April 14, 2021
L-2021-072
GL 2004-02

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

RE: Turkey Point Nuclear Plant, Units 3 and 4
Docket Nos. 50-250 and 50-251
Renewed Facility Operating Licenses DPR-31 and DPR-41

Supplement to Updated Final Response to NRC Generic Letter 2004-02

References:

1. Florida Power & Light Company Letter L-2017-149, Updated Final Response to NRC Generic Letter 2004-02, December 29, 2017 (ADAMS Accession No. ML17363A265)

2. Point Beach Nuclear Plant, Units 1 and 2; Seabrook Station, Unit No. 1; St. Lucie Plant, Units 1 and 2; and Turkey Point Nuclear Generating Units 3 and 4 - Audit Report Regarding Generic Letter 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors" Closure Methodology (EPID 2017-LRC-0000), December 2, 2019 (ADAMS Accession ML19217A003)

In Reference 1, Florida Power & Light Company (FPL) provided on behalf of Turkey Point Nuclear Units 3 and 4 (Turkey Point), an updated final response to Generic Letter (GL) 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors (ADAMS Accession No. ML042360586). Included within were FPL's statement of compliance with the *Applicable Regulatory Requirements* of GL 2004-02, a description of completed plant modifications and process changes, and an evaluation of the 16 issue areas identified in the NRC's 'Revised Content Guide for Generic Letter 2004-02 Supplemental Responses (ADAMS Accession No. ML073110389), including a summary of the significant margins and conservatisms utilized in supporting analyses to demonstrate that the risk of GL 2004-02 related failures at Turkey Point have been reduced to an acceptable level.

During January 15, 2019 through January 17, 2019, the NRC staff conducted an audit of the updated final response to GL 2004-02 at FPL's Juno Beach facility. In Reference 2, the NRC staff reported their audit results. For Turkey Point, Reference 2 identified open issues requiring additional information. This letter and its enclosures provide the requested additional information.

Enclosures 1 through 3 provide responses to NRC audit questions #9, #16 and #17 of Reference 2.

Enclosure 4 provides the updated in-vessel downstream effects analysis for Turkey Point. In Reference 1, FPL additionally provided on behalf of Turkey Point, an analysis of the effects of post-LOCA debris inside the reactor vessel based on the methodology in WCAP-17788-P, Revision 0, 'Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)' (ADAMS Accession No. ML15210A669). At the time of the Reference 1 submittal, WCAP-17788-P, Revision 0, was undergoing NRC review and as a result, the in-vessel effects analysis was not addressed during the January 2019 audit at FPL's Juno Beach facility. In September 2019, the NRC issued 'U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses' (ADAMS Accession No. ML19228A011), which outlined approaches acceptable to the NRC for conducting in-vessel effects evaluations. In addition, the Pressurized Water Reactor Owners Group (PWROG) issued PWROG-16073-P, 'TSTF-567 Implementation Guidance, Evaluation of In-Vessel Debris Effects, Submittal Template for Final Response to Generic Letter 2004-02 and FSAR Changes', Revision 0 (Proprietary). The revised analysis of this enclosure follows the NRC and PWROG guidance to demonstrate compliance

with 10 CFR 50.46(b)(5) with regard to post-LOCA debris effects inside the reactor vessel. The revised evaluation supersedes and replaces the in-vessel downstream effects evaluation provided in Reference 1.

In addition, two questions from the NRC's audit report (Reference 2) relate to licensing implications of the Turkey Point strategies. Section 6.0, "Alternate Evaluation" of NEI 04-07, Volume 1, (ADAMS Accession No. ML050550138) describes an alternate evaluation methodology for analyzing containment sump performance building upon risk insights and more realistic considerations relative to the highly conservative methodologies presented in earlier sections of Volume 1. In NEI 04-07, Volume 2 (ADAMS Accession No. ML050550156), the NRC authorized for use, Section 6.0 of NEI 04-07, Volume 1, subject to the conditions and limitations specified in the accompanying safety evaluation report (SER). FPL evaluated application of this alternative methodology for use in the containment sump performance analyses for Turkey Point Unit 3 and Unit 4 and concluded that neither an exemption from applicable regulatory requirements nor a license amendment is required. The enclosed audit report responses supporting GL 2004-02 closure reflect this determination.

Finally, as stated in Reference 1, changes to the Turkey Point licensing basis have been implemented which allowed FPL to complete plant modifications that enhanced Turkey Point's capabilities with respect to the information in GL 2004-02. Accordingly, the assumptions and inputs used to establish the bases for GL 2004-02 closure are consistent with the Turkey Point licensing basis and no new changes pursuant to 10 CFR 50.90 are being proposed as a result of this submittal. Upon NRC acceptance of FPL's closure of GL 2004-02, the updated final safety analysis reports (UFSARs) for Turkey Point Units 3 and 4 will be revised to reflect the final closure information in accordance with 10 CFR 50.71(e).

This letter contains no new regulatory commitments.

Should you have any questions regarding this submittal, please contact Robert Hess, Turkey Point Licensing Manager, at 305-246-4112.

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

Executed on the 14th day of April 2021.

Sincerely,

M. Pearce

Michael Pearce
Site Vice President, Turkey Point Nuclear Plant
Florida Power & Light

Enclosures:

1. Response to NRC Audit Question #9 Regarding Vortexing Evaluation
2. Response to NRC Audit Question #16 Regarding Flashing Evaluation
3. Response to NRC Audit Question #17 Regarding Void Fraction
4. Updated Resolution to In-Vessel Downstream Effects

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, Turkey Point Nuclear Plant
USNRC Senior Resident Inspector, Turkey Point Nuclear Plant
Ms. Cindy Becker, Florida Department of Health

Supplement to Updated Final Response to NRC Generic Letter 2004-02

Enclosure 1

Response to NRC Audit Question #9 Regarding Vortexing Evaluation

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1. Background and Purpose

During the Nuclear Regulatory Commission's (NRC) audit of the Turkey Point (PTN) Generic Letter (GL) 2004-02 submittal (Ref. 1), a concern was raised about using average approach velocities when comparing vortexing data from the St. Lucie Unit 1 (PSL-1) test strainer with the PTN Unit 3 (PTN-3) plant strainer (Ref. 2).

The purpose of this enclosure is to reevaluate the likelihood of vortexing at the PTN-3 plant strainer by comparing the Froude number for the strainer module with the maximum average approach velocity at PTN-3 to the strainer testing data from Diablo Canyon Power Plant (DCPP) and Vogtle Electric Generating Plant (VEGP). Numerous studies, including NUREG/CR-2760, have shown that increasing Froude numbers are correlated with an increased likelihood of vortexing and formation of more risk-significant vortex types. Additionally, Froude number similarity has been shown to predict plant scale sump behavior from scale testing.

