk include activities such as ultrasonic examination of bolts and/or bolt replacement.

CE-Designed Plants:

For the CE-designed plants, the components similar to the clevis insert bolts are the core stabilizing lug bolts. Neither these bolts, nor the core stabilizing lug shims (CE components similar to the Westinghouse clevis inserts) were screened in as part of MRP-191 (Reference 4). As a result, no inspection recommendations are provided for these components in MRP-227-A. However, considering these bolts are Alloy X-750, installed with similar preload stress, and at similar operating temperatures as the Westinghouse clevis insert bolts, these bolts are also potentially susceptible to PWSCC. Since the function of core stabilizing lugs is identical to that of the Westinghouse lower radial supports, similar inspections for looseness of the insert should be conducted to ensure function is maintained. Specific recommendations for inspection scope are as follows:

- Core support barrel snubber lug/core stabilizing lug shim interfacing surfaces (see Figure 5); look for aggressive or abnormal wear as compared to previous inspection, if available. The core support barrel snubber lug key does not make contact with the full depth of the core stabilizing lug shim. If wear is significant it would be visible as a step located toward the vessel side of the clevis insert as shown in Figure 5.
- Interface between the core stabilizing lug shim and the core stabilizing lug (see Figure 6); look for signs of looseness or dislocation. Looseness may be visible by gapping between the core stabilizing lug shim and the core stabilizing lug when looking radially outward toward the lug.

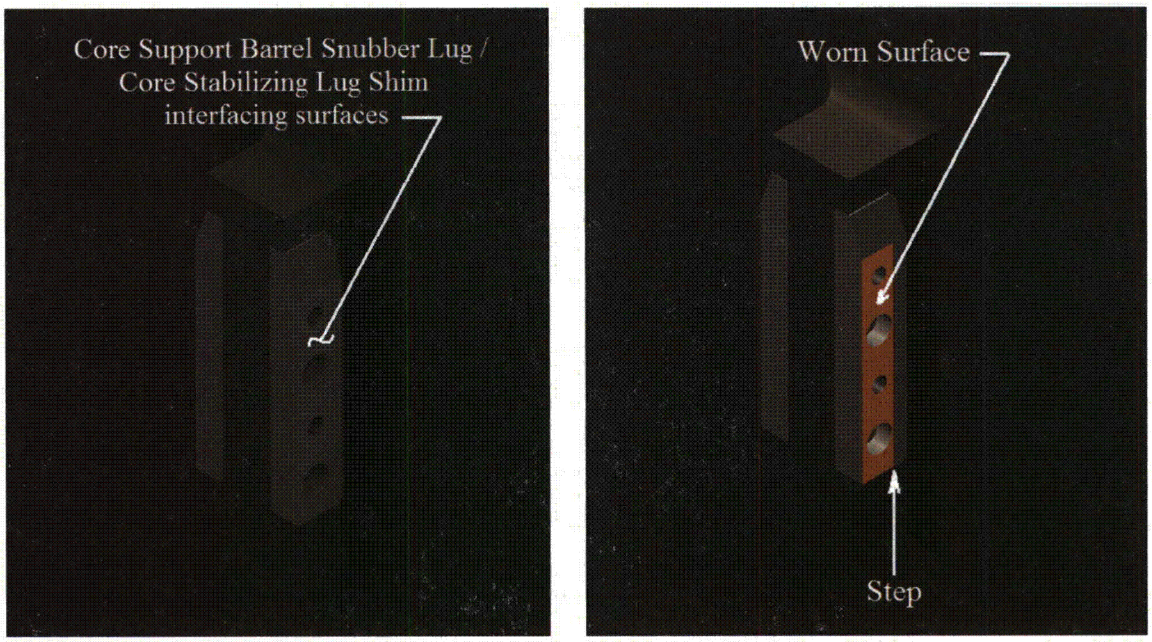

Figure 5 - Example of Wear at the Core Support Barrel Snubber Lug/Core Stabilizing Lug Shim Interfacing Surface

