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ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject:
Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Notification of Deviation from MRP-227, Revision 1-A, Section 7.5 Regarding the Disposition of Examination Results for Thermal Shield Flexures

In accordance with Nuclear Energy Institute (NEI) 03-08, "Guideline for the Management of Materials Issues," Revision 4, Vistra Operations Company LLC is notifying the Nuclear Regulatory Commission (NRC) of a deviation from Electric Power Research Institute (EPRI) Technical Report MRP-227, Revision 1-A, Section 7.5 that has been processed for Beaver Valley Power Station, Unit No. 1 (BVPS-1).

The following is a summary of the technical justification for deviating from the required methods for dispositioning examination results identified in Section 7.5 of MRP-227, Revision 1-A. The deviation is specifically related to the BVPS-1 thermal shield flexures that were examined during the spring 2024 refueling outage and the fall 2025 refueling outage. This information is being provided for information only, and Vistra Operations Company LLC is not requesting any action from the NRC staff.

During the spring 2024 refueling outage (1R29) MRP-227, Revision 1-A, reactor internal examinations, five of the six BVPS-1 thermal shield flexures were found to be fractured. Repairs of the BVPS-1 flexures were planned for the fall 2025 refueling outage (1R30), but it was necessary to defer the repairs to allow metallurgical testing and root cause evaluations to be completed for a similar repair that had been performed at another site which had subsequently degraded. The BVPS-1 reactor internals were examined during the fall 2025 refueling outage (1R30) with a combination of visual and volumetric exams and it was concluded that the condition remained unchanged from the spring 2024 (1R29) inspection results. Five of six thermal shield flexures remained fractured, thus failing the acceptance criteria in Table 5-3 of MRP 227, Revision 1-A. The flexure examination results were entered into the corrective action program. Due to the complexity of the evaluation, the example evaluation methodologies in Section 7.5 of MRP-227, Revision 1-A, were not viable or appropriate.

The Pressurized Water Reactor Owner's Group (PWROG) had previously performed a generic safety assessment for thermal shield support block cap screw and flexure degradation (PWROG-21021 Revision 2). This generic safety assessment concluded that "the degradation of any number of thermal shield support block cap screws or thermal shield flexures is not a safety concern for one cycle of operation following the discovery" if the "[d]egradation is detected during an MRP-227 inspection, which includes inspection of the core barrel welds, and the core barrel welds show no indications of core barrel cracking." In accordance with this guidance, MRP-227 examinations of the BVPS-1 core barrel were performed during the fall 2025 refueling outage, with no indications of cracking identified.

Additionally, Westinghouse reviewed the spring 2024 and fall 2025 BVPS-1 flexure examination results and provided an assessment of the degradation. The evaluation considered the risk presented by potential core barrel cracking, complete loss of thermal shield structural support, potential cascading degradation to the thermal shield support block bolts and baffle bolts, and the risk of introducing loose parts into the reactor coolant system. Similar assessments have been performed for several other PWR sites. The Westinghouse evaluation concluded, "(T)here are no nuclear safety concerns for one additional cycle of operation posed by the degradation observed at BVPS-1 during 1R30. Therefore, Beaver Valley Unit 1 can safely return to operation for one cycle following 1R30."

The deviation disposition report is enclosed and will be in effect until the next BVPS-1 refueling outage in the spring of 2027. Vistra Operations Company LLC is currently evaluating viable repair options and developing a plan to implement thermal shield structure repairs at BVPS-1.

There are no regulatory commitments contained in this submittal. If there are any questions, or if additional information is required, please contact Jack Hicks, Senior Manager, Licensing, at (254) 897-6725 or jack.hicks@vistracorp.com.

