OFFICE OF NUCLEAR REACTOR REGULATION STAFF ASSESSMENT OF ELECTRIC POWER RESEARCH INSTITUTE NEI 03-08, REVISION 2, "NEEDED" INTERIM GUIDANCE REGARDING BAFFLE-FORMER BOLT INSPECTIONS IN WESTINGHOUSE-DESIGN PRESSURIZED WATER REACTORS

1. INTRODUCTION

By letters dated July 27, 2016 (Ref. 1), and March 23, 2017 (Ref. 2), the Electric Power Research Institute (EPRI), Materials Reliability Program (MRP) transmitted to the Nuclear Regulatory Commission (NRC) NEI 03-08[1]interim guidance for baffle-former bolt (BFB) inspections in Westinghouse-design pressurized water reactors (PWRs). MRP 2016-022 contains, as an attachment, MRP 2016-021, which transmitted to the MRP members the interim guidance on BFB inspections for Westinghouse 4-loop plants operating in a downflow configuration (Tier 1 plants).

MRP 2017-011 contains, as an attachment, MRP 2017-009 (Ref. 4), which transmits guidance on initial BFB examination schedules for Westinghouse-design plants other than Tier 1 plants. It also contains guidance on BFB subsequent examination intervals for all Westinghouse-design PWRs. MRP 2017-011 states that the guidance supersedes MRP 2017-002, dated January 12, 2017, (Ref. 5) in its entirety. (MRP 2017-002 transmitted to NRC MRP 2016-033, dated September 29, 2016, which provided interim guidance for 2-loop and 3-loop downflow plants).

Tier 1 plants are defined in Westinghouse Nuclear Safety Advisory Letter (NSAL) 16-1, Revision 1, "Baffle-Former Bolts," August 1, 2016 (Ref. 6) as Westinghouse 4-loop PWRs operating in a downflow configuration. Tier 1 plants are further subdivided into Tier 1a plants, which have Type 347 stainless steel BFBs, and Tier 1b plants, which have Type 316 stainless steel BFBs.

Tier 2 plants are defined in NSAL-16-1, Revision 1 as Westinghouse 2-loop and 3-loop PWRs operating in a downflow configuration, and are further subdivided into Tier 2a (2-loop, Type 347 stainless steel BFBs), Tier 2b (3-loop, Type 347 stainless steel BFBs), and Tier 2c (3-loop, Type 316 stainless steel BFBs). Tier 3 plants are plants that began operation in a downflow configuration but have been converted to an upflow configuration, and Tier 4 are those plants that have always operated in an upflow configuration, which includes the two Combustion Engineering–design plants that have core shroud bolts in addition to the Westinghouse-design plants.

BFBs attach the baffle plates to the former plates of the reactor vessel internals (RVI) in many PWR designs. The main function of the baffle plates is to direct reactor coolant flow through the reactor core. In an extreme case of extensive BFB degradation, baffle plates could be detached

[1] NEI 03-08, Revision 3, "Guideline for the Management of Materials Issues," (Ref. 3) is a document issued by the Nuclear Energy Institute (NEI) to provide for overall coordination and oversight of materials issues for nuclear power plants. The implementation protocol of NEI 03-08, Rev 3 allows for issue programs such as the EPRI MRP to designate elements of work products such as inspection and evaluation guidelines reports as either "Mandatory", "Needed", or "Good Practice." Utilities have committed to follow the implementation protocol, and must take certain defined actions in order to deviate from "mandatory" or "Needed" guidance.

during design basis accidents. This could cause localized fuel damage, potentially jeopardizing core cooling and the ability to insert peripheral control rods.

BFB degradation is characterized by cracking due to irradiation-assisted stress corrosion cracking (IASCC) and fatigue. This is a known aging effect which was first observed in French PWRs in 1988 (Ref. 7) and in domestic PWRs in the late 1990's. For this reason, the industry's RVI aging-management guidelines in "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)" (Ref. 8) specified a baseline ultrasonic (UT) examination of the full population of BFBs for all Westinghouse-design PWRs between 25 and 35 effective full power years (EFPY) of operation, and subsequent examinations every 10 years thereafter.

EPRI issued the interim guidance in MRP Letter 2016-021 in response to the operating experience (OE) at two Westinghouse-design PWRs in 2016. The OE indicated a larger than expected percentage of BFBs were potentially degraded as compared to prior inspection findings.

The interim guidance in MRP 2016-021 specifies UT examination of all BFBs at the next refueling outage as "Needed" guidance for a subset of Westinghouse-design PWRs in accordance with NEI 03-08. The subset of plants for which UT examination is required at the next refueling outage is identified as Tier 1a in Westinghouse Nuclear Safety Advisory Letter (NSAL) 16-1, and comprises:

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

In addition, MRP 2016-021 specifies that Tier 1b plants perform visual testing (VT)-3 examination at the next refueling outage as "Needed" guidance in accordance with NEI 03-08[2], Rev 2. MRP 2016-022 states that if degradation is detected the plant shall complete actions consistent with Tier 1a plants (e.g., UT examination of all BFBs). If no degradation is detected during the visual VT-3 examination, MRP 2016-022 states that a UT consistent with Tier 1a plants' guidance shall be completed during the second refueling outage after issuance of the interim guidance. The Tier 1b plants consist of Sequoyah, Units 1 and 2.

The NRC staff notes that MRP 2016-022 endorses as interim guidance the recommendations of NSAL-16-1, Rev. 1, as NEI 03-08, Rev 2 "Needed" guidance for Tier 1 plants, and supplements the guidance in MRP-227-A.

The guidance in MRP 2017-009 is as follows:

> In the first refueling outage after March 1, 2018, domestic U.S. utility plants are to implement the following interim guidance as 'Needed' actions per NEI-03-08 protocol:

[2] MRP Letter 2016-021 does not reference a specific revision of NEI 03-08; however, NEI 03-08, Revision 2 (Ref. 9) was current at the time of issuance of the letter. NEI 03-08, Revision 3 was subsequently issued in February, 2017.

The existing MRP-227-A/Rev.1 Table 4-3 entry for "Examination Method/Frequency" of Baffle-Former-Bolts is modified as follows:

Current Requirement:

Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a 10-year interval.

Modified Requirement:

A. Baseline volumetric (UT) examination shall be performed as follows:

*1. NSAL-16-1 Rev.1 Tier 1 plants: per NSAL-16-1 Rev.1 and MRP-2016-021**

*2. NSAL-16-1 Rev.1 Tier 2 plants: no later than 30 EFPY**

3. Remaining plants: no later than 35 EFPY

**-initial baseline UT exams performed prior to 1/1/2018 are acceptable*

B. Subsequent volumetric (UT) examinations shall be performed on an interval established by plant-specific evaluation per MRP-227 Needed Requirement 7.5 as documented and dispositioned in the owner's plant corrective action. A reduced re-inspection interval has been determined to be an appropriate response to atypical or aggressive BFB degradation and shall satisfy the following criteria:

WEC Plant Design Type	***%UT Indications and Visually Failed BFBs***	***UT Re-Exam Period***
Down-Flow WEC[3] Plants	*<3% indications with no clustering(a)*	*not to exceed 10-years*
Down-Flow WEC Plants	*≥3% indications or clustering(a)*	*not to exceed 6-years(b)*
Upflow WEC Plants	*<5% indications with no clustering(a)*	*not to exceed 10-years*
Upflow WEC Plants	*≥5% indications or clustering(a)*	*not to exceed 6-years(b)*

[3] WEC = Westinghouse Electric Company

(a) *Note: Clustering defined per NSAL-16-1 Rev.1: three or more adjacent defective BFBs or more than 40% defective BFBs on the same baffle plate. Untestable bolts should be reviewed on a plant-specific basis consistent with WCAP-17096-NP-A for determination if these should be considered when evaluating clustering.*

(b) *A longer re-inspection interval, not to exceed 10-years, may be justified by plant-specific evaluation based on plant-specific exam findings. This evaluation may include additional justification from plant modifications and/or improvements (for example: replacements of BFBs, conversion to upflow, replacement of lower internals, etc.).*

C. As an alternative to performing UT inspections, a plant may perform proactive bolt replacements as preventative maintenance justified by plant-specific evaluation using established methodologies (for example, WCAP-15029-P-A[4] or equivalent). The plant-specific evaluation shall also establish and justify the UT re-examination period resulting from the bolt replacements performed.

NOTE: The MRP-227 Section 7.5, NEI 03-08, Rev 2 Needed requirement is unchanged with this interim guidance:

> *Examination results that do not meet the examination acceptance criteria defined in Section 5 of MRP-227 shall be recorded and entered in the owner's plant corrective action program and dispositioned. 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, WCAP-17096-NP or equivalent method).*

This guidance is supplemental to interim guidance previously promulgated by EPRI letter MRP-2016-021, dated July 25, 2016. However, this guidance supersedes MRP-2016-033, dated September 29, 2016, in its entirety.

The NRC staff assessed the guidance in MRP 2016-021 and MRP 2017-009 to determine if the recommendations for BFB inspections are appropriate to provide reasonable assurance that plants susceptible to BFB degradation are safe to operate until the scheduled inspections, considering the required examination scope, schedule and method. The NRC staff assessment has two main elements: 1) Determine if EPRI identified the appropriate population of plants that need accelerated inspections, and 2) Determine whether the schedule, scope, and method of examination are adequate to provide reasonable assurance of safe operation.

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[4] WCAP-15030-NP-A (Ref. 10) is the nonproprietary version of WCAP-15029-P-A, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions."

