



DEC 04 2020

L-2020-165
GL 2004-02

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

RE: St. Lucie Nuclear Plant, Units 1 and 2
Docket Nos. 50-335 and 50-389
Renewed Facility Operating Licenses DPR-67 and NPF-16

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

References:

1. Florida Power & Light Company letter L-2017-210, Updated Final Response to NRC Generic Letter 2004-02, December 20, 2017 (ADAMS Accession No. ML17362A108)
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 St. Lucie Nuclear Units 1 and 2 (St. Lucie), 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* section of GL 2004-02, a description of completed plant modifications and process changes, and an evaluation of the sixteen issue areas identified in the NRC's 'Revised Content Guide for Generic Letter 2004-02 Supplemental Responses' (ADAMS Accession No. ML073110278), including a summary of significant margins and conservatisms utilized in supporting analyses which demonstrate with high confidence that the risk of GL 2004-02 related failures at St. Lucie 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 St. Lucie, Reference 2 identified one open issue requiring additional information. Enclosure 1 to this letter provides the requested additional information regarding void fraction considerations in the net positive suction head required (NPSHr) determination for the emergency core cooling system (ECCS).

Enclosure 2 to this letter provides the updated in-vessel downstream effects analysis for St. Lucie. In Reference 1, FPL additionally provided on behalf of St. Lucie, 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-11090)' (ADAMS Accession No. ML19228A011). 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, January 2020 (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 the effects of post-LOCA debris inside the reactor

vessel. The revised evaluation supersedes and replaces the in-vessel downstream effects evaluation provided in Reference 1.

As stated in Reference 1, changes to the St. Lucie licensing basis have been implemented which allowed FPL to complete plant modifications that have enhanced St. Lucie's capability to withstand GL 2004-02 related failures and thereby assure compliance with the long-term reactor core cooling requirements of 10 CFR 50.46(b)(5). Accordingly, the assumptions and inputs used to establish the bases for GL 2004-02 closure are consistent with the St. Lucie 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 St. Lucie Units 1 and 2 will be reviewed to determine if further changes to the licensing basis are appropriate 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 Mr. Wyatt Godes, St. Lucie Licensing Manager, at (772) 467-7435.

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

Executed on Dec. 4, 2020

Sincerely,



Daniel D. DeBoer
Site Vice President, St. Lucie Nuclear Plant
Florida Power & Light

Enclosures:

1. White Paper to Address NRC Questions about NPSH Required Adjustment for Void Fraction Issues per Regulatory Guide 1.82
2. Updated Resolution to In-Vessel Downstream Effects

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, St. Lucie Nuclear Plant, Units 1 and 2
USNRC Senior Resident Inspector, St. Lucie Nuclear Plant, Units 1 and 2
Ms. Cindy Becker, Florida Department of Health

St. Lucie Nuclear Plant, Unit 1 and Unit 2
Docket Nos. 50-335 and 50-389

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Enclosure 1

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

ENCLOSURE 1

**White Paper to Address NRC Questions about NPSH Required Adjustment
for Void Fraction Issues per Regulatory Guide 1.82**

(4 pages follow)

St. Lucie Nuclear Power Plant
Supplement to Updated Final Responses to NRC Generic Letter 2004-02

Enclosure 1

**White Paper to Address NRC Question about NPSH Required Adjustment
for Void Fraction Issues per Regulatory Guide 1.82**

Background

In the audit report (Ref. 1) for the St Lucie responses to Generic Letter 2004-02 (Ref. 2), the Nuclear Regulatory Commission (NRC) staff posed the following concern. The staff believed that St Lucie should account for 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.