Sincerely,

Robert Kristophel

Enclosure:
Deviation Disposition Report

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

Enclosure
L-26-093

Deviation Disposition Report
(10 pages follow)

Utility: Vistra **Site/Unit:** Beaver Valley, Unit #1 **Condition Report:** CR-2025-0878

Applicable Issue Program (IP) Guideline(s): MRP-227-1-A Section 7.5 (NEI 03-08 "Needed" guidance)

Brief Description of Deviation:

This deviation disposition report (DDR) documents the technical basis for deviating from NEI 03-08 "Needed" guidance for dispositioning thermal shield flexure degradation at Beaver Valley Unit 1. The activities and evaluations described in this DDR constitute an acceptable alternative to performing the NEI 03-08 'Needed' guidance in MRP-227 Section 7.5, which requires the use of an NRC-approved evaluation methodology to disposition unsatisfactory inspection results.

During the 1R29 refueling outage (Spring 2024) MRP-227 Revision 1-A reactor internal examinations, it was determined that 5 of the 6 BVPS-1 thermal shield flexures were fractured. Repairs of the BVPS-1 flexures were planned for the 1R30 (Fall 2025) refueling outage, but it became necessary to defer the repairs to allow metallurgical testing and root cause evaluations to be completed for a similar repair that had been performed at another site and that had subsequently degraded. The BVPS-1 reactor internals were examined during the 1R30 (Fall 2025) refueling outage with a combination of visual and volumetric exams, and it was concluded that the condition remained unchanged from the 1R29 (Spring 2024) inspection results. 5 of 6 thermal shield flexures remained fractured, thus failing the acceptance criteria in Table 5-3 of MRP-227 Revision 1-A. Section 7.5 of the MRP document includes the following "Needed" NEI 03-08 guidance; "Engineering evaluations used to disposition an examination result that does not meet the examination acceptance criteria in Section 5, shall be conducted in accordance with NRC approved evaluation methods (i.e., ASME Code Section XI, PWR Owners Group topical report WCAP-17096-NP-A or equivalent method)."

The BVPS-1 Fall 2025 flexure examination results were entered into the plant's corrective action program and due to the complexity of the evaluation, it was determined that the example evaluation methodologies in Section 7.5 of MRP-227 revision 1-A were not viable or appropriate. The Pressurized Water Reactor Owner's Group (PWROG) had previously performed a generic safety assessment for thermal shield support block cap screw and flexure degradation (PWROG-21021 Rev 2). This generic safety assessment concluded that "the degradation of any number of thermal shield support block cap screws or thermal shield flexures is not a safety concern for one cycle of operation following the discovery" if the "[d]egradation is detected during an MRP-227 inspection, which includes inspection of the core barrel welds, and the core barrel welds show no indications of core barrel cracking." Additionally, Westinghouse reviewed the examination results and provided an assessment of the degradation (documented in DLW-RV010-TM-CG-000006). Similar assessments had been performed for several other PWR sites. The Westinghouse evaluation concluded, "(T)here are no nuclear safety concerns for one additional cycle of operation posed by the degradation observed at Beaver Valley Unit 1 during 1R30. Therefore, Beaver Valley Unit 1 can safely return to operation for one cycle following 1R30." The Evaluation section of this DDR provides a summary of the activities and evaluations that constitute an acceptable alternative to performing the NEI-03-08 'Needed' guidance in MRP-227 Section 7.5.

Position	Printed Name/Signature	Date
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Director - Site Engineering	Rob Klindworth Rob Klindworth	6/23/26
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V. P. - Fleet Engineering	Jay Lloyd	06/24/26

Evaluation and Technical Justification

Applicable IP Guideline(s)

MRP-227-1-A Section 7.5

Time Frame Deviation will be in Effect

1R30 (Fall 2025) refueling outage through the 1R31 (Spring 2027) refueling outage. The 1R31 refueling outage is currently scheduled to occur in Spring 2027.