The NRC staff also assessed whether designation of the BFB inspection guidance as "Needed" is adequate. The NRC staff met with EPRI on April 12, 2017, to discuss NRC questions related to MRP 2017-009. A summary of the meeting can be found in Reference 11. Reference 12 contains the NRC's list of questions related to MRP 2017-009. EPRI responded to the staff's question via Letter 2017-015, dated July 13, 2017 (Reference 13).

2. BACKGROUND

Design and Materials of the Baffle-Former Assembly

In Westinghouse-design PWRs, BFBs are part of the baffle-former assembly (Figures 1 and 2). BFBs secure the vertical baffle plates to the horizontal former plates, which in turn are bolted to the core barrel by the barrel-former bolts. The baffle plates thus form a shroud around the core that closely follows the outline of the core. The main function of the baffle plates is to direct and concentrate the flow of coolant through the core. Baffle plates span the whole height of the active core and vary in width. The baffle plates are made of 1.5-inch-thick, Type 304 stainless steel.

For a four-loop Westinghouse-design PWR, there are eight horizontal rows of BFBs, andthe number of bolts across the plate vary with the plate width, from 2 bolts on the narrowest plates to 12 on the largest plates. In other designs, the number of rows and bolts across the width of the plate vary. There are also edge bolts that secure the corners of the baffle plates together to help minimize gaps at the plate corners. Figure 3 shows a cross section of the baffle-former assembly and core barrel showing the locations of these various bolts. Figure 4 shows the appearance of a BFB head similar to the configuration used in 4-loop plants, as seen from inside the core baffle. A locking bar resting in a slot on the bolt head is tack welded to the baffle plates, to prevent the bolts from backing out and to retain the bolt head if the bolt shank fractures. Other Westinghouse plant designs have slightly different bolt head designs, with some designs employing a lock washer rather than a lock bar.

The main flow path for coolant into the reactor vessel is down the downcomer between the reactor vessel and the core barrel, then up through the lower core plate and through the core. A small portion of the flow (bypass flow) is directed between the core barrel and the core baffle for cooling purposes. In some plants, this flow enters through holes near the top of the core barrel and flows downward, joining the main core flow when it exits through a gap between the baffle plates and the lower core plate. This is referred to as a "downflow" configuration. In other plants, mostly newer Westinghouse units, the bypass flow enters at the bottom and flows upward, parallel to the main core flow ("upflow" configuration). Figure 5 shows both flow configurations. In addition, some plants originally constructed as "downflow" plants were converted to an "upflow" configuration as indicated in Figure 5 by plugging the holes near the top of the core barrel and providing new flow holes through the top of the highest former plate.

In domestic Westinghouse-design PWRs, BFBs were fabricated from two different stainless steels, Type 347 and cold-worked Type 316. The NRC staff understanding is that all Type 316 stainless steel BFBs used in Westinghouse-design RVI are cold worked; therefore, hereafter this material will be referred to simply as "Type 316."

Two Combustion Engineering (CE)-design PWRs have core shroud bolts which are similar to and perform the same function as BFBs. These bolts are annealed Type 316 stainless steel, and the CE units operate in an upflow configuration. Babcock & Wilcox (B&W) –design PWRs also have BFBs fabricated from Type 304 stainless steel, but are not addressed by the interim guidance.

Summary of NSAL-16-1

NSAL-16-1, Revision 1 (the NSAL, Ref. 5) was issued by Westinghouse to owners of Westinghouse-design PWRs on August 1, 2016. The NSAL provides the results of Westinghouse's evaluation under Title 10 of the Code of Federal Regulations, Part 21 (10 CFR Part 21), "Reporting of Defects and Noncompliance," of the BFB degradation. The NSAL also provides Westinghouse's recommendations for accelerated inspections of BFBs. Since the NSAL was submitted to the NRC for information only, the NRC staff did not review the analyses and evaluations supporting the NSAL. The NRC staff did use information on design and susceptibility from the NSAL to inform its review of OE related to BFBs.

The NSAL contains a summary of the OE related to BFB degradation. The NSAL also describes a technical evaluation of the safety implications of hypothetical BFB degradation in a Westinghouse 4-loop plant. Note that the NRC staff has not reviewed the calculations or analyses underlying this technical evaluation, and does not endorse the conclusions of the technical evaluation.

The NSAL technical evaluation assumes all the BFBs on one baffle plate are failed. A loss-of-coolant accident (LOCA) evaluation was then performed assuming a 4-inch line break. Westinghouse justified the assumption of a 4-inch line break based on the fact that all Westinghouse PWRs have successfully applied the leak-before-break (LBB) concept to the reactor coolant system main loop piping. In addition, many Westinghouse PWRs have applied LBB to the large branch lines, some down to 6 inches in diameter.

Therefore, the NSAL states that based on the LBB analyses already licensed, it is Westinghouse's engineering judgment that LBB can be successfully applied to the pressurizer surge line, residual heat removal lines, accumulator lines, and 6-inch safety injection lines for all operating U.S. Westinghouse reactors including 2-, 3- and 4-loop plants.

The LOCA evaluation summarized in the NSAL indicated a potential for some fuel grid deformation. However, the NSAL determined that a coolable core geometry would be maintained. With respect to seismic events, the NSAL determined that seismic effects would be bounded by LOCA effects. The evaluation also determined that control rod insertion would be maintained and core decay heat removal would not be compromised.

Therefore, the NSAL concluded that BFB degradation would not compromise the ability to cool the core, maintain reactor shutdown and long-term removal of decay heat after a LOCA. A loose parts assessment also determined that potential loose bolt heads and locking bars would not compromise safe operation of the plant. Based on these evaluations, the NSAL concluded that BFB degradation does not represent a substantial safety hazard as defined in 10 CFR Part 21.

The NSAL recommends actions for all Westinghouse-design PWRs, both domestic and foreign, plus the two CE PWRs with core shroud bolts, based on susceptibility to BFB degradation. The NSAL groups the plants into "tiers" with respect to susceptibility to BFB degradation. Factors considered in determining the groups were primary load (function of pressure and bolt/plate spacing), bolt design, and bolt material. The consideration of pressure was mainly a function of downflow versus upflow configuration, since the downflow configuration has a higher pressure differential across the baffle plates.

- Tier 1 is the most susceptible group, and consists of 4-loop plants currently operating in a downflow configuration and is further subdivided into Tier 1a, with Type 347 bolts, and Tier 1b, with Type 316 bolts. All the Tier 1 plants have relatively similar EFPY of operation of between 25 to 31 EFPY, thus EFPY was not used to further rank the plants within this tier.

  The NSAL indicates that the Type 347 bolt design has a sharper radius and a shorter bolt shank than the Type 316 bolt design, which results in a higher stress concentration factor for the Type 347 bolts. The NSAL further indicates that this design difference makes the Type 347 bolts more susceptible to IASCC even though the data does not indicate a great difference in inherent material susceptibility to stress-corrosion cracking. Thus, Tier 1a is considered the most susceptible group.

  - The domestic Tier 1 plants are:

    - Tier 1a
      - D.C. Cook, Units 1 and 2
      - Diablo Canyon, Unit 1
      - Indian Point, Units 2 and 3
      - Salem, Units 1 and 2

    - Tier 1b
      - Sequoyah, Units 1 and 2

- Tier 2 consists of the 2-loop and 3-loop downflow plants. The NSAL indicates that the magnitude of the pressure differential is smaller in the 2- and 3-loop downflow plants than the 4-loop downflow plants. The NSAL also states that the 2- and 3-loop plants have a larger number of bolts per square inch of baffle plate, thereby further reducing the pressure-induced stress on the bolts. Also, as with the 4-loop plants, the bolt design corresponding to the Type 347 material is expected to have a higher stress concentration at the head-to-shank transition. Tier 2 is further subdivided into Tier 2a (2-loop plants with Type 347 bolts), Tier 2b (3-loop plants with Type 347 bolts), and Tier 2c (2-loop and 3-loop plants with Type 316 bolts).

  - The domestic Tier 2 plants are:

    - Tier 2a
      - Prairie Island, Units 1 and 2
      - Ginna

    - Tier 2b
      - H.B. Robinson 2
      - Surry, Units 1 and 2

        - Turkey Point, Units 3 and 4

    - There are no domestic Tier 2c plants.

- Tier 3 consists of all converted upflow plants, 2-loop, 3-loop, and 4-loop. The NSAL states that plants that were originally operated in the downflow configuration can be converted to upflow through field modifications to the core barrel and former plates, and these modifications have been shown to reduce the incidence of baffle jetting damage to the fuel.

  The NSAL further states that the upflow conversion also reduces the bolt loads due to reduced pressure differentials across the baffle under both normal operating and expected faulted conditions. The NSAL states that although the upflow conversion is expected to have a positive impact on bolt life, there remains a potential for accelerated degradation during the original period of operation with the downflow configuration, but the overall condition of these converted upflow plants is better than equivalent plants operated continuously in a downflow configuration.

  The NSAL states that longer time operating in the downflow configuration is postulated to correspond to a higher potential for BFB degradation. The staff notes that Point Beach Units 1 and 2, the only Tier 3 plants identified by the NSAL as having Type 347 bolts, have already performed the initial MRP-227-A UT examination of BFBs.

    - The domestic Tier 3 plants are:
        - Point Beach, Units 1 and 2 (2-loop)
        - Farley, Units 1 and 2 (3-loop)
        - Beaver Valley, Unit 1 (3-loop)
        - North Anna, Units 1 and 2 (3-loop)
        - V.C. Summer, Unit 1 (3-loop)
        - Diablo Canyon, Unit 2 (4-loop)
        - McGuire, Unit 1 and 2 (4-loop)

Tier 4 consists of all plants that have been continuously operated in an upflow configuration.