The NRC staff requested NextEra/FPL to discuss whether the net positive suction head (NPSH) margin calculation accounts for void fraction at the pump suction below the two percent limit. NPSH required should be adjusted per Regulatory Guide (RG) 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident" (ADAMS Accession No. ML111330278). NextEra/FPL stated that Section 3.f.14 of its submittal shows less than two percent void fraction. NextEra/FPL did not account for void fraction at pump suction and stated that St. Lucie, Unit 1, is not committed to RG 1.82. NextEra/FPL also stated that Unit 2 uses RG 1.82, Revision 0, but is not committed to RG 1.82. In its degasification analysis, NextEra/FPL used 14.7 psia for temperatures below 212 degrees Fahrenheit (°F) 100 degrees Celsius (°C). NextEra/FPL assumed compression of bubbles due to pump submergence. The NRC staff reviewed the information provided and noted that it is acceptable for NextEra/FPL to use a method different from that in RG 1.82 to demonstrate acceptable pump performance. Additional information is required for this issue.

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

Licensing Basis

St Lucie is not licensed to the RG 1.82 requirement to de-rate the NPSH required (NPSH_r) for voids in 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. St. Lucie Unit 1 is not licensed to any version of RG 1.82 and Unit 2 is committed to RG 1.82 Revision 0.

Enclosure 1
White Paper to Address NRC Questions about NPSH Required Adjustment for Void Fraction Issues per Regulatory Guide 1.82

St Lucie is committed to General Design Criteria requiring emergency core cooling be maintained. A physical phenomenon that would appear to jeopardize emergency core cooling must be assessed and be shown to be manageable in order for the plants to meet their licensing requirements.

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. For St Lucie, the void fraction due to deaeration of fluid as it passes through the debris bed and strainer was evaluated at the mid-height of the strainer disks and was shown to be below the RG 1.82 limit of 2%. The evaluation did not credit any containment accident pressure and set the containment pressure at 14.7 psia since the peak sump temperature during recirculation is below 212°F. This subcooled pool condition and the relatively large submergence at the mid-height of the strainer helped reduce the void fraction values at the strainer.

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 will tend 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 and will collapse the bubbles. 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. 3, Section 2.4). While this process may be extended due to the different timing of the pressure recovery in the two scenarios, e.g., in the cavitation process "the elapsed times are so small (of the order of microseconds)" (Ref. 3, Section 3.4), it is still anticipated to be negligible relative to the transit time from the sump to the pump suction which is on the order of at least seconds. It is therefore concluded that bubbles formed at the strainer will collapse before reaching the pump suction and no adjustment to the pump NPSHr is required.

Conclusion

While not licensed to the RG 1.82 requirement for adjusting NPSHr due to void fraction at pump suction, St Lucie is obligated to ensure emergency core cooling be maintained and must address physical phenomena that could challenge this requirement. As summarized in this enclosure, St Lucie analyzed the void fractions at the mid-height of the strainer using a conservative method and showed that the resulting void fractions are below the acceptance limit in RG 1.82. As discussed in this enclosure, the voids will rapidly collapse as they transport to

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White Paper to Address NRC Questions about NPSH Required Adjustment for Void Fraction Issues per Regulatory Guide 1.82

the pump suction and experience higher pressures. Therefore, no adjustment to the pump NPSHr is necessary.

References

1. 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.
2. L-2017-210, St. Lucie Units 1 and 2, Docket Nos. 50-335 and 50-389, Renewed Facility Operating Licenses DPR-67 and NPF-16, Updated Final Response to NRC Generic Letter 2004-02, December 20, 2017, ADAMS Accession No. ML17362A108.
3. Brennen, Christopher Earls, Cavitation and Bubble Dynamics, Oxford University Press, 1995.

St. Lucie Nuclear Plant, Unit 1 and Unit 2
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Enclosure 2

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

ENCLOSURE 2

Updated Resolution to In-Vessel Downstream Effects

(9 pages follow)

**St. Lucie Nuclear Power Plant
Supplement to Updated Final Response to NRC Generic Letter 2004-02**

Enclosure 2

Updated Resolution to In-Vessel Downstream Effects

1. Background and Purpose

In December 2017, Florida Power & Light Company (FPL) provided an updated final response to Generic Letter (GL) 2004-02 for St. Lucie Units 1 and 2 (PSL1 and PSL2) (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 GL submittal for the applicability of fiber penetration testing but did not review the in-vessel evaluation (Reference 2, Page 8). 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 PSL1 and PSL2.