Deviation Description

During the 1R29 refueling outage (Spring 2024) MRP-227 Revision 1-A reactor internals examinations, it was determined that 5 of the 6 BVPS-1 thermal shield flexures were fractured. This condition was verified during the 1R30 (Fall 2025) refueling outage, thus failing the acceptance criteria in Table 5-3 of MRP-227 Revision 1-A. Section 7.5 of the MRP document includes the following "Needed" NEI 03-08 guidance; "Engineering evaluations used to disposition an examination result that does not meet the examination acceptance criteria in Section 5, shall be conducted in accordance with NRC approved evaluation methods (i.e., ASME Code Section XI, PWR Owners Group topical report WCAP-17096-NP-A or equivalent method)."

Section 3.4.1 of the NRC safety evaluation for MRP-227 Revision 1-A identifies the actions required when the evaluation methods described in Section 7.5 are not viable or appropriate for dispositioning examination results: "Section 6.0 [of MRP-227] also states that the NRC-accepted report WCAP-17096-NP-A provides approved methods for evaluating examination results that do not meet the examination acceptance criteria in Section 5, and that the WCAP-17096-NP-A methods are to be followed in accordance with the "Needed" requirement of 7.5.

The "Needed" guidance of Section 7.5 specifies that NRC-accepted evaluation methods (ASME Section XI, WCAP-17096-NP or equivalent) must be used for dispositioning examination results that do not meet the examination acceptance criteria in Section 5. The staff finds this needed guidance is sufficient to ensure that NRC-accepted evaluation methods are used, or if not, the NEI 03-08 deviation process will be followed." Based on this statement from the NRC safety evaluation for MRP-227 Revision 1-A, it was concluded that an NEI 03-08 deviation must be developed and submitted for the BVPS-1 thermal shield flexure degradation.

The BVPS-1 Fall 2025 vessel internals examination results were evaluated to characterize the risks associated with operating from 1R30 to 1R31 with the currently identified levels of thermal shield support degradation. The evaluation considered the conclusions reached in the generic PWROG safety assessment (documented in PWROG-21021-NP) and the Westinghouse plant-specific technical evaluation for BVPS-1 (documented in DLW-RV010-TM-CG-000006 and DLW-RV010-TM-CA-000003). Both documents contain proprietary information. This deviation disposition report provides a non-proprietary summary of the evaluations that were performed to determine the technical acceptability of continued operation with degraded flexures at BVPS-1 until 1R31 in Spring 2027.

Detailed Evaluation & Technical Acceptability

Examinations performed at BVPS-1 during the 1R30 refueling outage (Fall 2025) verified that 2 of the 12 TSSB bolts were degraded, and 5 of the 6 thermal shield flexures were degraded (no change from the previous examinations in Spring 2024). From a nuclear safety perspective, it was necessary to address three main areas of concern: 1) degraded flexures and TSSB bolts, 2) potentially degraded baffle bolts, and 3) concerns associated with loose parts in the reactor coolant system (RCS). Each area of concern is addressed in the following assessment.

1. Degraded Flexures and TSSB Bolts

There are three nuclear safety concerns associated with continued plant operation with degraded flexures and degraded TSSB bolts: 1) risk of core barrel cracking, 2) loss of structural support of the thermal shield (leading to separation and drop of the thermal shield), and 3) the thermal/hydraulic impacts of the open TSSB bolt holes. Each safety concern is discussed below.

1.A Core Barrel Cracking

- Regarding the nuclear safety risks presented by core barrel cracking, it should be noted that core barrel crack growth rates are generally expected to be slow under normal plant operating loads. Follow-up inspections at another three-loop Westinghouse thermal shield plant in Fall 2024 identified no additional crack growth in existing core barrel weld cracks over one cycle of operation in a plant with structurally sound thermal shield flexures. However, the worst case (bounding) scenario associated with core barrel cracking is crack growth driven by thermal shield vibration loads that ultimately leads to core barrel separation. Since the degraded flexures and TSSB bolts at BVPS-1 cause increased vibration loading to be transmitted from the thermal shield to the core barrel, operation in this condition increases the subsequent risk of core barrel cracking. PWROG-21021-NP affirms that even in the event of core barrel separation resulting from weld cracking, the structural integrity of the reactor pressure vessel and the core alignment functions of the reactor vessel internals would be maintained. Safe shutdown (control rod insertion and core coolabiltiy) could be achieved as long as the core barrel separation was detected and the plant was subsequently shutdown. However, during normal plant operation it is likely that a core barrel separation event would go un-detected for a significant amount of time. Long-term operation with a separated core barrel has not been fully analyzed, and since it is likely that a core barrel separation event would not be immediately detected and addressed during normal plant operation, from a nuclear safety perspective it was necessary to verify that the BVPS-1 core barrel will not separate over the next cycle of operation due to the thermal shield support degradation.

