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LTR-NRC-18-34
May 23, 2018

Subject: Notification of the Potential Existence of Defects Pursuant to 10 CFR Part 21

The following information is provided pursuant to the requirements of 10 CFR Part 21 to report a defect. It has been discovered that for Westinghouse nuclear steam supply system (NSSS) plants that have thermal sleeves in the control rod drive mechanism (CRDM) penetration tubes, wear of the thermal sleeve flange against the tube could have safety consequences that were not previously considered. Recent operating experience, during startup procedures at an Électricité de France (EdF) plant, has shown the possibility for the flange remnant from a thermal sleeve to interfere with control rod movement. The safety evaluations in report PWROG-16003-P, Rev. 1 and Technical Bulletin TB-07-2, Rev. 3 considered this interference to be highly unlikely and the inspection recommendations reflected a conservative time frame. Considering the new operating experience from EdF, the inspection guidelines that were based on the information available at that time may be non-conservative. While there have been no reported events of control rods failing to insert into the core during steady-state plant operation, Westinghouse is conservatively reporting this information as having the potential to create a substantial safety hazard.

(i) Name and address of the individual or individuals informing the Commission.

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(ii) Identification of the facility, the activity, or the basic component supplied for such facility or such activity within the United States which fails to comply or contains a defect.

The affected basic components are PWR Owners Group report PWROG-16003-P, Revision 1 and Westinghouse Technical Bulletin TB-07-2, Revision 3.

(iii) Identification of the firm constructing the facility or supplying the basic component which fails to comply or contains a defect.

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(iv) Nature of the defect or failure to comply and the safety hazard which is created or could be created by such defect or failure to comply.

The defect deals with inspection guidelines for the thermal sleeve flange, located in the reactor vessel internals. Figure 1 shows the location of the flange in the upper right. The label in Figure 1, "Thermal Sleeve Wear / Stuck Control Rod Location (EDF OE)" indicates the specific area of observed wear leading to interference with control rod movement.

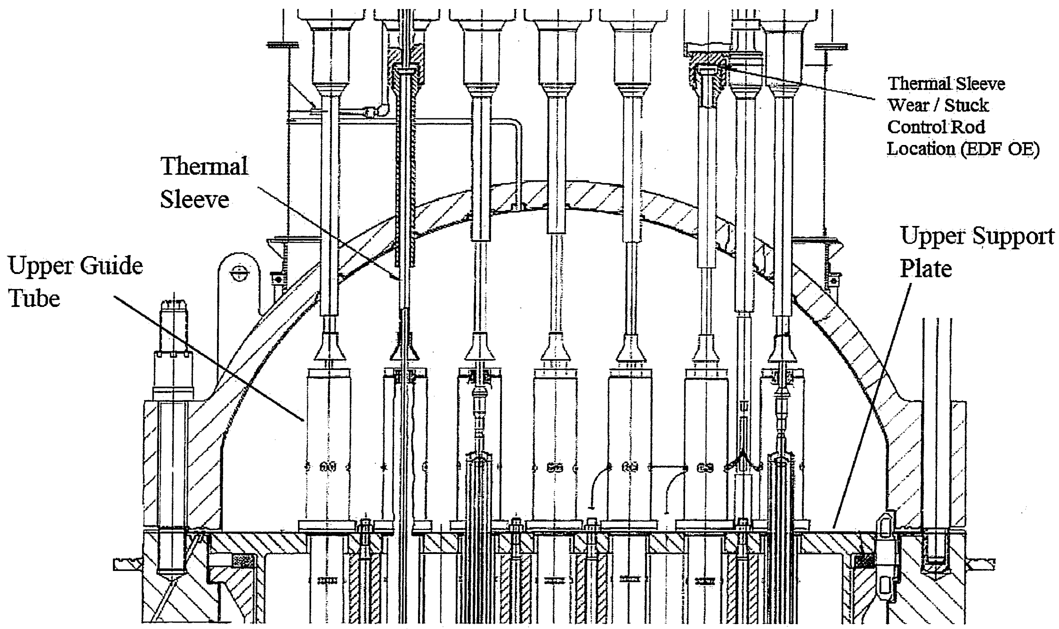

Figure 1 Thermal Sleeve Location in the Reactor Vessel Head

Westinghouse has determined that there is no immediate nuclear safety concern associated with this defect. A substantial safety hazard is only possible if there is interference with the movement of more than one control rod. If left uncorrected, the inspection guidelines in TB-07-2, Rev. 3 may not be sufficient to provide reasonable assurance that no more than one flange could separate in a reactor.

