Attachment 2
Westinghouse Final Response to U.S. NRC RAI 3.1.2.1-6 (a) and (b) on the Callaway Nuclear Plant Reactor Lower Radial Support Clevis Insert Bolts (Non-Proprietary)

NRC RAI 3.1.2.1-6 (a) and (b):

Background:

By letter dated October 24, 2012, Union Electric Company (Ameren Missouri) provided License Renewal Application (LRA) Amendment No. 13 which revised LRA Table 3.1.2-1, "Reactor Vessel, Internals, and Reactor Coolant System – Summary of Aging Management Evaluation – Reactor Vessel and Internals," to provide an amended aging management review (AMR) for the reactor vessel internals clevis insert bolts. In this amended AMR, the applicant confirmed that the clevis insert bolts are fabricated from nickel alloy materials and that potential cracking and loss of material due to wear will be managed by the applicant's American Society of Mechanical Engineers (ASME) Section XI Inservice Inspection (ISI), Subsections IWB, IWC, and IWD Program.

Issue:

Appendix A to Electric Power Research Institute (EPRI) Technical Report No. 1022863, "Materials Reliability Program: Pressured Water Reactor (PWR) Internals Inspection and Evaluation Guidelines (MRP-227-A)," (Reference 1) indicates that failures of nickel alloy (Alloy X-750) clevis insert bolts were reported by the licensee for one domestic Westinghouse-designed pressurized water reactor in 2010. The ASME Section XI visual VT-3 inspections of the clevis insert assemblies on a 10-year frequency may not be adequate to ensure the integrity of clevis insert assemblies during design basis events if multiple bolt failures occur prior to detection and a design basis event occurs.

Request:

a) Describe the configuration of clevis insert assemblies at Callaway Plant, Unit 1 (Callaway) including number of bolts in the assemblies. Specify the material of fabrication, including any applicable heat treatment, that were used for the design of the clevis insert bolts at Callaway.

b) Discuss and justify whether the operating experience associated with cracking of the clevis insert bolts is applicable to clevis insert assembly designs at Callaway.

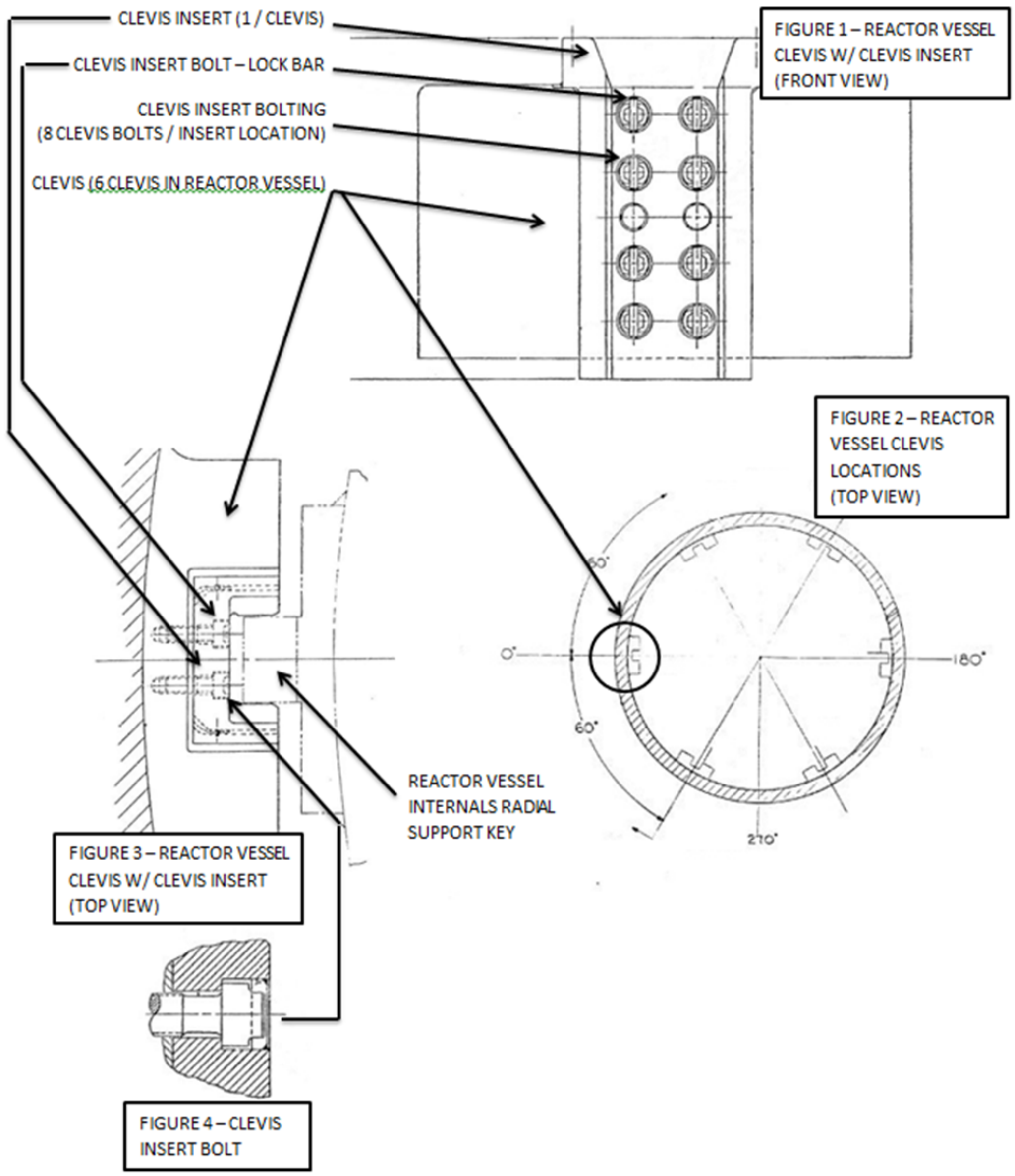
Response:

a) The clevis insert assemblies at Callaway are comprised of eight clevis insert cap screws fabricated from Inconel X-750, 2 dowel pins fabricated from Alloy 600 and an Alloy 600 hard-faced insert assembly that rests onto the clevis locations in the reactor vessel. In total there are six clevis locations (0, 60, 120, 180, 240, 300 degree locations). The clevis insert cap screws include a lock bar that is welded to the clevis insert face. The clevis insert cap screws are attached to the interior face of the clevis insert assembly.

The heat treatment sequence utilized for the Callaway clevis insert cap screws was:

- [ ]^{c}
- [ ]^{c}

The Callaway material used for the clevis insert cap screws is the same as used at the reference plant where cracking has been observed. The clevis insert design however is different than the reference plant in that the cap screw locations are included on the interior face of the clevis (parallel to the outside face of the radial insert key with the core barrel installed). This geometry is further defined in Figures 1-4 identified below.



b) The main function of the Lower Radial Support System (LRSS) is to prevent tangential or rotational motion of the lower internals assembly while permitting axial displacement and differential radial expansion. Callaway has six radial supports spaced at 60-degree intervals around the circumference of the vessel. Although labeled as radial supports, the supports actually support the core barrel only in the tangential direction because the tangential clearances between the core barrel keys and the vessel clevis inserts are much smaller than the radial clearances. This basic arrangement is the same for Callaway and the reference plant where clevis insert cap screw cracking was observed; however, the clevis insert designs are different.

The same number of eight cap screws is arranged in the same two vertical columns of four cap screws each. Two interference-fit dowel pins of the same size are located in-line with the cap screws in the same manner as the reference plant. The main design difference is that the Callaway reactor clevis insert is U-shaped, with the cap screws located inboard of the "U"; whereas the reference plant insert, while also being U-shaped, has flanges on either side where the cap screws are located. The tangential interference fit of the insert against the support lug is at the ends of these flanges for the reference plant design and on the sides of the "U" for the Callaway reactor design. Therefore, the tangential interference-fit compression stiffness of the two inserts is different.

Because of the small tangential clearance between the radial keys and the clevis insert, the keys are potentially subjected to flow-induced vibration loads and wear at the key-to-keyway (clevis) interface. These supports are designed to prevent excessive lateral and rotational displacement of the lower internals during seismic and loss-of-coolant accident (LOCA) conditions. The supports also limit displacements and misalignments in order to avoid overstressing the core barrel and to ensure that the control rods can be freely inserted. Therefore, assuming the clevis inserts remain in place as limited by the adjoining radial keys and support lugs, the design function of the LRSS will be maintained during seismic and LOCA conditions. Because the clevis insert cap screws for the Callaway units are of a similar design (except for the [ ]^{c} longer shank), of the same material, torqued to the same degree, and operated at a slightly hotter Tcold inlet temperatures as compared to the reference plant, it is possible that these cap screws can eventually crack in a manner similar to that of the reference plant in Reference 2. Therefore, the operating experience discussed above is applicable to Callaway. A recent root cause analysis summarizing metallurgical investigations of the degraded cap screws from the reference plant confirmed that primary water stress corrosion cracking (PWSCC) was the failure mechanism.

As discussed in Reference 2, structural evaluations performed to justify continued operation in the as-found condition demonstrated safe operation was acceptable for an additional fuel cycle. The only concern was possible long-term effects, such as the potential for vibratory loads to eventually cause loosening and wear of the insert and the subsequent increase in gaps between the insert, radial key, and support lug. For this evaluation, due to the difference in design of the clevis insert, a similar review of the structural adequacy of the Callaway clevis insert design was performed to determine if broken cap screws present a structural concern for safe operation. The structural aspects and loose parts assessment, as performed for Reference 2, are discussed in the following paragraphs.

