IPRenewal NPEmails

From: Molkenthin, James P <molkenjp@westinghouse.com>
Sent: Tuesday, August 09, 2016 11:57 AM
To: Poehler, Jeffrey; Holonich, Joseph
Cc: McFadden, John L; Paesano, Patricia C.; Wilson, Bryan M.; Lott, Randy G.
Subject: [External_Sender] Transmittal of Revised Westinghouse NSAL-16-1, Revision 1, Baffle-Former Bolts
Attachments: NSAL-16-1 R1.pdf

Importance: High

Jeff and Joe,

Please find attached revised Westinghouse NSAL-16-1, Revision 1, "Baffle-Former Bolts".

This revision is being issued to move Doel 1 & 2 from Tier 2a to Tier 2c and add Tihange 1 to Tier 3. Additionally, the discussion on the MRP-227 inspections for Tier 2c plants was removed.

If you have any questions or require additional information on the revised Westinghouse NSAL, please contact John McFadden at 412-374-2316 or mcfaddjl@westinghouse.com.

Please feel free to put the NSAL in ADAMS.

Regards,

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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
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Subject: **Baffle-Former Bolts**	Number: **NSAL-16-1 Revision 1**
Basic Component: Baffle-Former Bolts	Date: August 1, 2016
Substantial Safety Hazard or Failure to Comply Pursuant to 10 CFR 21.21(a) Transfer of Information Pursuant to 10 CFR 21.21(b) Advisory Information Pursuant to 10 CFR 21.21(d)(2)	Yes ☐ No ☒ N/A ☐ Yes ☐ Yes ☐

This revision is being issued to move Doel 1 & 2 from Tier 2a to Tier 2c and add Tihange 1 to Tier 3. Additionally, the discussion on the MRP-227 inspections for Tier 2c plants was removed. Changes are noted with revision bars in the margin.

SUMMARY

Recently, inspections of the reactor internals were performed at Indian Point Unit 2 and Salem Unit 1; both Westinghouse nuclear steam supply system (NSSS) units. Inspections at Indian Point Unit 2 were performed under the Material Reliability Program (MRP)-227-A, Revision 0 aging management guidelines. Salem Unit 1 performed visual inspections of baffle-former bolts in part due to the operational experience (OE) from Indian Point Unit 2, and when failed bolts were observed, a 100% ultrasonic (UT) inspection was performed. During these inspections, Indian Point Unit 2 and Salem Unit 1, found a larger-than-expected number of baffle-former bolts that were either failed or exhibiting UT indications. The pattern of degraded bolts was also concentrated, or clustered, more than anticipated based on OE gained from previous analyses and inspections.

The purpose of this Nuclear Safety Advisory Letter (NSAL) is to provide the results of the 10 CFR 21 evaluation performed for this issue, as well as the Westinghouse recommendations associated with this issue.

Additional information, if required, may be obtained from John L. McFadden, (412) 374-2316

Author:	Reviewer:	Manager:
John S. Galembush	William J. Smoody	James A. Gresham
Regulatory Compliance	Regulatory Compliance	Regulatory Compliance

ISSUE DESCRIPTION

During the spring 2016 Cycle 22 refueling outage at Indian Point Unit 2, the MRP-227-A baffle-former bolt inspections were performed. During the visual inspection, 31 baffle-former bolts were identified with protruding heads and lock bar damage, including two bolts with missing heads. Additionally, the subsequent UT inspection identified a total of 182 baffle-former bolts with indications and 14 non-testable bolts, for a total of 227 bolts with potential indications out of a total population of 832 bolts. Clustering of the degraded bolts was noted. A visual inspection of the baffle-to-baffle edge bolts and the baffle plates was performed and did not identify any damaged edge bolts or any abnormalities with the baffle plates. Additionally, there were no leaking fuel rods in the Indian Point Unit 2 core.

Subsequent to the Indian Point Unit 2 inspection, during the spring 2016 Cycle 24 refueling outage at Salem Unit 1, baffle-former bolt visual and UT inspections were performed. During the visual inspection, 18 baffle-former bolts were identified with missing bolt heads or protruding heads. The UT inspection identified 138 baffle-former bolts with indications and 18 non-testable bolts. Additionally, 8 baffle-former bolts were not UT inspected because they were identified to have visually cracked lock bar welds. This represents a total of 182 bolts with potential indications, out of a total of 832 bolts. An additional 7 failures were discovered while replacement work was being performed, bringing the total to 189. Clustering of the degraded bolts was noted. A visual inspection of the baffle-to-baffle edge bolts and baffle plates was performed and did not identify any damaged edge bolts or abnormalities in the baffle plates. There was one leaking fuel rod in the Salem Unit 1 core. The leak mechanism was debris fretting from a separated bolt head or lock bar and not from baffle jetting.