The comparison with DCPP is performed on debris laden vortex testing data of the front module of the DCPP test strainer (see Section 2.1). The comparison with VEGP is performed on clean strainer vortex testing data of the VEGP test strainer (see Section 2.2).

2. Approach

PTN-3 analyzed the non-uniform flow distribution at clean strainer conditions by balancing the pressure drop between different strainer modules and the sump strainer exit. The PTN-3 plant strainer features multiple modules, and each module has strainer disks mounted on one side of a plenum box. The analysis showed that strainer module A-1 has the highest approach velocity, and the parameters related to vortexing analysis for module A-1 are shown in the table below.

Table E1-1: PTN-3 Maximum Approach Velocity Strainer Module Parameters

Parameter	Value
Max ECCS Flowrate (piggy-back configuration)	3,446 gpm
Maximum Approach Velocity Module	A-1
Module A-1 Average Screen Approach Velocity (v)	0.00832 ft/s
Minimum Module A-1 Plenum Submergence (s) (based on minimum SBLOCA water level)	0.240 ft

The dimensionless Froude number (F_r) for the screen approach velocity across assembly A-1 at PTN-3 can be calculated as shown below:

$$F_r = \frac{v}{\sqrt{gs}} = \frac{0.00832}{\sqrt{32.2 \times 0.240}} = 0.00299$$

2.1. Diablo Canyon Power Plant Vortex Testing

Due to the similarity in physical design, the Froude number and vortexing performance of the front module of the DCPP test strainer is compared against the data from PTN-3's maximum approach velocity strainer module. Figure E1-1 shows the arrangement of the DCPP test strainer.

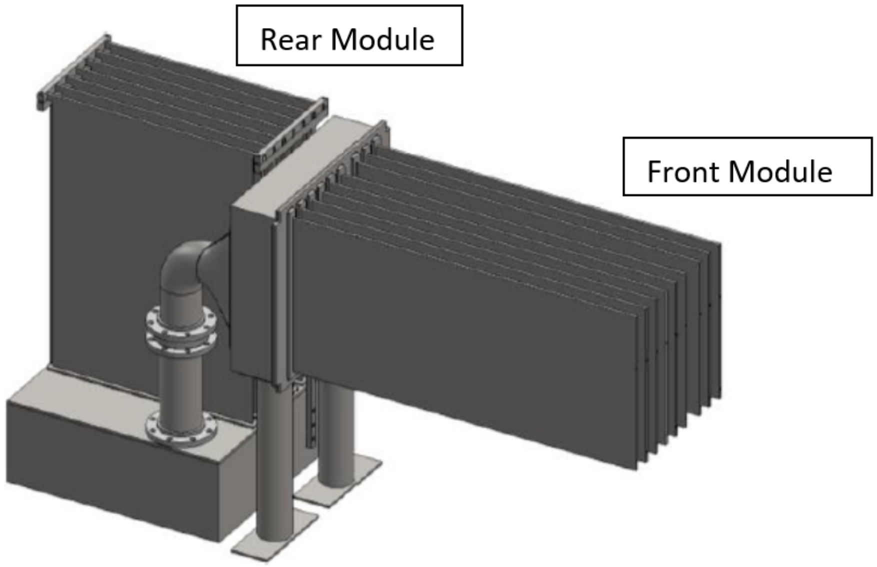

Figure E1-1: DCPP Test Strainer General Arrangement

The key parameters of the vortex testing of the front module of the DCPP test strainer are shown below in Table E1-2.

Table E1-2: DCPP Front Module Vortexing Test Parameters

Parameter	Value
Average Vortex Testing Flowrate	894.4 gpm
Front/Rear Module Flow Split for Debris Laden Conditions	49%/51%[(a)]
Front Module Disk Surface Area	186.8 ft^2
Front Module Average Approach Velocity	0.00523 ft/s
Maximum Front Strainer Module Vortex Testing Submergence	0.033 ft + 0.2725 ft = 0.306 ft[(b)]

[(a)] Debris laden testing showed a debris loaded front/rear module flow split of 49%/51%.
[(b)] A submergence of 0.033 ft was measured from the top of the rear strainer module. The top of the front strainer module was 0.2725 ft lower than the top of the rear strainer module during DCPP strainer testing.

The dimensionless Froude number for the front test strainer module at DCPP can be calculated as shown below:

$$F_r = \frac{0.00523}{\sqrt{32.2 \times 0.306}} = 0.00167$$

As seen in the DCPP Head Loss Technical Report, no substantial vortex formation was observed in the test loop throughout the vortex testing. No air entrainment from the strainer

assembly was observed even when the strainer submergence was further reduced, uncovering the rear strainer disk support cross-members. Accounting for this reduction in submergence, the maximum tested Froude number during the DCPP vortexing testing was:

$$F_r = \frac{0.00523}{\sqrt{32.2 \times 0.2725}} = 0.00177$$

This Froude Number is less than but at the same order of magnitude as that of the PTN-3 strainer.

2.2. Vogtle Electric Generating Plant Vortex Testing

Multiple vortexing tests were performed on the VEGP clean test strainer at various combinations of flow rate and strainer submergence (Ref. 3). The dimensionless Froude number is calculated for each test using the same equation as above. Table E1-3 presents the test parameters and calculated Froude number for each vortex test.

Table E1-3: Froude Number for VEGP Test Parameters

Test No.	Submergence		Approach Velocity (ft/s)	Froude No.
	(in)	(ft)		
1	3.625	0.302	0.0258	0.00827
2	3.625	0.302	0.0355	0.01138
3	4.175	0.348	0.0306	0.00914

No vortices were observed during VEGP testing for the test conditions shown above. Since the Froude numbers for the VEGP test conditions are greater than that of the PTN-3 strainer, it is reasonable to conclude that air-entraining vortexing will not be a concern for PTN-3.

3. Conclusions

Vortex testing at both DCPP (debris laden) and VEGP (clean strainer) has Froude numbers that either approximate within a factor of two (DCPP), or exceed (VEGP) the Froude number calculated for the highest approach velocity strainer module at PTN-3. As such, there is reasonable assurance that the vortexing characteristics of the maximum approach velocity PTN-3 strainer module are bounded by these tests. Since no air-entraining vortices were observed in the VEGP and DCPP tests, air entrainment due to vortexing will not impact the PTN-3 plant strainer modules during recirculation.