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. FPL has elected to use the Box 4 path to address in-vessel downstream effects for PSL1 and PSL2. This section evaluates the applicability of the methods and analytical results from WCAP-17788-P, Revision 1 for PSL1 and PSL2.

2.1. Sump Strainer Fiber Penetration Testing

As stated in the previous GL 2004-02 Submittal (Reference 1), sump strainer fiber penetration for PSL1 and PSL2 was evaluated based on plant-specific penetration testing, which was used to develop a model of fiber penetration through the strainer over time for each unit. No changes have been made to the testing or the fiber penetration model in this submittal.

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2.2. In-Vessel Fiber Load Analysis

The in-vessel fiber load analysis at PSL1 and PSL2 has not changed from the previous submittal (Reference 1). However, per the NRC latest review guidance, the flow diversion from the reactor core inlet through the AFPs within the reactor vessel (RV) is no longer credited in this submittal. All the fiber that reaches the RV is assumed to accumulate at the reactor core inlet. As a result, the reactor core inlet fiber load is equal to the total in-vessel fiber load shown in the previous submittal. For the base cases assuming all pumps operating, the core inlet fiber load is 25.4 g/FA and 19.7 g/FA for PSL1 and PSL2, respectively. For the core inlet failure cases with one containment spray (CS) pump assumed to fail, the core inlet fiber load is 59.3 g/FA and 39.7 g/FA for PSL1 and PSL2, respectively.

2.3. Applicability of WCAP-17788 AFP Analysis for PSL1 and PSL2

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 PSL1 and PSL2. To use this method, key in-vessel parameters of PSL1 and PSL2 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 E1-1.

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

Parameters	Values from WCAP-17788-P, Revision 1	PSL1 Values	PSL2 Values
Nuclear Steam Supply System (NSSS) Design	Various	Combustion Engineering (CE)	Combustion Engineering (CE)
Fuel Type	Various	Areva CE-14 HTP 14x14 fuel	Areva CE-16 HTP 16x16 fuel
Minimum Chemical Precipitation Time	t_{block} from WCAP-17788, Volume 1, Table 6-1 333 minutes	24 hours	24 hours
Maximum Hot Leg Switchover (HLSO) Time	N/A	6 hours	6 hours
Maximum Core Inlet Fiber Load for Hot Leg Break (HLB)	WCAP-17788, Volume 1, Table 6-3	59.3 g/FA	39.7 g/FA
Total In-Vessel Fiber Limit for HLB	WCAP-17788, Volume 1, Section 6.4	59.3 g/FA	39.7 g/FA
Minimum Sump Switchover (SSO) Time	20 minutes	26.48 minutes	21 minutes
Maximum Rated Thermal Power	3458 MWt	3020 MWt	3020 MWt
Maximum Alternate Flow Path (AFP) Resistance	WCAP-17788, Volume 4, Table 6-3	WCAP-17788, Volume 4, Table RAI-4.3-8	WCAP-17788, Volume 4, Table RAI-4.3-8
ECCS Flow per FA	3.8 – 11.4 gpm/FA	5.9 gpm/FA	6.5 gpm/FA

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

Response for PSL1

For PSL1, 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. PSL1 chemical precipitation time (t_{chem}) – Chemical precipitation is shown not to occur within 24 hours for containment sump temperatures above 82°F following the accident based on the autoclave testing in WCAP-17788, Volume 5 (Reference 6). A precipitation map was used to compare the sump aluminum concentration estimated with the WCAP-16530 (Reference 7) methodology with all NaOH (pH<10) group autoclave test results and the WCAP-17788 precipitation boundary equation (see Figure E1-1). Autoclave tests performed at a pH greater than 10 are omitted as non-representative of PSL1, which has a maximum containment sump pH of 9.66.

