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MRP Materials Reliability Program ____________________ MRP 2018-033

Date: September 5, 2018

PROJ 0689

To: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-001

From: Mike Hoehn II, Ameren-Missouri, MRP Research Integration Committee Chairman
Brian Burgos, EPRI, MRP Program Manager

Subject: Transmittal of NEI-03-08 "Needed" Interim Guidance for PWR CRDM Thermal Sleeve Wear

This letter transmits for NRC information recent NEI 03-08 Interim Guidance regarding the inspection of control rod drive mechanism (CRDM) thermal sleeves in PWR plants. As discussed during the 5/23/2018 public meeting with NRC staff, this guidance was developed by industry in response to recent operating experiences in 2017. This guidance provides immediate inspection for affected PWR plants as defined in Westinghouse's Nuclear Safety Advisory Letter (NSAL) 18-1. NRC recently published Information Notice (IN)-2018-10 on this topic to inform licensees of this operating experience.

EPRI notes that this is the first step in the development of broader guidance for the fleet in response to the recent operating experience. The joint industry team will continue its work on the development of additional guidance where appropriate, related to updates that are warranted associated with PWR Owners Group technical report PWROG-16003-P, as well as WCAP-17096-NP-A, and MRP-227-A.

If there are any questions or concerns, please contact Kyle Amberge, EPRI-MRP (kamberge@epri.com, 704-595-2039) or Chris Wax, APS (christopher.wax@aps.com, 623-393-6871).

Sincerely,

Mike Hoehn II, Ameren-Missouri MRP Research Integration Chair	Brian Burgos, Program Manager Materials Reliability Program

Enclosure: EPRI letter MRP 2018-027, dated 8/31/2018

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MRP Materials Reliability Program MRP 2018-027

Date: August 31, 2018

To: PMMP Members
MRP RIC Members

From: David M. Czufin, TVA, PMMP Chair
Brian Burgos, EPRI, MRP Program Manager

Subject: NEI 03-08 Needed Inspection Guidance for PWR CRDM Thermal Sleeve Wear

The purpose of this letter is to issue Interim Guidance regarding the inspection of reactor vessel closure head control rod drive mechanism thermal sleeve flanges. As discussed in a recent EPRI notification letter[1] to PWR plant sites, operating experience (OE) from international PWR plants related to wear of reactor vessel closure head control rod drive mechanism (CRDM) thermal sleeve flanges has resulted in control rod stoppage during plant restart operations after a refueling outage. Since the EPRI notification letter, Westinghouse has issued a 10CFR Part 21 report[2] and NSAL-18-1[3] with recommendations for the affected plants. NRC also issued Information Notice (IN) 2018-10 on this topic.

This Interim Guidance is "Needed" as defined in NEI 03-08[5]. The guidance was developed and approved by the joint MRP/PWROG utility members and PMMP-EC/PWROG-EC members to provide immediate guidance for the Table 1 and 2 plants as defined in the Westinghouse NSAL 18-1. This is the first step in the development of broader guidance[6-7] for the fleet in response to the recent operating experience.

Interim Guidance:

Based on the potential for this issue to result in a nuclear safety concern of non-functional control rods:

1. Units which are between 20 and 25 EFPY shall perform the dimensional measurements and/or visual inspections outlined in the "RECOMMENDED ACTIONS" section of NSAL-18-1 no later than the first refueling outage after 1/1/2019.
2. Units which have exceeded 25 EFPY shall perform the dimensional measurements and/or visual inspections outlined in the "RECOMMENDED ACTIONS" section of NSAL -18-1 during the next refueling outage after issuance of this interim guidance.

3. As an alternative, Thot units may implement the "RECOMMENDED ACTIONS" for Tcold plants, if desired.
4. Dimensional measurements and/or visual inspections performed prior to 1/1/2019 are acceptable.
5. There are no recommendations for units that have less than 20 EFPY on their original or replacement reactor vessel closure head.

International PWR units have identified significant wear and complete separation of thermal sleeve flanges[8] as early as 20 EFPY, thus, this recommendation for examination earlier than the 25 EFPY identified in the NSAL is considered appropriate.

If there are any questions or concerns related to this guidance, please contact the undersigned or Kyle Amberge (kamberge@epri.com or 704-595-2039).