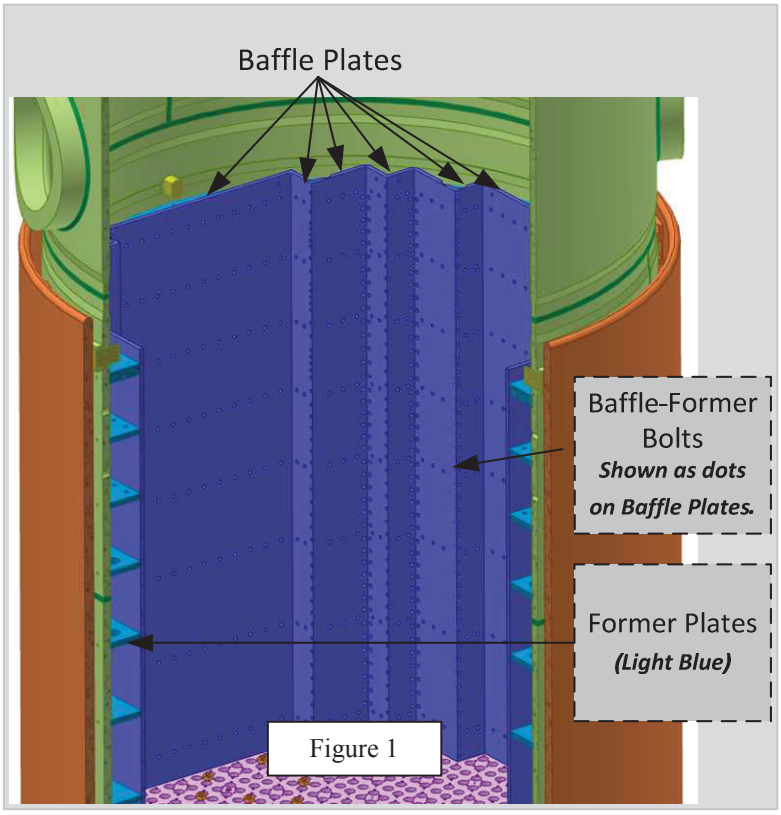

Figure 1

The baffle-former bolts are located in the reactor internals assembly. Figure 1 shows the interior of the lower internals, without fuel assemblies. The baffle plates are the vertical components that are next to the fuel, when the core is in place. The baffle-former bolts are the rows of small dots in Figure 1. These bolts attach the baffle plates to the former plates. The baffle-former assembly is a basic component and is part of the reactor internals structure. The function of the baffle-former assembly is to maintain the fuel assembly structural integrity to ensure that the control rods insert, maintain a coolable core geometry, and ensure a core configuration that supports long-term reactor shutdown.

Previous OE with clustered baffle-former bolt degradation included broken baffle-former bolts observed at D.C. Cook Unit 2 during the fall 2010 refueling outage (U2C19), as discussed in Westinghouse Technical Bulletin TB-12-5. Initial visual examinations at D.C. Cook Unit 2 identified 18 damaged bolts in the same baffle plate. A bolt replacement campaign subsequently revealed 24 additional failed bolts for a total of 42 in this baffle plate region. Based on metallurgical examinations of some of the removed

bolts, irradiation-assisted stress corrosion cracking (IASCC) was determined to be the apparent cause of the degradation. There was one leaking fuel rod in the D.C. Cook Unit 2 core. The leak mechanism was debris fretting from a separated bolt head or lock bar and not from baffle jetting.

There is also earlier OE related to pressurized water reactor (PWR) baffle-former bolt degradation, making this issue an area of focus for the industry for some time. Baffle-former bolt degradation has commonly been linked to IASCC. However, clustered failure patterns, such as those discussed herein, are a more recent discovery. Clustering of IASCC-induced bolt degradation is most likely a result of a two-stage process that began with a random distribution of degraded bolts and then began to cluster when increased loads on the surrounding bolts led to adjacent failures.

All three of the plants identified contain baffle-former bolts made of the same material (Type 347 stainless steel). Data on Westinghouse plants indicates that the bolt design used in Type 347 bolts has a higher propensity to IASCC than the improved designs used in Type 316 stainless steel bolts.