Thermal sleeve wear was first identified in 2007. At that time, the wear was primarily discovered on the outside diameter (OD) of the thermal sleeve at the elevation where the sleeve exits the control rod drive mechanism (CRDM) penetration housing and on the inside diameter (ID) just above the guide funnel due to contact with the drive rod. As such, thermal sleeve flange wear was not postulated as part of the extent of condition that was performed, nor was it evaluated.

In 2014, the first reported operating experience (OE) of thermal sleeve flange wear occurred. At two part-length CRDM locations at a U.S. plant, thermal sleeves were found to be failed at the flange due to wear through of the flange. Upon discovery of this issue, it was determined that the wear of the thermal sleeve flange could be correlated to a change in elevation of the thermal sleeve guide funnel elevations when compared to the as-designed condition. Subsequently, measurements were taken at all remaining thermal sleeve locations at the plant to evaluate the extent of condition. The measurements at other part-length locations showed significant but acceptable wear and all rodded locations showed low to moderate wear.

Westinghouse subsequently issued TB-07-2, Revision 2 which added details related to the OE on flange wear and provided associated recommendations. In addition, TB-07-2 also concluded that the flange wear was not likely to impact plant operation and therefore the safety significance was deemed to be low. TB-07-2, Revision 3, added clarifications. Westinghouse worked through the Pressurized Water Reactor Owners Group (PWROG) to develop acceptance criteria in PWROG-16003-P which could be used when measuring thermal sleeve wear in accordance with the TB-07-2 recommendations. PWROG-16003-P included a safety assessment supporting the conclusion of low safety significance summarized in the TB.

In December 2017, a 4-Loop 1300 MW EdF plant in France, Belleville Unit 2, experienced complete wear-through and separation of one of their thermal sleeve flanges at a rodded location (core location H-8). This is the first OE of a failure of a thermal sleeve at a rodded location due to flange wear. This condition was discovered while performing low-power physics testing (LPPT) and rod drop testing prior to returning to power. The issue was first identified due to difficultly stepping the rod into the core during LPPT. The rod was able to be freed by exercising the drive rod but then was 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 a ring in the CRDM penetration housing similar to the one discovered at the plant in 2014. This flange remnant is believed to be the cause of the rod stepping and dropping issue experienced by EdF at core location H-8.

Based on this new OE, which shows control rod functionality can potentially be impacted by thermal sleeve flange failure, the previous conclusions in TB-07-2 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. During steady state operation, little to no flow passes in the region of the flange remnant and therefore it is expected to remain in the pocket worn into the CRDM penetration housing. Under certain conditions, such as during a scram (rod drop) or a filling and venting operation prior to reactor start-up, there is a coolant flow up into the CRDM which can disturb this flange remnant and lift it from the wear pocket. If this occurs, the orientation of the flange remnant is uncertain and has a size capable of being caught by the drive rod lobes, pulled into the CRDM housing, and has the potential to impede the insertion of the rods. Although only possible under specific conditions, in this case due in part to flow in the region of the flange remnant, the Belleville Unit 2 event shows that these conditions can exist and jamming of the drive rod during both stepping and rod drop is possible. Therefore if no action is taken to monitor and correct this condition, flange wear failures could result in control rods failing to insert at multiple locations. Therefore, the lack of conservatism in the guidance could lead to a substantial safety hazard.

[Operability Assessment]

The data support justification for continued operation because (with the exception of the experience at EdF plants) wear observed to date has generally been moderate for rodded thermal sleeve locations. There is generally still significant margin to the acceptance criteria defined in PWROG-16003-P.