4. References

1. L-2017-149, Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251, Renewed Facility Operating Licenses DPR-31 and DPR-41, Updated Final Response to NRC Generic Letter 2004-02, December 29, 2017, ADAMS Accession No. ML17363A265.
2. NRC, Point Beach Nuclear Plant, Units 1 and 2; Seabrook Station, Unit No. 1; St. Lucie Plant, Units 1 and 2; and Turkey Point Nuclear Generating Units 3 and 4 – Audit Report Regarding Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors” Closure

Methodology (EPID 2017-LRC-0000), December 2, 2019, ADAMS Accession No. ML19217A003.

3. Vogtle Electric Generating Plant Unit 1 & 2, Supplemental Response to NRC Generic Letter 2004-02, July 20, 2018, ADAMS Accession No. ML18193B165.

Supplement to Updated Final Response to NRC Generic Letter 2004-02

Enclosure 2

Response to NRC Audit Question #16 Regarding Flashing Evaluation

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1. Background and Purpose

During the Nuclear Regulatory Commission's (NRC) audit of the Turkey Point (PTN) Generic Letter (GL) 2004-02 submittal (Ref. 1), a concern was raised about crediting the pre-accident air partial pressure to prevent flashing immediately downstream of the strainer (Ref. 3).

The purpose of this enclosure is to 1) reevaluate flashing for PTN Units 3 and 4 by crediting a small amount of containment accident pressure and 2) show that the containment pressure used for flashing analysis is much lower than the post-accident containment pressures from the PTN containment analysis.

2. Approach

The margin in post-accident containment pressure for preventing the occurrence of flashing immediately downstream of the strainer is evaluated for various post-accident containment and sump conditions, as shown in Table E2-1. For each set of conditions, the sump pool temperature and containment pressure are obtained from the design basis profiles evaluated for the double-ended guillotine break at the reactor coolant pump (RCP) suction with failure of one Emergency Diesel Generator (EDG). The strainer head loss is taken from the PTN GL 2004-02 submittal (Ref. 1). Note that the maximum total strainer head loss of 3.71 psi (including chemical effects), which bounds all PTN Unit 3 and Unit 4 breaks, is used in this analysis.

Flashing would occur if the pressure immediately downstream of the strainer is lower than the water vapor pressure at the sump temperature. The minimum containment pressure that is required to prevent flashing is calculated by adding the strainer head loss (3.71 psi, identified above) to the water vapor pressure at the sump temperature of interest and subtracting the strainer submergence. Finally, this minimum required containment pressure is compared with the post-accident containment pressure from the containment analysis to determine margin. The equations below illustrate the approach:

$$P_{Downstream} = P_{Containment} + \text{Submergence} - \text{Head Loss}$$
$$P_{Containment} + \text{Submergence} - \text{Head Loss} \geq \text{Vapor Pressure}$$
$$P_{Containment} \geq \text{Vapor Pressure} + \text{Head Loss} - \text{Submergence}$$

The evaluation contains the following conservatisms:

- One deviation was taken from the methodology discussed above when calculating the minimum containment pressure required to prevent flashing. The submergence of the strainer is conservatively neglected, as shown in Table E2-1. Including the submergence would reduce the minimum pressure required and increase the margin.
- As stated above, the maximum total strainer head loss for PTN Unit 4 Region II breaks (3.71 psi, including chemical effects) is used in the analysis. This value is greater than the maximum total strainer head losses for all breaks for Unit 3 (1.18 psi), and all Unit 4 Region I breaks (2.11 psi).
- The debris head losses used to determine the total strainer head losses were measured at a nominal temperature of 120°F (Ref. 1, pages E1-100, E1-112 and E1-127) and were not adjusted to the higher temperatures, at which the flashing analysis was performed

(see Table E2-1). This is conservative because adjusting the head loss to higher temperatures would result in lower head loss values due to lower fluid viscosity.

3. Results

The methodology described above is applied to various post-accident containment and sump conditions. The results are summarized in the table below.

Table E2-1: Margin in Containment Pressure for Preventing Flashing

Time (s)	Sump Pool T (°F)	Vapor Pressure (psia)	Strainer Head Loss (psi)	Min P Req'd to Prevent Flashing (psia)	Post-Accident Containment Pressure		Margin (psi)
					(psig)	(psia)	
4,752	237.78	23.98	3.71	27.69	27.30	42.00	14.31
5,362	236.18	23.29	3.71	27.00	28.60	43.30	16.30
19,503	220.58	17.38	3.71	21.09	22.70	37.40	16.31
53,503	202.92	12.24	3.71	15.95	16.70	31.40	15.45
434,003	162.81	5.07	3.71	8.78	8.00	22.70	13.92

Assuming that post-accident containment pressure is equal to water vapor pressure at the corresponding sump temperature, a maximum containment accident pressure of 3.7 psi needs to be credited to prevent flashing immediately downstream of the strainer. As shown in the table above, crediting this amount of accident containment pressure is reasonable and conservative, as there exist margins of at least 13 psi between the post-accident containment pressure from analysis and minimum pressure required to prevent flashing.

It is recognized that the margins shown in the above table are based on the temperature and pressure curves from the design basis containment analysis, which biased the inputs to maximize the containment pressure and sump temperature. Should more realistic inputs be used, the post-accident sump temperature and containment pressure could be lower. The PTN containment analysis compares the containment pressure curves of several different break scenarios (e.g., crossover leg break with the failure of an EDG, crossover leg break with the failure of a containment spray pump, crossover leg break with the failure of an intake cooling water pump, and a hot leg break with the failure of an EDG), and the maximum difference in containment pressure between these cases is approximately 10 psi at the start of recirculation. Two items should be noted.

- Even if this maximum difference in containment pressure is subtracted from the margins shown in the above table, there still exist margins of at least 3.9 psi between the post-accident containment pressure and minimum pressure required to prevent flashing. This approximation is conservative because it does not credit any variations in the sump temperature and the corresponding vapor pressure in the other curves.
- As shown in Table E2-1 for the crossover leg break with an EDG failure, the difference between the containment pressure from the containment analysis and the sump vapor pressure is approximately 18 psi at any given time during recirculation. This difference reveals the significant conservatism built into the assumption that the containment pressure is equal to water vapor pressure at the corresponding sump temperature for flashing analysis.

Vogtle used a similar approach to demonstrate applicability of using accident containment pressure to mitigate flashing. In their July 2018 submittal (Ref. 2), the minimum containment pressure required to prevent flashing was compared with the containment pressure from containment analysis to show margin. Vogtle performed the analysis using both the design basis and best estimate containment analysis results. The comparison showed that, when using the best-estimate sump temperature and containment pressure curves, the margins in the containment pressure for mitigating flashing (see Figures 3.f.14-2 and 3.f.14-3 in Ref. 2) are either greater than or comparable with those based on the design basis curves (see Figures 3.f.14-1 in Ref. 2). This is due mainly to the lower sump temperatures when best estimate conditions are used. While Vogtle has a different nuclear steam supply system (Westinghouse 4-loop) from PTN (Westinghouse 3-loop), similar comparison in containment pressure is expected, should PTN reperform their containment analysis using best estimate inputs.