- The domestic Tier 4 plants are:

    - Beaver Valley, Unit 2 (3-loop)
    - Shearon Harris, Unit 1 (3-loop)
    - A. W. Vogtle, Units 1 and 2 (4-loop)
    - Braidwood, Units 1 and 2 (4-loop)
    - Byron, Units 1 and 2 (4-loop)
    - Callaway, Units 1 and 2 (4-loop)
    - Catawba, Units 1 and 2 (4-loop)
    - Comanche Peak, Units 1 and 2 (4-loop)
    - Millstone, Unit 3 (4-loop)
    - Seabrook, Unit 1 (4-loop)
    - South Texas, Units 1 and 2 (4-loop)
    - Watts Bar, Units 1 and 2 (4-loop)
    - Wolf Creek (4-loop)

- Fort Calhoun, Unit 1 (CE-design)[5]
- Palisades, Unit 1 (CE-design)

The general recommendations applicable to all Tiers are:

- If visually damaged BFBs 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.
- Continue to follow the current MRP-227 guidelines and implement any revisions to the MRP-227 recommendations.

Table 1 summarizes the tiers defined in the NSAL and the recommendations for each:

Table 1 – Tiers for BFB Degradation Susceptibility from NSAL-16-1, Revision 1

Tier	Loops	Configuration	Stainless Steel Type	No. Units in U.S.	NSAL Recommendation
1a	4	Downflow	347	7	UT 100% of BFBs next RFO
1b	4	Downflow	316	2	VT-3 100% of BFBs next RFO. If indications are found, UT 100% of BFBs. If no indications in VT-3, UT 100% of BFBs during second RFO
2a	2	Downflow	347	3	Review previous UT inspection records for indications of clustering (3 adjacent failures or 40% or more degraded bolts on one plate). If clustering occurred, consider accelerated re-inspection
2b	3	Downflow	347	5	Same as Tier 2a
2c[6]	2, 3	Downflow	316	0	Same as Tier 2a

[5] Plant is now permanently shut down.
[6] There are no domestic Tier 2c reactors.

3	2,3,4	Converted upflow	All	11	If plant operated > 20 calendar years in downflow, evaluate need for accelerated inspections via comparison to Tier 1a design parameters
4	2,3,4	Upflow (original)	All	22	Follow guidance for general recommendations for all tiers

Staff Review of OE with BFB Degradation

Two U.S. plants found significant numbers of degraded BFBs during spring 2016 inspections. These plants are Indian Point, Unit 2 (IP2), which found 27 percent of BFBs potentially degraded (Ref. 14), and Salem, Unit 1, which found 22 percent of BFBs potentially degraded (Ref. 15).

Clustering of failed bolts occurred at both plants but was more severe at Salem 1, where three of eight octants showed degradation in most of the bolts, while other octants had only a few degraded bolts. Visual examinations of baffle-edge bolts in these plants found no evidence of degradation. Both IP2 and Salem 1 are Westinghouse 4-loop design, downflow plants with Type 347 stainless steel baffle-former bolts, which were designated Tier 1a in NSAL-16-1.

In 2010, one Westinghouse 4-loop plant, D.C. Cook Unit 2, conducted a visual examination and identified a cluster of 18 broken bolts on one large baffle plate (Ref. 16). Using Type 316 BFBs that had an enhanced design, the licensee replaced the broken bolts and some additional bolts, finding a total of 42 defective bolts on the plate. The licensee did not perform a UT examination of the BFBs.

In October 2016, UT examination of the full population of BFBs was conducted at D.C. Cook, Unit 2 in accordance with the guidance of MRP 2016-022. At D.C. Cook, Unit 2, approximately 22% of the BFBs were found to be degraded or potentially degraded either by the UT examination or visually (Ref. 17). Clustering of degraded bolts was also observed, similar to that seen at Indian Point, Unit 2, (IP2) and Salem, Unit 1.

Additionally, six Type 316 replacement bolts installed in 2010 and five baffle-edge bolts were found to be degraded at D.C. Cook, Unit 2 (Ref. 17). This finding represents the first degradation of replacement BFBs and baffle-edge bolts in a U.S. PWR.

Four more Tier 1a plants have performed initial UT examinations of BFBs as of October, 2017. Indian Point, Unit 3 found 31% of the BFBs potentially degraded, with clustering similar to that observed at IP2. At Salem, Unit 2, and Diablo Canyon, Unit 1, small numbers of bolts were found to be degraded by UT (9 bolts or 1.1% at Salem, Unit 2, 1 bolt or 0.1% at Diablo, Unit 1) (Ref. 18). D.C. Cook, Unit 1, found 52 potentially degraded bolts (which includes 4 untestable bolts), or 6.2%. All Tier 1a plants have now completed initial UT examinations in accordance with the interim guidance.

Most of the Tier 1a plants that found BFB degradation in 2016-2017 replaced all bolts that had potential UT indications plus additional bolts to provide margin against future BFB failures due to IASCC. Salem, Unit 1, replaced 189 out of 192 potentially degraded bolts, but qualified the as-left bolt pattern with an acceptable bolting pattern analysis.

Voluntary UT examinations of BFBs were performed at five Westinghouse-designed reactors in the late 1990s (Ref. 8, 17).. Two of these plants were 2-loop designs with Type 347 BFBs (Ginna and Point Beach, Unit 2), while two were 3-loop designs with Type 316 BFBs (Farley, Units 1 and 2). Ginna is a 2-loop downflow (Tier 2a) plant while Point Beach, Unit 2, converted to upflow in 1986-1987, thus is classified as Tier 3.

At the time of the examination, Farley, Unit 1 was a converted upflow plant (Tier 3) while Farley, Unit 2 was a downflow plant (Tier 2b). Farley, Unit 2 converted to upflow in 2002 so is now classified as Tier 3. The 2-loop plants found 5 to 10 percent of BFBs potentially degraded, while the 3-loop plants found no degraded bolts. Bolts were replaced at all four plants. All the potentially defective bolts were replaced at one 2-loop plant, and the defective bolts plus some additional bolts were replaced at the other 2-loop plant. The licensee of the 3-loop plants proactively replaced a subset of the bolts to achieve an acceptable bolting pattern with replacement bolts.

Voluntary examination of BFBs was performed at one B&W design reactor in 2005, which found essentially no degraded bolts.

No UT examinations of BFBs at U.S. plants were conducted in the 2005-2010 timeframe. In 2010, licensees began performing UT examinations in accordance with MRP-227-A. Three examinations were completed in 2010-2011 prior to NRC approval of MRP-227, but the examinations would have met the MRP-227-A guidance for two of these plants. UT examinations of BFBs have been performed at eight of eight U.S. Tier 2 plants since 2010. Three of these were Tier 2a plants and five were Tier 2b plants.

One of the Tier 2a plants, Ginna, performed only a partial UT examination, finding only one degraded bolt. The other two Tier 2a plants (Prairie Island, Units 1 and 2) found a maximum of 10% of the BFBs degraded. The four Tier 2b plants found a maximum of 8 degraded BFBs (1%). One of these plants only performed a partial examination due to equipment issues, but found no degraded bolts.

UT examinations have been performed at three Tier 3 (converted upflow) units since 2010 (Refs. 17, 20, 21). Point Beach, Unit 1 (2-loop) had <10% of the BFBs that did not produce relevant UT results[7] , but no indications of degradation in the bolts with relevant UT results. Point Beach, Unit 2, (2-loop) which replaced some bolts in 1999, had 15 original bolts with degradation, representing <3% of the remaining original bolts. North Anna, Unit 1 (3-loop) had < 1% of BFBs with indications (Ref. 17). Therefore, there is no current evidence of extensive BFB degradation in converted upflow plants.

UT examinations have therefore been performed at all of the 2-loop plants since 2010 (both downflow and converted upflow), and at five of five 3-loop downflow plants in the same time frame, representing a significant sample of these plant designs (Refs. 17, 20, 21). Per NSAL-16-1, all these plants have Type 347 bolts.

With the exception of one 2-loop and one 3-loop plant, at least 75% of the total BFB population at each individual unit was examined in accordance with MRP-227-A coverage requirements. These examinations found 10 percent or less of the BFBs potentially degraded[8] in 2-loop and

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7 Due to having a geometry different than that qualified for the UT technique (Ref. 12)

8 This number is cumulative, counting BFBs found during previous inspections in late 1990s.

less than 1 percent of the BFBs potentially degraded in 3-loop designs. The total number of degraded bolts in the 2-loop and 3-loop plants, whether downflow or converted upflow, does not represent a safety issue because it is bounded by acceptable bolting pattern analyses (ABPAs) for those plants.

An ABPA is an analysis to determine whether the baffle-former assembly meets design acceptance criteria under all design basis conditions, considering hypothetical or actual patterns of degraded and intact bolts. Licensees may prepare ABPAs prior to a planned examination of BFBs, then compare the as-found configuration of degraded and non-degraded bolts to the ABPA. Alternatively, licensees may perform an ABPA of the actual as-found configuration of degraded and non-degraded bolts.

With respect to non-Westinghouse U.S. PWRs, all but two CE design PWRs employ a welded, rather than bolted, baffle-former assembly (referred to as the core shroud in CE designs). UT examinations have not yet been conducted at the CE units with bolted core shrouds. These units are considered less susceptible to cracking, because annealed Type 316 bolts were used (Ref. 7).