Sincerely,

David M. Czufin, Tennessee Valley Authority
PMMP Chair

Brian Burgos, Program Manager
Materials Reliability Program

cc: J. Molkenthin-PWROG PMO

References:

1. EPRI letter MRP 2018-010, dated 4/20/2018, "Notification of Recent PWR CRDM Thermal Sleeve Flange Wear and Control Rod Motion Stoppage Operating Experience and Recommended Plant Actions"
2. WEC letter to NRC LTR-NRC-18-34, dated 5/23/2018 "Notification of the Potential Existence of Defects Pursuant to 10 CFR Part 21" (NRC ADAMS accession number ML18143B678)
3. WEC Nuclear Safety Advisory Letter (NSAL)-18-1, dated 7/9/2018 "Thermal Sleeve Flange Wear Leads to Stuck Control Rod"

4. NRC Information Notice (IN) 2018-10, dated 8/29/2018, entitled "Thermal Sleeve Flange Wear Leads to Stuck Control Rod at Foreign Nuclear Plant" (ADAMs Accession No. ML18214A710)
5. NEI-03-08 Revision 3, dated February 2017 "Guideline for the Management of Materials Issues"
6. Westinghouse Electric Company (WEC) Technical Bulletin (TB)-07-2, Revision 3, dated 12/7/2015, "Reactor Vessel Head Adapter Thermal Sleeve Wear"
7. Pressurized Water Reactor Owners Group Report, PWROG-16003-P, Rev. 1, "Evaluation of Potential Thermal Sleeve Flange Wear," August 2017. (Westinghouse Proprietary)
8. World Association of Nuclear Operators (WANO) Operating Event Paris Centre Report No. 2018-0110, dated 3/1/2018 "Blockage of a Control Rod due to Wear in the Thermal Sleeves of Vessel Heads in the 1300MW and N4 Series Fleet Belleville 2"

Westinghouse

Nuclear Safety Advisory Letter

This is a notification of a recently identified potential safety issue pertaining to basic components supplied by Westinghouse. This information is being provided so that you can conduct a review of this issue to determine if any action is required.

1000 Westinghouse Drive, Cranberry Township, PA 16066

Subject: **Thermal Sleeve Flange Wear Leads to Stuck Control Rod**	Number: **NSAL-18-1**
Basic Component: Thermal Sleeve in CRDM Reactor Head Penetration	Date: July 9, 2018
Substantial Safety Hazard or Failure to Comply Pursuant to 10 CFR 21.21(a)	Yes ☒ No ☐ N/A ☐
Transfer of Information Pursuant to 10 CFR 21.21(b)	Yes ☐
Advisory Information Pursuant to 10 CFR 21.21(d)(2)	Yes ☐

SUMMARY

In accordance with 10 CFR Part 21, Westinghouse reported an issue associated with thermal sleeve wear as a potential defect in May 2018 [1]. This NSAL provides details on the thermal sleeve flange issue to provide affected licensees a basis for operation and inspection recommendations.

Operating experience (OE) has shown that for Westinghouse nuclear steam supply system (NSSS) plants that have thermal sleeves in the control rod drive mechanism (CRDM) penetration tubes, the wear of the thermal sleeve flange against the tube could have potential consequences that were not previously considered. Recently, during a startup following a refueling outage at an Électricité de France (EdF) plant, Belleville Unit 2, a flange remnant from a separated thermal sleeve became cocked and interfered with control rod movement. The previous safety evaluations of separated sleeves and flanges, in topical report PWROG-16003-P [2] and Technical Bulletin TB-07-2, Revision 3 [3] considered this interference to be unlikely based on the information that was available at that time. Consequently, it was concluded that a stuck control rod was unlikely.

Considering the new OE from EdF [4], and the design similarities between the Belleville Unit 2 thermal sleeves and those used in Westinghouse NSSS plants and replacement reactor vessel heads, the inspection recommendations for thermal sleeve flanges in TB-07-2, Revision 3 may be insufficient. While there have been no reported events of control rods failing to insert into the core when required, Westinghouse reported this issue to the NRC under 10 CFR Part 21 because it had the potential to create a substantial safety hazard.

This NSAL supersedes portions of TB-07-2, Revision 3 related to the thermal sleeve flanges. The other information in TB-07-2, Revision 3, associated with the outer diameter/inner diameter (OD/ID) sleeve issue remains valid.