- To verify that the BVPS-1 core barrel will not experience cracking that could lead to separation of the core barrel over the next 18 months of operation, the areas of the core barrel susceptible to cracking were visually examined using EVT-1 examination methods during the Fall 2025 (1R30) refueling outage. The core barrel upper girth weld (OD and ID) was examined with no recordable indications identified. Additionally, both the OD and the ID surfaces of the core barrel around the thermal shield support blocks were visually examined, with no indications of degradation identified. These areas of the core barrel were chosen for inspection because they are in the direct load path for thermal shield vibration loading since the added vibration loads resulting from flexure degradation are transferred from the thermal shield support block elevation through the core barrel to the reactor vessel flange. Additionally, these areas of examination were chosen because core barrel cracking OE from similar plants identified that the upper girth weld is the most likely location for core barrel cracking to occur.
- PWROG-21021-NP Rev 2 states that "...if no core barrel cracking is found during the MRP-227 inspection of the core barrel welds, it is not expected that core barrel separation will occur within one additional fuel cycle even with the core barrel experiencing increased loading as the result of degraded thermal shield supports. This is because the loads reacted by the core barrel are expected to be bounded by the faulted loads evaluated in WCAP-17684-P which demonstrated that a core barrel can survive a faulted event with existing cracking. Additionally, previous analysis of core barrel flaws has demonstrated that flaw growth is predicted to be negligible over one fuel cycle." Since the BVPS-1 core barrel showed no indications of cracking during the Fall 2025 examinations, it is concluded that the structural integrity of the BVPS-1 core barrel is maintained and that it is acceptable to operate for an additional cycle until the Spring of 2027. PWROG-21021-NP concluded that "If no core barrel cracking is found during the aforementioned inspections...the guidance in Section 4.4 of this report can be referenced directly as the technical basis for safely operating an additional cycle in the degraded state."

1.B Loss of Thermal Shield Structural Support

Regarding the loss of structural support of the thermal shield, the worst case (bounding) scenario would be the complete failure of the thermal shield support system and the subsequent detachment of the thermal shield from the core barrel. Based on the design of the BVPS-1 thermal shield support structure and operating experience from other sites, complete failure of the support structure and detachment of the thermal shield is considered to be highly unlikely. Significant additional thermal shield support degradation would need to occur before the thermal shield could become disconnected from its support assembly. Even if all of the TSSB bolts and flexures were degraded (BVPS-1 currently has 10 structurally sound TSSB bolts and 1 structurally sound flexure), the 6 welded dowel pins would need to degrade, and the thermal shield would need to rotate about the core barrel to fall off of the 6 support blocks. Even though complete failure of the thermal shield support structure is considered to be highly unlikely, it remains the bounding condition for this degradation and as such was evaluated. The individual technical concerns associated with a failed thermal shield support structure are addressed below.