B&W-designed PWRs use solution annealed Type 304 BFBs, and have an upflow configuration (Ref. 19). UT examinations of the full population of BFBs have been performed at four B&W design PWRs since 2012, finding only a very limited number of degraded bolts.

The higher susceptibility to IASCC in the downflow plants compared to upflow plants may be in part due to the higher pressure differential from the inside to the outside of the baffle plates present in the downflow configuration, which increases the stress in the BFBs. Another contributor to the higher susceptibility to bolt degradation in 4-loop designs compared to 3-loop designs is that 4-loop designs have the same number or fewer BFBs than 3-loop designs, but a larger area of baffle plates. This larger area would result in higher stresses in the bolts.

The NRC staff notes that only two Westinghouse-design PWRs in the U.S. with Type 316 BFBs have performed a UT examination of the BFBs, and this examination was performed earlier in plant life. However, UT examinations of French and Belgian PWRs with Type 316 BFBs, have been periodically performed since the early 1990's. While the overseas plants found some degraded BFBs, the overall percentages of degraded bolts found are more in line with the U.S. 2-loop and 3-loop plants with Type 347 BFBs.

The NRC staff assessment of the observed degradation of a few Type 316 replacement BFBs at D.C. Cook, Unit 2, finds that the degradation appears to be related to conditions unique to that plant, rather than indicating a generic susceptibility of Type 316 replacement BFBs. This is discussed below in more detail. Further, the NSAL explained that the design of the Type 347 bolts makes them more susceptible to IASCC. Therefore, the NRC staff agrees that plants with Type 347 bolt material should be considered more susceptible to BFB degradation than those with Type 316 bolts.

Based on the OE summarized above, significant bolt degradation, as well as clustering of degraded bolts, has been limited to Westinghouse 4-loop designs with a downflow configuration and Type 347 bolts (Tier 1a plants). Therefore, the NRC staff agrees with the NSAL categorization of plants with these characteristics as Tier 1a, the most susceptible group. Since the 4-loop downflow plants with Type 316 bolts (Tier 1b in the NSAL) share two of the three characteristics of the Tier 1a plants, the staff agrees that the Tier 1b plants are the second most susceptible category, and the accelerated BFB inspections of Tier 1b plants are appropriate.

Sufficient examinations have been performed of BFBs in 2-loop and 3-loop design plants with 25-35 EFPY to provide reasonable assurance that these plants are not experiencing a high percentage of BFB degradation. The NRC staff is not aware of any incidence of clustering of degraded bolts in 2-loop or 3-loop plants, as defined in NSAL-16-1 (3 or more adjacent bolt failures or 40% or more degraded on a single plate).

Degradation of replacement Type 316 BFBs has only been observed at D.C. Cook, Unit 2, a Tier 1a plant. In contrast, two 2-loop plants and two 3-loop plants have had replacement Type 316 BFBs in service since the late 1990's with no degradation observed in these bolts. The NRC staff believes that the degradation of replacement bolts at D.C. Cook, Unit 2, was the result of high stresses on the replacement bolts due to failures of nearby original bolts. This may have been possible because in 2010, bolts were only replaced in a limited area of the baffle where there was a visual indication of degradation, and a UT examination was not performed on any bolts.

Therefore, any degradation of additional bolts outside the area of bolts replaced would not have been detected in 2010 unless the bolts were completely failed, resulting in visual indications of failure. UT results from 2016 show a large cluster of more than 40 degraded bolts located near the degraded replacement bolts. Undetected degradation of these bolts likely resulted in a transfer of loads to the replacement bolts, which led to stresses in these bolts high enough to initiate IASCC. Additionally, the replacement bolts could have accrued sufficient neutron fluence in four 18-month cycles (six years) to exceed the fluence threshold for IASCC, if the stresses were sufficiently high.

Similarly, at D.C Cook, Unit 2, the degraded baffle-edge bolts were located on a seam between two plates in the middle of the same large cluster of degraded bolts. Degradation of BFBs on the plates adjacent to the seam with the degraded baffle-edge bolts could have led to increased stresses on these baffle-edge bolts. The NRC staff notes that the examinations specified for baffle-edge bolts in MRP-227-A is a visual VT-3 examination, with the baseline examination between 20 and 40 EFPY and subsequent examinations on a 10-year interval. The specified examination coverage is bolts and locking devices on high-fluence seams, 100% of components accessible from the core side. The EPRI interim guidance did not make any changes to the MRP-227-A guidance for baffle-edge bolts.

If UT of 100% of BFBs is performed and all degraded bolts are replaced, as was done at the three Tier 1a plants that conducted initial examinations in 2016, there could be a few undetected degraded bolts since the probability of detection of the UT technique for BFBs is not 100%. However, any such undetected bolts would tend to be randomly distributed rather than grouped in a cluster. Development of clustering from isolated degraded bolts would require several operating cycles. Therefore, these plants should not be at risk of extensive degradation or clustered failures of original bolts in the near term.

The licensees of all three Tier 1a plants that performed UT examination of BFBs in 2016 have indicated they either plan to perform a follow-up UT or visual examination of 100% of the BFBs at the next refueling outage. Therefore they will only operate one cycle before inspecting. Without clustering of degraded original bolts, it is unlikely that replacement bolts could be sufficiently stressed to develop IASCC in one cycle.

Staff's Risk-Informed Evaluation of BFB Degradation

In response to the OE with extensive BFB degradation in 2016, the NRC staff performed a risk-informed evaluation (Ref. 23) of the safety significance of recently identified reactor vessel BFB degradation, in accordance with Office of Nuclear Reactor Regulation (NRR) Office Instruction LIC-504, Revision 4, "Integrated Risk-Informed Decision-Making Process for Emergent Issues," (Ref. 24). The LIC-504 takes into account the five key principles of risk-informed regulation:

(1) Compliance with Existing Regulations
(2) Consistency with the Defense-in-Depth Philosophy
(3) Maintenance of Adequate Safety Margins
(4) Demonstration of Acceptable Levels of Risk
(5) Implementation of Defined Performance Measurement Strategies

The staff's evaluation identified the following options for the most susceptible Tier 1a group, Westinghouse 4-loop downflow plants with Type 347 BFBs:

- Option 1: Require immediate shutdown and inspection of the BFBs.
- Option 2: Allow continued operation until the next refueling outage, at which time the plants are required to examine all accessible BFBs.
- Option 3: Issue a generic communication to gather additional information to support a future regulatory decision.
- Option 4: Maintain the status quo, under which plants would inspect the BFBs consistent with the current recommended schedule in MRP-227-A.

With respect to LOCA, the NRC staff evaluation determined that only medium and large-break LOCAs create a significant possibility of detachment or deflection of a baffle plate with degraded BFBs that could lead to core damage. Therefore, the risk of core damage due to LOCA was driven by the generic estimated large and medium LOCA frequencies for PWRs, which are relatively low. For seismic events, a bounding seismic assessment for U.S. plants was performed and found that a core damage frequency of $1x10^{-3}$ per reactor-year would only be approached at a 75% reduction in structural capacity of the baffle assembly. This corresponds to a much higher level of degradation than has been seen in any plant.

In the LIC-504 evaluation, the NRC staff recommends Option 2 because the level of risk represented by operation for only one cycle is acceptable, with an associated core damage frequency less than $1x10^{-3}$ per reactor-year and a large early release frequency less than $1x10^{-4}$ per reactor-year with very conservative assumptions on the reduction in structural capacity of the BFB assembly. While this is also true for Option 1, the NRC staff determined immediate shutdown would place an unnecessary burden on the plants. The NRC staff eliminated Options 3 and 4 because they would extend the time frame for inspections or other corrective actions and increase the uncertainties related to risk.

Implementation of NEI 03-08, Rev 2 "Needed" Interim Guidance

NEI 03-08, Revision 3 (Ref. 3) provides for overall coordination and oversight of materials issues for nuclear power plants. The formal industry material initiative began in 2003 with the objective of assuring safe, reliable and efficient operation of the U.S. nuclear power plants in the management of materials issues. All nuclear utilities made a voluntary commitment (not to be confused with a regulatory commitment) to follow the initiative.

Appendix B to NEI 03-08, Rev 3 describes the implementation protocol for the initiative. The implementation protocol allows issue programs (such as the EPRI MRP) to designate certain elements of work products (such as reports providing guidance for inspection and evaluation) as "Mandatory," "Needed", or "Good Practice."

"Mandatory" and "Needed" are the two highest categories of NEI 03-08, Rev 3 guidance. The criteria for designation of an element of a work product as "mandatory" are:

- Element substantively affects the ability of structures, systems and components to perform their intended safety function.
- Element would be highly risk significant as determined by the responsible issue programs if not implemented.
- Element poses a significant threat to continued operation of the affected plants, including economic threats that could reasonably lead to protracted plant shutdown or retirement.
- A consensus of the responsible materials issue program believes the element should be designated as "Mandatory".

The criteria for designation of an element of a work product as "Needed" are:

- Element substantively affects the ability of structures, systems or components to reliably perform their economic function.
- Element would be moderately risk significant as determined by the responsible issue program if not implemented.
- Element addresses a material degradation mechanism that has significant financial impact on the entire industry, especially where failure at one plant could affect many other plants.
- A consensus of the responsible materials issue program believes the element should be designated as "Needed".

The NRC staff agrees that the examination interim guidance for BFBs in MRP 2016-021 is appropriately classified as "Needed" based on the above criteria. In particular, the NRC staff risk assessment in the LIC -504 evaluation supports the assessment that the element would be moderately risk significant if not implemented. With respect to the guidance for Tier 2, 3, and 4 plants in MRP 2017-009, the NRC staff opinion is that designation of this guidance as "Needed" is conservative since these plants appear to be at lower risk of extensive BFB degradation than Tier 1 plants, and, therefore, is acceptable.

