ENCLOSURES 5, 6, AND 7 CONTAIN PROPRIETARY INFORMATION
WITHHOLD FROM PUBLIC DISCLOSURE IN ACCORDANCE WITH 10 CFR 2.390

1717 Wakonade Drive
Welch, MN 55089

May 24, 2019 L-PI-19-001

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

Prairie Island Nuclear Generating Plant, Units 1 and 2
Docket Nos. 50-282 and 50-306
Renewed Facility Operating License Nos. DPR-42 and DPR-60

Voluntary Submittal of Plant-Specific Evaluation to Extend the Re-Inspection Interval for Baffle Former Bolts

References:
1) Materials Reliability Program Letter, MRP 2017-009, " Transmittal of NEI-03-08 'Needed' Interim Guidance Regarding Baffle Former Bolt Inspections for PWR Plants as Defined in Westinghouse NSAL 16-01 Rev. 1," March 15, 2017.
2) U.S. Nuclear Regulatory Commission Document, "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." [ADAMS Accession Number: ML17310A861]

Per Reference 2, Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), is providing to the U.S. Nuclear Regulatory Commission (NRC) the enclosed plant-specific evaluations that support extending the re-inspection interval for baffle former bolts at the Prairie Island Nuclear Generating Plant (PINGP) Unit 1 and Unit 2. Per the staff recommendation in Reference 2, this information is being provided voluntarily for information and no action is requested.

The evaluations in the enclosures provide the technical basis for extending the PINGP Unit 1 and Unit 2 baffle former bolt inspection intervals from six years to ten years, which are summarized as follows:

- Enclosure 1 is Westinghouse letter LTR-AMLR-18-55-NP, "Prairie Island Units 1 and 2 Baffle Former Bolt Predictive Evaluation," which is a plant-specific evaluation to estimate future degradation. Enclosure 1 is a non-proprietary version of Enclosure 5.
- Enclosure 2 is NSPM PINGP Evaluation 608000000371, "Baffle Bolt Core Bypass Flow Predictive Modeling." Evaluation 608000000371 addresses core bypass flow and supplements information developed in Westinghouse letter LTR-AMLR-18-55-P

(Enclosure 5). In the evaluation, NSPM concluded that a 10-year reinspection interval for both Unit 1 and Unit 2 provides reasonable assurance that core bypass flow design limits will continue to be met at the time of the next inspection. Enclosure 2 is the non-proprietary version of Enclosure 6.

- Enclosure 3 is Westinghouse letter LTR-AMLR-19-18-NP, “Prairie Island Unit 2 – Baffle-Edge bolt Sensitivity Study,” which is a baffle edge bolt sensitivity study regarding the impact of baffle-edge bolts on core bypass flow under degraded baffle-former bolt conditions. This sensitivity study further supports the acceptability of a 10-year reinspection interval. Enclosure 3 is a non-proprietary version of Enclosure 7.
- Enclosure 4 is an affidavit from Westinghouse that declares the basis for proposing the proprietary information be withheld from public disclosure. Enclosures 5, 6, and 7 contain information proprietary to Westinghouse Electric Company LLC ("Westinghouse"), which are supported by an Affidavit signed by Westinghouse, the owner of the information. The Affidavit sets forth the basis on which the information may be withheld from public disclosure by the Nuclear Regulatory Commission ("Commission") and addresses with specificity the considerations listed in paragraph (b)(4) of Section 2.390 of the Commission's regulations. Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.
- Enclosure 5 provides a copy of Westinghouse letter LTR-AMLR-18-55-P, “Prairie Island Units 1 and 2 Baffle Former Bolt Predictive Evaluation.” Westinghouse letter LTR-AMLR-18-55-P has been determined by Westinghouse to be proprietary and is requested to be withheld from public disclosure in accordance with 10 CFR 2.390.
- Enclosure 6 is NSPM PINGP Evaluation No. 608000000371-P, “Baffle Bolt Core Bypass Flow Predictive Modeling. Enclosure 6 contains information that has been determined by Westinghouse to be proprietary to Westinghouse and is requested to be withheld from public disclosure in accordance with 10 CFR 2.390.
- Enclosure 7 provides a copy of Westinghouse letter LTR-AMLR-19-18-P, “Prairie Island Unit 2 – Baffle-Edge bolt Sensitivity Study.” Westinghouse letter LTR-AMLR-19-18-P has been determined by Westinghouse to be proprietary and is requested to be withheld from public disclosure in accordance with 10 CFR 2.390.

Summary of Commitments

This letter makes no new commitments and no revisions to existing commitments.

If there are any questions or if additional information is needed, please contact Mr. Jeff Kivi at 612-330-5788.

Scott Sharp

Scott Sharp
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosures:

1. Westinghouse Letter LTR-AMLR-18-55-NP, Rev. 1, “Prairie Island Units 1 and 2 Baffle Former Bolt Predictive Evaluation,” October 16, 2018 – NON-PROPRIETARY.
2. Redacted NSPM PINGP Evaluation No. 608000000371, “Baffle Bolt Core Bypass Flow Predictive Modeling,” dated May 2, 2019 – NON-PROPRIETARY.
3. Westinghouse Letter LTR-AMLR-19-18-NP, Rev. 0, “Prairie Island Unit 2 – Baffle-Edge Bolt Sensitivity Study,” March 21, 2019 – NON-PROPRIETARY
4. Affidavit
5. Westinghouse Letter LTR-AMLR-18-55-P, Rev. 1, “Prairie Island Units 1 and 2 Baffle Former Bolt Predictive Evaluation,” October 16, 2018 – PROPRIETARY.
6. NSPM PINGP Evaluation No. 608000000371-P, “Baffle Bolt Core Bypass Flow Predictive Modeling,” dated May 2, 2019 – PROPRIETARY.
7. Westinghouse Letter LTR-AMLR-19-18-P, Rev. 0, “Prairie Island Unit 2 – Baffle-Edge Bolt Sensitivity Study,” March 21, 2019 - PROPRIETARY

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC

ENCLOSURE 1

Westinghouse Letter LTR-AMLR-18-55-NP,
“Prairie Island Units 1 and 2 Baffle Former Bolt Predictive Evaluation”
(Non-Proprietary)

21 pages follow

Westinghouse

To:	Nathan Lang Reactor Internals Design & Analysis I	Date:	December 19, 2018
From:	Louis Turicik Aging Management & License Renewal	Your ref:	N/A
Ext:	412-374-6072	Our ref:	LTR-AMLR-18-55-NP, Rev. 1

Subject: **Prairie Island Units 1 and 2 Baffle Former Bolt Predictive Evaluation**

Xcel Energy (Xcel) completed ultrasonic testing (UT) examination of the baffle-former bolts (BFBs) at Prairie Island Units 1 and 2 during the fall 2014 and fall 2013 refueling outages, respectively. Westinghouse has performed a probabilistic predictive evaluation of the BFBs for Prairie Island Units 1 and 2, detailed in CN-AMLR-18-9 [1]. The results of this evaluation support determination of BFB re-inspection intervals and planning for potential contingency options. This letter summarizes the results of that evaluation.

Revision 1 addresses customer comments regarding revision 0 of this letter (original issue).

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[1]**Electronically approved records are authenticated in PRIME.**

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*** This record was final approved on 12/19/2018 2:54:19 PM. (This statement was added by the PRIME system upon its validation)

1.0 Background

Xcel performed volumetric UT inspections of the Prairie Island Units 1 and 2 BFBs during the fall 2014 and fall 2013 refueling outages, respectively. Approximately 5.5% of the 728 Prairie Island Unit 1 BFBs were found to be with UT indication of degradation [2]. Approximately 10.3% of the Unit 2 BFBs were found with UT indications of degradation [3]. Figure 1 and Figure 2 depict these inspection results. No replacement activity was conducted by Xcel after the inspections.

Baffle-former bolt degradation is a known occurrence in pressurized water reactor operating experience and believed to be initiated by Irradiation Assisted Stress Corrosion Cracking (IASCC). The Prairie Island Units 1 and 2 baffle-former bolt material (Type 347 stainless steel) [12, 13] is susceptible to IASCC [14, 15, 16, 17]. The observed degradation of BFBs led to Westinghouse development of the acceptable bolting pattern analysis (ABPA) methodology [18], which has since been applied to several operating plants. In terms of BFB degradation, Prairie Island Units 1 and 2 were designated "Tier 2a," in [4] since both are 2-Loop downflow plants. The recommendations of [4] were used to create interim guidance letters [5] and [6] (Assessed by the NRC in [7]) for MRP-227-A [8] and were incorporated into NEI-03-08 "Needed" Requirements. An ABPA can be used to assess whether a baffle-former assembly with degraded BFBs is acceptable for continued operation. However, it does not directly consider future degradation rates of bolts or the locations where degradation may occur. Due to the quantity of indications seen during the BFB inspection of Prairie Island Units 1 and 2, the degradation category has been classified as "Accelerated." As a plant-specific evaluation to estimate future degradation, a BFB predictive evaluation is required [10].

			14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39
	Q1	G	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1
Failed	11	F	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1
Total	182	E	1	1	1	1	0	1	1	1	1	0	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1
% Fail	6.0%	D	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
		C	1	1	1	1	1	0	0	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
		B	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1
		A	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1

			40	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65
	Q2	G	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
Failed	2	F	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
Total	182	E	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1
% Fail	1.1%	D	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
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		B	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1
		A	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1

			66	67	68	69	70	71	72	73	74	75	76	77	78	79	80	81	82	83	84	85	86	87	88	89	90	91
	Q3	G	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
Failed	13	F	1	1	1	0	0	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1
Total	182	E	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	0	1	1	1	0	1
% Fail	7.1%	D	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
		C	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
		B	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1
		A	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	0	0	1	1	1	1

			92	93	94	95	96	97	98	99	100	101	102	103	104	1	2	3	4	5	6	7	8	9	10	11	12	13
	Q4	G	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
Failed	14	F	1	1	1	0	1	1	1	1	1	0	1	1	1	1	1	0	0	1	1	1	1	1	1	1	1	1
Total	182	E	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	0	1	1	1	1	1	1	1	1	1
% Fail	7.7%	D	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
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		B	1	1	1	1	0	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
		A	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	0	0	1	1

Figure 1: Prairie Island Unit 1 As-Found Pattern of Bolts with Ultrasonic Test Indications [1]

			14	15	16	17	18	19	20	21	22	23	24	25	26	27	28	29	30	31	32	33	34	35	36	37	38	39
	Q1	G	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
Failed	13	F	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	0	1	1	1	1	0	1	1	1	1
Total	182	E	1	1	1	1	1	1	1	1	0	0	1	1	1	1	1	1	0	0	1	1	1	1	1	1	1	1
% Fail	7.1%	D	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
		C	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
		B	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	0	1	1	1	0	1	1	1	1
		A	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	0	1	1	0	1	1	1	1	1	1

			40	41	42	43	44	45	46	47	48	49	50	51	52	53	54	55	56	57	58	59	60	61	62	63	64	65
	Q2	G	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
Failed	19	F	1	1	1	1	1	1	1	1	0	0	1	1	1	1	1	1	0	0	1	1	1	1	1	1	1	0
Total	182	E	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	0	1	1	1	0	1	1	1	1
% Fail	10.4%	D	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1
		C	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
		B	1	1	1	1	1	1	0	1	1	0	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1
		A	1	1	1	1	0	0	1	1	0	1	1	1	1	1	1	1	1	0	1	0	0	0	1	1	1	1

			66	67	68	69	70	71	72	73	74	75	76	77	78	79	80	81	82	83	84	85	86	87	88	89	90	91
	Q3	G	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
Failed	27	F	1	1	1	0	0	0	0	1	1	0	1	1	1	1	1	1	0	0	1	1	1	1	1	1	1	1
Total	182	E	1	1	1	1	0	1	1	1	0	0	1	1	1	1	1	1	0	0	1	1	1	0	1	1	1	1
% Fail	14.8%	D	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
		C	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
		B	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	0	1	0	1	0	1	1	1	0
		A	1	1	1	0	0	0	0	1	0	1	1	1	1	1	1	1	1	0	1	1	0	0	1	1	1	1

			92	93	94	95	96	97	98	99	100	101	102	103	104	1	2	3	4	5	6	7	8	9	10	11	12	13
	Q4	G	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
Failed	16	F	1	1	1	0	0	1	1	1	0	0	1	1	1	1	1	1	0	0	1	1	1	1	1	1	1	1
Total	182	E	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1
% Fail	8.8%	D	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1
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		B	1	1	1	1	1	0	0	1	1	0	1	1	1	1	1	1	0	1	1	0	1	1	1	1	1	1
		A	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1

Figure 2: Prairie Island Unit 2 As-Found Pattern of Bolts with Ultrasonic Test Indications [3]

2.0 BFB Predictive Evaluation Overview

The Westinghouse BFB predictive methodology used in this evaluation is summarized in [20] and detailed in [21]. This methodology simulates the degradation of BFBs due to IASCC by characterizing the evolution of stress in the baffle-former assembly and performing probability-based predictions of which bolts will fail. Empirically-validated Weibull parameters are utilized to determine the probability of IASCC initiation at each bolt location, and as bolts fail due the cumulative effects of stress, reactor environment, neutron dose, and time, the acceleration in bolt failure rate caused by stress redistribution from neighboring bolt failures is addressed. The bolt degradation model within the evaluation is exercised as a Monte Carlo simulation to evaluate a range of plausible scenarios from which trends of bolt failure rates and patterns can be determined. The predictive evaluation was developed from industry operating experience, laboratory testing, and a plant-specific finite element analysis that quantifies stress at each bolt location throughout the reactor operating history.