With respect to the extent of condition, all plants with baffle-former bolts are potentially susceptible to IASCC; however, degraded baffle-former bolts to the extent observed at Indian Point Unit 2 and Salem Unit 1 has been limited to downflow reactor designs with Type 347 bolts. This condition may be attributed to a higher pressure drop across the baffle plate, baffle-former bolt pre-load relaxation, and a more susceptible bolt design. Therefore, the combination of a downflow reactor design configuration, Type 347 bolts, and 4-loop configuration is considered an indication of susceptibility to clustered degradation.

All 2- and 3-loop Westinghouse NSSS reactors have more baffle-former bolts per square inch of baffle plate than the 4-loop Westinghouse NSSS reactors, resulting in a lower primary stress on the 2- and 3-loop downflow baffle-former bolts relative to 4-loop downflow baffle-former bolts. A majority of the 2- and 3-loop Westinghouse NSSS downflow reactors have UT inspection data within the past 6 years. The observed bolt UT indications were randomly distributed. UT inspections at these plants have identified significantly fewer indications than UT inspections at 4-loop downflow plants. At D.C. Cook Unit 2, Indian Point Unit 2, and Salem Unit 1, which are all 4-loop downflow plants, the clustered nature of the baffle-former bolt degradation has necessitated bolt replacement campaigns.

While 2- and 3-loop plants with downflow designs are not excluded from potential clustered baffle-former bolt degradation, based on the available information, the 4-loop downflow plants have been determined to be the most susceptible. The 4-loop Westinghouse NSSS downflow plants are:

- D.C. Cook Units 1 and 2
- Diablo Canyon Unit 1[1]
- Indian Point Units 2 and 3
- Salem Units 1 and 2
- Sequoyah Units 1 and 2

Sequoyah Units 1 and 2 are the only 4-loop downflow plants with Type 316 baffle-former bolt material and design. Because of this difference, baffle-former bolts at Sequoyah Units 1 and 2 are expected to be more resistant to IASCC. Therefore, the summary of the evaluation that follows focuses on D.C. Cook Units 1 and 2, Diablo Canyon Unit 1, Indian Point Units 2 and 3, and Salem Units 1 and 2. The evaluation for the 4-loop Westinghouse NSSS downflow design bounds the other Westinghouse NSSS

[1] Diablo Canyon Unit 2 is an upflow converted design.

plant types. In addition, the CE-designed plants that have bolted core shrouds are similar to the Westinghouse NSSS upflow-designed plants and are bounded by this evaluation. The remaining CE-designed plants do not have bolted core shrouds and are not affected by this issue.

TECHNICAL EVALUATION

Assumptions

The first step in evaluating the safety implications of a degraded baffle-former assembly is to define a limiting condition for the degraded structure and limiting loading condition. Based on recent OE, it is conservative to postulate that all of the baffle-former bolts holding any one baffle plate to the former plates could fail. If these bolts fail, then the baffle plates would be held in place in two ways: (1) the baffle-to-baffle edge bolts and the corner edge bracket baffle to former bolts and (2) the overlapping configuration of the plates as seen in Figure 2. The only inspection that can be performed on the edge bolts is visual (and in some locations even visual inspection is not possible), but the lack of damage observed in the visual inspections of the baffle-to-baffle edge bolts and lack of baffle-jetting evidence suggests that most of these bolts can be assumed to continue to perform their intended functions. Larger baffle plates are either fully or partially restrained due to the overlapping of other baffle plates. To be conservative, only partial credit is taken for the edge bolts or the corner edge bracket baffle to former bolts. This is a conservative extrapolation of the condition found at Salem Unit 1, which identified a large number of UT indications on baffle-former bolts in one quadrant, but identified no visual indications of damage to baffle-to-baffle edge bolts.