4. References

1. L-2017-149, Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251, Renewed Facility Operating Licenses DPR-31 and DPR-41, Updated Final Response to NRC Generic Letter 2004-02, December 29, 2017, ADAMS Accession No. ML17363A265.
2. Vogtle Electric Generating Plant Units 1 and 2 Supplemental Response to NRC Generic Letter 2004-02, July 10, 2018, ADAMS Accession No. ML18193B163 and ML18193B165.
3. NRC, Point Beach Nuclear Plant, Units 1 and 2; Seabrook Station, Unit No. 1; St. Lucie Plant, Units 1 and 2; and Turkey Point Nuclear Generating Units 3 and 4 – Audit Report Regarding Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors” Closure Methodology (EPID 2017-LRC-0000), December 2, 2019, ADAMS Accession No. ML19217A003.

Supplement to Updated Final Response to NRC Generic Letter 2004-02

Enclosure 3

Response to NRC Audit Question #17 Regarding Void Fraction

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1. Background and Purpose

During the Nuclear Regulatory Commission's (NRC) audit of the Turkey Point (PTN) Generic Letter (GL) 2004-02 submittal (Ref. 1), a concern was raised about the impact of void fraction on the net positive suction head (NPSH) required for the pumps taking suction through the sump strainers, using the methodology in Regulatory Guide (RG) 1.82 (Ref. 2).

> *Describe how the potential for void fraction at the pump inlet at less than the 2 percent limit was evaluated for effect on NPSH margins. RG 1.82 provides guidance for increasing NPSH required as void fraction at the pump suction increases from 0-2 percent. If the RG 1.82 guidance is not followed, provide assurance that the pumps will operate for the required time period considering the void fractions predicted at the pump suction.*

The purpose of this enclosure is to address the NRC's assertion by considering two points: First, the physical aspect of what is being asserted is discussed. Then, the PTN licensing basis is reviewed for the requirement of compliance with RG 1.82.

2. Physical Phenomena

The quantities of voids that are expected to be developed within the sump are the result of two phenomena: dissolved air coming out of solution and water flashing due to a change in pressure (at a fixed temperature) as the water crosses the screen and debris bed during the recirculation phase following a loss of coolant accident (LOCA). PTN analyzed the maximum void fraction at the predicted peak post-LOCA sump temperature of 237.8°F using Henry's Law. Table E3-1 shows the resulting void fractions at the midpoint of the strainer disks for both units.

Table E3-1: PTN Void Fraction at Strainer Mid-Height

Parameter	PTN3	PTN4 Region I	PTN4 Region II
Void Fraction	0.94%	1.50%	1.60%

One significant conservatism in this analysis is the use of minimum containment pressure as input. The containment pressure was assumed to be the saturation pressure at the sump temperature (23.99 psia) plus a small amount of containment accident pressure (1.5 psi for PTN3; 3 psi for PTN4 Region I; and 6 psi for PTN4 Region II). The PTN containment analysis showed that the containment pressure corresponding to this peak sump temperature is 42 psia, which is at least 12 psi higher than the containment pressures assumed in the void fraction analysis for PTN3 and PTN4. Therefore, the amounts of containment accident pressure credited are acceptable. It is recognized that the containment analysis biased the inputs to increase the containment pressure. However, should more realistic inputs be used in the containment analysis, both the sump temperature and containment pressure would decrease, and the above justification is expected to remain valid. It should also be noted that the void fractions shown above would be greatly reduced if the full containment pressure from the containment analysis (42 psia) is used, as done in other GL 2004-02 submittals (Ref. 3, Page 92; Ref. 4, Page 24; Ref. 5, Page 3-109).

There are several other conservatisms in the PTN void fraction analyses:

1) The strainer head losses used included a calculated clean strainer head loss and measured debris head losses at higher flow rates and lower temperatures than the conditions used for the void fraction analysis. This conservatively increases the head loss and void fraction.
2) The submergence at the strainer mid-height was calculated based on the minimum SBLOCA water level while the analysis used the maximum LBLOCA strainer head losses. Combining these inputs in the analysis is conservative, as they would not happen simultaneously.

When the voids formed at the strainer are transported to the pump, the increased elevation head generated in moving the fluid down to the suction of the pump overcomes the head loss through the combination of the piping, strainer, and debris bed. The increasing pressure as the voids flow to the pump suction tends to compress and collapse the bubbles. An analogous situation would be cavitation where water near its saturation point experiences a rapid pressure drop, for example, flow through a valve. Bubbles form rapidly due to the pressure drop in the throat of the valve. As the fluid slows and the pressure recovers downstream of the valve, the bubbles rapidly collapse; they do not persist to be delivered further downstream after the pressure has recovered. Similarly, the water in the sump transits the screen and debris bed, and experiences a pressure drop that allows bubbles to form. As the fluid is then transported downward to the suction of the pumps, the change in elevation head increases the fluid pressure. For PTN, the increase in pressure is over 5 psi. The analysis of bubble dynamics using the Rayleigh-Plesset equation for spherical bubbles without thermal effects found that "The growth [of the bubbles] is fairly smooth and the maximum size occurs after the minimum pressure. The collapse process is quite different. The bubble collapses catastrophically" (Ref. 6, Section 2.4) as they enter the region of increasing pressure. While this process of bubble collapse may be extended in the sump due to the different timing of the pressure recovery, compared with the cavitation process where "the elapsed times are so small (of the order of microseconds)" (Ref. 6, Section 3.4), it is still anticipated to be negligible relative to the flow transit time from the sump to the suction of the RHR pump which is on the order of at least seconds.

3. Licensing Basis

PTN is not licensed to the RG 1.82 requirement to de-rate the NPSHr for voids at the pump suction. That requirement is presented in RG 1.82 Revisions 1 through 4 as "When air ingestion is 2% or less, compensation for its effects may be achieved without redesign if the 'available' NPSH is greater than the 'required' NPSH plus a margin based on the percentage of air ingestion." Revision 0 of RG 1.82 does not discuss air ingestion or its impact on NPSH requirements for pumps. PTN3 and PTN4 are not licensed to any versions of RG 1.82.

4. Conclusion

While not licensed to the RG 1.82 requirement for adjusting NPSHr due to void fraction at pump suctions, PTN is obligated to ensure Emergency Core Cooling be maintained and must address physical phenomena that could challenge this requirement. As summarized in this enclosure, PTN analyzed the strainer void fractions using a conservative method and showed that the resulting void fractions at the sump strainer are below the acceptance limit of 2% in RG 1.82. Based on the analysis for cavitation, the voids that could form at the strainer will rapidly collapse

as they transport to the pump suction when experiencing higher pressures and therefore will not impact pump operability.