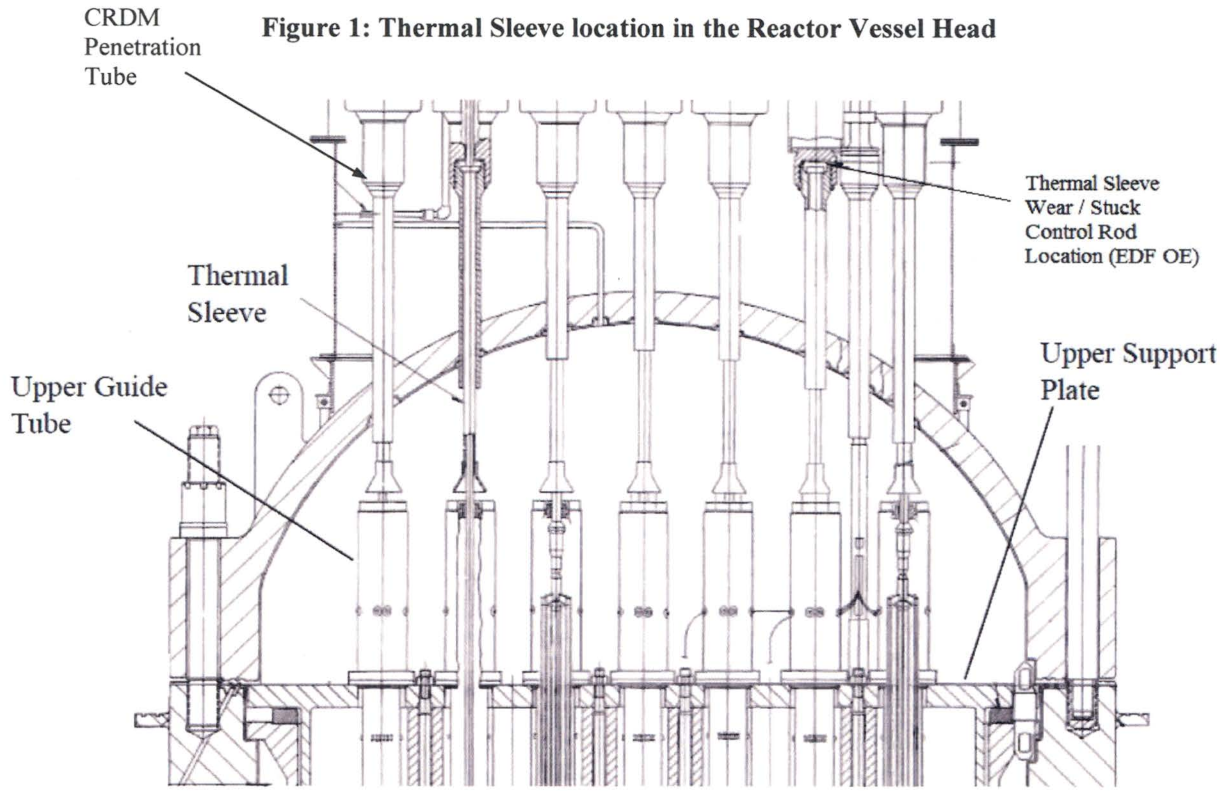
Additional information, if required, may be obtained from Nicholas A. Szweda, (412) 374-4105

Author:	Reviewer:	Manager:
William J. Smoody I&C Licensing & Regulatory Support	Steven T. Slowik I&C Licensing & Regulatory Support	Korey L. Hosack I&C Licensing & Regulatory Support
Verifier: Bryan M. Wilson Reactor Internals Design & Analysis I	Verifier: Eric M. Benacquista Reactor Internals Design & Analysis I	

ISSUE DESCRIPTION

As described in the preceding Summary, the reported defect pertains to inspection guidelines for the thermal sleeve flange, located in the upper reactor vessel internals. Figure 1 depicts the label – Thermal Sleeve Wear/Stuck Control Rod Location (EDF OE) – in the upper right portion which identifies the specific area of observed wear which led to flange separation and interference with control rod movement.

Figure 1: Thermal Sleeve location in the Reactor Vessel Head



In 2014, the first reported OE of thermal sleeve flange wear occurred. TB-07-2, Revision 2 was issued in February 2015, which added details related to the OE on flange wear and provided associated recommendations[1]. Based on information available at the time, TB-07-2, Revision 2 concluded that the flange wear was not likely to impact control rod movement and therefore, the safety significance was low. TB-07-2, Revision 3 added clarifications to the recommendations and affected plant susceptibility lists, but did not change the conclusions related to the safety significance. Westinghouse subsequently worked with the Pressurized Water Reactor Owners Group (PWROG) to develop acceptance criteria that is contained in PWROG-16003-P which could be used by the participating members when measuring thermal sleeve flange wear in accordance with the TB-07-2, Revision 3 recommendations.