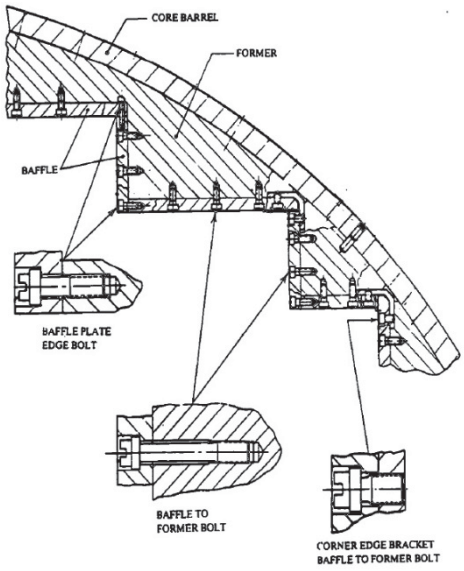

Figure 2

Leak-before-break (LBB) technology has been successfully applied to all U.S. Westinghouse NSSS plants for the primary loop piping. Many of the U.S. Westinghouse NSSS plants have also successfully applied LBB to the large branch lines connected to the reactor coolant system (RCS); e.g., the pressurizer surge line, residual heat removal (RHR) lines, accumulator lines (14, 12, and 10 inches), and 6-inch safety injection (SI) lines. Based on the LBB analyses already licensed, it is Westinghouse's engineering judgment that LBB can be successfully applied to the pressurizer surge line, RHR lines, accumulator lines, and 6-inch SI lines for all operating U.S. Westinghouse NSSS reactors including 2-, 3- and 4-loop plants. Therefore, the 10 CFR 21 evaluation for the 4-loop downflow plants considered a 4-inch line break as the initiating event.

Line Break Blowdown Forces

Westinghouse performed a hydraulic forces analysis to generate reactor vessel (RV) loss-of-coolant accident (LOCA) forces and baffle plate pressure differentials for a Westinghouse NSSS 4-loop reactor with a downflow barrel-former region. A review of the branch lines for Westinghouse NSSS 4-loop reactors with downflow barrel-former regions identified that the limiting branch line break would be a 4-inch Schedule 120 break, located in the cold leg at approximately 8.78 feet from the RV inlet nozzle elbow. Potential hot leg and pump suction leg branch line breaks were also reviewed and were determined to be less limiting when compared to the 4-inch cold leg break. The limiting RCS operating conditions for the 4-loop downflow plants at 100% power were reviewed and shown to be adequately represented in the analysis. The analysis used the methodology that is typically used for determining the licensing basis LOCA loads and for analyses that determine acceptable baffle-former bolting patterns.

Reactor Internals Structural Assessment
Westinghouse developed a conservative assessment of the structural effects that a degraded baffle-former assembly could have on the reactor internals components and fuel assemblies. For conservatism, the assessment assumed that all of the baffle-former bolts holding any one baffle plate to the former plates were degraded. This assumption bounds the number of baffle-former bolts identified as degraded at Indian Point Unit 2 and is more concentrated (i.e., clustered) than those identified at Salem Unit 1.

With respect to the structural impact of a degraded baffle-former assembly, the main areas of interest are related to the fuel assemblies, internals components/assemblies that could interact directly with the baffle plates, and internals components/assemblies that could be impacted by changes in stiffness resulting from the degraded condition. Westinghouse reviewed the structural impacts on the baffle-former-barrel assembly, fuel assembly alignment pins, and core barrel and determined that the degradation of the baffle-former bolts would not impact the safety-related functions of these components. The impacts on core plate motions and LOCA hydraulic loads were also reviewed and determined to be acceptable.

Seismic Considerations
The evaluation considered seismic conditions and determined that for baffle-former bolt degradation, seismic effects are bounded by the LOCA effects. When a baffle plate is loose, there is no significant mechanism to excite the plate. There will be minor pressure fluctuations in the baffle-former-barrel region due to the relative motion of components; however, these pressure fluctuations have been shown to be very small. Therefore, any pressure results are bounded by those analyzed in the LOCA analysis. Vertical seismic effects are not expected to be significant and intermittent impacts with the upper or lower core plates will not have significant inertia to damage the core plates when considering safe shutdown earthquake (SSE). The upper internals holddown forces have significant margin to accommodate any incidental impacts. Therefore, for the purposes of this evaluation, seismic effects were considered decoupled from a LOCA event.

Fuel Assemblies
The evaluation considered conservative fuel grid impact loads for 4-loop Westinghouse NSSS downflow plants. For these plants, the fuel assemblies that contain rod control cluster assemblies (RCCAs) were evaluated to determine any core locations where grid impact loads exceeded grid strengths, potentially leading to reduced flow areas and degraded coolability. This evaluation concluded that there is a potential for grid deformation on the core periphery, and that there is also a potential for limited deformation in the inboard fuel assemblies. The assessment of these potential impacts to core coolability is discussed in the "LOCA Assessment" section that follows.

Secondly, all RCCAs were evaluated and concluded to demonstrate complete RCCA insertion into the fuel assemblies at all RCCA locations in these plant designs considering the potential grid deformation discussed above. RCCAs were evaluated to demonstrate thimble tube structural integrity, flexibility of the control rodlets, and insignificant distortion of the grid.