This stochastic, semi-empirical model is constructed as a probabilistic network/influence diagram, as illustrated in Figure 3.

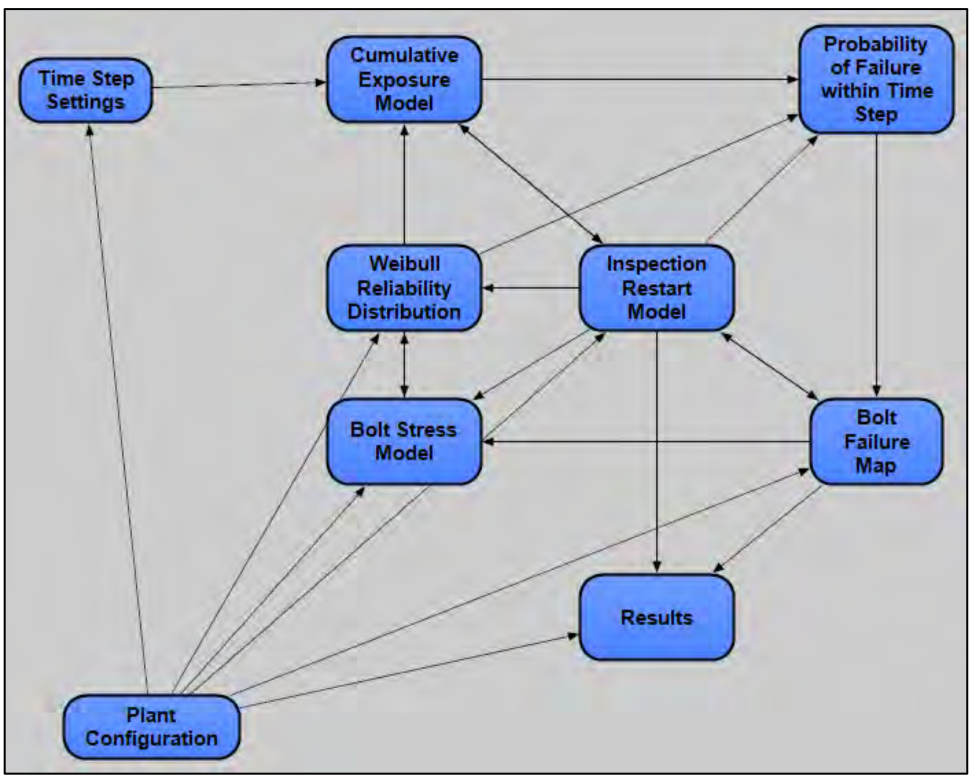


Figure 3: Influence Diagram of BFB Predictive Reliability Model

The bolt degradation model within the generic predictive evaluation was calibrated to Prairie Island Units 1 and 2 operating experience and modified to include the following plant-specific inputs:

- Prairie Island 1 and 2 baffle plate and BFB geometry
- Prairie Island 1 and 2 preload and pressure stresses at each bolt location

- Prairie Island 1 and 2 bolt design stress concentration factor including head-to-shank fillet radius
- Prairie Island 1 and 2 differential pressure across the baffle plates
- Prairie Island 1 and 2 stress relaxation due to accumulated dose at each BFB location
- Prairie Island 1 and 2 UT inspection results

The analysis includes the following two cases:

- Prairie Island Unit 1 prediction given 2014 inspection results [1]
- Prairie Island Unit 2 prediction given 2013 inspection results [3]

The data from each simulation run are displayed in plots of the predicted overall fraction of failed bolts versus time, expressed as effective full-power years (EFPY). Heat maps of bolt failure probability at each location for a specific point in time (six and ten years of operation following the inspection) are also provided. Additionally, each of the specific bolting patterns generated by the predictive model is evaluated at future time steps to determine when the plant will be at risk of having an unacceptable bolt pattern.

3.0 Prairie Island Units 1 and 2 BFB Predictive Evaluation Results

The BFB predictive model evaluates the unit on a quadrant basis. The baffle-former-assembly quadrant locations are defined in Figure 4. The predictive model quadrants are centered at 0°, 90°, 180°, and 270° locations around the baffle assembly. The intent of this orientation is to avoid dividing the baffle plate with the largest span of BFBs (six BFB across) between quadrants. For example, quadrant 1 is defined in the predictive model as octants 2 and 3.

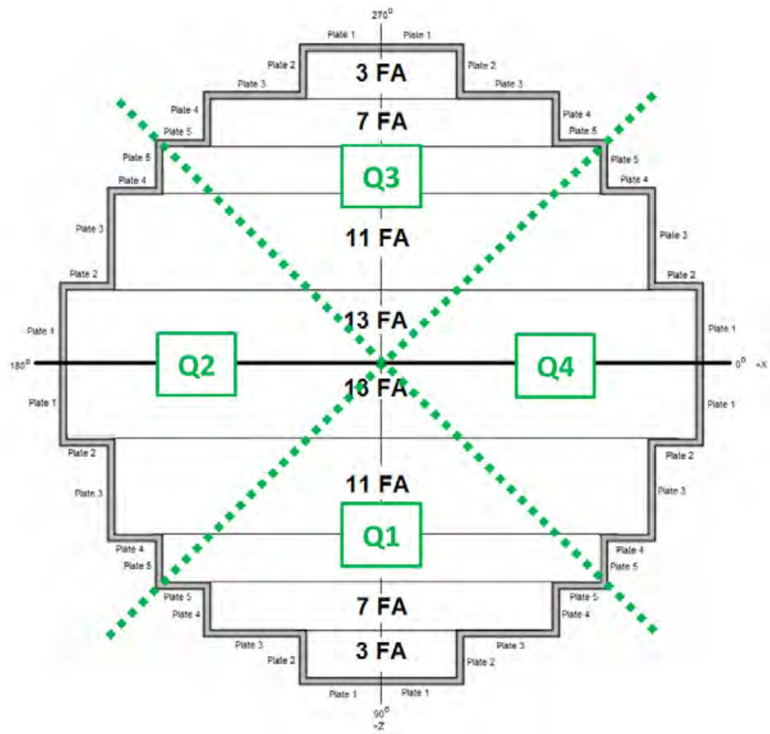


Figure 4: Quadrant Definition for the PINGS Predictive Model

Figure 5 through Figure 12 depict the predicted proportion of failed bolts for each quadrant, given the inspection results. Each figure also includes a predicted bolt failure probability heat map at the end of six and ten years of operation following the inspections that were conducted in 2014 and 2013 for Units 1 and 2 respectively. In the heat maps, locations that display a red cell indicate a bolt that had a UT indication during inspection, and is considered failed.

The proportion failed plots provided in this section give the 50th percentile, median prediction along with higher and lower percentile predictions. These prediction lines indicate the percentage of the Monte Carlo results that were above or below the given line. For example, the 75th percentile line indicates that 75 percent of the results were at or below that line, while 25 percent of the results were above that line.

a,c

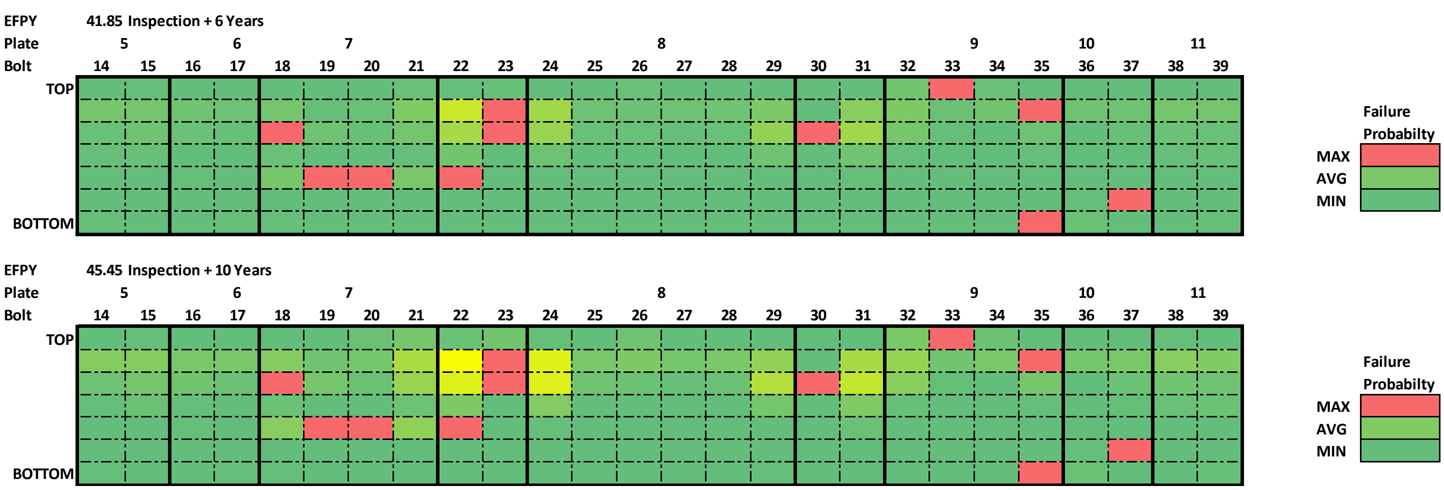
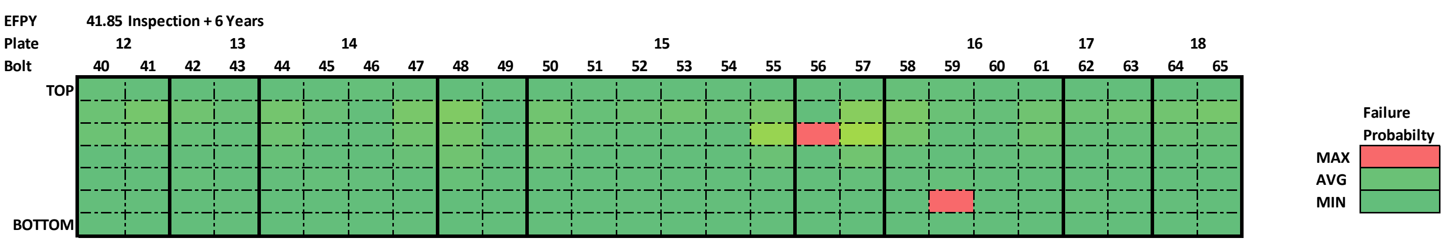


Figure 5: Prairie Island Unit 1 Quadrant 1 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map Following Six and Ten Years of Operation After Inspection

a,c



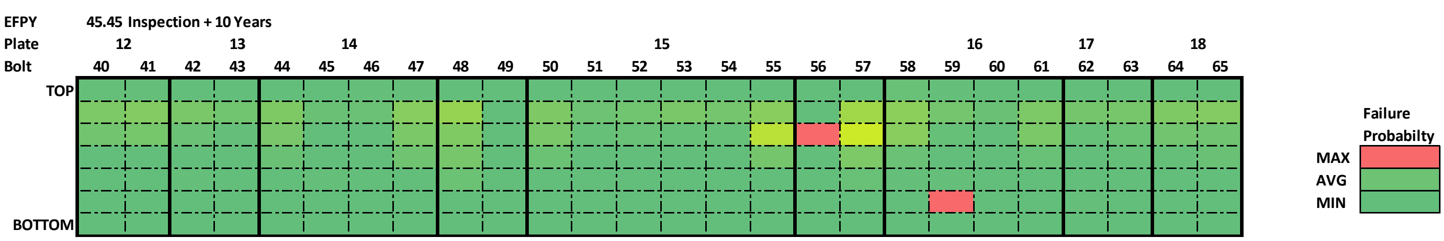


Figure 6: Prairie Island Unit 1 Quadrant 2 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map Following Six and Ten Years of Operation After Inspection

a,c

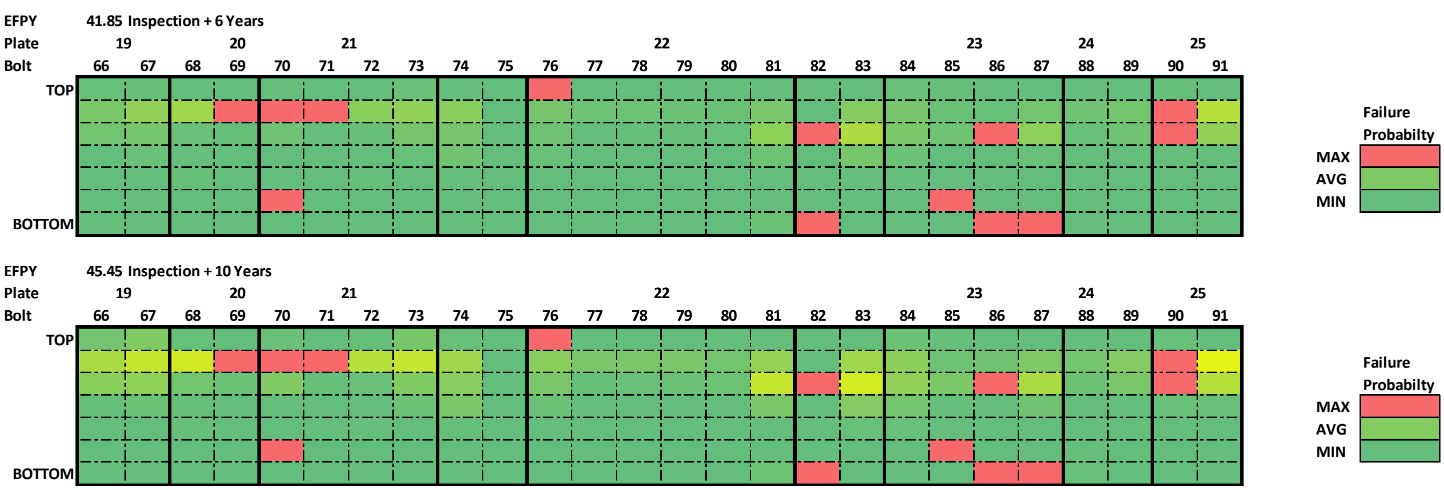
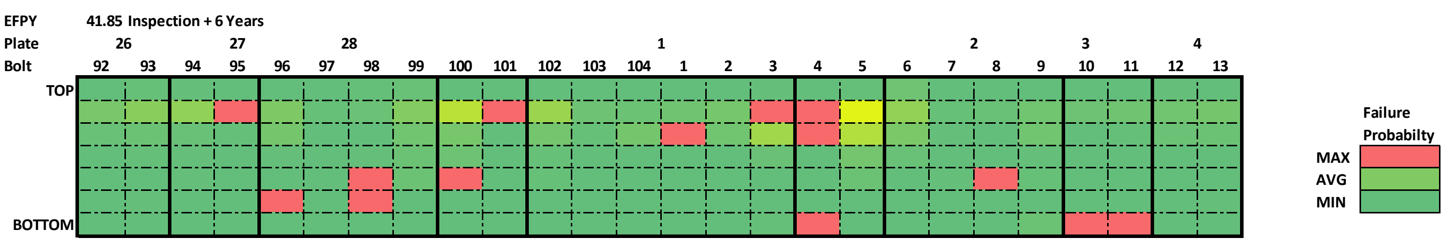