Examination of the failed sleeve in 2014 showed that the upper thermal sleeve flange, which rests inside the CRDM penetration tube, had worn through and become separated. Further inspection of the failed thermal sleeve and the CRDM penetration tube concluded that a mutual wear mechanism existed between the two components. Figure 2 shows the original CRDM housing surface (A), the worn CRDM penetration tube surface (B), and the thermal sleeve flange remnant (C). The shaded regions of the figure represent material removed by the wear. The result was separation of the remaining remnant (dotted red outline) of the upper flange and a worn pocket in the adapter tube.

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[1]The issue of thermal sleeve wear was first identified in 2007 at a U.S. plant.

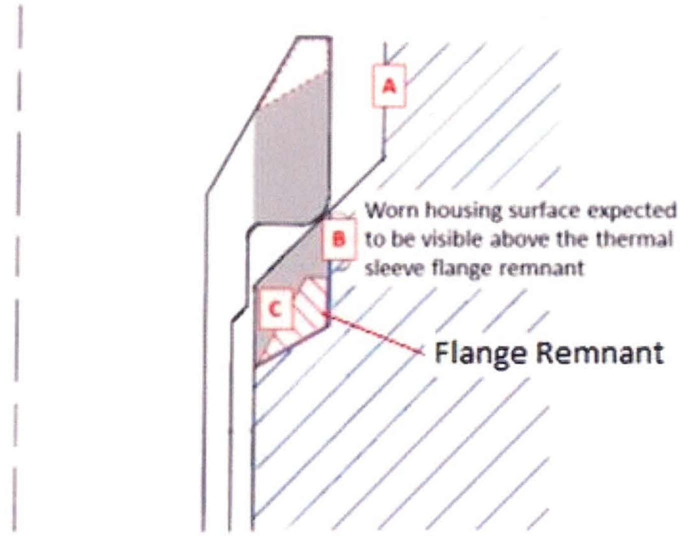

Figure 2: Worn Thermal Sleeve Flange and Image of Worn CRDM Housing

As thermal sleeve flange to CRDM penetration tube wear occurs, visual indication of the extent of wear develops. OE indicates that the height of the thermal sleeve guide funnel relative to various reactor vessel head design features can be used to identify the amount of thermal sleeve flange to CRDM penetration tube wear. As the flange and tube surfaces wear away, the thermal sleeve will move lower and become closer to the top of the upper guide tube (UGT). Figure 3 shows the lowering of the thermal sleeve and the potential for contact between the guide funnel and the top of the UGT. The amount of flange wear required to make contact and the actual visual indications seen will depend on the guide tube configuration used at the plant.

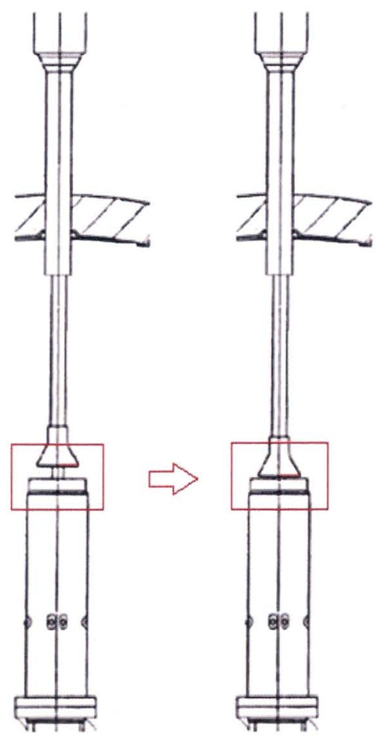

Figure 3: Potential Contact between the Guide Funnel and UGT

For plants with 14x14 guide tubes, the upper guide tube configuration is an open box-shaped tube with a top plate recessed from the top of the tube. These designs have either a fixed or a removable insert installed in this top plate which results in the top surface of the insert being flush or slightly recessed from the top edge of the tube. The width of the box tube for these guide tubes is smaller than the diameter of the funnel, so any contact would likely show up as intermittent shiny marks on the top edge of this tube. Furthermore, for the majority of the plants with 14x14 guide tubes (exceptions noted in Table 2) the spacing between the guide funnel and the upper guide tube is smaller than the lowering required for separation to occur. Therefore, with the exception of the two plants noted in Table 2, the wear markings on 14x14 guide tubes are likely to occur earlier in operation but are not necessarily an indication of imminent flange separation. For the two exceptions noted in Table 2, the spacing between the guide funnel and the upper guide tube is nearly the same distance as required for flange separation. Therefore, wear markings on the upper guide tube may or may not signify complete separation of the thermal sleeve flange; however, it is a clear indication that significant wear of the thermal sleeve flange has occurred and will likely require action.