The funnel lowering rates from the available inspections were calculated resulting in a 95% upper bound rate of approximately 0.03 in/EFPY (inches / effective full power year) and a 99% upper bound rate of approximately 0.04 in/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 in/EFPY based on informal communications. This is consistent with historical data for other components, such as guide card wear, which is driven by flow-induced vibration (FIV) and pump-induced vibration (PIV) wear related phenomena.

Given the lower wear amounts, the lower wear rates being experienced at domestic and international Westinghouse plants compared to Belleville Unit 2, and the unique geometric and operations scenarios which are required to impede a drive rod, it is considered unlikely, within the context of one or two operating cycles from now, for thermal sleeve flange wear to result in a condition where more than one control rod is unable to insert. Analyses have shown 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 a credible event prior to the next available inspection opportunity for the impacted plants because is it highly unlikely that more than one flange remnant would be present simultaneously. Therefore, Westinghouse considers continued operation is justified for all domestic and international Westinghouse-NSSS plants.

Furthermore, considering the most limiting thermal sleeve lowering criteria of 1-inch 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 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 half of an inch).

In summary, based on the events experienced 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 of these foreign objects is not considered a credible event prior to the next available inspection point for the impacted plants. Therefore, Westinghouse considers continued operation is justified for all domestic and international Westinghouse-NSSS plants.

(v) The date on which the information of such defect or failure to comply was obtained.

The Westinghouse president was informed of this issue on May 22, 2018.

(vi) In the case of a basic component which contains a defect or fails to comply, the number and location of these components in use at, supplied for, being supplied for, or may be supplied for, manufactured, or being manufactured for one or more facilities or activities subject to the regulations in this part.

For this issue Westinghouse-NSSS plants may be categorized by reactor vessel head temperatures; i.e., T-Cold, T-Cold Capable and T-Hot. Fluid flow via nozzles in the upper internals determines which category. T-Cold Capable plants are a hybrid of T-Hot and T-Cold with some flow nozzles plugged and some open.

The plants designated as T-Cold or T-Cold capable (Tier 1 Plants) are expected to be more susceptible to thermal sleeve flange wear. T-Hot plants (Tier 2 Plants) are expected to experience thermal sleeve flange wear at a significantly lower rate and are not expected to have near term susceptibility to this issue. We have included all impacted Westinghouse-NSSS plants, both domestic and international, based on the best available information.

Tier 1 Plants (T-Cold or T-Cold Capable Plants):

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*	Wolf Creek
Kori 3 & 4	

* **T-Cold Capable**

Tier 2 Plants (T-Hot Plants):

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
Diablo Canyon 1	Ringhals 3 & 4
Indian Point 2 & 3	Salem 1 & 2
Mihama 1 & 2	Surry 1 & 2
North Anna 1 & 2	Takahama 1
Ohi 1 & 2	Turkey Point 3 & 4

(vii) The corrective action which has been, is being, or will be taken; the name of the individual or organization responsible for the action; and the length of time that has been or will be taken to complete the action.

A Westinghouse communication will be supplied to affected licensees to inform them that this defect has been reported and to provide updated recommendations concerning future inspection guidance. The communication will recommend that affected licensees perform inspections on a more accelerated schedule than presented in TB-07-2, Revision 3.

(viii) Any advice related to the defect or failure to comply about the facility, activity, or basic component that has been, is being, or will be given to purchasers or licensees.

Affected licensees will be informed that this defect is being reported. As noted earlier, Westinghouse determined that there is no immediate nuclear safety concern associated with the defect and, therefore, there is time for corrective actions to be taken by the affected licensees. A Westinghouse communication will be issued to affected licensees to recommend that inspections be performed on a more frequent basis than was recommended in TB-07-2, Revision 3.

(ix) In the case of an early site permit, the entities to whom an early site permit was transferred.

N/A

Very truly yours,

James A. Gresham, Secretary
Westinghouse Safety Review Committee

cc: E. Lenning (NRC MS O-11-F1)