To deviate from NEI 03-08, Rev 3 "Needed" guidance, the following actions are required from the utility:

- Documented in accordance with the plant's corrective action program
- Independent review performed (may be internal or external to the utility)
- Concurrence from the responsible utility executive

For deviations from "Mandatory" or "Needed" guidance, the utility must submit the justification for the deviation to the issue program within 45 days and must also notify the NRC via a letter within the same time frame. Therefore, although NEI 03-08, Rev 3 "Needed" guidance is not an

NRC regulatory requirement, deviations require a high visibility decision by the utility along with prompt notification to the NRC.

3. ASSESSMENT

Changes to Initial UT Examination Schedules

Tier 1 Plants

Based on its review of OE, the NRC staff finds that Tier 1a (Westinghouse 4-loop, downflow, Type 347 bolts) is the most susceptible category of plants for BFB degradation. The NRC staff review of OE suggests that the recommended initial examination schedule for BFBs of 25-35 EFPY in MRP-227-A needs to be modified for Tier 1a plants. The NRC staff LIC-504 evaluation concluded that the risk of plant operation until the next refueling outage for these plants is acceptable. Therefore, the NRC staff finds that the interim guidance for Tier 1a plants to perform a UT examination of the BFBs at the next refueling outage is acceptable.

With respect to the recommendation of UT as the examination method for Tier 1a plants, the NRC staff finds it acceptable because UT is the method that can detect BFB degradation prior to complete failure of the bolt shank with the most complete characterization of BFB integrity.

With respect to Tier 1b plants, since these plant have Type 316 BFBs, the NRC staff expects that these plants would be somewhat less susceptible to extensive BFB degradation than Tier 1a plants. For Tier 1b plants, the interim guidance specifies visual VT-3 examination of the BFBs at the next refueling outage, followed by UT examination at the second refueling outage if no degradation is noted by the VT-3 examination. VT-3 has proven effective to detect BFB degradation when bolts have completely fractured, which often leads to subsequent failure of the bolt locking devices and protruding or missing bolt heads, and most plants with extensive degradation had a significant number of visually observable failed BFBs.

Further, the interim guidance requires plants to perform UT during the same refueling outage if the VT-3 examination detects any evidence of degradation. The staff therefore finds that VT-3 examination at the next refueling outage will provide reasonable assurance that extensive BFB degradation would be detected via visually observable failures followed by UT of the entire baffle-former assembly. The staff also finds UT examination at the second refueling outage (given acceptable VT-3 results) is appropriate since the Tier 1b share two of the three characteristics of the most susceptible group (4 loops and downflow configuration), but are somewhat less susceptible to IASCC due to the use of Type 316 bolts, based on an assessment of OE with Type 316 BFBs.

Tier 2 Plants

The interim guidance in MRP 2017-009 recommends revised schedules for initial BFB examination in some categories of Westinghouse PWRs. For Tier 2 plants, MRP 2017-009 recommends initial examination no later than 30 EFPY. MRP 2017-009 also states that initial baseline examinations performed before January 1, 2018, are acceptable. The NRC staff finds this recommendation appropriate because Tier 2 plants are downflow plants. Thus, Tier 2 plants have higher stresses on the BFBs than plants operating in an upflow configuration. Plus they also all have Type 347 BFBs, which may make these plants somewhat more susceptible to BFB degradation than plants operating in upflow with Type 316 BFBs.

OE has shown some BFB degradation in Tier 2 plants, although not to the extent seen in Tier 1a plants. Since some Tier 2 plants may already have exceeded 30 EFPY, the staff asked EPRI at the April 12, 2017, public meeting to clarify the initial examination schedule for Tier 2 plants that have already exceeded 30 EFPY as of the issue date of MRP 2017-009 (Question 1). The EPRI July 13, 2017, response to Question 1 stated that the Tier 2 U.S. plants that have not yet performed their initial baseline inspection would have to complete the examination in the next outage or provide a deviation disposition.

The NRC staff finds this response acceptable because the staff considers the Tier 2 plants to be bounded with respect to susceptibility to BFB degradation by the Tier 1a plants. The NRC staff LIC-504 evaluation determined it was acceptable to perform the initial UT examination at the next refueling outage. Question 1 is thus resolved.

Tier 3 and 4 Plants

For the remaining plants, which consist of converted upflow (Tier 3) and original upflow plants (Tier 4), MRP 2017-009 recommends the initial examination no later than 35 EFPY. This is essentially no change from the current initial examination schedule recommended by MRP-227-A of 25 to 35 EFPY for all Westinghouse PWRs. For original upflow (Tier 4) plants, the NRC staff finds this guidance acceptable because this group of plants has the lowest susceptibility to BFB degradation.

According to NSAL-16-1, Revision 1, all the domestic Tier 3 plants have Type 316 BFBs except for Point Beach, Units 1 and 2, which have already performed initial UT examinations of BFBs. With the Type 316 bolts plus the upflow configuration, these plants should be less susceptible to BFB degradation than Tier 1 plants, and Tier 2 plants.

NSAL-16-1, Revision 1, contained a recommendation that Tier 3 plants which operated in downflow for twenty years or more should evaluate the need for an accelerated baseline examination schedule for BFBs. This recommendation was not incorporated into the interim guidance of MRP 2017-009. In MRP 2017-009, the baseline UT schedule for Tier 3 plants is included under A.3, “Remaining Plants,” and initial baseline UT examination would be required no later than 35 EFPY, which is the same maximum EFPY as the MRP-227-A recommendation.

Tier 3 plants with 4 loops, such as Diablo Canyon, Unit 2, and McGuire, Units 1 and 2, are probably the most susceptible among the Tier 3 plants. The 4-loop Tier 3 plants should be bounded in susceptibility by Tier 1b plants, which have 4-loops, downflow configuration, and Type 316 BFBs. Both Tier 1b plant have performed a VT-3 examination of all BFBs in accordance with MRP 2016-021, and found no degraded bolts (Ref. 18). The NRC staff was concerned that, since some Tier 3, 4-loop plants may have operated for a significant time period in a downflow configuration before converting to upflow, those plants may have IASCC susceptibility approaching that of Tier 1 plants. Therefore, at the public meeting held on April 12, 2017, the staff asked EPRI to clarify the basis for not recommending an accelerated baseline examination schedule for 4-loop Tier 3 plants (Question 2).

In its July 13, 2017, response to Question 2, EPRI stated that it was decided to word the recommendation in this manner because it only applied to one plant. That plant has unique limitations in place due to future plans for station closure prior to entering the period of extended operation. The NRC staff identified this plant as Diablo Canyon, Unit 2, which plans to shut down by 2025. Further, all Tier 3 4-loop plants have Type 316 bolts, which should make them less susceptible to bolt degradation than 4-loop plants with Type 347 bolts with the same EFPY.

The NRC staff finds EPRI's response to Question 2 acceptable because the one Tier 3, 4-loop plant with 20 years or more operation in a downflow configuration plans to shut down early. Also the plant has Type 316 bolts which reduces the susceptibility to IASCC.

Question 2 is thus resolved.

Subsequent Examination Intervals

MRP 2017-009 states that subsequent volumetric (UT) examinations shall be performed on an interval established by plant-specific evaluation per MRP-227 "Needed" Requirement 7.5 as documented and dispositioned in the owner's plant corrective action program. MRP 2017-009 further states that a reduced reinspection or subsequent examination interval has been determined to be an appropriate response to atypical or aggressive BFB degradation and shall satisfy criteria in the table.

In addition, the table in MRP 2017-009 provides revisions to the maximum subsequent examination intervals for both downflow and upflow plants. The subsequent examination interval is shortened to a maximum of six years depending on the percentage of BFBs with indications, whether clustering is present, and whether the plant is downflow or upflow configuration. For downflow plants, the subsequent examination interval is shortened for plants having ≥ 3% of BFBs with indications, while for upflow plants, it is shortened for plants with ≥ 5% of BFBs with indications. For both downflow and upflow plants, clustering is defined as 3 or more adjacent BFBs with indications or greater than 40% of bolts degraded on a single baffle plate.

The NRC staff finds the guidance to establish the subsequent volumetric examination interval by a plant-specific evaluation to be acceptable. The basis for this is that it would be difficult to establish this generically based on the many different percentages and patterns of degraded BFBs that may be found during examinations, whether degraded BFBs are replaced, etc. MRP 2017-009 does not provide any guidance for the methodology of the plant-specific evaluations. However, the NRC staff notes that several vendors have developed tools such as probabilistic models to predict degradation rates of BFBs. The staff encourages licensees to use such tools to support their plant-specific evaluations of subsequent examination intervals for BFB examinations. These tools are generically summarized in the "BFB Predictive Analysis White Paper" published by EPRI in MRP 2017-010 (Ref. 25).

During the April 12, 2017, public meeting, the NRC staff asked EPRI to discuss the basis for the six-year interval and whether it was supported by a generic analysis (Question 3).

EPRI's July 13, 2017, response to Question 3 stated that the basic goal of these screening thresholds from the interim guidance (downflow plants with ≥ 3% indications or clustering and upflow plants with ≥ 5% indications or clustering) was to avoid a potential clustering event. The response further stated that the industry requested three consultants to provide BFB prediction evaluations, and the evaluations used current OE data and existing IASCC initiation laboratory data. The response stated that the three consultants used significantly different approaches for predicting the degradation. The response also stated that while the models are proprietary, the results from all three vendors provided generally the same results, which were used as input in determining these screening thresholds for the interim guidance. EPRI's response also provided additional details on the three models.