Figure 7: Prairie Island Unit 1 Quadrant 3 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map following Six and Ten Years of Operation After Inspection

a,c



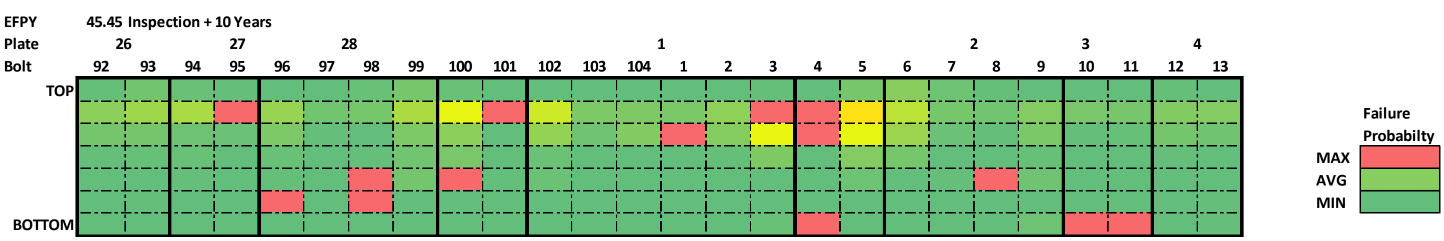


Figure 8: Prairie Island Unit 1 Quadrant 4 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map following Six and Ten Years of Operation After Inspection

a,c

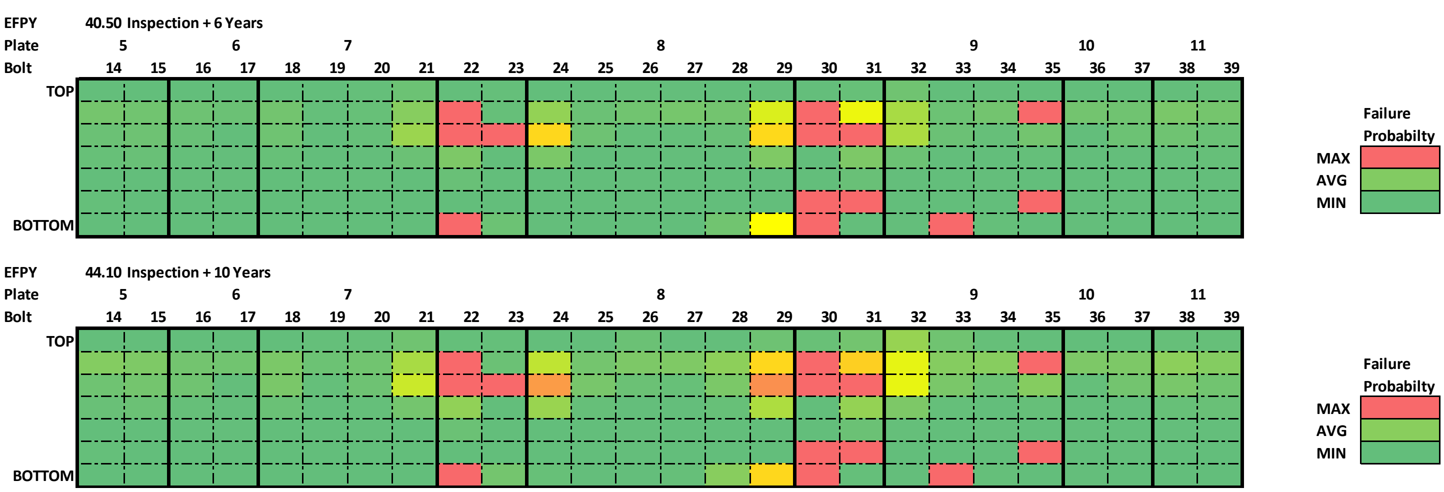


Figure 9: Prairie Island Unit 2 Quadrant 1 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map Following Six and Ten Years of Operation After Inspection

*** This record was final approved on 12/19/2018 2:54:19 PM. (This statement was added by the PRIME system upon its validation)

a,c

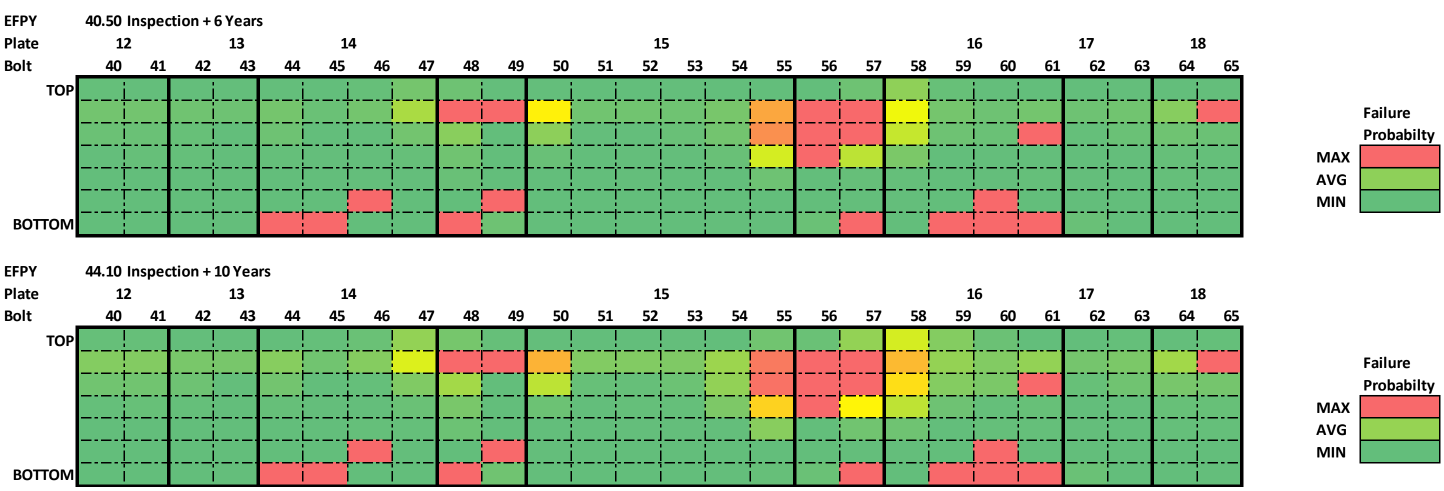


Figure 10: Prairie Island Unit 2 Quadrant 2 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map Following Six and Ten Years of Operation After Inspection

a,c

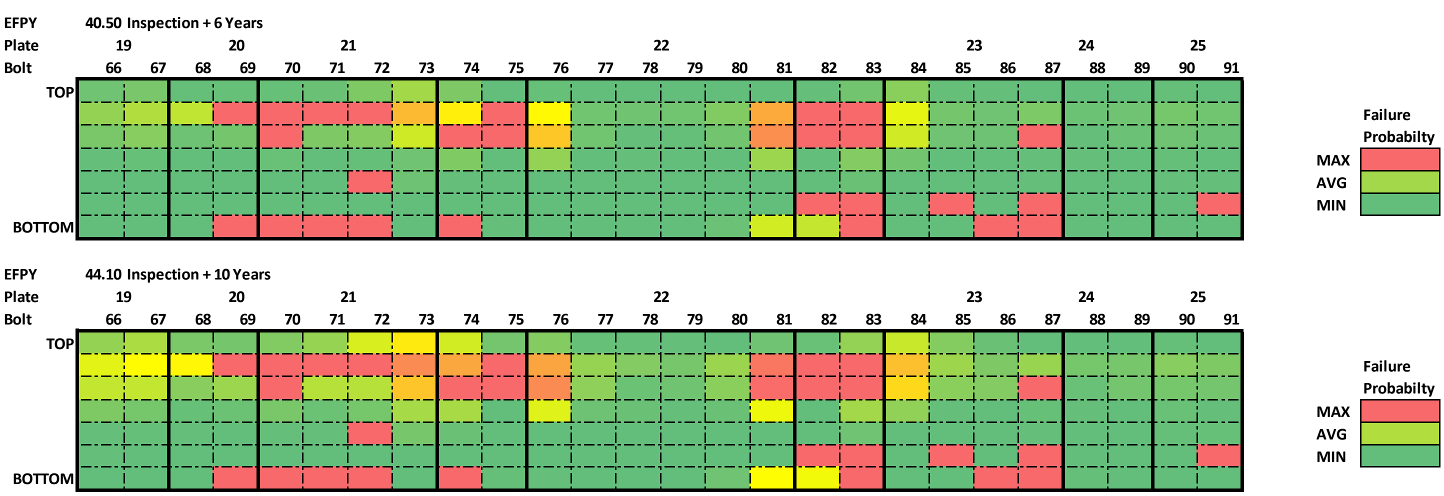


Figure 11: Prairie Island Unit 2 Quadrant 3 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map Following Six and Ten Years of Operation After Inspection

a,c

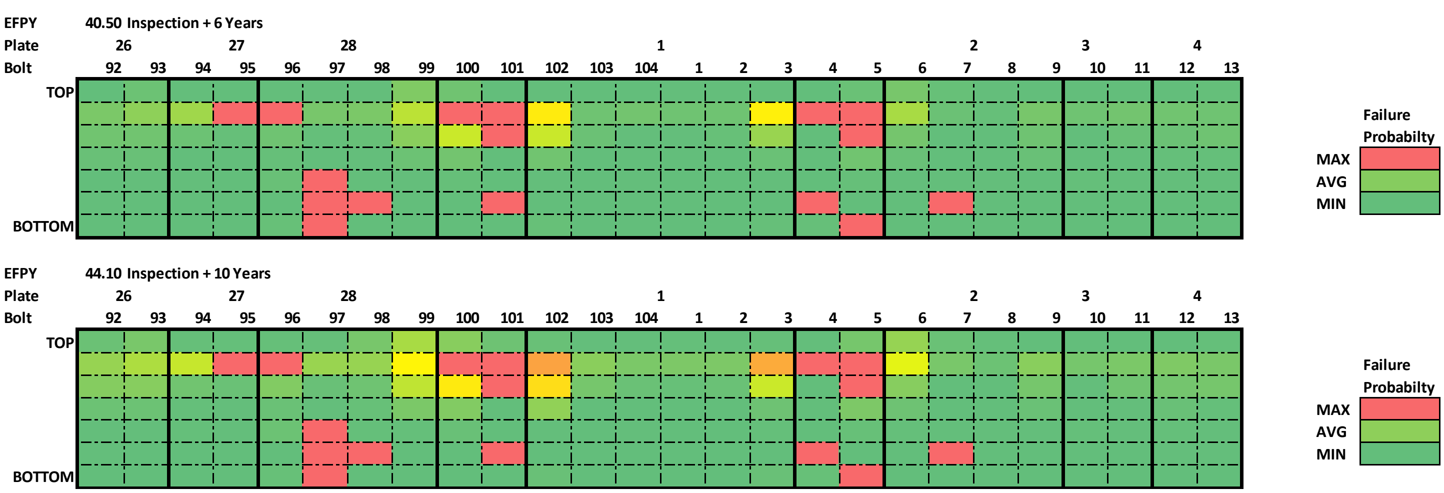


Figure 12: Prairie Island Unit 2 Quadrant 4 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map following Six and Ten Years of Operation After Inspection

Table 3-1 and Table 3-2 summarize the predicted proportion of failed bolts following six and ten years of operation at Prairie Island Units 1 and 2 respectively.

Table 3-1: Prairie Island Unit 1 Predicted Percent of Total Bolts Failed (Inspection +6 and +10 Years of Operation)

a,c

Table 3-2: Prairie Island Unit 2 Predicted Percent of Total Bolts Failed (Inspection +6 and +10 Years of Operation)

a,c

4.0 Assessing the Likelihood of an Acceptable Bolting Pattern

4.1 Acceptability Criteria

The proportion of failed bolts in each quadrant or in the full assembly is an indicator of the long-term operability of Prairie Island Units 1 and 2, but it does not account for the specific patterns of failed bolts. Per the baffle-former bolt acceptance criteria methodology of WCAP-17096-NP-A [9] and the interim guidance of PWROG-17071 [10], both the number and pattern of degraded bolts must be assessed for acceptability.