Safety Analysis
The impacts of baffle-former bolt degradations were also evaluated generically for the impact on the plant-specific safety analyses contained in the Final Safety Analysis Report. The results of these evaluations are discussed in the following sections.

Non-LOCA Safety Analyses
The postulated baffle-former bolt failures could result in deformation of the baffle plates, and increase the total bypass flow by 1.40% during normal steady-state operation. The increase in core bypass flow would potentially result in small changes to the core average and core outlet temperatures.

For non-LOCA events, the key impact of increased bypass flow is a reduction in core flow, which can negatively impact the departure from nucleate boiling (DNB) margin. For the impacted plants for which DNB is the safety analysis acceptance criterion, it has been confirmed that DNB ratio (DNBR) margin is available to offset the penalty associated with the reduced core flow, such that DNB conclusions for the applicable licensing basis DNB analyses remain valid. Therefore, the current overtemperature ΔT and overpower ΔT reactor trip setpoints assumed in the licensing basis analyses remain valid.

For non-DNB events, or events for which other safety analysis acceptance criteria are used for the impacted plants, the effects of the reduced core flow have been evaluated with respect to the applicable safety analysis acceptance criteria. The primary/secondary overpressure and pressurizer overfill safety analysis acceptance criteria continue to be met for the impacted plants with the postulated increased bypass flow. Also, the increased bypass flow has little or no impact on the events that use fuel melt limits, peak clad temperature, and margin-to-hot-leg saturation as the safety analysis acceptance criteria.

LOCA Assessment
The effects of the potentially degraded baffle-former bolts were assessed considering a postulated LOCA. The LOCA assessment addressed the LOCA hydraulic forces, large break (LB) and small break (SB) LOCA, and LOCA long-term cooling analyses.

An increase in bypass flow of 1.4% during normal operation is insignificant to the LOCA hydraulic forces analysis; therefore, the current analyses supporting operation of the affected plants are still valid.

Considering the LBLOCA and the current available margin to the 2200°F cladding temperature limit, and margins in the Westinghouse LOCA evaluation models, core cooling is maintained for the predicted grid deformation at the affected plants as it relates to LBLOCA analyses.

The SBLOCA event is a low velocity, quasi-stratified transient, and the assumed increase in core bypass flow of 1.4% is expected to have little impact on the SBLOCA thermal-hydraulic response. The conservative assumptions of the baffle-former bolt degradation, previously discussed, result in some fuel assembly locations experiencing grid deformation during the postulated SBLOCA. Some of the fuel assemblies with deformed grids contain RCCAs. It has been confirmed that control rod insertability is maintained; therefore, the standard SBLOCA analysis methodology assumption that control rods insert remains valid, thus ensuring that the core is subcritical. Additional margin is also available with respect to core subcriticality due to the injection of highly borated water for the duration of the event following a postulated SBLOCA, but this was not credited in the analysis.

As stated previously in the discussions on the fuel assemblies, there is potential for grid deformation on the core periphery, as well as potential for limited deformation in the inboard fuel assemblies. A core coolable geometry is maintained for assemblies on the core periphery with grid deformation; this is based on taking credit for the low power generation in the peripheral assemblies, and the observation that any flow redistribution which may occur would tend to benefit the inboard assemblies.

Considering that some inboard fuel assemblies could also experience grid deformation, the result would be grid area reduction/single cell flow area reduction. However, the overall core flow area is unchanged with only a small difference in cooling geometry. Specifically, the geometry continues to be vertical tube bundles, only now with unsymmetrical local grid loss coefficients; thus, the overall geometry to be cooled

is principally unchanged. Therefore, it is concluded that the increase in core bypass and the amount and location of the grid deformation would have little impact on the SBLOCA transient. Based on this, and considering the available margin to the cladding temperature limit and the margins in the SBLOCA evaluation model, core cooling is maintained for the grid deformation due to the potentially degraded baffle-former bolts.

Long-term cooling was reviewed with respect to fuel assembly grid deformation. Grid deformation, should it occur, could reduce the rod-to-rod spacing; however, significant margins in the licensing basis analyses exist. Slight changes in core geometry such as those resulting from fuel assembly grid deformation would not significantly affect long-term heat removal as coolant flow velocities are low, fuel assembly crossflow would not be fully impeded, and the overall core flow area would be preserved. Baffle-former bolt degradation would not increase the core decay heat. Therefore, the ability to remove decay heat in the long-term after a LOCA would not be compromised by baffle-former bolt degradation.