By letter dated August 4, 2017 (Ref. 26), EPRI submitted to NRC for information only a Summary 'White Paper' of the Baffle-Former Bolt Prediction Results Provided by Structural Integrity Associates, AREVA, and Westinghouse (Ref. 25). The white paper contains additional details on the predictive models prepared by the three vendors. The NRC staff reviewed the white paper and notes that the Westinghouse model shows that acceleration of the failure rate due to clustering occurs at around 10% of the BFBs degraded.

The NRC staff further notes that for a degradation level 3% for downflow plants or 5% for upflow plants, all three models support a 6-year interval for subsequent examination to avoid reaching 10 % degradation. However, as degradation levels approach 10%, the models indicate that a subsequent examination interval shorter than 6 years is necessary to ensure a degradation level of 10%, with the associated acceleration of degradation and risk of clustering, is not reached before the next examination. The models described in the white paper are generic.

With respect to the maximum interval for subsequent examination for downflow plants, MRP 2017-009 establishes a low threshold for a maximum interval of 6 years of ≥ 3% BFBs with indications, or any clustering. This is a much smaller percentage of degraded bolts than has been observed in Tier 1a plants that have had significant degradation, and is also a smaller percentage of degraded BFBs than has been observed in 2-loop Tier 2 plants. The NRC staff therefore finds the threshold in terms of the percentage of degraded bolts to be appropriate.

Also, a shortened interval is appropriate if clustering is observed, since clustering is an indicator the BFB degradation is rapidly progressing. For upflow plants, MRP 2017-009 allows a 10-year maximum subsequent examination interval provided the percentage of BFBs with indications is <5%, and a 6-year maximum interval if the percentage of BFBs with indications is ≥ 5%, or clustering is present. The slightly higher threshold for upflow plants is appropriate because the likelihood of rapidly increasing degradation is much less in these plants.

However, the interim guidance in MRP 2017-009 also states that subsequent volumetric UT examinations shall be performed on an interval established by plant-specific evaluation per MRP-227 "Needed" Requirement 7.5 as documented and dispositioned in the owner's plant corrective action program. Based on the generic models, the staff considers it likely that plants exceeding the thresholds of 3% or 5% for a shortened subsequent examination interval could determine intervals shorter than 6 years for subsequent examinations.

MRP 2017-009 states on page 1 that this guidance is not intended to modify the acceptance criteria of WCAP-17096-NP-A [“Reactor Internals Acceptance Criteria Methodology and Data Requirements” (Ref. 27)], nor the expansion criteria associated with MRP-227, Section 5. These criteria may be adjusted in the future via other guidance.

However, the NRC staff points out that the maximum UT re-examination period in the table effectively does modify the acceptance criteria of WCAP-17096-NP-A, which allow a 10-year interval for subsequent UT examination if less than 50% of the margin bolts are degraded, since 3% or 5% of the bolts would generally be less than 50% of the margin bolts. At the April 12, 2017, public meeting, the NRC staff discussed this issue with EPRI (Question 4).

In its July 13, 2017, response to Question 4, EPRI stated that the degradation percentages in MRP 2017-009 were established as thresholds to require a heightened scrutiny of the re-inspection interval based on as-found conditions from the UT examinations. They were not intended to serve as acceptance criteria that supersede any criteria in WCAP-17096-NP-A. The response also stated the WCAP-17096-NP-A criteria must continue to be adhered to but a more conservative reinspection or subsequent examination interval was deemed appropriate while the industry sought to better understand the recent OE and account for it in updated acceptance criteria methodology guidance.

The response further stated that, as a result of recent BFB OE, the PWROG Materials Committee is currently working on a project that will address any changes needed to the guidance in WCAP-17096-NP-A. Finally the response stated that until such time that WCAP-17096-NP-A is revised, the industry team will be considering the need for interim guidance to the re-inspection or subsequent examination interval to address any apparent conflicts between the guidance in MRP 2017-009 and the current guidance in WCAP-17096-NP-A.

The NRC staff finds EPRI’s response to Question 4 acceptable because it clarifies that both criteria must continue to be met. Also, the NRC staff notes that the MRP 2017-009 criteria are generally more restrictive than the WCAP-17096-NP-A criteria. The former are likely to result in a shorter subsequent examination interval in cases where the WCAP-17096-NP-A guidance would allow a 10-year subsequent examination interval.

MRP 2017-009 states that the interim guidance is to be implemented in the first refueling outage after March 1, 2018, and also does not specify the examination coverage for the subsequent examinations. Therefore, the NRC staff requested the following clarifications (Question 5, Items 1 and 2) to the subsequent examination intervals given in the table during the public meeting on April 12 2017:

1. Does this table [page 2 of MRP 2017-009] apply to all UT examination results for BFBs or only those after March 1, 2018? For example, are the table recommendations retroactive to a plant that performed its MRP-227-A baseline UT examination of BFBs in 2012?
2. Is the coverage for the subsequent UT examination the same as the initial UT examination (100% of accessible BFBs)?

In its July 13, 2017, response to Item 1 of Question 5, EPRI stated that consistent with NSAL-16-1, which requires Tier 2 plants to review their previous MRP-227 examination results for clustering, this guidance would apply to previous MRP-227 inspections. So if a plant failed the criteria for a standard 10-year interval, they would need to plan for a 6-year re-inspection. The NRC staff finds this response acceptable because it is appropriate for plants to apply the latest guidance, regardless of when the actual inspection was performed.

In its July 13, 2017, response to Item 2 of Question 5, EPRI stated that the requirements of MRP-227 for coverage are not modified by the interim guidance unless requirement C is used as an alternative. Thus, plants performing MRP-227 reinspections or subsequent examinations are expected to obtain 100% of the accessible bolting population. Plants could justify alternatives using plant-specific evaluation in the plant's corrective action program. The NRC staff finds this response acceptable, since there is no reduction in coverage for subsequent examinations unless proactive bolt replacements are performed as allowed by requirement C of MRP 2017-009.

Question 5 is thus resolved based on the above.

Note (b) to the table applies to plants with a 6-year maximum reinspection or subsequent examination interval, and states that a longer interval, not to exceed 10-years, may be justified by plant-specific evaluation based on plant-specific exam findings. This evaluation may include additional justification from plant modifications and/or improvements (for example, replacements of BFBs, conversion to upflow, replacement of lower internals, etc.).

MRP 2017-009 also states that, as an alternative to performing UT inspections, a plant may perform proactive bolt replacements as preventative maintenance justified by plant-specific evaluation using established methodologies (for example, WCAP-15029-P-A ["Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions,"(Ref. 28[9])] or equivalent). The letter states: "The plant-specific evaluation shall also establish and justify the UT re-examination period resulting from the bolt replacements performed."

Since a plant may replace a number of original bolts less than the number that would be required for an acceptable bolt pattern with only replacement bolts, some original bolts may still be relied upon for structural integrity of the baffle-former assembly. Also, for plants that replaced sufficient bolts to constitute an acceptable bolting pattern with only replacement bolts, the interval for subsequent UT examination is not specified, so the interval is assumed to be a maximum of 10 years.

In addition, there are a few plants that replaced some BFBs during the late 1990's, such as Point Beach, Unit 2, Gina, and Farley Units 1 and 2. It is not clear if these early replacements could be credited under MRP 2017-009 Item C as an alternative to initial UT examination. Therefore, at the public meeting on April 12, 2017, the staff asked EPRI about the following (Question 6, Items 1 – 3):

1. Clarify the initial examination schedule for original bolts if the plant has a mixture of original and replacement bolts.

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[9] Westinghouse Report, WCAP-15030-NP-A, Rev. 0, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions under Faulted Load Conditions," Reference 29, is the non-proprietary version of WCAP-15029-P-A.

2. Clarify the schedule for subsequent examinations for a plant that has replaced sufficient BFBs to constitute an acceptable bolting pattern with only replacement bolts.
3. For plants that replaced BFBs earlier in plant life, is there any adjustment to the initial UT examination schedule or coverage in consideration of these replacements?

In its July 13, 2017, response to Item 1 of Question 6, EPRI stated that if bolts are proactively replaced to less than an acceptable pattern, UT currently would still be required according to their applicable initial examination schedule. The response further stated that a plant may be able to justify reduced inspection coverage to only original bolts in this case, and that furthermore, if a plant replaces bolts in phases and completes replacement to an acceptable pattern prior to the required baseline UT inspection, a plant-specific evaluation may be used to justify modifying the UT inspection schedule for the bolts.

The staff finds this response acceptable because replacement bolts are less susceptible to IASCC and probably would not develop IASCC for many years, if ever during plant operation. Severe undetected degradation of nearby original bolts could cause accelerated degradation of replacement bolts, but should not be present since original bolts are still being examined by UT.

In response to Item 2 of Question 6, EPRI confirmed that there potentially are scenarios where a plant could justify a re-inspection longer than 10 years, but there are too many plant-specific variables to write concise generic guidance to that effect. The response further stated that key considerations would be whether these BFB replacements were conducted proactively or in response to significant degradation and any other site specific actions performed (e.g., upflow modification or other mitigating strategies).

In its July 13, 2017, response to Item 3 of Question 6, EPRI stated that, in general, currently there is no adjustment to the initial UT exam schedule or coverage requirements due to the replacements early in plant life.