5. References

1. L-2017-149, Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251, Renewed Facility Operating Licenses DPR-31 and DPR-41, Updated Final Response to NRC Generic Letter 2004-02, December 29, 2017, ADAMS Accession No. ML17363A265.
2. NRC, Point Beach Nuclear Plant, Units 1 and 2; Seabrook Station, Unit No. 1; St. Lucie Plant, Units 1 and 2; and Turkey Point Nuclear Generating Units 3 and 4 – Audit Report Regarding Generic Letter 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors” Closure Methodology (EPID 2017-LRC-0000), December 2, 2019, ADAMS Accession No. ML19217A003.
3. Salem Nuclear Generating Station Units 1 and 2 Docket Nos. 50-727 and 50-311 Generic Letter 2004-02 Updated Supplemental Response for Salem, ADAMS Accession No. ML12129A389.
4. U.S. Nuclear Regulatory Commission Staff Review of the Documentation Provided by PSEG Nuclear, LLC for Salem Nuclear Generating Station, Units 1 and 2 Concerning Resolution of Generic Letter 2004-02 Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors, ADAMS Accession No. ML14113A221.
5. Palo Verde Nuclear Generating Station (PVNGS) Units 1, 2, and 3 Docket Nos. STN 50-528, 50-529, and 50-530 Revision 2 to Supplemental Response to NRC Generic Letter 2004-02, Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized-Water Reactors, December 18, 2013, ADAMS Accession No. ML13357A218.
6. Brennen, Christopher Earls, Cavitation and Bubble Dynamics, Oxford University Press, 1995.

Supplement to Updated Final Response to NRC Generic Letter 2004-02

Enclosure 4

Updated Resolution to In-Vessel Downstream Effects

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1. Background and Purpose

In December 2017, Florida Power & Light Company (FPL) provided an updated final response to Generic Letter (GL) 2004-02 for Turkey Point Units 3 and 4 (PTN3 and PTN4) (Reference 1). That submittal included an evaluation of in-vessel downstream effects based on the methodology in WCAP-17788-P Revision 0 (Reference 5). However, at the time of the submittal, the WCAP was still under review by the Nuclear Regulatory Commission (NRC). As a result, the staff reviewed the submittal for the applicability of fiber penetration testing but did not review the in-vessel evaluation (Reference 2, Page 13). In September 2019, the NRC issued the staff review guidance for in-vessel downstream effects (Reference 3). The review guidance outlined approaches that the NRC would deem acceptable to demonstrate compliance with the requirements of 10 CFR 50.46(b)(5) for addressing in-vessel effects. The guidance also developed criteria to determine the level of plant-specific review activity needed to establish compliance (Reference 3).

This enclosure follows the NRC staff review guidance (Reference 3) and Pressurized Water Reactor Owners Group (PWROG) implementation guidance (Reference 4) to describe the in-vessel analysis, establish in-vessel acceptance criteria, and demonstrate that the criteria are met for PTN3 and PTN4.

2. Resolution of In-Vessel Debris Effects

The NRC Review Guidance for the resolution of in-vessel downstream effects (Reference 3) provided four different paths (identified as Box 1 through Box 4 paths) that PWR licensees can use to resolve the issue based on the alternate flow path (AFP) analysis in WCAP-17788-P, Revision 1 (Reference 6). The Box 1 path is for plants with low in-vessel fiber loads based on WCAP-16793 which is unachievable for PTN3 and PTN4. The Box 2 and Box 3 paths are only applicable for B&W units and plants with upper plenum injection, respectively. FPL has elected to use the Box 4 path to address in-vessel downstream effects for PTN3 and PTN4. This section evaluates the applicability of the methods and analytical results from WCAP-17788-P, Revision 1 for PTN3 and PTN4.

2.1. Sump Strainer Fiber Penetration

<u>Response for PTN3</u>

Sump strainer fiber penetration for PTN3 was quantified using the penetration test data from Diablo Canyon Power Plant (DCPP). A comparison of the PTN3 and DCPP sump strainers and justification for using the DCPP penetration testing have been provided in the previous submittal (Reference 1). As discussed in Section 2.2 of this enclosure, a conservative method was used to quantify the PTN3 core-inlet fiber load when applying the DCPP testing data to account for the differences in debris composition between PTN3 and DCPP testing.

<u>Response for PTN4</u>

As stated in the previous GL 2004-02 Submittal (Reference 1), sump strainer fiber penetration for PTN4 was evaluated based on plant-specific penetration testing, which was used to develop a model of fiber penetration through the strainer over time. No changes have been made to the testing or the fiber penetration model in this submittal.

2.2. In-Vessel Fiber Load Analysis

The quantification of fiber that passes through the strainer and reaches the reactor core is discussed below. Only the changes from the previous submittal are discussed.

Response for PTN3

Different from the previous submittal, a more conservative method is used to quantify the reactor core inlet fiber load for PTN3 to account for the difference in fiber composition between the DCPP test and PTN3 plant condition. Instead of the time-dependent analysis based on the WCAP-17788 methodology, constant fiber penetration fractions derived from the DCPP test were applied to the total transported fiber load from the worst case PTN3 break.

The analysis accounted for both prompt fiber penetration, which refers to fiber passing through strainer perforation as soon as it reaches the strainer, and shedding penetration, which refers to fiber that already accumulated on the strainer migrating through the debris bed and strainer perforation.

- Prompt fiber penetration for PTN3 was quantified by multiplying the maximum PTN3 transported fiber load by the maximum prompt penetration fraction (11.3%). This maximum prompt penetration fraction was derived from the DCPP fiber penetration model for a clean strainer. This approach does not credit any reduction in prompt penetration fraction as more fiber accumulates on the strainer. As shown by the DCPP testing, the average prompt penetration fraction drops to 4.4% for the last fiber batch.

- Shedding fiber penetration for PTN3 was calculated by multiplying the maximum PTN3 transported fiber load by the average shedding rate and the hot leg recirculation switchover (HLSO) time of 6.5 hours. Using the 6.5-hour duration to quantify fiber shedding is reasonable because, following transfer to hot leg recirculation, there is sufficient coolant entering the top of the core via the hot leg such that coolant flow through the core inlet is not necessary to adequately remove decay heat.