Therefore, the ability to cool the core, maintain reactor shutdown, and remove decay heat in the long-term after a LOCA, would not be compromised by baffle-former bolt degradation.

Loose Parts Assessment
Degraded baffle-former bolts have the potential to create loose parts in the RCS. Operation with loose parts in the RV and RCS primary side is undesirable; however, Westinghouse has performed primary side loose parts evaluations for at least 40 plants for a broad variety of loose parts. These evaluations included loose parts assessments performed specifically for baffle bolts and lock bars that evaluated their potential impacts on the reactor internals, fuel, and RCCA insertion, in addition to components and systems including the steam generators, reactor coolant pumps, RCS piping, pressurizer, and auxiliary equipment and systems (to address postulated transport of the lock bars or an unrecovered bolt head being carried by flow out of the reactor). All of these prior evaluations concluded that the presence of baffle bolts and lock bars as primary side loose parts would not adversely impact safe operation of a plant, and would not adversely impact the capability to shut down the reactor or indefinitely maintain it in cold shutdown.

If fretting between a detached bolt head or lock bar and a fuel rod were to occur, it could result in the potential for leaking fuel rods, which would cause an increase in coolant activity. The consequence of loose part-related fretting is reduced by monitoring the coolant activity and taking appropriate action if elevated levels are reached. Compliance with plant Technical Specifications for RCS specific activity will limit the activity in the reactor coolant so that the licensing basis accident analyses meet the dose acceptance limits.

SAFETY SIGNIFICANCE

The baffle-former assembly, containing baffle-former bolts, is a basic component and is part of the reactor internals structure. The function of the baffle-former assembly is to maintain the fuel assembly structural integrity to ensure that the control rods insert, maintain a coolable core geometry, and ensure a core configuration that supports long-term reactor shutdown. Based on the evaluation performed to address degraded baffle-former bolts, this situation does not represent a potential defect, this issue does not create a substantial safety hazard (SSH) if left uncorrected, and continued operation of the unit(s) in consideration of this issue is acceptable.

AFFECTED PLANTS

All Westinghouse-designed NSSS plants with baffle-former bolts and CE-designed plants with bolted core shrouds are potentially affected by this issue. The Westinghouse **AP1000**® [2] plant design does not utilize baffle-former bolts and is not affected by this issue.

NRC AWARENESS

The NRC has been made aware of the most recent baffle-former bolt issues through various means including, but not limited to:

- NRC Event Report No. 51902: Salem Unit 1 report pursuant to 10 CFR 50.72(b)(3)(ii)(B), regarding anomalies identified on baffle-former bolts while conducting a scheduled visual inspection, May 3, 2016.
- NRC Event Report No. 51829: Indian Point Unit 2 report pursuant to 10 CFR 50.72(b)(3)(ii)(B), regarding anomalies identified on baffle-former bolts while conducting a scheduled visual inspection, March 29, 2016.
- Westinghouse senior management has discussed the most recent baffle-former bolt OE with the NRC.

RECOMMENDED ACTIONS

As previously discussed in Westinghouse Technical Bulletin TB-12-5, the most likely cause of clustered baffle-former bolt degradation is IASCC crack initiation and growth, followed by propagation to adjacent bolts. IASCC susceptibility is a strong function of stress and radiation dose. All plants with baffle-former bolts are potentially susceptible to baffle-former bolt failures due to IASCC; however, significant clustered failures have not been observed in plants with upflow and converted upflow reactor designs. The upflow design results in a reduced pressure drop across the baffle plate and a correspondingly lower primary stress on the baffle-former bolt, resulting in a lower driving force for IASCC, as well as subsequent adjacent baffle-former bolt degradation. Therefore, the downflow reactor design configuration is considered a primary risk factor for susceptibility to clustered baffle-former bolt degradation.

Other factors contributing to IASCC susceptibility are baffle-former bolt design features, such as the head-to-shank radius and the material. At this time it is difficult to quantify the relative importance of these contributing factors because there is insufficient information available to provide a direct comparison. Therefore, primary load (pressure and bolt/plate spacing), bolt design, and material are used to establish the tiers of potential susceptibility based on qualitative arguments. Westinghouse is actively engaged in the development of a predictive model of baffle-former bolt behavior that will address both apparently random and clustered failures. The model combines OE with structural models of the bolt failure process to provide a technical basis for a more quantitative assessment of the potential susceptibility of each tier. The predictive model will be used as the basis for more precisely targeted recommendations for each tier of susceptibility.