For plants with 15x15 guide tubes, the upper guide tube configuration is similar to the design of the 14x14 guide tube, except that the tube width is slightly greater than the diameter of the guide funnel. These designs also have either a fixed or a removable insert installed which sit flush or slightly recessed from the top of the tube. For these plants, contact would likely show up as either intermittent shiny marks on the top edge of this tube or shiny marks on the top outer diameter of the insert in the top of the UGT. The spacing between the guide funnel and the upper guide tube for the 15x15 guide tubes is nearly the same distance as required for flange separation. Therefore, wear markings on the upper guide tube or insert for this design may or may not signify complete separation of the thermal sleeve flange; however, it is a clear indication that significant wear of the thermal sleeve flange has occurred and will likely require action.

For plants with 17x17 and 16x16 guide tubes, actual contact of the guide funnel with the UGT top housing plate will likely only occur after the flange has completely separated from the thermal sleeve. In December 2017, a 4-loop 1300 MW EdF plant in France, Belleville Unit 2, experienced complete wear-through and separation of one of the thermal sleeve flanges at a rodded location (Core Location H-8). Figure 4, taken from Belleville Unit 2, illustrates the wear marking ("shiny ring") that may be visible on the UGT top housing plate after the reactor vessel head is removed, if a plant has been operating with a failed thermal sleeve.

Figure 4 also shows the anti-rotation stud design used on the EdF guide tubes. Westinghouse NSSS plants have an anti-rotation stud/nut configuration on the top housing plate which results in the stud protruding slightly higher and the nut extending more into the diameter of the wear ring. This difference in configuration may result in slightly different wear markings on the stud and nut as compared to what is shown in Figure 4. While signs of contact between the guide funnel and the anti-rotation stud/nut may be visible before flange separation occurs, this wear pattern indicates that significant thermal sleeve flange wear has occurred.

Figure 4: "Shiny Ring" Evidence of Guide Funnel Contact Wear on the UGT at Belleville Unit 2

TECHNICAL EVALUATION

The recent experience at Belleville Unit 2 is the first OE of a failure of a thermal sleeve in a rodded location (Core Location H-8) due to flange wear. This condition was discovered while performing low-power physics testing (LPPT) and rod drop testing following a refueling outage. The issue was first identified due to difficulty in stepping the rod into the core during LPPT. The rod was able to be freed by exercising the drive rod, but was again stopped prior to full insertion during the subsequent rod drop testing. Evaluations of the condition following this discovery showed that the thermal sleeve had worn through the flange, leaving the remnant flange ring in the CRDM penetration housing similar to the one discovered in an unrodded core location in 2014. EdF subsequently determined that this thermal sleeve flange remnant was the cause of the stuck rod stepping and dropping issue.

The new OE identifies that control rod functionality can potentially be impacted by thermal sleeve flange separation. Comparing the thermal sleeve designs used in Westinghouse-designed plants and replacement reactor vessel heads identified many similarities between the Westinghouse thermal sleeve designs and the thermal sleeve that separated at Belleville Unit 2. Therefore, the previous conclusions in TB-07-2, Revision 3 and PWROG-16003-P related to the safety significance of thermal sleeve flange wear failures and the frequency of the associated inspection recommendations may not be sufficiently conservative.

Wearing of the thermal sleeve flange to the point of separation can result in a loose part (flange remnant) positioned at the top of the CRDM penetration housing just below the CRDM latch assembly. If unidentified, or identified and left uncorrected, flange wear failures could result in control rods failing to insert at multiple locations. Therefore, the lack of conservatism in the existing Westinghouse inspection guidance could result in a substantial safety hazard.

SAFETY SIGNIFICANCE

Based on the Belleville Unit 2 OE, flange wear failures at multiple core locations could occur and result in a potential safety concern of multiple stuck control rods, if no action is taken to monitor and correct potential flange wear.

As described below, the available data supports continued operation because, aside from the new EdF OE, wear observed to date has generally been moderate in rodded thermal sleeve locations. Therefore, we can expect there is generally still significant margin to the acceptance criteria defined in PWROG-16003-P. The acceptance criterion in PWROG-16003-P is derived to assure flange separation does not occur and interfere with rod insertion.