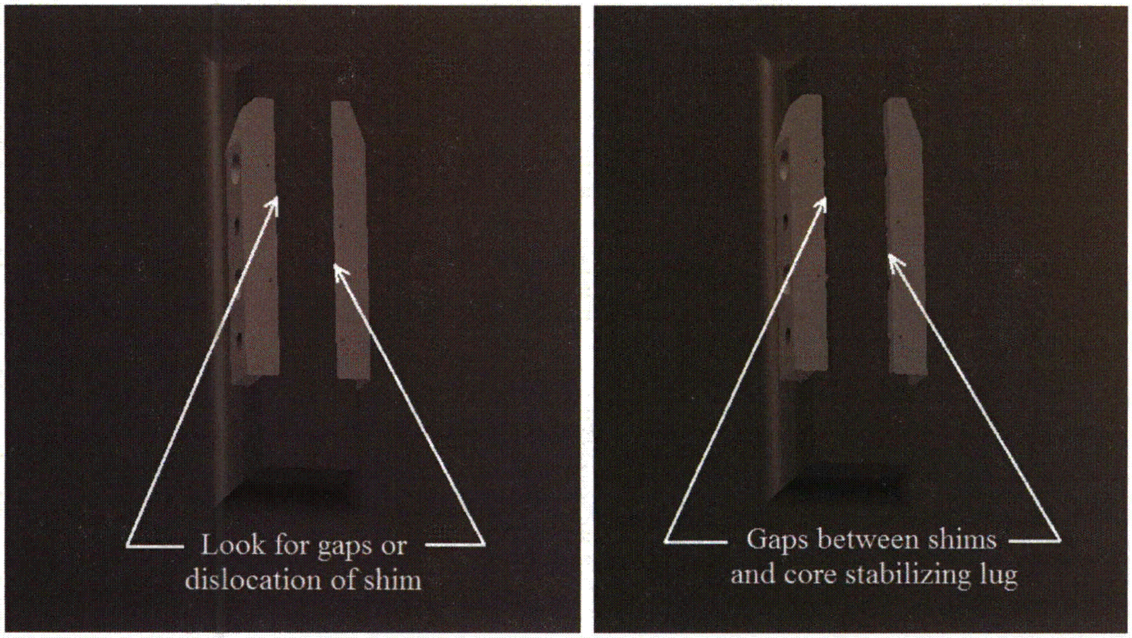

Figure 6 - Example of Dislocation of the Shim from the Core Stabilizing Lug

While managing the functional performance is the primary goal of the recommended inspections, it is recommended that the scope of the inspection also include the following as a means of reducing maintenance risk associated with clevis insert bolt degradation (Figure 7):

- Bolt heads; look for wear between the bolt head and the shim and/or bolt head dislocation.
- Bolt threaded end; look for signs of the bolt threads backing. Bolt threads should generally be recessed from the face of the shim; look for differences in the amount of recess between the two bolts threaded into the shim being viewed.
- Locking pins; look for dislocation, wear, or failure of the locking pin.

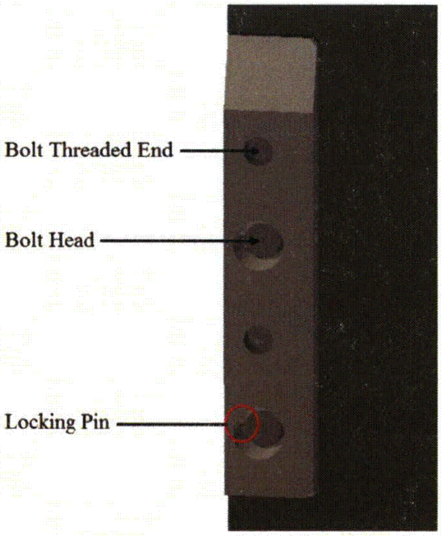

Figure 7 - Identification of Hardware

As previously discussed, these methods may be less than adequate in detecting bolt failure, however they may manifest themselves as a result of bolt failure and prior to any notable loosening of the core stabilizing lug shim or increased wear between the core support barrel snubber lug and core stabilizing lug shim.

Similar to the recommendations for the Westinghouse-designed plants, optional methods such as ultrasonic examination of bolts or bolt replacement could be used to improve the reliability of crack detection and enhance the management of economic risk.

CONCLUSION

Other than the single occurrence discussed, Westinghouse is unaware of any other occurrences of clevis insert bolt degradation. Based on the engineering evaluations performed to date and considering information made available through the investigations conducted on removed failed bolts, there are no safety or operability concerns to be communicated to the industry. The recommendations for inspections of clevis insert bolt-related degradation to enhance focus on the conditions that may result from clevis insert bolt failures, is based on available OE. Westinghouse will continue to monitor the results of clevis insert bolt inspections conducted throughout the industry and will communicate to the industry any relevant OE which changes the recommendations provided herein.

REFERENCES

1. Westinghouse InfoGram, IG-10-1, Revision 0, "Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation," March 31, 2010.
2. "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," EPRI, Palo Alto, CA: 2011. 1022863.
3. Nuclear Energy Institute, NEI 03-08, Rev. 2, "Guideline for the Management of Materials Issues," January 2010
4. "Materials Reliability Program: Screening, Categorization, and Ranking of Reactor Internals Components for Westinghouse and Combustion Engineering PWR Design (MRP-191)," EPRI, Palo Alto, CA: 2006. 1013234.