  The shedding rate was derived from the DCPP testing data. For each of the 10 filter bags that were used to collect shedding fiber, the shedding rate was taken to be the mass ratio between the collected shedding fiber in the filter bag and the fiber on the strainer when the filter bag was placed online, divided by the online duration of the filter bag. The calculated shedding rate decreased as more fiber collected on the strainer, ranging between 2.88×10^{-6} s^{-1} and 9.64×10^{-8} s^{-1}. The average shedding rate (9.76×10^{-7} s^{-1}) was used in the PTN3 analysis.

  Applying the shedding rate to the total transported fiber load conservatively assumes that all the transportable fiber is on the strainer and available for shedding at the start of the recirculation. Additionally, the amount of prompt penetration is not subtracted when quantifying shedding penetration.

Aside from the conservatisms discussed above, the analysis assumed that all the fiber that passes through the strainer accumulates at the core inlet, conservatively neglecting any diversion of fiber away from the reactor by the in-service containment spray (CS) pumps. Additionally, no credit was taken for flow diversion from the reactor core inlet through the AFPs, which is consistent with the latest NRC review guidance (Reference 3).

Since the previous submittal, the transported fiber load for the worst-case PTN3 break was refined by applying a PTN3-specific density of 8 lbm/ft^3 for Kaowool (vs. the generic maximum Kaowool density of 12 lbm/ft^3 based on NEI 04-07). This reduced the total transported fine fiber debris load to 159.39 lbm, including Kaowool, Nukon and latent fiber.

The calculated prompt and shedding fiber penetration quantities for PTN3 were combined, before being divided by the number of fuel assemblies. The resulting reactor core-inlet fiber load is 62.55 g/FA.

Response for PTN4

The in-vessel fiber load analysis at PTN4 has not changed from the previous submittal (Reference 1). No credit was taken for flow diversion from the reactor core inlet through the AFPs, which is consistent with the latest NRC review guidance. The reactor core inlet fiber load is 20.12 g/FA for PTN4.

2.3. Applicability of WCAP-17788 Alternate Flow Path Analysis for PTN

As discussed above, FPL has elected to use the Box 4 path from the NRC review guidance (Reference 3) to address in-vessel downstream effects for PTN3 and PTN4. To use this method, key in-vessel parameters of PTN3 and PTN4 need to be compared with those assumed in the WCAP-17788 analysis to demonstrate applicability, as required in the NRC Review Guidance. Table E1-1 compares the plant parameters with those used in the WCAP-17788. More detailed discussions of the comparison are presented following the table.

Table E1-1 – Summary of In-Vessel Effects Parameters

Parameters	Values from WCAP-17788-P, Revision 1	PTN3 Values	PTN4 Values
Nuclear Steam Supply System (NSSS) Design	Various	Westinghouse	Westinghouse
Fuel Type	Various	Westinghouse 15 x 15 Upgrade Assemblies	Westinghouse 15x15 Upgrade Assemblies
Barrel/Baffle Configuration	Various	Downflow	Downflow
Minimum Chemical Precipitation Time	t_{block} from WCAP-17788, Volume 1, Table 6-1 260 minutes	8 hours	8 hours
Maximum HLSO Time	N/A	6.5 hours	6.5 hours
Maximum Core Inlet Fiber Load for Hot Leg Break (HLB)	WCAP-17788, Volume 1, Table 6-3	62.55 g/FA	20.12 g/FA
Total In-Vessel Fiber Limit for HLB	WCAP-17788, Volume 1, Section 6.4	62.55 g/FA	20.12 g/FA
Minimum Sump Switchover (SSO) Time	20 minutes	14.99 minutes	14.99 minutes
Maximum Rated Thermal Power	2951 MWt	2644 MWt	2644 MWt
Maximum AFP Resistance	WCAP-17788, Volume 4, Table 6-2	WCAP-17788, Volume 4, Table RAI-4.2-24	WCAP-17788, Volume 4, Table RAI-4.2-24
ECCS Flow per FA	8 – 40 gpm/FA	11.6 – 12.7 gpm/FA	11.6 – 12.7 gpm/FA

Comparison of PTN3 and PTN4 Chemical Precipitation Time with HLSO Time and t_{block}

For PTN3 and PTN4 chemical precipitation was shown to occur after the latest HLSO time and after the time that complete core inlet blockage can be tolerated, which is defined in WCAP-17788 as t_{block}.

1. PTN3 and PTN4 chemical precipitation time (t_{chem}) – Chemical precipitation is shown not to occur within at least 8 hours for containment sump temperatures above 160°F following the accident based on the autoclave testing in WCAP-17788, Volume 5 (Reference 6).

   Two different methods were used to determine the precipitation timing for the minimum and maximum pH conditions. For minimum pH conditions (at a pH of 7.111), key chemical precipitation parameters and values for PTN (see Table E1-2) were compared to and were shown to be bounded by the testing parameters of Test OB 29-03 in WCAP-17788, which used the sodium tetraborate (NaTB) buffer. Note that Test OB 29-03 used much more aluminum than that in the PTN3 and PTN4 containment. Test OB 29-03 did not show precipitation through the initial 8 hours using a filtration test temperature of 160°F.

   Table E1-2 – Key Parameters for Chemical Precipitation Timing

Parameter	PTN3 Values	PTN4 Values
Buffer	Sodium Tetraborate	Sodium Tetraborate
Sump pH (Long-term)	7.111 – 8.048	7.111 – 8.048
Minimum Sump Volume	36,214 ft^3	36,214 ft^3
Maximum Sump Pool Temperature	276.4°F	276.4°F
Maximum Calcium Silicate	48,308 g	650,005 g
Maximum E-Glass	111,313 g	820,284 g
Maximum Silica	0 g	0 g
Mineral Wool	0 g	0 g
Maximum Aluminum Silicate	92,534 g	0 g
Maximum Interam™	0 g	0 g
Aluminum (total combined)	39,379 ft^2	37,367 ft^2

   For maximum pH conditions (at a pH of 8.048), a precipitation map was used to compare the sump aluminum concentration estimated with the WCAP-16530 (Reference 7) methodology with the results of all NaTB test groups and the WCAP-17788 precipitation boundary equation (see Figure E1-1). Using the maximum sump aluminum concentration at 8 hours and a sump pH of 8.048, the pH + p[Al][*] was calculated to be 10.44, which crosses the precipitation boundary at a temperature just under 135°F, conservatively determined using the precipitation boundary curve at the pH of 7.111, as shown in Figure E1-1. Therefore, at the maximum containment sump pool pH of 8.048, aluminum precipitation will not occur prior to 8 hours at temperatures above 135°F.

[*] p[Al] is the negative log to the base 10 of the aluminum concentration [Al] in mol/L.