Clevis Support Lug Primary Stress

The clevis insert, if completely loose to slide radially inward, is captured in a manner similar to the reference plant and is restrained by a similar radial gap before it contacts the radial key. This condition would require the two interference-fit dowel pins to also lose restraint. With the clevis insert displaced fully inward, the primary stresses on the clevis support lugs remain acceptable relative to the reactor vessel original ASME, 1971 Edition (with 1972 Winter Addenda), code of construction under plant-specific maximum

upset and faulted condition loads due to seismic and LOCA conditions. These loads include the maximum impact loads that occur against the clevis inserts.

Clevis Insert Primary Plus Secondary Stress
The bending stress of the insert is maximized if it is assumed that one entire column of cap screws is broken and the other column of screws is intact. This forces the loose side of the insert to expand and contract to a greater extent relative to the support lug. With the maximum resulting interference during heatup and maximum tangential and radial loadings during steady state and cooldown, when a small clearance can exist, the resulting stress range remains within the primary plus secondary stress range analyzed in the generic analysis of record for this clevis insert design. Therefore, the increase in insert stress due to broken cap screws remains acceptable.

Cap Screw Primary Plus Secondary Stress
This scenario uses the same cap screw arrangement as discussed above where one column of cap screws is entirely broken. In this case, during cooldown and potentially during steady-state, when the insert is not tangentially preloaded against the support lug, the entire applied radial load on the insert is reacted by the intact cap screws. The resulting cap screw stress produced by this prying load on the insert is acceptable with four intact screws. However, with three or less intact screws, the allowable stress intensity can be potentially exceeded. During heatup, the clevis insert remains preloaded against the support lug, and this type of loading on the intact screws will not occur.

Clevis Insert Restraining Force (No Cap Screws)
If all of the cap screws are broken, and no restraint by the dowel pins is assumed, it is possible that the clevis insert can lose preload during steady state operation but it will still maintain its ability to perform its intended function. The clevis insert is restrained tangentially by the support lugs and restrained radially by the limited clearance with the radial key. In addition, the insert has a thick upper flange that prevents it from falling downward, and the downward force from the downcomer flow will prevent it from working upward. Operating experience with damaged bolts and one dowel pin, as described in Reference 2, showed no discernible change in the clevis insert wear surfaces after operation for two additional cycles. It is fully expected that with the design installed at Callaway, longer operation can be maintained before discernable degradation occurs.

During core barrel removal at cold conditions, the interference fit of the insert provides greater frictional force than the applied frictional force produced by the key sliding upward against the insert. The two dowel pins will also provide additional vertical constraint of the insert. Therefore the clevis insert design prevents separation of the insert during core barrel removal operations if the cap screws (and dowel pins) are non-functional.

Loose Parts Assessment
As discussed above, loss of the insert itself will not occur. Although over time, it may slowly displace radially inward toward the core barrel key, it will not move any further. The remaining engagement of the insert in the support lug will maintain adequate support of the core barrel against any normal, upset, or faulted condition loads. The insert cap screws have the same head design and locking device design as the reference plant. A lock bar is installed in a groove in the cap screw head and the bar is welded to the insert

counterbore where the cap screw is inserted. If a cap screw head should separate, the lock bar can, over time, wear and separate, causing the cap screw head to be loose in the counterbore recess. The as-built radial gaps measured between the core barrel radial keys and the inserts are all less than the height of the cap screw heads. Therefore, the cap screw heads remain captured, unless over a long period of time, wear of the heads reduces the height of the heads by this amount. The cap screw head wear is expected to be small because the cap screw material is much harder than the clevis insert and radial key material.

Evaluations were performed on the potential for loose parts with failed clevis insert cap screws for the reference plant (Reference 2). Lock bars at the degraded cap screw locations have experienced wear-related degradation; therefore, the potential for loose parts from the lock bars to affect other locations in the reactor vessel was also evaluated. Callaway and the reference plant have similar lower internals design so the effects of where these loose parts would be captured or would impact against the lower internals are the same. Therefore, no significant degradation of mechanical components is expected as a result of the potential presence of loose parts from the lock bars in the primary system.

References

1. *Materials Reliability Program: Pressurized Water Reactors Internals Inspection and Evaluation Guidelines, (MRP-227-A).* EPRI, Palo Alto, CA: 2011. 1022863.
2. Westinghouse InfoGram, IG-10-1, "Reactor Internals Lower Radial Support Clevis Insert Cap Screw Degradation," March 31, 2010.