The funnel lowering rates from the available inspections were calculated resulting in a 95% upper bound rate of approximately 0.03 inch/EFPY (effective full power year) and a 99% upper bound rate of approximately 0.04 inch/EFPY. This rate is well below the rate calculated from the Belleville Unit 2 OE, which is approximated to be on the order of 0.13 inch/EFPY based on informal communications with EdF.

Given the lower wear amounts, the lower wear rates being experienced at Westinghouse plants compared to Belleville Unit 2, and the unique geometric and operations scenarios which are required to impede drive rod motion, it is unlikely during one operating cycle for thermal sleeve flange wear to result in a condition where more than one control rod is unable to insert. The safety analyses demonstrate that the reactor can be safely shut down with the highest worth control rod stuck in the fully withdrawn position. Additionally, sticking of two or more control rods as a result of these loose parts is not considered to be a credible event prior to the next inspection opportunity for the impacted plants because is it highly unlikely that more than one flange remnant would be present simultaneously. Therefore, continued operation is justified for all Westinghouse NSSS plants.

Furthermore, considering the most limiting thermal sleeve lowering separation criteria of approximately 1-inch [2] and the 99% upper bound wear rate of 0.04 inch/EFPY, separation is not expected for plants with fewer than 25 EFPY on their reactor vessel head. Note that the cited separation criterion of 1-inch does not define actual flange separation, but rather the wear at which a seismic event may cause distortion of the flange. Actual separation would not be expected before significantly more lowering occurs (conservatively, more than an additional half of an inch).

In summary, based on the OE at Belleville Unit 2 and the position of the potential loose part, it is possible that if no action is taken to monitor and correct this condition, flange wear failures at multiple locations could occur. However, sticking of two or more control rods as a result is not a credible event prior to the next available inspection (i.e., the next refueling outage) for the impacted plants. Therefore, continued operation is justified for all Westinghouse NSSS plants.

AFFECTED PLANTS

The wear of thermal sleeves (primarily OD and ID wear) is a known issue that the industry has successfully managed for over 10 years. Reported OD and ID wear has consistently shown that thermal sleeve wear at plants with higher upper head bypass flow, referred to as "T-Cold" head plants, tends to be greater than thermal sleeve wear experienced at plants with lower upper head bypass flow, referred to as "T-Hot" head plants. T-Cold and T-Cold capable[2] plants are potentially more susceptible to experiencing thermal sleeve flange wear and are listed as plants with higher susceptibility in Table 1.

T-Hot plants are expected to experience flange wear at a significantly slower rate and, therefore, are considered to have lower susceptibility to thermal sleeve flange wear.
Table 2 lists the plants with T-Hot heads. These tables list the Westinghouse plants supplied with reactor internals that included thermal sleeves. Some plants may be permanently shut down or are planning to do so. Tables 1 and 2 should be reviewed by the Licensee to confirm whether their reactor head is T-Cold or T-Hot. The tables exclude the Westinghouse NSSS plants that have eliminated thermal sleeves in their replacement heads.

Table 1: Plants with Higher Susceptibility to Thermal Sleeve Flange Wear

A.W. Vogtle 1 & 2	Maanshan 1 & 2
Ascó I & II	McGuire 1 & 2
Braidwood 1 & 2	Millstone 3
Byron 1 & 2	Seabrook
Callaway	Sequoyah 1 & 2
Catawba 1 & 2	Shearon Harris
Comanche Peak 1 & 2	Sizewell B
Diablo Canyon 2	South Texas 1 & 2
Doel 4*	Tihange 3*
Hanbit 1 & 2	Watts Bar 1 & 2
Kori 2*, 3 & 4	Wolf Creek

* T-hot but T-cold capable

[2] T-Cold capable plants are essentially a hybrid of T-Hot and T-Cold with some head spray cooling nozzles plugged and some open (not quite T-Cold). In this configuration there is uncertainty as to whether the local jets will cause local flow conditions similar to T-Cold; therefore, these plants are conservatively assumed to have a higher susceptibility to thermal sleeve flange wear.