Using the maximum sump aluminum concentration at 24 hours and minimum containment sump pH of 8.14, the $pH + p[Al]$ was calculated to be 12.36 which crosses the precipitation boundary at a temperature 82°F as shown in Figure E1-1. Therefore, aluminum precipitation will not occur within 24 hours at containment sump temperature above 82°F. Containment sump temperatures below 82°F by 24 hours would be indicative of a significantly less severe accident than simulated using the WCAP-16530 methodology. Therefore, for PSL1, t_{chem} is 24 hours.

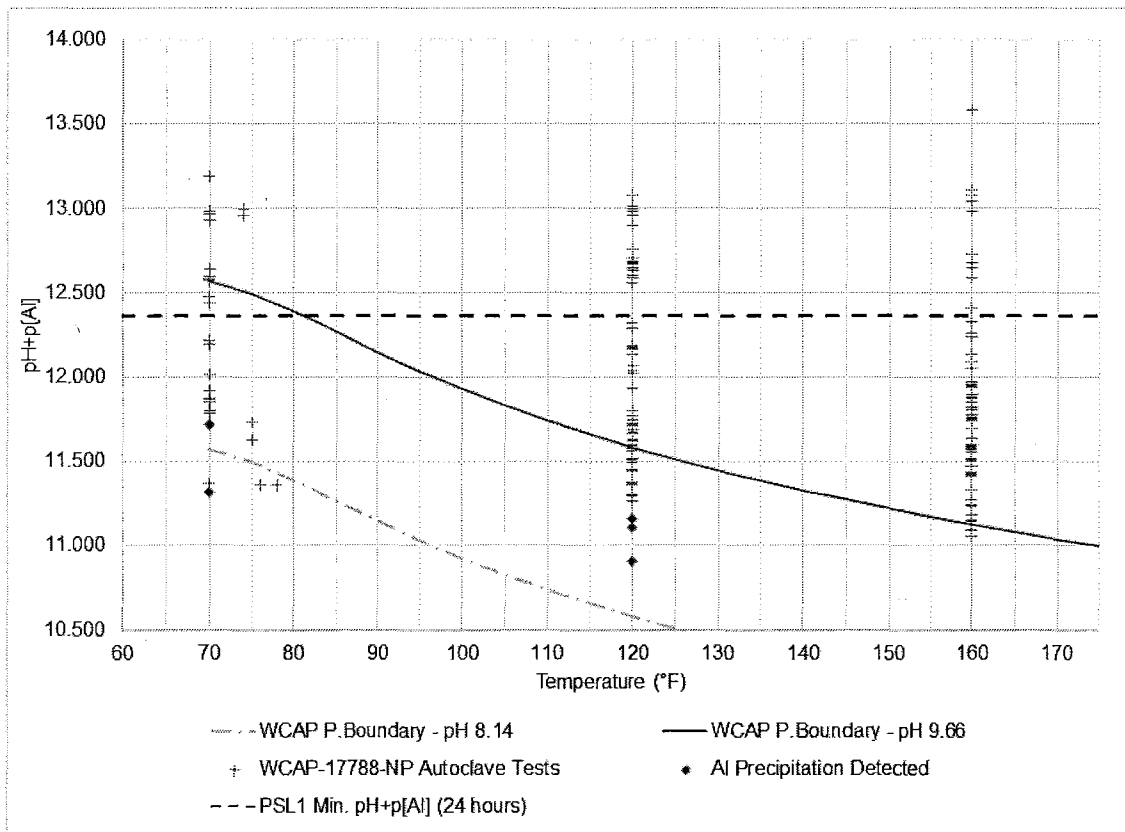


Figure E1-1. Aluminum Precipitation Map for NaOH Buffer (pH < 10)

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2. PSL Unit 1 HLSO time – PSL1 maximum HLSO time is 6 hours after the event.
3. Time of t_{block} used in WCAP-17788 – PSL1 is a Combustion Engineering (CE) NSSS plant design. WCAP-17788 used a t_{block} of 333 minutes.