- A detached thermal shield could become misaligned and could cause localized decreased reactor coolant system flow (affecting core cooling). However, the movement of the thermal shield is limited by its interaction with the core barrel and the radial clevis inserts of the reactor vessel. Addressing the consequences of a dropped thermal shield, PWROG-21021-NP Rev 2 states: "The thermal shield is not expected to tilt any way to block the flow entering the lower plenum. The thermal shield will be closer to the lower plenum, which will cause an expansion closer to the lower plenum region; however, this is not expected to cause any adverse effects to the core bypass flow. The core will remain coolable while the thermal shield rests on the LRSS clevises." Since a dropped thermal shield will not block coolant flow into the lower plenum region, it is concluded that the core will remain coolable even with a detached thermal shield.
- A detached thermal shield would no longer serve its passive function of reducing neutron fluence. PWROG-21021-NP states: "The thermal shield is a passive structure whose function is to reduce the neutron fluence on the reactor pressure vessel in the core region." Since the thermal shield drop is limited by the lower radial restraints, the passive function of the thermal shield would still be maintained for the majority of the core height. The space previously occupied by the dropped thermal shield would be immediately replaced with water which also provides neutron shielding for the reactor vessel even though it is not as efficient as stainless steel. Additionally, reactor vessel embrittlement concerns are a long-term issue. Operating the reactor temporarily with the thermal shield lower than the design position may result in the need to revise the reactor vessel integrity calculations if such an event were to occur but such a condition poses no risk to immediate plant safety as there is currently significant margin in the BVPS-1 reactor vessel integrity calculations. BVPS-1 is currently at approximately 37.5 EFPY and the current pressure and temperature limits in Section 5.2 of the BVPS-1 license requirements manual are applicable to 50 EFPY.
- A detached thermal shield may damage other components as it falls. The damage could affect the integrity of the reactor coolant pressure boundary or it could affect the alignment and insertability of the control rods (and subsequently the ability to shutdown the reactor). As documented in PWROG-21021-NP Rev 2, Westinghouse evaluated the impact load of the thermal shield landing on the lower radial restraints by equating the kinetic energy of the thermal shield to the work done on all of the components during impact. PWROG-21021-NP concluded: "It was found that the stress intensity in the LRSS [Lower Radial Support System] clevis and reactor vessel shell junction was under ASME allowable stress intensities for faulted conditions. The load transferred to the reactor vessel supports was determined to be bounded by previously evaluated LOCA forces. In summary, the impact load caused by a detached thermal shield landing on the LRSS clevises is not expected to grossly damage the LRSS clevises, reactor pressure vessel shell, or the reactor pressure vessel supports. Therefore, it can be concluded that the impact of the detached thermal shield is not a safety concern since the primary pressure boundary integrity will be maintained and not impede the subsequent safe shutdown of the reactor."
- It is therefore concluded that even in the limiting condition of complete thermal shield support failure and subsequent detachment of the thermal shield, core geometry (cooling and control rod

insertability) will be maintained for BVPS-1. Additionally, the structural integrity of the reactor coolant pressure boundary will be maintained because the impact loads on the pressure vessel are below the ASME allowable stress intensities for faulted conditions.

1.C Thermal/Hydraulic Impact of Open TSSB Bolt Holes

Operating experience indicates that degraded thermal shield flexures could cause additional TSSB bolts to fail due to the added vibration loadings. The safety assessment in PWROG-21021-NP covered "the degradation of any number of thermal shield support block cap screws or thermal shield flexures..." so it is concluded that the safety assessment in this document bounds further degradation of TSSB bolts at BVPS-1 over the current operating cycle. Additionally, one of the two BVPS-1 TSSB bolts at the 195-degree support block is missing and the other is degraded. It was therefore necessary to perform an evaluation to determine the acceptability of the radiation impact, hydraulic impact, and thermal impact resulting from operation with open TSSB bolt holes. This evaluation was documented in Westinghouse letter DLW-RV010-TM-CG-000003 and it was concluded that continued operation with open TSSB bolt holes at the 195-degree location is acceptable considering the expected changes in radiation exposure, core bypass flow rates, jetting impingement on other reactor internals components, and thermal gradients.