Based on EPRI's response to Question 6, the NRC staff understands that the interim guidance does not generically allow modifications to the initial examination schedule for BFBs for plants that previously replaced BFBs to less than an acceptable pattern. In addition, the NRC staff understands that the interim guidance does not allow a subsequent examination interval longer than 10 years for replacement BFBs. However, both of these options could be implemented by a plant if a plant-specific evaluation is performed justifying the change in examination schedule.

The NRC staff understanding is that such plant-specific changes would be identified as deviations from the NEI 03-08, Rev 2 "Needed" guidance in MRP 2017-009. The NRC staff finds EPRI's response to Question 6 acceptable since it clarifies that the interim guidance does not generically allow modifications to the initial or subsequent examination schedules based on replacement of bolts.

At the conclusion of the April 12, 2017, meeting, the NRC staff expressed two additional concerns regarding the interim guidance (Ref. 11):

1. Under Item B of the interim guidance, subsequent UT examination shall be performed on an interval established by plant-specific evaluation per MRP-227-A, "Needed" Requirement 7.5 as documented and dispositioned in the owner's plant corrective action. However, the interim guidance does not provide any requirements or guidance

for the plant-specific evaluation. Therefore, the NRC staff is concerned that the level of rigor of these plant-specific evaluations may vary widely.

2. The second concern is related to Note (b) to the table and is described in Question 7 below.

Based on its concerns, the NRC staff added a new Question 7:

Note (b) to the table in MRP 2017-009 states:

> (b) A longer re-inspection interval, not to exceed 10 years, may be justified by plant-specific evaluation based on plant-specific exam findings. This evaluation may include additional justification from plant modifications and/or improvements (for example, replacements of BFBs, conversion to upflow, replacement of lower internals, etc.)

The NRC staff is concerned that Note (b) could allow plants to exceed the maximum subsequent examinations intervals without having to deviate from the NEI 03-08, Rev 2 “Needed” guidance of MRP 2017-009. Therefore, the NRC staff would not be informed of plants exceeding the table intervals and would not have an opportunity to review the plant-specific evaluations supporting the longer interval. MRP 2017-009 also does not provide any guidance for the methodology of the plant-specific evaluations. Therefore, the level of rigor of these evaluations could vary widely.

In Question 7, the NRC staff therefore requested that EPRI consider removing note (b) from the interim guidance, or adding more detailed guidance for the methodology of the plant-specific evaluation of the subsequent examination interval.

In its July 13, 2017, response to Question 7, EPRI stated that, consistent with the response to Question 4, the guidance relating to the re-inspection criteria (specifically Note (b)) will be revised by incorporating the Interim Guidance of MRP 2017-009 into MRP 227, Rev 1-A should the NRC staff find MRP-227, Rev 1 acceptable for use. Alternately, EPRI identified that the revision could be done in the revised guidance for W-ID: 7 in WCAP-17096-NP-A (potentially via interim guidance) to require submittal consistent with guidance imposed for several CE and Westinghouse core barrel welds, designated CE-ID: 6 and 7 and W-ID: 3, 3.1, 4, and 5.

The guidance for these welds state: “Any proposal for extension of the verification period beyond a single refueling cycle or use of an alternative verification process would require a technical basis to be submitted to the regulator.” The response further stated that similar language consistent with the above will be added such that the plant-specific evaluation described in Note (b) to justify a longer re-inspection interval will be submitted to NRC for information within one year after any BFB inspection or bolt replacement activity for which the results trigger the reduced reinspection or subsequent examination interval. If the evaluation is completed after this one year timeframe it shall be submitted within 90 days of completion of the evaluation.

The NRC staff finds EPRI’s proposal to modify the guidance related to subsequent examinations to require submittal of any plant-specific evaluations to the NRC, to be partially acceptable. Submission within one year of the BFB inspection or bolt replacement activity that triggers the reduced subsequent examination interval is acceptable, since it would give the NRC staff an opportunity to review this evaluation well before the end of the reduced subsequent examination interval.

However, the language allowing the evaluation to be submitted within 90 days of completion could allow these evaluations to be submitted very close to the end of the 6-year interval. This would not allow sufficient review time for the NRC staff prior to the end of the reduced subsequent examination interval.

During a conference call on July 25, 2017, EPRI clarified that the language allowing submission of the evaluation within 90 days of completion of the evaluation if completed after the one year timeframe, was intended to give licensees the flexibility to decide whether to perform this evaluation more than one year from the initial examination. For example, a licensee whose initial examination results dictated a six-year maximum subsequent examination interval might decide three years into the six-year interval to perform a plant-specific evaluation to extend the interval.

However, the NRC staff preference would be to receive these evaluations a minimum of one year before the current scheduled subsequent examination. Therefore, the NRC staff recommends that the final guidance incorporated into MRP-227, Rev. 1 or a revision to WCAP-17096-NP-A, require submitting to the NRC staff for information the plant-specific evaluations to justify extending the subsequent examination interval a minimum of one year prior to the end of the subsequent examination interval.

As previously noted, based on its review of the generic BFB predictive models in the white paper (MRP Letter 2017-010, Ref. 25), the NRC staff considers it likely that some plants exceeding the thresholds of 3% or 5% for a shortened subsequent examination interval may need subsequent examination intervals of less than 6 years in order to ensure accelerated degradation and clustering do not occur. The NRC staff therefore considers it appropriate for the plant-specific evaluations for such plants to be submitted to the NRC for information. Therefore, the NRC recommends the following change be implemented in the final guidance:

> If the table in MRP 2017-009 indicates that the subsequent inspection interval is not to exceed 6 years (e.g., downflow plants with ≥ 3% BFBs with indications or clustering, or upflow plants with ≥ 5% of BFBs with indications or clustering), the plant-specific evaluation to determine a subsequent inspection interval should be submitted to the NRC for information within one year following the outage in which the degradation was found. If the licensee later decides to revise the initial plant-specific evaluation to extend the previously determined interval, the revised evaluation should be submitted to the NRC for information at least one year prior to the end of the current applicable interval for BFB subsequent examination. This recommendation should be incorporated into the final version of MRP-227, Rev. 1 should the NRC staff find it acceptable for use, and/or a revision to WCAP-17096-NP Rev. 2-A.

4. CONCLUSIONS

The NRC staff concludes that the guidance with respect to initial examination schedules, and the maximum limits on subsequent examination intervals in EPRI MRP 2016-021 and MRP 2017-009, as modified by the responses to the NRC staff questions in EPRI's July 13, 2017, letter, provides for acceptable aging management of BFBs in Westinghouse and Combustion Engineering design RVI. However, the NRC staff recommends the following:

> If the table in MRP 2017-009 indicates that the subsequent inspection interval is not to exceed 6 years (e.g., downflow plants with ≥ 3% BFBs with indications or clustering, or upflow plants with ≥ 5% of BFBs with indications or clustering), the plant-specific evaluation to determine a subsequent inspection interval should be submitted to the NRC for information within one year following the outage in which the degradation was found. If the licensee later decides to revise the initial plant-specific evaluation to extend the previously determined interval, the revised evaluation should be submitted to the NRC for information at least one year prior to the end of the current applicable interval for BFB subsequent examination. This recommendation should be incorporated into the final version of MRP-227, Rev. 1 should the NRC staff find it acceptable for use, and/or a revision to WCAP-17096-NP Rev. 2-A.

This assessment is not binding to licensees, and compliance with the recommendation above is strictly voluntary.

The NRC staff conclusions in this assessment will be superseded, as necessary, based on the findings related to BFB aging management in the safety evaluation of MRP-227, Rev. 1 should the NRC staff find it acceptable for use.

5. REFERENCES

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2. Letter from Bernie Rudell and Brian Burgos dated March 23, 2017, Transmittal of NEI 03-08, "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for U.S. PWR Plants as Defined in Westinghouse NSAL 16-01" (MRP 2017-011) (ADAMS Accession No. ML17087A107).
3. NEI 03-08, "Guideline for the Management of Materials Issues," Revision 3, Nuclear Energy Institute, February 2017 (ADAMS Accession No.).
4. NEI-03-08, "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for PWR Plants as Defined in Westinghouse NSAL 16-01 Rev.1. (MRP 2017-009), March 15, 2017 (ADAMS Accession No. ML17087A106).
5. NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt inspections for Tier 2 Plants as Defined in Westinghouse NSAL 16-01, January 12, 2017 (MRP 2017-002) (ADAMS Accession No. ML17017A165)

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7. Information Notice 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants," March 25, 1998.
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12. List of Questions Related to MRP 2017-009 (April 3, 2017), April 11, 2017 (ADAMS Accession No. ML17101A421).
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14. Licensee Event Report (LER) 2016-004-00, "Unanalyzed Condition due to Degraded Reactor Baffle-Former Bolts," Indian Point Unit No. 2, Docket No. 50-247, DPR-46, May 31, 2016 (ADAMS Accession No. ML16159A219).
15. LER 272/2016-001-00, "Unanalyzed Condition due to Degraded Reactor Baffle to Former Bolts," Salem Nuclear Generating Station Unit 1, Renewed Facility Operating License No. DPR-70, NRC Docket No. 50-272, July 5, 2016 (ADAMS Accession No. ML16187A330).
16. Westinghouse Technical Bulletin TB-12-5, "Baffle Bolt Degradation in a Westinghouse NSSS Plant with Downflow Reactor Internal Design," March 7, 2012. (NOT PUBLICALLY AVAILABLE).
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19. "Baffle-Former Bolt Overview of Industry Experience," presentation by Heather Malikowski at the July 19, 2016 Public Meeting between NRC and the Baffle-former Bolt Focus Group (ADAMS Accession No. ML16203A025).