Table 2: Plants with Lower Susceptibility to Thermal Sleeve Flange Wear

Almaraz 1 & 2	Point Beach 1* & 2*
Angra 1	Prairie Island 1* & 2*
Beaver Valley 1 & 2	R.E. Ginna*
D.C. Cook 1 & 2	Ringhals 2, 3 & 4
Diablo Canyon 1	Salem 1 & 2
Doel 1** & 2**	Surry 1 & 2
Indian Point 2 & 3	Takahama 1
Mihama 1*	Tihange 1
North Anna 1 & 2	Turkey Point 3 & 4
Ohi 1 & 2	

* The indicated plants have 14x14 guide tubes with gaps between the guide funnel and upper guide tube that will limit flange wear and prevent flange separation. Therefore, the recommendations in this NSAL do not apply to these plants.
** The indicated plants have 14x14 guide tubes and gaps between the guide funnel and the upper guide tube which are nearly the same as the distance required for flange separation.

NRC AWARENESS

The NRC was informed of this issue by Westinghouse letter LTR-NRC-18-34, May 23, 2018 [1].

RECOMMENDED ACTIONS

Based on the potential for this issue to result in a nuclear safety concern of stuck control rods, the affected plants should perform the following if they have exceeded 25 EFPY[3] on their original or replacement reactor vessel closure head. There are no recommendations for plants that have less than 25 EFPY on their original or replacement reactor vessel closure head.

T-Cold and T-Cold Capable Plants

- Prior to performing measurements of flange wear, plants should establish measurement acceptance criteria to prevent thermal sleeve flange separation.
    - For the applicable participating members, the acceptance criteria established in PWROG-16003-P can be used for assessing the wear condition. At the time that the criteria in PWROG-16003-P were developed, the thermal sleeve flange separation was not considered a safety significant event. As such, the criteria were developed to prevent separation during normal/upset operation conditions and did not consider faulted conditions. However, a preliminary evaluation shows that load increases associated with faulted conditions can likely be offset by reductions in analysis conservatisms such that these criteria will remain applicable upon inclusion of faulted loads. The PWROG is planning an update to PWROG-16003-P to include faulted conditions in the acceptance criteria.
    - For the other plants that did not participate in PWROG-16003-P, acceptance criteria can be established on a plant-specific basis using plant design information.

[3] Although it was calculated that separation is not expected prior to achieving 25 EPFY on the reactor vessel closure head, performing inspections prior to this time provides some additional margin to reduce the potential of taking immediate remedial action at the time of inspection.

- During the first refueling outage following the issuance of this NSAL:
    - o Perform measurements of the lowering of all thermal sleeves using the methods discussed in the "General Recommendations" section below. This will establish a baseline for the plant which can be used to assess the current condition of the thermal sleeves and to determine plant-specific wear rates which can be used in establishing the frequency of conducting subsequent measurements.
        - Using the results of the baseline measurement, determine if the thermal sleeve flange wear has exceeded the established acceptance criteria, or if flange separation has occurred.
        - If the acceptance criteria have not been exceeded, establish a re-inspection frequency based on the baseline measurements that provides reasonable assurance that compliance with the licensing basis will be maintained between inspections.
            - For the applicable participating members, the wear projection methodology established in PWROG-16003-P can be used for determining the re-inspection interval.
            - For the other plants that did not participate in PWROG-16003-P, the re-inspection frequency can be determined using the wear rates established from the baseline measurement and plant-specific acceptance criteria.
        - If the acceptance criteria in PWROG-16003-P have been exceeded, but flange separation has not occurred, it may be possible to develop plant-specific acceptance criteria that takes into account plant-specific wear rates and/or design data.
            - If plant-specific criteria are developed and the Licensee is successful in confirming the acceptability of the as-found condition, then a re-inspection frequency may be established based on the measurements taken for the baseline and any plant-specific data used to develop the acceptance criteria.
            - If the Licensee cannot demonstrate the acceptability of the as-found condition with either the general or plant-specific criteria (if applicable), then the issue should be mitigated. The mitigation strategy should consider actions needed to ensure the plant remains consistent with its licensing basis.
        - If flange separation has occurred, the issue should be mitigated. The mitigation strategy should consider actions needed to ensure the plant remains consistent with its licensing basis.
- In the event of a plant shutdown between refueling outages, without head removal and prior to the first measurement of thermal sleeve flange wear:
    - o Prior to restart, perform rod exercises to ensure that the rods move freely in and out of the core. For example, the standard Technical Specifications require exercising each individual control rod by moving each control rod ten steps in either direction.