While not explicitly required by PWROG-17071, the standard evaluation of the bolting pattern output by the predictive model bases pattern acceptability on criteria derived from the methodology outlined in WCAP-15029-P-A [11], including all of the normal and upset criteria. The baffle-former bolt predictive model produced hundreds of likely bolt patterns, which correspond to potential degraded states at the end of the re-inspection interval at Prairie Island Units 1 and 2. These patterns were evaluated in terms of the likelihood of passing an ABPA. In order to evaluate the resultant patterns for acceptability, a series of prescriptive and formulaic checks are performed on each predictive pattern. Bolting pattern attributes considered include the following.

[ ]a,c

These checks are based on an understanding of the results and behavior of the four acceptable bolting patterns that were recently analyzed for the updated ABPA for Prairie Island in WCAP-17586-P [19]. The four acceptable patterns were evaluated explicitly against six ABPA criteria (Bolt Stress, Grid Impact, Low-Cycle Fatigue, High-Cycle Fatigue, Momentum Flux, and Core Bypass) using the methodology detailed in WCAP-15029-P-A [11]. Multiple iterations of preliminary analysis on the ABPA patterns were required in order to obtain acceptable results and, as a result, small variations from the given ABPA patterns have the potential to yield an unacceptable result for at least one of the six ABPA criteria. The preliminary analyses showed that the highest sensitivity was related to failure of bolts along the baffle plate seams, which can lead to exceeding the core bypass limit (and potential baffle jetting). Other areas of concern for reduced bolting patterns have historically been the number of clustered failures (as operating experience has shown that clusters of failed baffle-former bolts tend to grow in size in downflow plants like Prairie Island Units 1 and 2) as well as the overall number of intact bolts on each baffle plate.

With the basis of the formulaic checks established, each predicted future bolting pattern was checked and assigned a weighted value corresponding to its likelihood of passing an ABPA. These values were aggregated to generate respective probabilities for each unit of passing a future ABPA. In order to continue operation to the end of a specified time interval, the probabilistic evaluation must show acceptability for at least 95% of the predicted patterns [10].

4.2 Pattern Acceptability Considering all Faulted and Normal/Upset Criteria

Prairie Island Units 1 and 2 were each analyzed after six and ten years of operation after the UT inspections conducted in 2014 and 2013, respectively. The results are shown in Tables 4-1 and 4-2 below.

Table 4-1: Prairie Island Unit 1 Pattern Acceptability

a,c

Table 4-2: Prairie Island Unit 2 Pattern Acceptability

a,c

Prairie Island Unit 1 meets the requirement of 95% or greater pattern acceptability at ten years of operation following inspection. Using this approach, it could not be shown that Prairie Island Unit 2 meets the requirement at six or ten years of operation following the 2013 inspection. It was determined that core bypass flow was the limiting criterion affecting the ABPA evaluation.

4.3 Pattern Acceptability Excluding Consideration of Core Bypass

4.3.1 Conservatisms in Core Bypass Evaluation Criteria

Core bypass flow presents numerous technical challenges in evaluating predictive model results, leading to the application of several conservatisms. One such conservatism stems from the fact that core bypass

*** This record was final approved on 12/19/2018 2:54:19 PM. (This statement was added by the PRIME system upon its validation)

flow is a phenomenon that occurs as an aggregate for the entire reactor, but the predictive model evaluates individual quadrants of the assembly independently. This results in the conservative assumption that the entire assembly fails the core bypass criterion if even a single quadrant is judged to exceed acceptable levels of degradation. Secondly, in lieu of inspection results that would indicate otherwise, the deterministic ABPA methodology conservatively assumes that all edge bolts are inactive, or failed.

4.3.2 Evaluation

Unit 2 was re-evaluated with consideration of only the faulted load structural stability and grid impact criteria, on the basis that core bypass flow will be addressed separately.

In addition, sensitivity studies were used to relax the ABPA ranking criteria in targeted areas, such as total percentage of intact bolts and minimum percentage of intact bolts per baffle plate. The results of this study for Unit 2, which are limited to consideration of faulted load structural stability and grid impacting, can be seen below.

- Prairie Island Unit 2: Inspection plus six years of operation – Approximate **[ ]^{a,c}** probability of passing an ABPA assessment.
- Prairie Island Unit 2: Inspection plus ten years of operation – Approximate **[ ]^{a,c}** probability of passing an ABPA assessment.

5.0 Summary and Conclusions

5.1 Results Considering all Faulted and Normal/Upset Criteria

Prairie Island Unit 1: Based on the evaluation of the predicted failure patterns, there is an approximate **[ ]^{a,c}** probability of the bolting pattern in the limiting quadrant being acceptable after six years of operation following the 2014 inspection. There is an approximate **[ ]^{a,c}** probability of the bolting pattern in the limiting quadrant being acceptable after ten years of operation. This meets the requirements of [10], and provides a technical basis to support ten years of operation prior to the next UT inspection.

Prairie Island Unit 2: Based on the evaluation of the predicted failure patterns, there is an approximate **[ ]^{a,c}** probability of the bolting pattern in the limiting quadrant meeting all evaluation criteria after six years of operation following the 2013 inspection. There is an approximate **[ ]^{a,c}** probability of the bolting pattern in the limiting quadrant meeting all evaluation criteria after ten years of operation.

5.2 Results Excluding Consideration of Core Bypass

Prairie Island Unit 2: Based on the evaluation of the predicted failure patterns using the modified methodology detailed in Section 4.3, there is an approximate **[ ]^{a,c}** probability of the bolting pattern in the limiting quadrant being acceptable after six years of operation following the 2013 inspection. There is an approximate **[ ]^{a,c}** probability of the bolting pattern in the limiting quadrant being acceptable after ten years of operation.

As discussed in Section 4.3, this assessment excludes consideration of core bypass flow, which is to be addressed separately when justifying an extended UT inspection interval.

6.0 References

1. Westinghouse Calculation Note, CN-AMLR-18-9, Rev. 0, "Baffle Former Bolt Predictive Reliability Analysis for Prairie Island Units 1 and 2," October 19, 2018. (Proprietary)
2. WesDyne Inspection Report, WDI-PJF-1312663-FSR-001, Rev. 0, "Baffle Bolt Ultrasonic Examination For Prairie Island 1R29," December 2014. (Proprietary)
3. WesDyne Inspection Report, WDI-PJF-1309762-FSR-001, Rev. 0, "Prairie Island Unit 2 RFO28 Baffle-Former Bolt Ultrasonic Inspection Field Service Report," December 2013. (Proprietary)
4. Westinghouse Nuclear Safety Advisory Letter, NSAL-16-1, Revision 1, "Baffle-Former Bolts," August 1, 2016.
5. Materials Reliability Program Letter, MRP 2016-021, "Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for Tier 1 Plants as Defined in Westinghouse NSAL 16-01," July 25, 2016.
6. Materials Reliability Program Letter, MRP-2017-009, Rev. 0, "Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for PWR Plants as Defined in Westinghouse NSAL 16-01 Rev. 1," March 15, 2017.
7. U.S. Nuclear Regulatory Commission Document, "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." *[ADAMS Accession Number: ML17310A861]*
8. *Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)*. EPRI, Palo Alto, CA: 2011. 1022863.
9. Westinghouse Report, WCAP-17096-NP-A, Rev. 2, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," August, 2016.
10. PWROG Document, PWROG-17071, Rev. 0, "WCAP-17096-NP-A Interim Guidance," March 2018.
11. Westinghouse Report, WCAP-15029-P-A, Rev. 1, "Reactor Internals Acceptance Criteria Methodology and Data Requirements," December 1998. (Proprietary)
12. Westinghouse Report, WCAP-13266, Revision 1, "Baffle-Former Bolt Program for the Westinghouse Owners Groups Phase 1: Plant Categorization," July 1993. (Proprietary)
13. Westinghouse Drawing, 499B436, Rev. 3, ".625 Socket Head Cap Screw (Undercut)." (Proprietary)
14. *Hot Cell Testing of Baffle/Former Bolts Removed from Two Lead Plants: (PWRMRP-28)*, EPRI, Palo Alto, CA: 2000. 1000971.
15. *Materials Reliability Program: Hot Cell Testing of Baffle/Former Bolts Removed from Two Lead PWR Plants (MRP-51),* EPRI, Palo Alto, CA: 2001. 1003069.
16. McKinley, J., et al., "Examination of Baffle-Former Bolts from D.C. Cook Unit 2," 16th International Conference on Environmental Degradation of Materials in Nuclear Power Systems, Asheville, North Carolina, USA, August 11-15, 2013.
17. Somville, F., et al., "Ageing Management of Baffle Former Bolts in Belgian Nuclear Power Plants," Fontevraud 8, Avignon, France, September 15-18, 2014.
18. Westinghouse Report, WCAP-15029-P-A, Revision 1, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Barrel Bolting Distributions Under Faulted Load Conditions," January 1999. (Proprietary)
19. Westinghouse Report, WCAP-17586-P, Rev. 2-A, "Determination of Acceptable Baffle-Barrel Bolting for Prairie Island Units 1 and 2," August 2018. (Proprietary)

20. Westinghouse Letter, LTR-RIAM-17-9, Rev. 0, "Summary Report for the Baffle-Former Bolt Predictive Evaluations Prepared for EPRI," February 1, 2017.
21. Westinghouse Report, WAAP-10181 PAPER, Rev. 2, "Predictive Modeling of Baffle Former Bolt Failures in Pressurized Water Reactors," July 2017.

ENCLOSURE 2

Redacted NSPM PINGP Evaluation No. 608000000371,
“Baffle Bolt Core Bypass Flow Predictive Modeling”
(Non-Proprietary)

36 pages follow

1.0 Purpose and Summary Result

As required by the Prairie Island Reactor Vessel Internals Program, H74, and the NSPM Materials Degradation Management Program, CD 5.36, a plant specific evaluation was performed by Westinghouse Electric Company to determine the acceptable re-inspection interval for the Unit 1 and 2 Baffle-Former Bolts. The type of analysis used was a probabilistic predictive model as described in PWROG-17071, Interim guidance to WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements". PWROG-17071 specifies that results of a Monte Carlo simulation should show a 95% reliability of the baffle-former assembly for maintaining structural stability and fuel grid integrity, which protects core coolability and rod insertability during faulted conditions. Both PINGP Unit 1 and 2 demonstrated the required reliability to justify a 10-year re-inspection interval. However, Unit 2 could not be shown to have 95% reliability for meeting the core bypass flow design limit at 10 years using the methodology employed by Westinghouse.

Deterministic baffle-bolt pattern analysis follows the methodology of WCAP-15029-P-A, which in addition to faulted conditions, specifies evaluations for normal/upset cases. These include an evaluation of baffle gap openings and potential to exceed design core bypass flow (CBF) limits. Westinghouse extended these deterministic evaluations to the analysis of the predictive Monte Carlo results, but technical challenges caused several unrealistic conservatisms to be employed. Consequently Westinghouse could not show as high a reliability (i.e. 95%) for all 4 quadrants of the internals for meeting the core bypass flow limits, as was demonstrated for the structural stability and fuel grid impact criteria. Since PWROG-17071 does not specify acceptance limits for other than the structural criteria, core bypass flow will instead be addressed qualitatively using the quantitative insights generated by the predictive analysis. The purpose of this evaluation is to review the results of the Westinghouse predictive model with respect to core bypass flow and determine their acceptability in regard to a 10-year re-inspection interval.

A second purpose is to support the Prompt Operability Determination and Operable but Nonconforming condition documented under Prairie Island CAP 501000000150. The operable but nonconforming (OBN) condition exists because of the use of methodology for the Acceptable Bolt Pattern Analysis (ABPA) that differs from the NRC-approved methodology. The OBN relies on the industry work that justified 6 years as a suitable re-inspection interval, and therefore indicates a 6-year duration for the validity of the Prompt Operability Determination (POD). Because it is expected that the nonconforming condition may continue to exist slightly beyond the 6 year mark from Unit 2's last inspection, this evaluation, by establishing a new validity period of 10-years, will serve as partial justification for continued operability when the POD is revised or reissued. The current 6-year allowable re-inspection interval, which was established using similar probabilistic predictive models as what were recently employed on a plant-specific basis for PINGP, was conservatively taken as a time limit for the POD. The plant-specific predictive modeling, by using less conservative loadings and a ABPA that demonstrates fewer required bolts, will serve to extend the period of validity of the POD. This EC does not establish any other elements of the Operability Determination beyond the justification for the period of validity.

Summary Result:

Prairie Island Units 1 and 2 are both predicted to meet the structural stability and fuel grid impact limits of an ABPA with at least 95% probability at the 10-year point from the baseline inspection. Therefore, the requirements of PWROG-17071 and WCAP-17096 Rev. 3 (draft) are met. However, core bypass flow design limits must be additionally addressed. [

][a,c] A sensitivity study demonstrated that many of the predictive model results that were counted as failures based on algorithmic scoring criteria for CBF would have instead been categorized as acceptable under a rigorous baffle gaps and thermal hydraulic calculation. Those predictive model results that could not be shown to pass a rigorous CBF calculation occur very infrequently in the Monte Carlo simulation, and this frequency was judged to be acceptably small. Secondly, the presence of edge bolts has been shown to mitigate baffle gap leakage in conjunction with worst-case bounding predictive model outputs. Finally, previous reports published by Westinghouse have concluded that fuel fretting failures would be expected to occur at baffle gap flows much lower than those that would exceed core bypass flow design limits, and thus be self-revealing. Therefore, it is concluded that there is a reasonable expectation that core bypass flow design limits will continue at all times up until the time of next inspection, 10 years from the baseline examination.