2) Potential for Baffle Bolt Degradation

Industry operating experience indicates that the added vibration loads that result from thermal shield flexure degradation can increase failure rates for baffle bolts. The 1,088 BVPS-1 baffle bolts were volumetrically examined during the 1R28 (fall 2022) refueling outage. 12 non-testable bolts (due to fabrication surface geometry) were identified, indicating that BVPS-1 is not likely to be susceptible to significant baffle bolt degradation. Also, general visual examinations of the baffle bolts were performed during the 1R30 (Fall 2025) refueling outage with no degradation identified. However, because of the risk that flexure degradation could cause baffle bolts to degrade, it was necessary for the Westinghouse plant specific evaluation to address the potential risks associated with baffle bolt degradation.

The baffle bolts are part of the baffle-former structure that surrounds the nuclear fuel. The function of the baffle plates is to provide support to the fuel assemblies, direct reactor coolant flow though the core, and control bypass flow. The evaluation of baffle bolt degradation in the BVPS-1 plant-specific technical evaluation assumed the extreme condition that all of the baffle bolts on a single plate had failed and that the plate was no longer connected to the baffle formers. This level of degradation is well beyond that observed in the operating nuclear fleet and is not expected for plants with MRP-227 aging management programs. Even with the assumed advanced levels of baffle bolt degradation, it was determined that 1) the structural integrity of the remaining reactor internals assembly would be maintained, 2) control rod insertability would be acceptably maintained even with the potential for fuel assembly grid crush, and 3) core geometry (cooling) would be maintained.

3) Loose Parts Assessment

During the Fall 2025 (1R30) refueling outage, FME searches were performed in the lower head of the reactor vessel and the lower core plate of the core barrel. Additionally, visual and volumetric examinations were performed to understand the extent of the reactor vessel internals degradation. After the 1R30 FME search and reactor internals examination activities, the following loose parts potentially remain in the BVPS-1 reactor coolant system.

- 193° TSSB bolt lock bar
- 197° TSSB bolt partial shank
- 197° banana pin
- Lower left assembly cap screw, head, head/shank and threads, or head and shank/threads, and lock bar
- Upper right assembly cap screw, head, head/shank and threads, or head and shank/threads, and lock bar
- Flexure fracture surface fragments

The flexure and TSSB bolt dowel pin were not included in the loose parts assessment because it was determined that it is not credible that the flexure or the dowel pin would become loose parts. Since visual examinations identified that the flexures were fully separated, there is no longer a load path through the failed flexures and as a result further cracking that would cause the failed flexures to become loose parts is not deemed to be credible. Additionally, the cracked welds at the 195-degree dowel pin indicate that the dowel pin retained its interference fit in the core barrel which would keep the pin from falling out. The dowel pin weld remnants would also preclude the dowel pin from falling out. It is therefore concluded that neither the failed flexure remnants nor the TSSB dowel pin will become loose parts prior to Spring 2027.

An evaluation of potential loose parts generated by TSSB bolt degradation and thermal shield flexure degradation is documented in Section 4 of PWROG-21021-NP. Loose parts generated as a result of the TSSB bolt and flexure degradation could potentially cause impact damage to equipment, become wedged between components as gaps change due to thermal cycles, block coolant flow, cause wear between adjacent parts, and inhibit rod cluster control assembly (RCCA) control rods from fully inserting.

Regarding impact effects from the postulated loose parts, PWROG-21021-NP considered the safety issues associated with the impact of generated loose parts on the structural integrity of the reactor coolant pressure boundary and the structural integrity of the core support structures. Regarding the postulated loose parts from degraded TSSB bolts and degraded flexure bolts, PWROG-21021-NP concluded that "none of these postulated loose parts would be expected to cause significant damage due to impact with the RPV and internals components since the individual weight of each loose part is less than the maximum allowable loose part weight..."