T-Hot Plants

- Continue to monitor the industry OE on this issue.
- During the next refueling outage after issuance of this NSAL and each subsequent refueling outage, perform a visual inspection of the tops of the upper internal UGTs per the guidance below to identify if any sleeves have lowered significantly or are in a failed state:

  - For plants with 15x15 guide tubes and the two plants with 14x14 guide tubes noted in Table 2, look for shiny marks on the top edge of the upper guide tube enclosure.
  - For plants with 17x17 and 16x16 guide tubes, look for the "shiny ring" shown in Figure 4.
- During the next under-head inspection (i.e., the volumetric inspections required by ASME Section XI Code Case N-729-4 [5] and 10 CFR 50.55a), and at subsequent under-head inspections:
  - Perform a visual inspection of the bottom of the thermal sleeve guide funnels to look for any shiny surfaces on the bottom surface of the guide funnel that would indicate that the thermal sleeve guide funnels have dropped to a point where they are in contact with the top of the guide tube.
  - Perform a visual inspection of thermal sleeve guide funnel elevations to identify whether any sleeves are noticeably lower than others.
  - If any of the above inspections indicate advanced wear of a thermal sleeve flange (or separation), inspections and mitigation strategies consistent with the T-Cold plants are recommended to be implemented.

General Recommendations

As previously discussed, thermal sleeve flange wear is evidenced by a lowering of the thermal sleeve relative to the rest of the closure head. Therefore, measurement for thermal sleeve flange wear should involve a technique that is capable of determining the relative distance between a known reference point on the head, such as the bottom surface of the penetration nozzle, or the closure head mating surface and the bottom surface of the thermal sleeve guide funnel. Laser scanning is one such measurement method that is capable of determining this distance.

Because of the variability in the as-built location of the penetration housing and tolerances associated with the penetration housing and thermal sleeves, it may be difficult to positively assess thermal sleeve flange wear by a visual inspection or a comparative inspection of the elevation of one sleeve to another. For most Westinghouse-designed heads, the as-built elevation of the penetration housing relative to the mating flange is a known point which can be used to determine the original elevation of the thermal sleeve; therefore, in this case, either the closure head mating surface or the bottom surface of the penetration nozzle would serve as acceptable reference points for determining the elevation of the thermal sleeves.

For cases where as-built information is not available, or the availability of this data is uncertain, an acceptable method is to measure the location of the thermal sleeve guide funnel bottom surface relative to the bottom of the penetration nozzle. Using the bottom surface of the penetration nozzle as the reference point reduces the uncertainty in determining the change in elevation relative to the original position of the thermal sleeve.

Having a measurement of the greatest wear will greatly aide in assessing a plant-specific wear rate that will ensure an acceptable approach for addressing the extent and progression of wear. Westinghouse requests that the inspection results be provided to Westinghouse 90 days or sooner after the end of the outage. This information will be added to a wear experience database that will support future and on-going characterization and refinement of the Westinghouse fleet wear condition.

If a plant in either Table 1 or 2 has performed baseline inspections and established a plant-specific monitoring program or has taken proactive mitigating actions to address this issue prior to the release of these recommendations, such actions should satisfy these recommendations assuming the monitoring program is based on conservative assumptions similar to those used in PWROG-16003-P.

The susceptibility assessment and recommendations provided herein are applicable to Westinghouse NSSS plants with original reactor vessel heads and plants where Westinghouse designed the replacement head. Plants with replacement heads not designed by Westinghouse should contact their replacement head designer for confirmation of applicability.

REFERENCES

1. Westinghouse Letter LTR-NRC-18-34, "Notification of the Potential Existence of Defects Pursuant to 10 CFR Part 21," May 23, 2018 [NRC Accession Number ML18143B678].
2. Pressurized Water Reactor Owners Group Report, PWROG-16003-P, Rev. 1, "Evaluation of Potential Thermal Sleeve Flange Wear," August 2017.
3. Westinghouse Technical Bulletin, TB-07-2, Rev. 3, "Reactor Vessel Head Adapter Thermal Sleeve Wear," December 7, 2015.
4. EdF Notice: "Déclaration d'un événement de niveau 1 (échelle INES) lié au risque de blocage d'une grappe de commande, dans les centrales de Belleville-sur-Loire (Cher) et de Saint-Alban (Isère)", Publié le 26/02/2018. [*English translation:* Declaration of a Level 1 event (INES scale) linked to the risk of blockage of a control cluster, at the Belleville-sur-Loire (Cher) and Saint-Alban (Isère) plants.]
5. ASME Section XI Code Case N-729-4, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds," approval date June 22, 2012.