2.0 References

2.1 WCAP-17586 Rev. 2, Determination of Acceptable Baffle-Barrel Bolting for Prairie Island Units 1 and 2

2.2 LTR-AMLR-18-55, Rev. 1, Prairie Island Units 1 and 2 Baffle Former Bolt Predictive Evaluation

2.3 WCAP-15029-P-A, "Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Bolting Distributions Under Faulted Load Conditions"

2.4 PWROG-17071, Rev.0, "WCAP-17096-NP-A Interim Guidance"

2.5 WCAP-17096 NP-A, Rev. 2, “Reactor Internals Acceptance Criteria Methodology and Data Requirements”

2.6 WCAP-17096, Rev. 3 (draft), “Reactor Internals Acceptance Criteria Methodology and Data Requirements”

2.7 EPRI MRP-2017-009, “Transmittal of NEI-03-08 “Needed” Interim Guidance Regarding Baffle Former Bolt Inspections for PWR Plants as Defined in Westinghouse NSAL 16-01 Rev.1”

2.8 WCAP-15425, “Determination of Acceptable Baffle-Barrel Bolting for Kewaunee and Prairie Island Unit 1 and 2”

2.9 WCAP-15036, Rev.1, “Determination of Acceptable Baffle-Barrel-Bolting for Two-Loop Westinghouse Domestic Plants”

2.10 LTR-AMLR-19-18-P, Rev. 0, “Prairie Island Unit 2 – Baffle Edge Bolt Sensitivity Study”

2.11 Email dated 3/13/19 from Nathan Lang, Westinghouse, Subject: Prairie Island PIN-003 Draft Deliverables

2.12 CAP 501000000150, “PI ABPA not per NRC approved methodology”

2.13 EPRI MRP-227-A, “Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines”

2.14 NEI 03-08, Rev. 3, “Guideline for the Management of Materials Issues”

2.15 NSAL 16-1, Rev. 1, “Baffle Former Bolts”

3.0 Methodology

Baffle bolt degradation can result in several effects. The chief effects that must be protected against are a reduction in the number and arrangement of intact bolts to a point where in the event of a LOCA (Loss of Coolant Accident) or SSE (Safe Shutdown Earthquake), the remaining bolts break under load and the plates are free to impact the fuel, or even if the bolts remain intact, the spacing is insufficient to prevent the plates from flexing enough to impact the fuel. If grid impacting does occur, then the grid strength must be sufficient to prevent deformation or “grid crush”, which could result in loss of coolable geometry or control rod insertability.

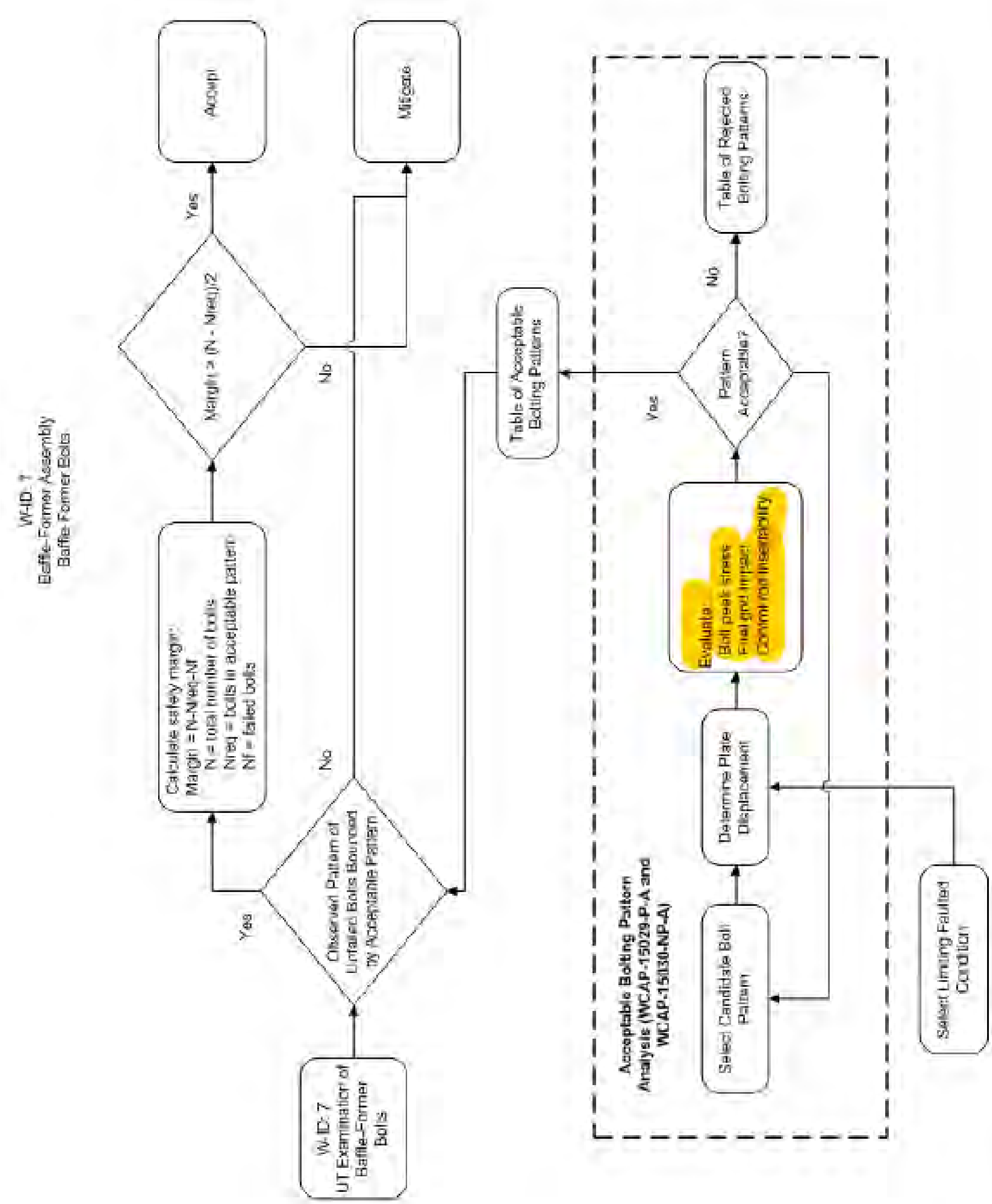
WCAP-17096-NP-A, Rev.2, Reactor Internals Acceptance Criteria Methodology and Data Requirements, which has been reviewed and approved by the NRC, specifies loss of structural stability as the failure criteria to be prevented in evaluations of BFB degradation.

Refer to the following excerpts:

W-ID: 7	**Baffle-former Assembly**
	Baffle-former Bolts
Category:	Primary Applicability: All plants
Degradation Effect:	Cracking (IASCC, fatigue), Aging Management (IE)
Expansion Link:	Lower support column bolts, barrel-former bolts
Function:	The baffle-former bolts attach the baffle plates to the formers.
Inspection	
Method:	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a 10-year interval.
Coverage:	100% of accessible bolts. Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See MRP-227-A Figures 4-23 and 4-24, and Table 4-3, Note 3.
Observable Effect:	UT will detect bolts with large cracks (approx. 30%) through of cross-sectional area. Fractured bolts should be captured by locking devices – no visible indication.
Failure	
Failure Mechanism:	Known IASCC cracking of similar highly irradiated bolts has been reported.
Failure Effect:	Loss of structural stability
Failure Criteria:	Require an acceptable bolting pattern

PINGP EVALUATION

NUMBER:	**608000000371**	**REVISION:**	0	**MINOR REVISION:**	0	**SHEET NO:**	7 OF 36
TITLE:	**BAFFLE BOLT CORE BYPASS FLOW PREDICTIVE MODELING**					**DATE: 5/2/19**	
						COMP. BY:	CRK



WCAP-17096-NP-A

August 2016
Revision 2

This is carried forward into the Interim Guidance report, PWROG-17071, which requires that predictive models demonstrate a 95% probability of retaining a pattern at the time of the next inspection that remains structurally stable and prevents unacceptable grid impact. The probabilistic methods are incorporated into the draft of Revision 3 of WCAP-17096, which will also be submitted for NRC approval. All of these documents are silent on Core Bypass Flow as an acceptance criterion, which is a normal/upset rather than faulted criterion. It is instructive to note that another normal/upset criterion from WCAP-15029, "momentum flux", which is used to assess the likelihood of baffle jetting related fuel failures, [

]a,c

In the evaluation that follows, we discuss the justification for using a predictive model acceptance limit below the 95%th percentile for core bypass flow results in the assessment of a 10-year re-inspection interval. As noted, this is not precluded by the industry guidance. This will entail reviewing the [ ]a,c results of the Westinghouse predictive model and employing semi-quantitative and qualitative arguments to show that there is reasonable expectation that design limits for core bypass flow will continue to be met at all times up to the time of next inspection. This evaluation meets all industry guidance and regulatory requirements.

As background, it is important to recognize the chronology and relationship between the issuance of the industry guidance.

WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements", was prepared to establish the minimum standards to justify a 10 year re-inspection interval under MRP-227-A. It was based on having at least 50% of the margin remaining between the 100% design condition and the minimum acceptable pattern from an ABPA at the time of inspection. It did not consider concentration or spacing of indications, thus implying a somewhat random distribution of failures.

Following observation of extensive large clusters of failures at Indian Point 2, Salem 1, and D.C. Cook 2, the industry issued interim guidance under MRP 2017-009, "Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for PWR Plants as Defined in Westinghouse NSAL-16-1 Rev.1." The guidance limited downflow plants with >=3% of bolts with indications, or clusters of failures, to no more than 6 years between inspections, unless a "plant specific evaluation" justified longer.

PWROG-17071, "WCAP-17096-NP-A Interim Guidance" was issued to describe the requirements for the plant specific evaluation. It contains the methodology and acceptance levels for a probabilistic predictive evaluation.

WCAP-17096, Revision 3, was recently issued in draft, incorporating the PWROG-17071 guidance, and will be submitted to NRC for approval.

The following are excerpts of the controlling guidance documents that describe the plant-specific evaluation that can be used to establish baffle bolt re-inspection intervals.

From EPRI MRP 2017-009, "Transmittal of NEI-03-08 "Needed" Interim Guidance Regarding Baffle Former Bolt Inspections for PWR Plants as Defined in Westinghouse NSAL-16-1 Rev.1":

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 reinspection 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 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]*
Up-Flow WEC Plants	*<5% indications with no clustering[a]*	*not to exceed 10-years*
Up-Flow WEC Plants	*≥5% indications or clustering[a]*	*not to exceed 6-years[b]*

Note: (a) 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 reinspection 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 up-flow, replacement of lower internals, etc.).

From PWROG-17071, Rev.0, "WCAP-17096-NP-A Interim Guidance":

Modified Requirement

Methodology

Goal: Must demonstrate that bolting patterns provide sufficient margin to acceptance criteria for continued operation through the re-inspection interval.

Data Requirements:
Loads
UT inspection results
Test data for empirical model (as necessary)
Baffle-former assembly design information including any applicable modifications
Fast neutron (dpa) distribution in the baffle-barrel region (if needed)

Analysis:

1. Structural integrity of the baffle-former assembly must be demonstrated by an evaluation of the bolting pattern. One approach to accomplish this is to perform an acceptable bolting pattern analysis (ABPA) per the methodology described in WCAP-15029-P-A [6], which has been reviewed and accepted by the NRC via Safety Evaluation.

The bolting pattern considers functional bolts to include those with satisfactory UT inspection results or replacement bolts, as well as any portion of untestable bolts determined to be acceptable per step 2. Otherwise, the un-inspectable bolts and those that show visual or UT indication of failure are treated as non-functional. This evaluation must consider any plant-specific variations of operating conditions from analysis assumptions, and may credit plant modifications such as upflow conversion to reduce bolt load.

2. The degradation is categorized as either "Typical" or "Accelerated" based upon plant design and the inspection results. The percentage of bolts with indications is based upon the fraction of the total quantity. This categorization considers the likelihood of additional bolt failures due to IASCC over a typical 10-year re-inspection interval. The degradation category for the observed and assumed failed bolts shall be determined using the following table:

Plant Design	Inspection Results(2)	Degradation Category
Down-Flow	Less than 3% indications with no clustering(1)	Typical
	3% or greater indications or clustering(1)	Accelerated
Upflow and Converted Upflow	Less than 5% indications with no clustering(1)	Typical
	5% or greater indications or clustering(1)	Accelerated

Notes:
1. Clustering is defined as three or more adjacent defective BFBs in the vertical or lateral directions on the same baffle plate, or more than 40% defective BFBs on the same baffle plate [4]. Untestable bolts should be reviewed on a plant-specific basis for determination if these should be credited as structural members when classifying clustering.
2. These criteria are considered applicable to subsequent inspections until an appropriate degradation rate threshold is established through operating experience.

3. Additional margin to account for potential degradation through the next re-inspection interval must be provided. Plants that exhibit "Accelerated" degradation must apply the Probabilistic Model, but those with "Typical" degradation may apply either the Probabilistic Model or Margin Ratio method as defined in the Acceptance Criteria.

Note: Any plant-specific evaluation used to extend the re-inspection interval beyond those defined in MRP-2017-009 [4] is to be submitted to the NRC for information at least one year prior to the end of the current applicable interval for BFB subsequent examination [12].

Acceptance Criteria: 1. The analysis requirements as outlined above are applied to ensure structural integrity of the bolting pattern.