The PTN3/PTN4 containment analysis showed that the containment sump temperature stays above 211°F after switchover to sump recirculation up to 8 hours following the accident for a double ended pump suction break. Sump temperatures below 160°F by 8 hours would be indicative of a significantly less severe accident than simulated in Test OB 29-03 or modeled using the WCAP-16530 methodology. Therefore, t_{chem} is 8 hours for PTN3 and PTN4.

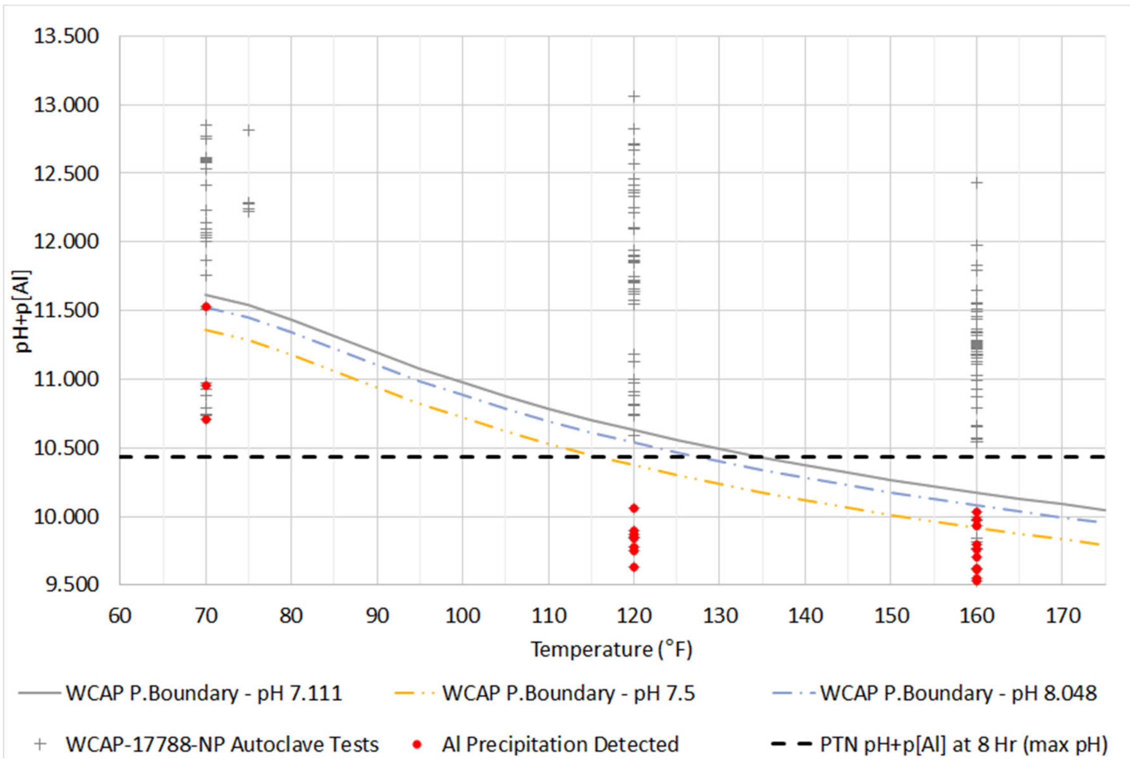

Figure E1-1 – Aluminum Precipitation Map for NaTB Buffer

2. PTN3 and PTN4 HLSO time – PTN3 and PTN4 maximum HLSO time is 6.5 hours after the event

3. Time of t_{block} used in WCAP-17788 – PTN3 and PTN4 are Westinghouse NSSS plants with a downflow barrel/baffle design. WCAP-17788 used a t_{block} of 260 minutes (Reference 6, Table 6-1) for this reactor design.

Comparison of PTN3 and PTN4 In-Vessel Fiber Load with WCAP-17788 Limit

Response for PTN3

The maximum amount of fiber that may arrive at the core inlet for PTN3 exceeds the core inlet fiber limit but is less than the total in-core fiber limit presented in WCAP-17788, Volume 1.

1. WCAP-17788 core inlet fiber limit – The core inlet fiber limit that is applicable for PTN3 (i.e., Westinghouse NSSS with a downflow barrel/baffle design and Westinghouse fuel) is in Table 6-3 of WCAP-17788 Volume 1. Since the PTN3 fuel assembly has the same pitch

as the fuel assembly pitch used in WCAP-17788 Volume 1, no adjustment to this fiber limit is necessary (Reference 6).

2. WCAP-17788 total in-core fiber limit – The total in-core fiber limit is in Section 6.4 of WCAP-17788, Volume 1 (Reference 6).
3. PTN3 in-vessel fiber load – The PTN3 core inlet fiber load is 62.55 g/FA.

The WCAP-17788 core inlet fiber limit is based on the assumption that debris accumulates uniformly at the core inlet. In reality, the debris bed at the core inlet will not be uniform due to non-uniform flow distribution, and it would take more debris than determined by WCAP-17788 to completely block the core inlet and activate the AFPs (Reference 3, Appendix B). Because of the expected non-uniform debris loading, the debris head loss at the core inlet would be lower than predicted in WCAP-17788. Lower head loss would allow additional fiber accumulation beyond the core inlet fiber limit where complete core blockage is predicted to occur in the WCAP. By definition, if the head loss at the core inlet is not high enough to activate flow through the AFPs, the core is continuing to receive sufficient flow for core cooling through the core inlet. As described in WCAP-17788, long term core cooling (LTCC) is assured as long as the total amount of fiber to the reactor coolant system (RCS) remains below the total in-core fiber limit. Therefore, it is reasonable to use the total in-core fiber limit as the acceptance criterion for HLBs. For PTN3, the maximum quantity of fiber predicted to reach the reactor core (62.55 g/FA) is lower than the WCAP-17788 total in-core fiber limit. As a result, the accumulation of fiber inside the reactor core will not challenge LTCC.

Response for PTN4

The maximum amount of fiber that may arrive at the core inlet for PTN4 is less than the core inlet fiber limit and less than the total in-core fiber limit presented in WCAP-17788.

1. WCAP-17788 core inlet fiber limit – Same as that shown above for PTN3.
2. WCAP-17888 total in-core fiber limit – Same as that shown above for PTN3.
3. PTN4 in-vessel fiber load– The PTN4 fiber load is 20.12 g/FA.

Comparison of PTN3 and PTN4 SSO Time with that Assumed in WCAP-17788

The earliest SSO time for PTN3 and PTN4 is shorter than that assumed in the WCAP-17788 analysis.