Plants in each tier have been identified based on available Westinghouse records. Each utility should verify classification into the appropriate tier prior to acting on these recommendations.

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Tier 1: 4-loop plants currently operating in downflow

As previously discussed, the downflow configuration creates the pressure differential across the baffle plates, which is believed to be the driving force for the clustering of broken baffle-former bolts. This differential, which occurs at the top of the baffle and corresponds to the pressure drop through the core, is expected to be the largest in the 4-loop downflow plants. All current 4-loop downflow plants have similar effective full-power years (EFPY) of operation in the range of approximately 25-31 years; therefore, the length of operation is not a differentiator for susceptibility to IASCC for these plants. Furthermore, as compared to the original baffle-former bolts fabricated from Type 316 stainless steel for this grouping of plants, Type 347 stainless steel baffle-former bolts generally have a sharper radius at the head-to-shank and a shorter bolt shank. Although that data does not indicate a sharp difference in the IASCC susceptibility of these two stainless steel alloys, the differences in bolt design produce a higher stress concentration factor at the bolt head-to-shank transition for Type 347 baffle-former bolts. Therefore, Tier 1 can be broken into two sub-tiers based on the baffle-former bolt design:

Tier 1a: 4-loop plants currently operating in a downflow configuration with Type 347 stainless steel baffle-former bolts. The three plants with observed clusters of baffle-former bolt failures fall into Tier 1a. This group is expected to have the highest potential for failure patterns similar to those observed at D.C. Cook Unit 2, Indian Point Unit 2, and Salem Unit 1.

D.C. Cook 1 & 2	Diablo Canyon 1	Indian Point 2 & 3	Salem 1 & 2

Tier 1b: 4-loop plants currently operating in downflow with originally-designed Type 316 stainless steel baffle-former bolts. There are only two units at one site that fall into Tier 1b. MRP-227 volumetric inspections of baffle-former bolts have not been performed in these units. It is difficult to attribute any potential benefit directly to the material properties because there are no direct laboratory comparisons of IASCC initiations in Type 316 and Type 347 materials and only two Westinghouse NSSS 3-loop plants have completed UT inspections of Type 316 baffle-former bolts. Although those two plants did not report any indications, French plants have reported failures in Type 316L baffle-former bolts. The potential benefits of Type 316 bolting may be attributed to differences in design and manufacturing of the Type 316 baffle-former bolts that make them more resistant to IASCC. The design used for the Type 316 baffle-former bolts produces a lower stress concentration under the bolt head. While there is a slightly lower priority on the inspection of these Tier 1b units, the potential exists for failure patterns similar to those at D.C. Cook Unit 2, Indian Point Unit 2, and Salem Unit 1. The 4-loop downflow plants with Type 316 bolting were included in the Tier 1 inspection requirements because they were considered to be at risk for bolt failures.

Sequoyah 1 & 2

Tier 2: Remaining 2-loop and 3-loop downflow plants
Although a pressure differential across the baffle plates is expected in all downflow plants, the magnitude of the differential is smaller in the 2- and 3-loop downflow plants. These plants also have a larger number of bolts per square inch of baffle plate, thereby further reducing the pressure induced stress on the bolts. As with the 4-loop plants, the bolt design corresponding to the Type 347 stainless steel is expected to have a higher stress concentration at the head-to-shank transition. Tier 2 has been broken down into three sub-tiers based on reactor design and bolt material:

Tier 2a: 2-loop plants currently operating in a downflow configuration with Type 347 stainless steel baffle-former bolts. All three of the 2-loop downflow plants in the U.S. have completed MRP-227 inspections of baffle bolts. As compared to the 3-loop plants, UT examinations of the 2-loop plants have identified a higher number of failures, although significant clustering of these failures has not been seen to date at these plants.

Beznau 1 & 2	Prairie Island 1 & 2	R.E. Ginna

Tier 2b: 3-loop plants currently operating in a downflow configuration with Type 347 stainless steel baffle-former bolts. Three-of-the-five Tier 2b plants have completed MRP-227 inspections, with 9 or less indications; another one of these plants has completed a partial inspection (305 of 1088) with no indications.

H.B. Robinson 2	Surry 1 & 2	Turkey Point 3 & 4

Tier 2c: 2- and 3-loop plants currently operating in a downflow configuration with Type 316 stainless steel baffle-former bolts.