2. The degradation category is established consistent with industry guidance.

3. The evaluation must include margin by applying one of the following methods:

<u>Probabilistic Model (Typical or Accelerated Degradation)</u>

A. An empirical model of the degradation effect is developed from plant-specific design information and test data. The model is calibrated to plant-specific bolt inspection results, or otherwise applies bolt inspection results from similar plants after accounting for relevant differences as appropriate.

B. Determine patterns of predicted failed bolts at the end of the re-inspection interval based upon the likelihood of failure. Justification shall be provided that an appropriate degradation rate has been applied. The evaluation is allowed to credit bolt replacements, or plant improvements such as upflow conversion.

C. A probabilistic reliability analysis is used to demonstrate reasonable assurance that the pattern of failed bolts will be acceptable at the end of the re-inspection interval. These techniques must consider both the total proportion of failed bolts and the spatial distribution, or pattern, of bolt failures. When predicting the total proportion of failed bolts, it is acceptable to consider results associated with the 95th fractile. For the potential spatial distributions of bolt failures, an evaluation showing acceptability of at least 95% of the predicted patterns at the end of the re-inspection interval can be used. It is anticipated that this method will be used to demonstrate a 10-year re-inspection interval. Re-inspection intervals beyond 10-years may be justifiable in certain circumstances (e.g., acceptable bolting pattern relies on only replacement bolts and the plant is in, or has converted to, an upflow configuration).

<u>Margin Ratio (Typical Degradation Only)</u>

Less than 50% of the initial margin may be consumed. Therefore, the quantity of bolt failures upon re-inspection must be less than or equal to the number of bolt failures that has occurred to date.

$$N_f < (N - N_{req}) / 2$$

where

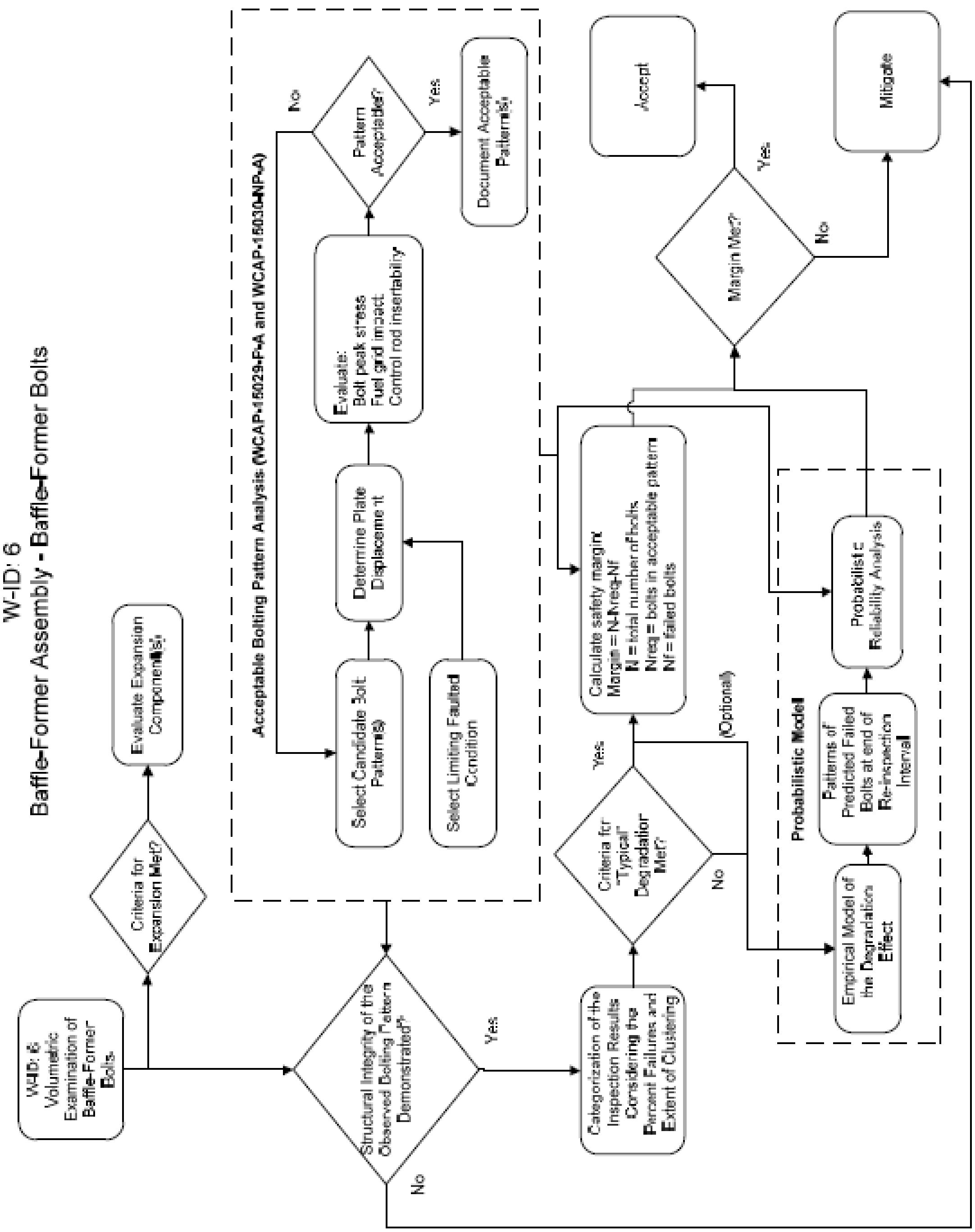
N = total number of baffle-former bolts
Nreq = number of baffle-former bolts in acceptable pattern
Nf = number of failed bolts

The margin (M) is:

$$M = N - Nreq - Nf$$

This degradation model may be applied at subsequent re-inspections, up to a 10-year re-inspection interval, when the "Typical" degradation category remains applicable.

From WCAP-17096, Rev. 3, “Reactor Internals Acceptance Criteria Methodology and Data Requirements”:



From WCAP-15029-P-A, “Westinghouse Methodology for Evaluating the Acceptability of Baffle-Former-Bolting Distributions Under Faulted Load Conditions”:

3. NORMAL/UPSET LOAD ANALYSIS

The faulted load analysis approaches described in Section 2.0 are the primary means for assessing the acceptability of revised baffle-barrel-bolting distributions. The methods used to evaluate N/U conditions are the same regardless of the numbers and distributions of baffle-barrel-bolting; therefore, the details of the N/U analysis methods used to assess the acceptability of reduced bolting distributions are not included here. However, it is possible that reduced bolting could lead to undesirable N/U performance. The N/U analyses are performed for this reason, and are described below.

3.1 THERMAL GROWTH AND STEADY-STATE PRESSURE LOADS

Thermal growth of the baffle plates relative to the core barrel and steady-state pressure loads have two potentially adverse effects: 1) Opening of the baffle/baffle gaps, and 2) thermal cycling and fatigue of the baffle-barrel-bolting. The purpose of performing this analysis is to determine the extent to which these effects are influenced by reduced bolting, particularly reduced edge bolting. These phenomena may lead to: 1) increases in the number of cracked bolts, and/or 2) opening the baffle-baffle gaps enough to increase bypass flow beyond design limits.

The thermal growth analysis is performed in two stages: 1) analysis of baffle-barrel temperature differences, and 2) analysis of former temperature distributions. The results of these analyses are then combined into a structural model, from which bolt stresses and baffle gaps are determined. The final steps in the analysis sequence are to take the structural analysis results and use them to perform baffle-jetting and bypass flow calculations as well as a bolt low cycle fatigue evaluations.

4.0 Acceptance Criteria

It must be demonstrated, within reasonable expectations, that the design limits for core bypass flow will be met at all times until the next inspection.

5.0 Assumptions

5.1 Unvalidated Assumptions: None

5.2 Validated Assumptions:

5.2.1 It is assumed that because PWROG -17071 and WCAP-17096 do not explicitly apply a 95% reliability acceptance limit to the Core Bypass Flow criteria, that CBF may be addressed qualitatively, and that the 95% reliability limit need not be quantitatively demonstrated.

5.2.2 It is assumed that an assessment of core bypass flow, which has design basis criteria for normal and upset conditions, can be treated in the same manner as momentum flux for the same conditions. Neither have specific acceptance criteria requirements in PWROG -17071 and WCAP-17096 for establishing an inspection interval. However, both can be shown to meet acceptance criterion, with a reasonable expectation, through a qualitative evaluation. [

]a,c A similar evaluation was performed in this EC for core bypass flow.

5.2.3 It is assumed that the supposition of 100% edge bolt failures is extremely conservative and that consideration of some amount of intact edge bolts is justifiable, given appropriate engineering judgement

5.2.4 These assumptions are validated by imposing additional 3rd party reviews to this evaluation (i.e. Expert Elicitation).

6.0 Inputs

6.1 It is taken as an input that WCAP-17586 Rev. 2, “Determination of Acceptable Baffle-Barrel Bolting for Prairie Island Units 1 and 2”, found the following four patterns of degradation acceptable for structural and core bypass flow criteria. Note that these are hypothetical patterns that were selected to be bounding of any potential future state and evaluated to show the minimum acceptable number and configuration of bolts that would pass all criteria, including normal, upset, and faulted. Note the extent of degradation that was shown to be tolerable within design, and recognize that the ABPA assumes that each octant all around the assembly is represented by this pattern of failures.

a,c

a,c

a,c

a,c

The ABPA analyzed the above patterns for Core Bypass Flow relative to "Case 1" which was the all-bolts-intact case (minus edge bolts). The below table represents the results. [

]a,c

[ ]a,c

6.2 The following results from LTR-AMLR-18-55, Rev. 0, “Prairie Island Units 1 and 2 Baffle Former Bolt Predictive Evaluation”, are taken as inputs. These are a visual representation of a rollup of all of the predictive simulations, showing on average, the probability of a given bolt having failed at the 10-year future state. These are relied upon in the Section 7.0 evaluation to show the significant differences between the quadrants. Note that while the letter report contains heat maps for both the 6- and 10-year cases, only the 10-year heat maps are presented here.

a,c

Figure 9: Prairie Island Unit 2 Quadrant 1 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map Following Six and Ten Years of Operation After Inspection

a,c

Figure 10: Prairie Island Unit 2 Quadrant 2 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map Following Six and Ten Years of Operation After Inspection

a,c

Figure 11: Prairie Island Unit 2 Quadrant 3 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map Following Six and Ten Years of Operation After Inspection

a,c

Figure 12: Prairie Island Unit 2 Quadrant 4 Predicted Fraction of Failed Bolts and Bolt Failure Probability Heat Map following Six and Ten Years of Operation After Inspection

The below table of results represents the overall fraction of bolts determined to be failed in the median (50th percentile) and 95th percentile of the simulations in each of the four quadrants.

Table 3-2: Prairie Island Unit 2 Predicted Percent of Total Bolts Failed (Inspection +6 and +10 Years of Operation)

a,c

The below table represents the results of the evaluation of each future state in the simulation on a quadrant-by-quadrant basis. It was subsequently concluded that the criterion that was most often resulting in the future state being categorized as unacceptable were the core bypass flow rules. In the Section 7.0 Evaluation, the quadrant differences are used to establish a probability of acceptability of the entire core.

Table 4-2: Prairie Island Unit 2 Pattern Acceptability a,c

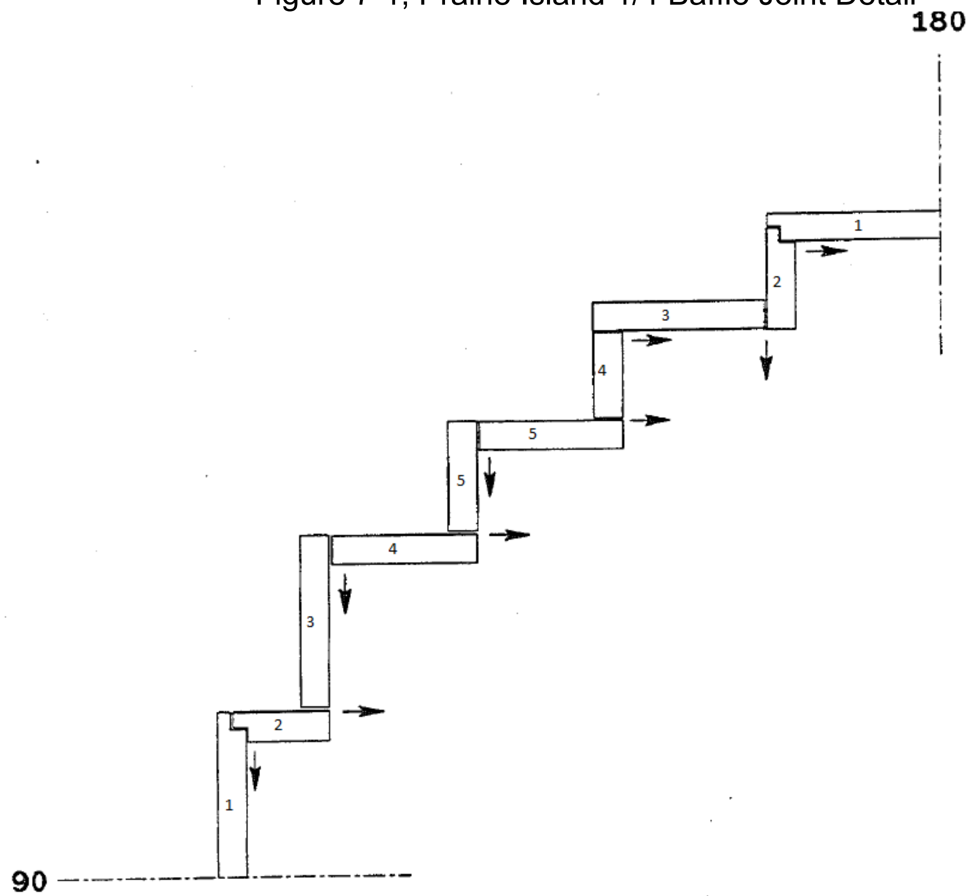
7.0 Evaluation

This evaluation will consist of the following elements:

1. Discussion of the disparity between predictive model scoring for CBF versus ABPA results and the results of detailed baffle gaps and thermal hydraulic calculations (sensitivity study).
2. Discussion of the treatment of core bypass flow in the relevant guidance documents;
3. Description of how core bypass flow is assessed deterministically in an ABPA;
4. Assessment of the following conservatisms in the Westinghouse predictive model treatment of CBF, including:
   - [
   -]a,c
5. Evaluation of the relationship between baffle jetting related cladding failures and CBF,
6. Discussion of how edge bolts are needed for BFB clustered failures to propagate,
7. Conservatism of [ ]a,c and;
8. Discussion of modeling anomalies and impact on results.