Response for PSL2

For PSL2, 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. PSL2 chemical precipitation time (t_{chem}) – Aluminum precipitation is shown not to occur within 24 hours for containment sump temperatures above 73°F following the accident based on the autoclave testing in WCAP-17788, Volume 5. A precipitation map was used to compare the sump aluminum concentration estimated with the WCAP-16530 methodology with representative trisodium phosphate (TSP) group autoclave test results and the WCAP-17788 precipitation boundary equation (See. Figure E1-2). Tests 41-03 and IBOB 41-03 are omitted as non-representative of PSL2 conditions as described in WCAP-17788, Volume 5, Section G.12. The pH of these two tests (9.13) well exceed the maximum PSL2 pH of 8.102.

Using the maximum sump aluminum concentration at 24 hours and minimum containment sump pH of 7.081, the $pH + p[Al]$ was calculated to be 11.65 which crosses the precipitation boundary at a temperature 73°F as shown in Figure E1-2. Therefore, aluminum precipitation will not occur within 24 hours at containment sump temperature above 73°F. Containment sump temperatures below 73°F by 24 hours would be indicative of a significantly less severe accident than simulated using the WCAP-16530 methodology.

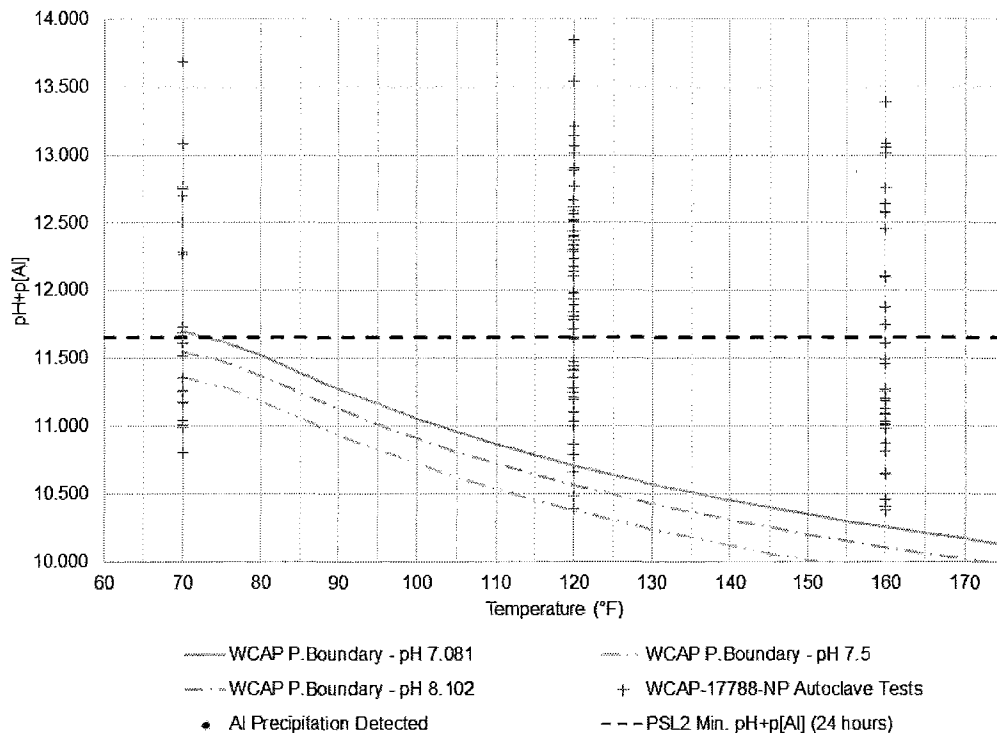


Figure E1-2: Aluminum Precipitation Map for TSP Buffer

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WCAP-17788, Volume 5 concludes that calcium precipitation within the first 24 hours does not need to be screened for conditions that it defines as representative of a US PWR. Comparison of the key calcium precipitation parameters for Groups 30 and 41 confirm that the PSL2 conditions fit this definition (see Table E1-2), and calcium precipitation will not occur within the first 24 hours following a LOCA.