2-Loop	3-Loop
Angra 1	Ringhals 3
Doel 1 & 2	

Tier 3: Converted upflow plants
Plants that were originally operated in the downflow configuration can be converted to upflow through field modifications to the core barrel and former plates. These modifications have been shown to reduce the incidence of baffle jetting damage to the fuel. The upflow conversion also reduces the bolt loads due to pressure differentials across the baffle under both normal operating and expected faulted conditions. Although the upflow conversion is expected to have a positive impact on bolting life, there remains a potential for accelerated degradation during the original period of operation with the downflow configuration. The overall condition of these converted upflow plants is better than equivalent plants operated continuously in a downflow configuration. Longer time operating in the downflow configuration is postulated to correspond to a higher potential for baffle-former bolt degradation. Point Beach Units 1 and 2 have Type 347 bolts; the remaining plants have Type 316 bolts.

2-Loop	3-Loop	4-Loop
Kori 1	Almaraz 1	Diablo Canyon 2
Krsko	Beaver Valley 1	McGuire 1 & 2
Point Beach 1 & 2	Farley 1 & 2	Ohi 1 & 2
	North Anna 1 & 2	
	Ringhals 2	
	Takahama 1	
	Tihange 1	
	V.C. Summer 1	

Tier 4: All plants continuously operated in upflow configuration
This tier includes all plant designs and bolt configurations. All plants in this tier have Type 316 bolts.

2-Loop	3-Loop	4-Loop	CE-designed plants
Kori 2	Almaraz 2	A.W. Vogtle 1 & 2	Fort Calhoun 1
	Asco 1 & 2	Braidwood 1 & 2	Palisades 1
	Beaver Valley 2	Byron 1 & 2	
	Doel 4	Callaway 1	
	Hanbit 1 & 2	Catawba 1 & 2	
	Kori 3 & 4	Comanche Peak 1 & 2	
	Maanshan 1 & 2	Millstone 3	
	Ringhals 4	Seabrook 1	
	Shearon Harris 1	Sizewell B	
	Tihange 3	South Texas 1 & 2	
	Vandellos 2	Watts Bar 1 & 2	
		Wolf Creek	

General Recommendations for all Tiers
If visually damaged baffle-former bolts or lock bars are detected, it is recommended that the fuel assemblies that were adjacent to the baffle in the previous cycle, and are scheduled for use in the next cycle, be inspected for fretting wear on the face that was adjacent to the baffle.

It is recommended that the plant continues to follow the current MRP-227 guidelines and implement any revisions to the MRP-227 recommendations.

Recommendations by Tier
Tier 1a: It is recommended that the plant completes a UT volumetric inspection of the baffle-former bolts at the next scheduled refueling outage. In preparation for this inspection, the plant should consider developing an acceptable bolting pattern analysis and be prepared to replace any baffle-former bolts with visible damage or UT indications prior to starting the unit up from the refueling outage.

While this issue does not represent a SSH, recent OE has shown a high likelihood that the plants in this tier may have to replace bolts. As such, the plants in this tier should consider other mitigation strategies. These could include an upflow conversion and preemptive bolt replacements.

Performing an upflow conversion and replacing baffle-former bolts to a specified acceptable bolting pattern maintains design requirements while minimizing re-inspections to monitor the progression of this issue.

Tier 1b: It is recommended that the plant completes a VT3 inspection of the baffle-former bolts at the next scheduled refueling outage. If any visual indications are found, it is recommended that the plant completes a UT volumetric inspection of the baffle-former bolts. The plant should continue to monitor baffle-former bolt OE to determine if a more rigorous inspection is warranted.

If no visual indications are found, it is recommended that the plant completes a UT volumetric inspection of the baffle-former bolts prior to the completion of the second refueling outage after the issuance of this NSAL.

Tier 2a, 2b, and 2c: Tier 2 plants that have previously completed UT inspections should review the inspection records to identify any indication of the onset of clustering in the bolt failure patterns before

the next scheduled refueling outage. Clustering is defined as 3 or more adjacent bolts or a total number of failures in a single baffle plate greater than 40% of the total number of bolts on that baffle plate. Any indication of clustering should result in the consideration of an accelerated re-inspection schedule.

Tier 3: 4-loop plants that have operated in a downflow configuration for more than 20 calendar years should evaluate the need to perform a UT volumetric inspection of baffle-former bolts on an accelerated schedule considering the plant-specific condition and design parameters compared to the Tier 1a plants. All other Tier 3 plants should follow the guidance in the "General Recommendations for all Tiers" section.

Tier 4: These plants should follow the guidance in the "General Recommendations for all Tiers" section.

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