The below figure shows the arrangement of one quadrant of baffle plates with the plate numbering that will be referred to in the evaluation.

Figure 7-1, Prairie Island 1/4 Baffle Joint Detail



Predictive Model Results vs. Detailed Baffle Gaps and CBF Calculations

A brief review of the predictive model results will frame the need for a detailed investigation of the conservatisms in the Westinghouse methodology. The predictive model determined that at the 95th percentile confidence level, [ ]^{a,c} For the sake of comparison, the patterns analyzed in the ABPA (i.e,. Cases 2 thru 5 of Input 6.1) had [ ]^{a,c} This is almost double the degradation predicted by the model output at 95 percentile. Notably, the ABPA patterns postulated in cases 2 thru 5 passed all structural, fatigue, and core bypass flow criteria. Because the ABPA patterns also contained significant clustering, prevalence of clustering in the predictive results would not be expected to account for this much of a disparity in what is considered an acceptable level of bolt failures. It is non-intuitive that the predicted output patterns are judged to fail the core bypass criteria, with around half or less of the degradation that was shown to be acceptable when analyzed explicitly under ABPA rules.

A visual comparison of the predictive model heat maps to the ABPA patterns analyzed in WCAP-17586, Rev.2 does not at once reveal the reason for the low reliability reported by the predictive model report. With the driving force for plate deflection coming from the differential pressure across the baffle plates, with the higher pressure on the exterior, the joint configuration and intact bolting configurations, have a strong influence on core bypass flow. For instance exterior joints, such as the 1-2, the 3-4, and the 5-5, tend to be closed by differential pressure. This leaves the interior joints (the 2-3 and the 4-5), as the important joints. Also, since differential pressure is highest in the upper half of the assembly, a given gap opening will result in higher bypass flow if it is located in the upper part of the assembly. Gap openings towards the bottom of the assembly would not tend to result in much bypass flow, because there is no pressure driving force. Also, leakage through the baffles at the bottom bypasses less of the core.

Given the relationship of the plates to one another, it is understood that failures on plate 1 and 3 are relatively unimportant to core bypass flow. This is because, as noted above, plate 1 only contributes to an exterior joint. And for plate 3, it can be seen that the way it abuts plate 2 means that it would have to hinge or deflect by an amount greater than its own thickness before the displacement would actually begin to open the 2-3 joint gap. Such deflections are not possible due to the proximity to the adjacent fuel bundle. So the structural criteria of grid crush would be exceeded before failures on plate 3 would substantially increase core bypass flow. Failures on plate 2 in the top half are much more significant to the leakage from the 2-3 joint, since displacements at the edge of plate 2 start to increase the gaps for flow immediately. However, due to the restraint afforded by the overlap with plate 1, which stiffen plate 2 from bowing, the ABPA results demonstrate that [ ]^{a,c}

Bolts on the upper half of plates 4 and 5 are the most important, depending on whether the plate 4 bolts are predominately holding the 4-5 joint closed, or the plate 5 bolts are predominately holding the joint closed. This is because unlike the relationship between plates 1 and 2, there is nothing stiffening the affected plate from bowing. This arrangement alternates around the assembly, with only the plate 4 or 5 bolts (but not both) being the most important across any given joint (note: it is unclear whether the Westinghouse scoring algorithms reflect this fact). The explanation for this is the same as for why the bolts on plate 3 are unimportant for prevention of core bypass flow. One of the plates (4 or 5) along the 4-5 joint would have to deflect by an amount greater than its own thickness before core bypass flow would be increased, and this is not possible without first having crushed peripheral fuel (i.e. already failing the structural criteria). With respect to the heat map discussion below, on the left side of the heat map, it is the bolts on plate 5 that are critical to prevention of 4-5 joint opening, and on the right side, it is the plate 4 bolts that are critical to keeping the joint closed.

Although there are limitations in assessing the reliability based on the heat maps alone, Quadrants 1 and 4 show distributions of failures that would be very likely to pass an ABPA (i.e. >95%), by engineering judgement. This judgement is based on the numerical probabilities on the heat maps which show, for example, [ ]a,c A similar review of Quadrant 4 was performed, and the probabilities were similar except for the left-hand Plate 5, which showed [ ]a,c but since one portion of one joint alone would not represent a core bypass flow challenge, this probability was averaged with the other three important joints in making this assessment.

Quadrants 2 and 3 are more heavily degraded than Q1 and Q4 and have more clustered distributions, but the distributions on the critical plates for CBF are not consistent with the reported reliability percentiles. Quadrant 2 was scored in the Westinghouse results to be likely to fail an ABPA in [ ]a,c of trials. However, a review of the heat maps would indicate [ ]a,c by engineering judgement, even with the symmetry and edge bolt assumptions (discussed later). Similar to what was done for Quadrants 1 and 4, the bounding probabilities for clusters to form on plates 2, 4, and 5 were reviewed, and were largely bounded by the previous (Q1 and Q4) values. The only outlier was one Plate 2 cluster that [ ]a,c However, the cluster had only a [ ]a,c chance of involving the top former, and ABPA patterns “Customer A” and “Customer B” have very nearly encompassed this condition with passing results (despite the symmetry assumption applied in an ABPA). Therefore the [ ]a,c probability was adjusted and then averaged with the other joint probabilities in estimating the overall Q2 probability.

Quadrant 3 has the most highly degraded 10-year heat map and is [ ]a,c Aside from the predicted behavior of the bolts around one 4-5 joint (discussed in detail later), the heat map of this quadrant would indicate no more than around a [ ]a,c likelihood of failing an ABPA, by engineering judgement. Again, this involved looking at plates 2, 4, and 5, which showed [ ]a,c For the same reasons discussed above (the realistic need for more than one joint to leak to exceed a whole quadrant limit), the [ ]a,c probability of the left-hand 4-5 joint opening up can be averaged with the probabilities of the other joints opening up [ ]a,c in judging the whole of Quadrant 3, and this results in the estimate of [ ]a,c failure probability. Specifics of the predictions on the Q3 left-hand plate 5 are reviewed later in this paper in detail.

[ ]a,c Because of the types of concerns identified in the paragraphs above regarding the disconnect between the scoring of the predictive results and what engineering judgement would estimate based on ABPA information, it was elected to perform a core bypass sensitivity study. It was shown through follow-on study (Ref. LTR-AMLR-19-18-P, Rev. 0) that [ ]a,c Specifically, when Westinghouse selected a composite pattern (see below, extracted from the reference) intended to be bounding of the worst predictive patterns [ ]a,c and this pattern was explicitly analyzed for baffle gaps and CBF using THRIVE, this worst case pattern was found to have acceptable CBF within design limits.

Figure 7-2, "Worst Case" Pattern Shown to be Acceptable for Core Bypass Flow

a,c

From the results of the sensitivity study in LTR-AMLR-19-18-P, Rev.0, it is evident that excess conservatism in the scoring algorithms would have resulted in many of the predictive model output patterns to be classified as failures when they would have passed based on more detailed analysis. The Westinghouse letter report concluded that "were an explicit core bypass flow evaluation performed for all of the predictive model pattern results, the percentage of acceptable patterns for Prairie Island Unit 2 at the end of a 10-year re-inspection interval would be greater than the [ ] a,c reported in [the predictive analysis]." Based on the outcome of the sensitivity study for the "worst case", the engineering judgement applied above to estimate the passing percentiles from the heat maps is given substantial validation.

At the other end of the spectrum, a bounding "worst-case" pattern was investigated in the sensitivity study [ ] a,c shown below, and this pattern could not be shown to pass the core bypass flow criteria without edge bolt credit. However, [ ] a,c per Reference 2.10, and this frequency has been judged to be acceptably low (specifics of Quadrant 3, the worst quadrant are described later). The sensitivity study did demonstrate that with all edge bolts intact, even this bounding worst case did have acceptable core bypass flow results.

Figure 7-3, "Worst-case" Pattern Shown to be Unacceptable for Core Bypass Flow (note top former failures on plates 2 and 5)

a,c

Core Bypass Flow in Regulatory and Industry Guidance for Predictive Modeling

Several documents impose regulatory and industry guidance specific to baffle bolt re-inspection interval determinations. As discussed in Section 3.0, neither of the acceptance criteria documents WCAP-17096 or PWROG-17071 specifies the limits for normal/upset criteria in setting baffle bolt re-inspection intervals. Nevertheless, even though WCAP-15029 primarily addresses the methodology for performing a faulted analysis, it does also briefly describe several normal/upset criteria which are used to evaluate patterns in an Acceptable Bolt Pattern Analysis (ABPA), without going into detail about the methodology employed. These include Core Bypass Flow, Momentum Flux (Baffle Jetting), and Low-cycle and high-cycle fatigue life. As previously noted, [

]a,c

but this is not treated as a disqualifier for a pattern in an ABPA. Core bypass flow is evaluated to ensure during normal operation, leakage through the baffle gaps does not exceed the design value that is input to the transient analysis, since bypass flow does not contribute to heat removal from the fuel and prevention of DNB in a transient.

Core Bypass Flow Assessment in ABPA

[

]a,c

Core Bypass Flow Assessment in Predictive Modeling

[

]a,c

Assumption of One-Eighth Symmetry

[

]a,c

Uncredited Edge Bolts

Another issue with applying core bypass flow criteria to predictive model results is [

]a,c This is a required simplification in the deterministic ABPA analysis, because there is no other basis (such as inspection data) on which to determine the number of edge bolts that are intact. However, far more likely is that failure of significant numbers of edge bolts would be linked to failure of nearby baffle bolts. In the Unit 2 predictive results, for instance, three of the quadrants are relatively less degraded (i.e. showing small probabilities of large clusters of failures), and so there should be no driving force for large amounts of edge bolts to fail in those less degraded quadrants, as there should not be substantial transfer of loads to them. Thus, in the absence of clustered baffle-bolt failures resulting in load redistribution, edge bolt failures would be expected to remain random, driven by preload and normal operating loads only, and would not be expected to permit large baffle gaps to open. In the presence of large clusters of BFB failures, it might also be reasonable that the edge bolts along those clusters might see significant clustered failures as well, although this has not been observed in the relevant industry operating experience. For example, there has never been a confirmed edge bolt failure in the industry, despite widespread visual failures and loose parts generated from BFB failures at Indian Point 2, Salem, and Cook 2. These plants did not suffer baffle jetting related fuel failures, as would be expected where significant lack of edge bolt functionality existed (Note: Cook had a jetting-related failure, but through a bolt hole previously left empty, not a baffle-baffle seam). In the cases where there were fuel cladding leaks, it was due to loose parts fretting. Moreover, these downflow plants replaced BFBs but no edge bolts in their replacement campaigns. However, they did not suffer baffle jetting afterwards, which suggests that there were no large clusters of edge bolt failures from their events.

Baffle-Jetting as Leading Indicator

Several previous evaluations have determined that it is not credible for large leakage flows through multiple baffle gaps to occur and exceed the core bypass criteria without detectable fuel cladding failures occurring beforehand. WCAP-17586 Rev. 2 demonstrated in the Momentum Flux (Baffle jetting) evaluation that [

]a,c which would result in elevated RCS activity levels and be detectable promptly. To exceed the core bypass criteria, not only do many edge bolts have to fail, which would by themselves have resulted in jetting, but then many baffle bolts around the full periphery of the baffle former assembly must also be failed to allow for substantial opening of baffle-baffle joints.

Both WCAP-15036, Rev.1 and WCAP-15425 contain the following statement in their Sections 5.4:

[

]a,c

Edge Bolts Required for Cluster Growth Across Interior Joints

[

]a,c

Conservatism of Treating Worst Quadrant as Limiting of Whole Assembly

[

]a,c

<u>Impact of Modeling Anomalies and Other Considerations</u>

It should be noted that following the initial predictive model development and analysis, there were significant efforts made to devise an improved or more accurate method to quantitatively determine core bypass flow from a bounding representation of the worst-case predictive output patterns. These efforts were hampered because [

][a,c]

During the course of the follow-on sensitivity study effort, two specific contributors were identified as, at least in part, leading to the unfavorable results for Unit 2. In a finite number of cases where the [

][a,c] However, Customer Pattern A and Customer Pattern B from the ABPA analysis demonstrated that a [

][a,c]

A second unusual observation that is unfavorably affecting the reported reliability of Unit 2 Quadrant 3 is related to Plate 5 at the left edge of the figure below:

a,c

In 2013, Plate 5 had no failures in the UT data, and Plate 4, adjacent to it, had only 2 widely separated UT indications. The predictive model is showing [

][a,c] This was exactly the case that was selected for CBF analysis as the worst-case for the sensitivity study described earlier in this section, which could not be shown to pass, even with partial edge bolt credit, so this pattern of predicted bolt failures produces unfavorable results.