Table E1-2: Key Parameters for Chemical Precipitation Timing

Parameter	PSL2 Value
Buffer	Trisodium Phosphate
Sump pH (Long-term)	7.081 – 8.102
Minimum Sump Volume	58,100 ft ³ (3,624,911 lb _m)
Maximum Sump Pool Temperature	215°F
Maximum Calcium Silicate	51.5 kg
Maximum E-Glass	1,646 kg
Maximum Silica	0 kg
Mineral Wool	0 kg
Maximum Aluminum Silicate	0 kg
Maximum Concrete	Not Determined*
Maximum Interam™	0 kg
Aluminum	63.42 ft ²
Galvanized Steel	Not Determined**

* Concrete is not a significant contributor to chemical precipitation

** Galvanized steel has no impact on calcium precipitation

In conclusion, for PSL2, t_{chem} is no less than 24 hours.

2. PSL2 HLSO time – PSL2 maximum HLSO time is 6 hours after the event.
3. Time of t_{block} used in WCAP-17788 – PSL2 is a CE NSSS plant design. WCAP-17788 used a t_{block} of 333 minutes.

Comparison of In-Vessel Fiber Load with WCAP-17788 Limits

The maximum amounts of fiber that may arrive at the core inlet for PSL1 and PSL2 exceed the core inlet fiber limit but is less than the total in-core fiber limit presented in WCAP-17788.

1. WCAP-17788 core inlet fiber limit for PSL1 – The core inlet fiber limit that is applicable for PSL1 (i.e., CE NSSS and Areva CE14HTP 14x14 fuel) is in Table 6-3 of WCAP-17788 Volume 1 (Reference 6). As shown in Section 4.4 of Reference 4, the fiber limit from WCAP-17788 Volume 1 was reduced by applying a scaling factor for the CE14HTP fuel in Table 4.4-2 to account for the difference in fuel assembly pitch between PSL1 and that analyzed in the WCAP.
2. WCAP-17788 core inlet fiber limit for PSL 2 – The core inlet fiber limit that is applicable for PSL2 (i.e., CE NSSS and Areva CE16HTP 16x16 fuel) is in Table 6-3 of WCAP-17788 Volume 1 (Reference 6). As shown in Section 4.4 of Reference 4, the fiber limit from WCAP-17788 Volume 1 was reduced by applying a scaling factor for the CE16HTP fuel

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in Table 4.4-2 to account for the difference in fuel assembly pitch between PSL2 and that analyzed in the WCAP.

3. 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).
4. In-vessel fiber load – The maximum core-inlet and total in-core fiber load is 59.3 g/FA for PSL1 and 39.7 g/FA for PSL2.

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, 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 LTCC through the core inlet. As described in WCAP-17788, LTCC is assured as long as the total amount of fiber to the 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. The maximum quantity of fiber predicted to reach the reactor core (59.3 g/FA for PSL1 and 39.7 g/FA for PSL2) 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.

Comparison of SSO Time with that Assumed in WCAP-17788

The earliest SSO times for PSL1 and PSL2 are greater than that assumed in the WCAP-17788 analysis.

1. PSL1 and PSL2 SSO time – The SSO time marks the beginning of sump recirculation and fiber accumulation inside the reactor vessel. The shortest duration for injection from the refueling water tank (RWT) is 26.48 minutes and 21 minutes for PSL1 and PSL2, respectively.
2. The SSO time assumed in the WCAP-17788 analysis is 20 minutes (Reference 6, Volume 4, Table 6-3).