1. The SSO time assumed in the WCAP-17788 analysis is 20 minutes (Reference 6, Volume 4, Table 6-2).
2. PTN3 and PTN4 SSO time – The SSO time marks the beginning of sump recirculation and fiber accumulation inside the reactor vessel. For PTN3 and PTN4, the shortest duration for injection from the Refueling Water Storage Tank (RWST) is 14.99 minutes. Significant conservatisms were used when determining this time, as shown below.

    a. The evaluation assumed the maximum numbers of pumps are taking suction from the RWST for the entire duration of the injection phase: 2 CS pumps, 2 residual heat removal (RHR) pumps, 3 charging pumps, and 4 high head safety injection (HHSI) pumps from both units. This is not consistent with the emergency operating procedure (EOP) network, which would sequence pump flow based on the accident conditions.

b. The maximum pump flow rates were used for the entire duration without accounting for the effect of varying RCS or containment pressure.
c. No operator action delays or pump start-up times were credited.
d. The available RWST volume used in the analysis was the minimum Technical Specification volume of one RWST minus twice the level transmitter uncertainty. This accounts for the instrument uncertainties at the initial RWST water level and the low level setpoint in a conservative fashion to reduce the RWST draindown time.

Although the earliest SSO time for PTN is less than that assumed in the WCAP, the WCAP analysis is still applicable for PTN following the PWROG guidance (Reference 4) because the decay heat at the earliest SSO for PTN is lower than that used in the WCAP analysis. PTN has a rated thermal power of 2644 MWt. At the 14.99 minutes (or 900 sec) SSO time, the PTN decay heat is 67.7 MWt using the 10 CFR 50 Appendix K decay heat model (1971 ANS decay heat standard plus 20%, see Table 4.5-1 in Reference 4). This decay heat is approximately 4% lower than that used in the WCAP-17788 AFP analysis (see Table 4.5-2 in Reference 4) based on the assumed SSO time (20 minutes) and thermal power (2951 MWt) for Westinghouse downflow plants in the WCAP. Therefore, the PTN plant condition is bounded by the WCAP analysis.

It should be noted that a simulator run performed by PTN showed an SSO time of beyond 40 minutes when following the EOP steps and using more realistic inputs, such as number of operating pumps, RWST volume, pump flow rates and operator action time. The Simulator accurately models the plant's response to accident conditions and is required to do so under ANSI/ANS 3.5-2009. Periodic testing is performed to ensure the Simulator maintains fidelity and certification in accordance with the NRC guidance of ANSI/ANS 3.5-2009.

<u>Comparison of PTN3 and PTN4 Maximum Thermal Power with that Assumed in WCAP-17788</u>

PTN3 and PTN4 maximum rated thermal power is less than the analyzed power level in WCAP-17788 for a Westinghouse NSSS with a downflow barrel/baffle design.

1. PTN3 and PTN4 rated thermal power – Maximum rated thermal power for PTN3 and PTN4 is 2644 MW_t.
2. Thermal power assumed in WCAP-17788 – The WCAP analysis used a thermal power of 2951 MW_t for a Westinghouse downflow barrel/baffle plant design as shown in Table 6-2 of WCAP-17788, Volume 4 (Reference 6).

<u>Comparison of PTN3 and PTN4 Reactor AFP Resistance with that Assumed in WCAP-17788</u>

The PTN3 and PTN4 reactor AFP resistance is less than that analyzed in WCAP-17788.

1. PTN3 and PTN4 reactor AFP resistance – The PTN3 and PTN4 AFP resistance is presented in Table RAI-4.2-24 of WCAP-17788, Volume 4 as "Total <u>Adjusted</u> K/A^2" (Reference 6).
2. Maximum AFP resistance assumed in WCAP-17788 – Two different AFP resistances were used for the WCAP analysis, as presented in Table 6-2 of WCAP-17788, Volume 4 (Reference 6)

As discussed in the PWROG guidance, comparing the PTN3 and PTN4 total adjusted AFP loss coefficient with the AFP resistance used in the WCAP-17788 analysis is appropriate because the adjusted loss coefficient accounted for the difference in thermal power between PTN3/PTN4 and that assumed in the WCAP AFP analysis. The "adjusted" PTN3/PTN4 AFP loss coefficient, when applied with the thermal power assumed in the WCAP analysis for downflow plants (2,951 MWt), results in the pressure drop for PTN3/PTN4 AFPs. This thermal power is also used in the WCAP analysis, along with an assumed AFP resistance to ensure that the resulting pressure drop across the AFP in the WCAP analysis bounds all the downflow plants, including PTN3 and PTN4.

Comparison of PTN3 and PTN4 ECCS Flow Rate with that Analyzed in WCAP-17788

The PTN ECCS flow per fuel assembly is within the range of flow rates analyzed in WCAP-17788.

1. PTN3 and PTN4 ECCS flow rate – The PTN3 and PTN4 ECCS flow rate per fuel assembly is between 11.6 gpm/FA and 12.7 gpm/FA, calculated by dividing the ECCS flow rate by the number of fuel assemblies
2. ECCS flow rates analyzed in WCAP-17788 – For a Westinghouse NSSS with a down flow barrel/baffle design, the analyzed ECCS flow rate is 8 gpm/FA to 40 gpm/FA (Reference 6, Volume 4, Table 6-2).

Based on the comparisons shown above, the AFP analysis in WCAP-17788 for the Westinghouse Downflow plants is applicable for and bounds PTN3 and PTN4, following the Box 4 resolution path in the NRC review guidance for in-vessel downstream effects. As a result, accumulation of debris inside the reactor core will not challenge LTCC at PTN3 or PTN4.

3. References

1. FPL Letter L-2017-149 "Turkey Point Units 3 and 4 Docket Nos. 50-250 and 50-251 Renewed Facility Operating Licenses DPR-31 and DPR-41 Updated Final Response to NRC Generic Letter 2004-02," December 29, 2017 (ADAMS Accession No. ML17363A265).
2. NRC "Audit Plan for NextEra Methodologies for Closure of Generic Letter 2004-02," November 26, 2018 (ADAMS Accession No. ML18331A033).
3. NRC "U.S. Nuclear Regulatory Commission Staff Review Guidance for In-Vessel Downstream Effects Supporting Review of Generic Letter 2004-02 Responses," September 4, 2019 (ADAMS Accession No. ML19228A011)
4. PWROG-16073-P, "TSTF-567 Implementation Guidance, Evaluation of In-Vessel Debris Effects, Submittal Template for Final Response to Generic Letter 2004-02 and FSAR Changes," Revision 0.
5. WCAP-17788-P, Volumes 1 – 6, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," Revision 0.
6. WCAP-17788-P, Volumes 1 – 6, "Comprehensive Analysis and Test Program for GSI-191 Closure (PA-SEE-1090)," Revision 1.
7. WCAP-16530-NP-A, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191", March 2008