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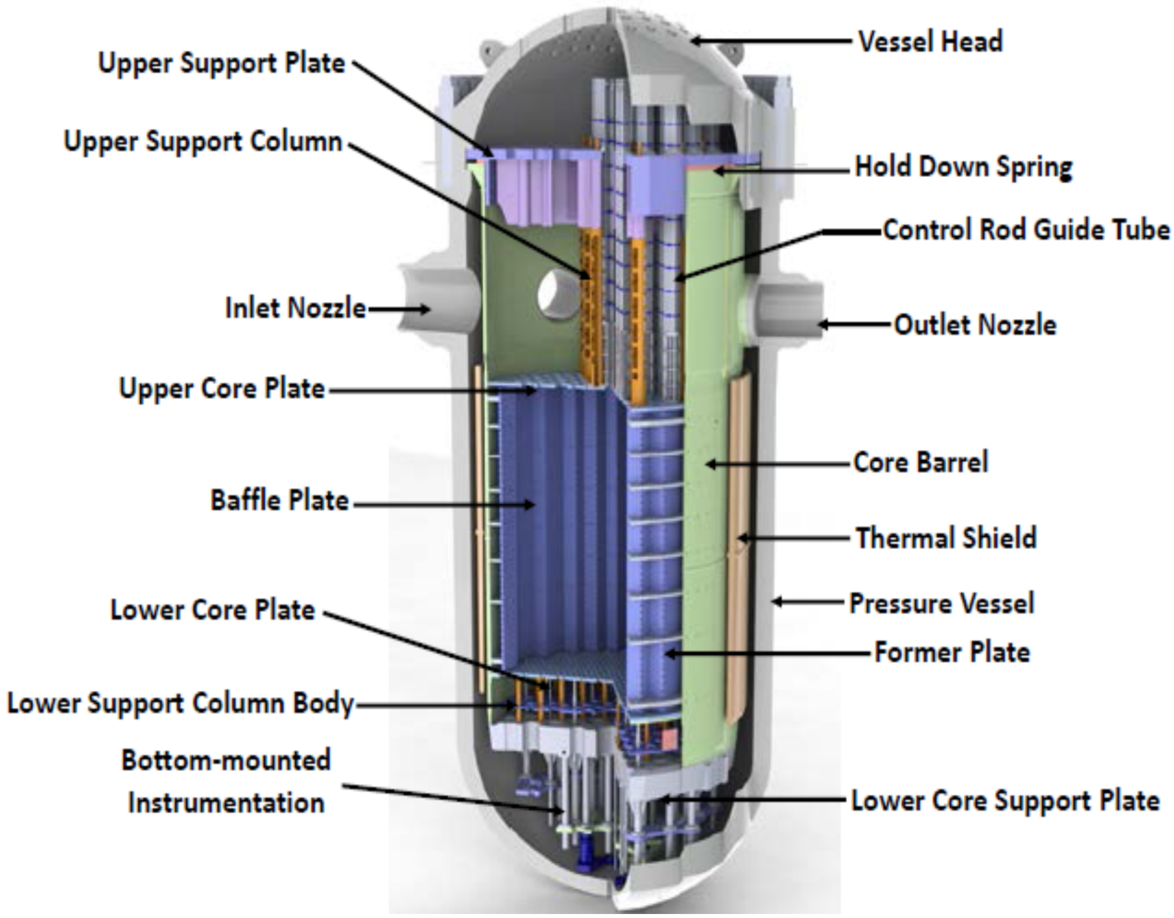

Figure 1. Typical Westinghouse reactor vessel internals general arrangement and major components

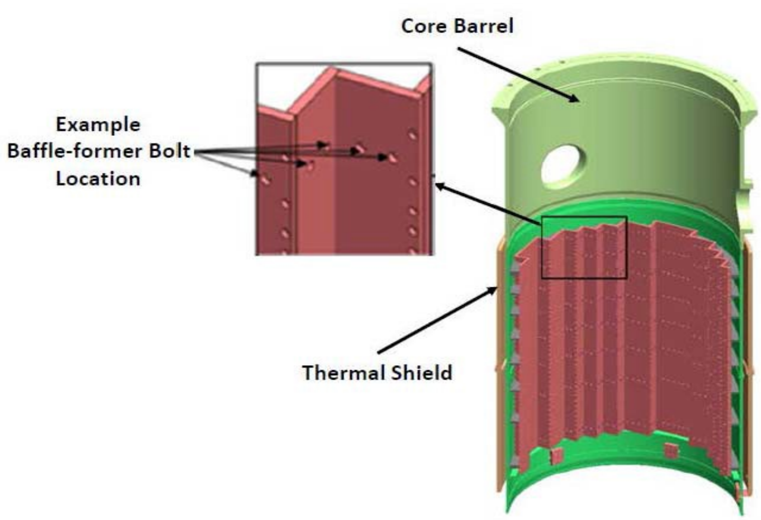

Figure 2. Typical Westinghouse baffle-former assembly

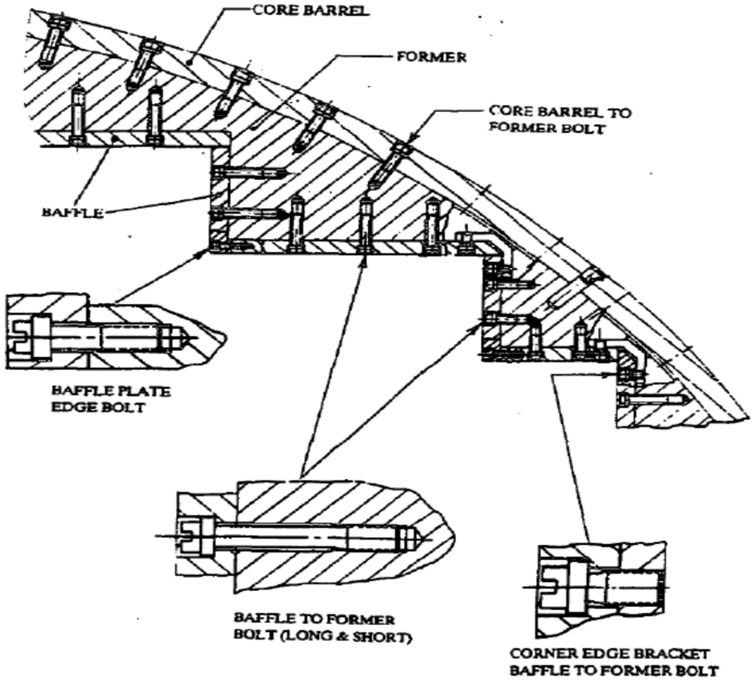

Figure 3. Cross section of one octant of a typical Westinghouse baffle-former assembly showing locations of bolt types

Figure 4. Appearance of BFB head as seen from the core side

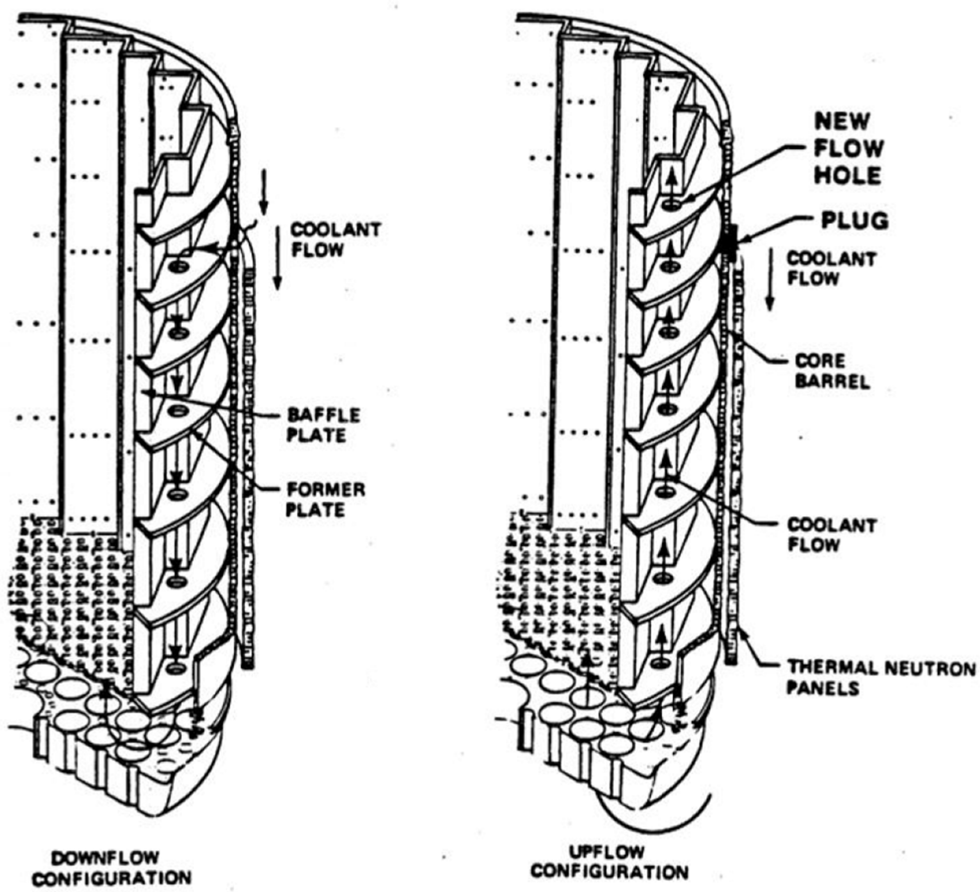

Figure 5. Schematic of flow paths in Westinghouse-designed reactor internals, comparing downflow and upflow configurations