As a follow-on activity in the sensitivity study, Westinghouse assessed the physical and analytical drivers for this phenomenon. A contributing factor could be the large cluster on plate 3 former level 5-6. However, there is no satisfactory explanation why plate 4 is seemingly less impacted than plate 5, despite being closer to the cluster, especially as it pertains to cluster expansion into the top former and further down the plate. While postulated as a contributor, a review of relative radiation dose also failed to explain the difference. [

][a,c] This is because any random failures on the line of symmetry are immediately modeled as a cluster of two across the joint due to the symmetry. This artificiality results in a cascading effect. This effect is evident in that plate 5, in all octants of the U2 predictive results, showed anomalously high failure probabilities on formers 5 and 6, regardless of proximity to existing as-found UT indications. This effect is also present in the Unit 1 results, as visible in the heat maps, thus confirming that this is a modeling artifact.

Given the totality of the information, the formation of a large cluster on plate 5 that leaves several contiguous former levels completely unsupported is not consistent with the as-found data (UT data for this and other quadrants and the other unit). In other words, if plate 5 had an increased propensity for failures for some reason, it should be showing up in the plant data, but it is not. For these reasons, the model outputs showing a large cluster spontaneously developing on plate 5 between 2013 and 2023 timeframe are likely spurious. Since the 4-5 joint is a large contributor to core bypass flow and extremely sensitive to any kind of clustering, this modeling anomaly is a key contributor to the low reported reliability of Quadrant 3 at the 10-year mark. Within the predictive model results, [

][a,c] This was by far the most frequently impacted plate by this phenomenon. On the contrary, it is judged to be unrealistic for plate 5 (with no failures at time zero) to undergo such substantial degradation over the inspection interval with no apparent relationship to the as-found data. Nonetheless, [
][a,c] so even though the prediction is unrealistic, the 95% reliability target applied to the faulted structural criteria is almost satisfied.

Summary

Prairie Island Units 1 and 2 are both predicted to meet the structural stability and fuel grid impact limits of an ABPA with at least 95% probability at the 10-year point from the baseline inspection. Therefore, the requirements of PWROG-17071 and WCAP-17096 Rev. 3 are met. However, Core Bypass Flow is a design limit that needs to be considered, as we have done herein. In summary, a detailed assessment of the Westinghouse predictive model results does provide reasonable expectation that the core bypass flow design limit will be satisfied at the 10 year post-inspection point. This assessment takes into consideration the significant conservatism involved with [ ]^{a,c} A sensitivity study demonstrated that even with no edge bolts, many of the predictive model results that were counted as failures based on [ ]^{a,c} would have instead been categorized as acceptable under a rigorous baffle gaps and thermal hydraulic calculation. Those predictive model results that could not be shown to pass a rigorous CBF calculation occur very infrequently in the Monte Carlo simulation. This frequency was judged to be acceptably small. There is reasonable expectation that the physical plant configuration will remain in compliance with the design basis until the time of next inspection in 2023. Edge bolt failures are a prerequisite for baffle gap opening, and extensive edge bolt failure has been shown to be tolerable without challenging core bypass flow limits even for a worst-case pattern bounding of the predictive model outputs. Finally, as defense in depth, it has been concluded in both WCAP-15036 and WCAP-15425 that leakage through the baffle gaps is a self-revealing phenomenon due to baffle jetting, and that this would occur well before challenging the core bypass flow design limits.

8.0 Conclusion

Given the above-described conservatisms, the Westinghouse predictive results provide reasonable expectation that core bypass flow will remain within design limits at all times up to the time of next inspection 10-years from the last Unit 2 inspection, which occurred in Fall 2013.

9.0 Attachments

9.1 [ ]^{a,c}

PINGP EVALUATION

a,c

PINGP EVALUATION

NUMBER:	**608000000371**	**REVISION:**	0	**MINOR REVISION:**	0	**SHEET NO:**	36 OF 36
TITLE:	**BAFFLE BOLT CORE BYPASS FLOW PREDICTIVE MODELING**					**DATE: 5/2/19**	
						COMP. BY:	CRK

a,c

ENCLOSURE 3

Westinghouse Letter LTR-AMLR-19-18-NP,
“Prairie Island Unit 2 – Baffle-Edge Bolt Sensitivity Study”
(Non-Proprietary)

4 pages follow

Westinghouse

To: Nathan Lang
Reactor Internals Design & Analysis I

Date: March 21, 2019

From: Louis Turicik
Aging Management & License Renewal

Your ref: N/A

Ext: 412-374-6072

Our ref: LTR-AMLR-19-18-NP, Rev. 0

Subject: **Prairie Island Unit 2 – Baffle-Edge Bolt Sensitivity Study**

References:

1. Westinghouse Project Impact Notice, PIN-003, Rev. 1, "Prairie Island Unit 1 & 2 BFB Inspection Interval Evaluation," February 1, 2019. *(electronically attached to this letter, see PRIME)*
2. Westinghouse Calculation Note, CN-AMLR-18-9, Rev. 1, "Baffle Former Bolt Predictive Reliability Analysis for Prairie Island Units 1 and 2," March 2019. (Proprietary)
3. Westinghouse Report, WCAP-17586-P, Rev. 2, "Determination of Acceptable Baffle-Barrel Bolting for Prairie Island Units 1 and 2," October, 2018. (Proprietary)
4. Xcel Energy Design Information Transmittal (DIT), DIT 601000000463-05, Rev. 0, "Prairie Island Owner-Elected Partial Edge Bolt Pattern for PIN-003 Edge Bolt Sensitivity Study," March 14, 2019.

Xcel Energy has requested a sensitivity study regarding the impact of baffle-edge bolts on core bypass flow under degraded baffle-former bolt (BFB) conditions (Reference 1). The BFB predictive model pattern results from Reference 2 have been re-evaluated in order to select two representative patterns which are intended to approximate bounding (i.e. "worst case") patterns with regards to core bypass flow at the end of a hypothetical 10-year re-inspection interval for Prairie Island Unit 2. The selected patterns, Case 1 and Case 2, can be seen in Figure 1. These two patterns were evaluated for the effect of BFB degradation on baffle plate gaps. The baffle gaps structural finite element model from the Prairie Island ABPA (Reference 3) was modified to include the explicit representation of baffle-edge bolts along the baffle plate seams. One customer-supplied baffle-edge bolt pattern from Reference 4 was considered (see Figure 2), as well as cases with all edge bolts intact and no credit given to any edge bolts. These three baffle-edge bolt sensitivity study cases were evaluated in conjunction with each of the two selected degradation patterns from the BFB predictive model, resulting in a total of six baffle gap analyses. The results of each baffle gap analysis were used as input to the THRIVE code to evaluate core bypass flow. The results of each THRIVE run includes the core bypass flow percentage (which can compared to the Prairie Island Unit 2 design limit of **[ ]**^{a,c} as listed in Reference 3) and any adverse flow conditions, if identified. These results are summarized in Table 1.

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[ ]a,c

Figure 1: Selected Patterns for Core Bypass Flow Sensitivity Study

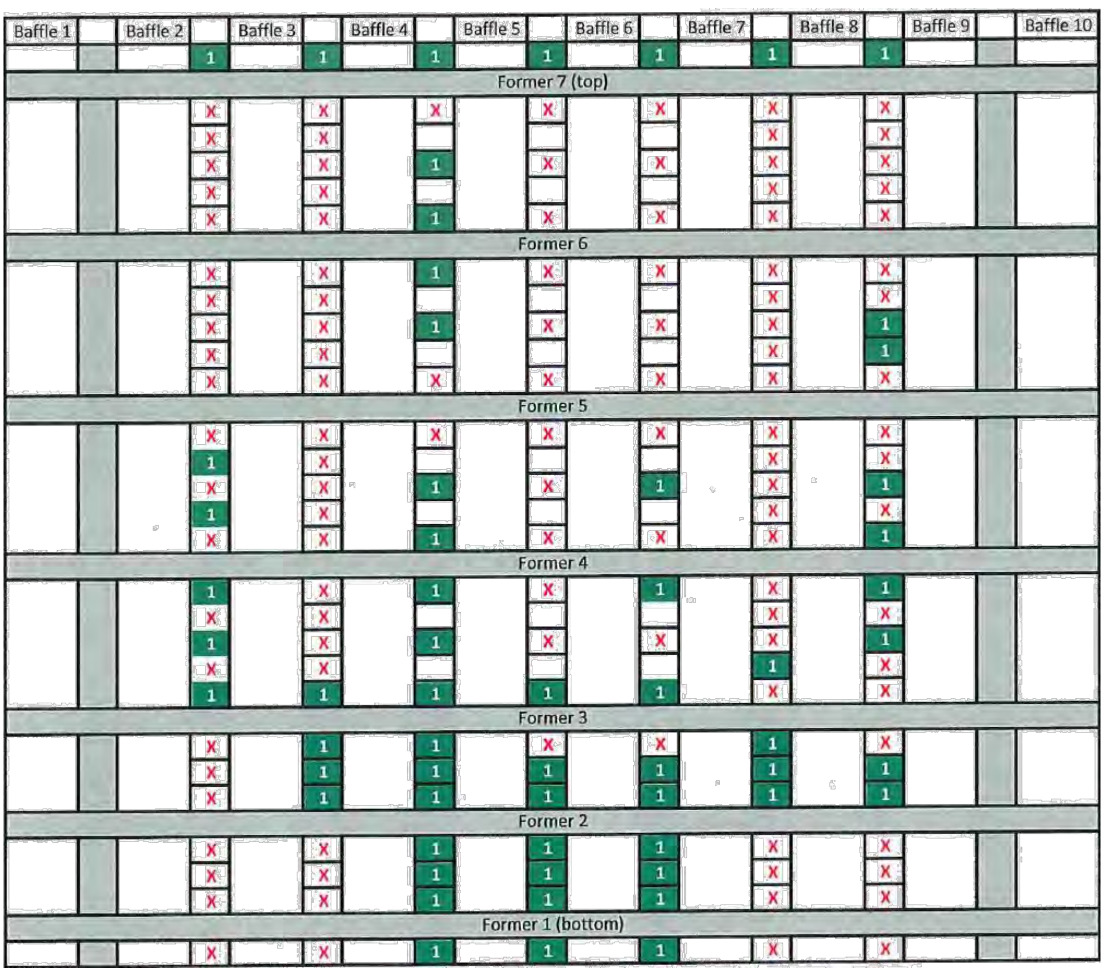

Figure 2: Partial Edge Bolt Pattern for Sensitivity Study (Reference 4)

Table 1: Core Bypass Flow Results

[ ]a,c

*** This record was final approved on 3/22/2019 10:21:09 AM. (This statement was added by the PRIME system upon its validation)

As can be seen in Table 1, the Case 1 bolting pattern resulted in acceptable core bypass flow results for the run with all edge bolts intact, but resulted in a non-converged solution in THRIVE with partial edge bolts, and a non-converged solution of the structural baffle gaps evaluation (prior to THRIVE) in the case with no edge bolts intact. As such, explicit core bypass flow percentages could not be determined for these runs.

The Case 2 bolting pattern showed acceptable core bypass flow results for all runs performed, but resulted in a condition of "reverse flow" within the baffle-barrel region for the partial edge bolt and no edge bolt cases. Reverse flow occurs when all the coolant that enters the baffle-barrel region exits the baffle-barrel region before reaching the lower core plate. The effects of such a flow distribution remain unknown, as this condition is beyond the design basis for Westinghouse reactor internals. A subsequent run with all baffle-former bolts intact and no edge bolts was also performed in Reference 2, which likewise resulted in a condition of reverse flow. However the reverse flow condition does not occur when all baffle-former bolts and edge bolts are included (i.e. the original design condition). This demonstrates that the reverse flow phenomenon is more likely a result of the assumed edge bolt degradation, rather than the baffle-former bolt pattern itself.

In summary, through the evaluations performed herein, it was found that one of the two BFB predictive model patterns, which were intended to approximate bounding core bypass flow conditions, met the design core bypass flow limit. This indicates that, were an explicit core bypass flow evaluation performed for all of the predictive model pattern results, the percentage of acceptable patterns for Prairie Island Unit 2 at the end of a 10-year re-inspection interval would be greater than the **[ ]**^{a,c} reported in Reference 2.

If you have any questions, please contact the undersigned.

Authored by: ELECTRONICALLY APPROVED[1]
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*** This record was final approved on 3/22/2019 10:21:09 AM. (This statement was added by the PRIME system upon its validation)

ENCLOSURE 4

Westinghouse Affidavit

4 pages follow

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:
COUNTY OF BUTLER:

(1) I, Nancy B. Closky, have been specifically delegated and authorized to apply for withholding and execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse).

(2) I am requesting the proprietary portions of LTR-AMLR-18-55-P Rev. 1, 608000000371-P Rev. 0, and LTR-AMLR-19-18-P Rev. 0 be withheld from public disclosure under 10 CFR 2.390.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged, or as confidential commercial or financial information.

(4) Pursuant to 10 CFR 2.390, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse and is not customarily disclosed to the public.

(ii) Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

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(5) Westinghouse has policies in place to identify proprietary information. Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage (e.g., by optimization or improved marketability).

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(6) The attached documents are bracketed and marked to indicate the bases for withholding. The justification for withholding is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These

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lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (5)(a) through (f) of this Affidavit.

I declare that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on: May 22, 2019

Nancy B. Closky
Nancy B. Closky
Plant Licensing & Engineering

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of a document, furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

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