Wedging effects from the postulated loose parts were addressed by reviewing the reactor internals structure to determine locations where loose parts could potentially become wedged between the reactor vessel and internals. Three locations were identified: 1) core barrel nozzle-to-vessel nozzle gap, 2) radial key-to-clevis insert gap, and 3) the secondary core support base plate-to-vessel lower head gap. PWROG-21021-NP concluded that the first two gaps were small enough that loose parts of significant size could not enter and as a result only insignificant loads would be applied to the subject components if wedging were to occur. For the secondary core support base plate-to-vessel lower head gap, it is possible for loose parts of significant size to enter the gap during cold shutdown conditions. However, PWROG-21021-NP identified that a loose part wedged in this area would "not produce enough force to yield the absorber column, uplift the core barrel flange from the vessel ledge, or overstress the reactor vessel head."

Regarding blocked flow effects from the postulated loose parts, PWROG-21021-NP identified that loose parts from the flexures and TSSB bolts would be carried to the lower plenum regions and would become trapped in the bottom part of the lower internals structure or fuel assembly debris nozzle filters. It was concluded that "The size of the loose parts would not cause enough blockage to significantly impact flow. Therefore, the effects of the loose object on flow blockage can be considered negligible, and there will be no negative impact on the RVI components. It can be concluded that potential for blocked flow from the loose parts generated from TSSB cap screw and flexure degradation is not a safety concern."

The postulated loose parts could cause wear to occur on the reactor vessel cladding and fuel assembly surfaces. It was concluded that the reactor vessel base metal corrosion rate is low as identified by operating experience and previous evaluations. Wear through the cladding would be detected during the reactor vessel ISI exams, so cladding wear is not a concern for one additional fuel cycle of operation. Additionally, any fretting between loose parts and the fuel rods "may result in leaking fuel rods which would cause an increase in coolant reactivity. However, this is not a safety concern as the plant still can maintain the ability of a safe shutdown." Also, in the case of debris fretting failure, Technical Specification 3.4.16 will ensure that the coolant activity stays within the prescribed limits. Any leak in a fuel rod would be detected and appropriate action would be taken.

The risk of loose parts inhibiting control rod insertion was considered. Any loose parts generated as a result of TSSB bolt and flexure degradation would be trapped in the fuel debris nozzles or between the upper core plate and top former. Any loose part that was small enough to get through the fuel debris nozzles would need to orient itself across one of the RCCA spider clearance slots in the guide card to prevent the RCCA from fully inserting. PWROG-21021-NP concluded: "it is considered a remote possibility that any loose part could affect the insertion of a RCCA. As such, the sticking of two or more control rods as a result of these foreign objects is not considered a credible event."

Regarding the impact of loose parts on downstream RCS equipment, it is possible that small, fractured pieces of the TSSB bolts (such as a lock bar) could travel out of the reactor vessel and into downstream components such as the steam generators or reactor coolant pumps. Based on previous loose parts evaluations, it was concluded that: "Fragments of similar size to the fragments of a TSSB cap screw lock bar were determined to be not a safety concern to the downstream RCS components."

In addition to the loose parts assessment documented in PWROG-21021-NP, Westinghouse performed a plant specific loose parts assessment for BVPS-1. This assessment concluded that "The potential loose parts from degraded BFB/TSSBB bolts are similar to those that have been previously evaluated at other Westinghouse-designed plants. The loose parts assessment summarized herein supports safe continued plant operation with the loose parts in the primary side of the RCS and the auxiliary systems."

Issue Program Precedence

Similar evaluations have been performed by Westinghouse to provide technical justification for other plants to operate with degraded thermal shield supports until repairs were performed.

Regulatory Perspective

Section 7.5 of MRP-227 Revision 1-A identifies WCAP-17096-NP-A as an acceptable NRC-approved evaluation method for dispositioning an examination result that does not meet the examination acceptance criteria identified in Section 5 of MRP-227 Revision 1-A. In section 3.3.11 of the NRC safety assessment for WCAP-17096 Rev 3 (ML23248A258), the NRC acknowledged that "In many cases, the additional actions for applicable components in the subject TR may specify performance of a specific type of analysis for the specified components. Accordingly, the need for performing these supplemental activities fall within the scope of the NRC licensee's own corrective action program,..." and that "...the specific type of analysis that may be referenced in an additional action for an applicable component in the subject TR constitutes only one way of dispositioning levels of age-related degradation that exceed the levels for initiating corrective actions per the criteria that are defined for the applicable component type in MRP-227, Revision 1-A. However, consistent with the implementation criteria in Chapter 7 of MRP-227, Revision 1-A, the licensee has the option of performing and using any type of engineering analysis that differs from the analysis specified in the subject TR. The licensee may also justify further service of the components with proposed component repair or replacement activities and repair/replacement schedules that would ensure the intended functions of the components."