Comparison of Maximum Thermal Power with that Assumed in WCAP-17788

The maximum rated thermal power for PSL1 and PSL2 is less than the analyzed power level in WCAP-17788 for a CE NSSS design.

1. PSL1 and PSL2 rated thermal power – The PSL1 and PSL2 maximum rated thermal power is 3020 MWt.
2. Thermal power assumed in WCAP-17788 – The WCAP analysis used a thermal power of 3458 MWt for a CE plant design as shown in Table 6-3 of WCAP-17788, Volume 4 (Reference 6).

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Comparison of Reactor AFP Resistance with that Assumed in WCAP-17788

The PSL1 and PSL2 reactor AFP resistance is less than that analyzed in WCAP-17788.

1. PSL1 and PSL2 reactor AFP resistance – The PSL1 and PSL2 AFP resistance is presented in Table RAI-4.3-8 of WCAP-17788, Volume 4 (Reference 6) as “Total Unadjusted K/A^2 (ft⁴)”.
2. AFP resistance assumed in WCAP-17788 – The AFP resistance used in the WCAP-17788 analysis that is applicable for PSL1 and PSL2 (i.e., CE Plant) is provided in Table 6-3 of WCAP-17788, Volume 4 (Reference 6) as “Barrel/Baffle Total K/A^2 ” for the Max Resistance Cases.

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

Response for PSL1

The PSL1 ECCS flow per fuel assembly is within the range of flow rates analyzed in WCAP-17788.

1. PSL1 ECCS flow rate – The recirculation flow configuration used by the limiting in-vessel analysis includes two high pressure safety injection (HPSI) pumps supplying flow to the cold legs at a flowrate of 640 gpm per pump. One low pressure safety injection (LPSI) pump is manually restarted to provide hot leg injection several hours after the start of recirculation. However, the LPSI pump flow rate is not included when calculating the ECCS flow rate per fuel assembly. Therefore, the total flow rate delivered to the reactor vessel is 1280 gpm from two HPSI pumps. As a result, the PSL1 ECCS flow rate per fuel assembly utilized in the limiting in-vessel analysis is 5.9 gpm/FA.
2. ECCS flow rates analyzed in WCAP-17788 – For a CE NSSS design, the analyzed ECCS flow rate is 3.8 gpm/FA to 11.4 gpm/FA (Reference 6, Volume 4, Table 6-3).

Response for PSL2

The PSL2 ECCS flow per fuel assembly is within the range of flow rates analyzed in WCAP-17788.

1. PSL2 ECCS flow rate – The recirculation flow configuration used by the limiting in-vessel analysis includes two HPSI pumps supplying flow to the cold legs at a flow rate of 700 gpm per pump. Therefore, the total flow rate delivered to the reactor vessel is 1400 gpm. As a result, The PSL2 ECCS flow rate per fuel assembly utilized in the limiting in-vessel analyses is 6.5 gpm/FA.
2. ECCS flow rates analyzed in WCAP-17788 – For a CE NSSS design, the analyzed ECCS flow rate is 3.8 gpm/FA to 11.4 gpm/FA (Reference 6, Volume 4, Table 6-3).

Based on the comparisons shown above, in-vessel downstream effects due to accumulation of debris inside the reactor core will not challenge LTCC at PSL.

3. References

1. FPL Letter L-2017-210 “St. Lucie Unit 1 and 2 Nuclear Plant Docket Nos. 50-335 and 50-389 Renewed Facility Operating Licenses DPR-67 and NPF-16 Updated Final Response

Enclosure 2
Updated Resolution to In-Vessel Downstream Effects

- to NRC Generic Letter 2004-02," December 20, 2017 (ADAMS Accession No. ML17362A108).
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)
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 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