From the statement above the Regulator acknowledged that there may be cases where unsatisfactory examination results would need to be dispositioned using a method that was different than the methods specifically identified in Section 7.5 of MRP-227, and that such dispositioning would be accomplished under the Licensee's own corrective action program.

Conclusion

Based on the facts presented above that are summarized from PWROG-21021-NP and the plant-specific technical evaluation performed by Westinghouse, it is concluded that it is technically acceptable to operate BVPS-1 for one additional cycle until the 1R31 (Spring 2027) refueling outage. These analyses and condition monitoring activities planned for the upcoming cycle demonstrate the technical acceptability of operating until 1R31 and also demonstrate that deviating from the NEI 03-08 guidance to use NRC-approved engineering evaluations is justified. Even for the bounding scenarios of thermal shield support degradation (complete detachment of the thermal shield), the structural integrity of the reactor pressure vessel will be maintained. Additionally, since it has been verified through examinations and crack growth rate evaluations that core barrel separation will not occur over the next cycle, core geometry will be maintained to ensure control rod insertability and core cooling are not adversely affected. The thermal and hydraulic characteristics of the reactor coolant flow are acceptable even considering the flexure and TSSB bolt degradation. Any loose parts that are expected to be generated as a result of the degraded flexures and TSSB bolts would not be expected to have an adverse effect on the operation of the reactor coolant system or any components in the reactor coolant system. DLW-RV010-TM-CG-000006 concluded that "The safety assessment in Section 3.0 demonstrate[s] that that there are no nuclear safety concerns posed by the TSSBB degradation observed at Beaver Valley Unit 1 for operation from 1R30 to 1R31 with the identified degradation of the TS support hardware. Therefore, Beaver Valley Unit 1 can safely return to operation for one fuel cycle following 1R30 for one additional cycle of operation."

Furthermore, 1R29 remote visual examination results of the BVPS-1 thermal shield support structure confirmed degradation levels that were previously suggested by Westinghouse analysis of neutron noise data from 2019, 2022, 2023, and early in plant life. Based on the demonstrated ability of neutron noise monitoring to provide evidence of progressing thermal shield support hardware degradation, an automated neutron noise monitoring system has been installed at BVPS-1. Automated neutron noise monitoring will be performed periodically, and the frequency will be adjusted as necessary based on the monitoring results. The periodic neutron noise monitoring will provide current information regarding the condition of the BVPS-1 thermal shield support structure over the current operating cycle. Any significant changes from previous analyses will initiate additional corrective actions. The last BVPS-1 neutron noise data was recorded in Spring 2026. This data can be compared with neutron noise data recorded during the current cycle to verify that significant degradation of the BVPS-1 reactor vessel internals is not continuing to occur.

References:

1. MRP-227 Revision 1-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," June 2020.
2. PWROG-21021-P Revision 2, "Thermal Shield Support Block Cap Screw and Flexure Degradation Assessment," November 2022.
3. DLW-RV010-TM-CG-000004 Revision 0, "Beaver Valley Unit 1 Spring 2024 Outage – Disposition of As-Found Conditions for the Reactor Vessel Internals and Justification for Return to Service," May 2024.
4. NEI 03-08, Revision 4, "Guideline for Management of Materials Issues," October 2020.
5. PWROG-17044-P, Revision 0, "Baffle Bolting Generic Safety Assessment," March 2018
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