FPL Energy
Seabrook Station

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

Seabrook Station
Safety Evaluation Report Discrepancies

FPL Energy Seabrook, LLC (FPL Energy Seabrook) provides in the attachment a markup of the Safety Evaluation Report (SER) for Amendment 101 to the Seabrook Station Operating License identifying apparent discrepancies. The discrepancies were previously discussed with Seabrook's NRC Project Manager. FPL Energy Seabrook understands that the NRC will review the discrepancies and reissue corrected SER pages as necessary.

Should you require further information regarding this matter, please contact Mr. James Peschel, Regulatory Programs Manager, at (603) 773-7194.

Very truly yours,

FPL Energy Seabrook, LLC

Mark E. Warner
Site Vice President

cc: S. J. Collins, NRC Region I Administrator
V. Nerses, NRC Project Manager, Project Directorate I-2
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SBK-L-05101

ATTACHMENT 1

- Regulatory Guide (RG) 1.105, Revision 3, "Setpoint for Safety-Related Instrumentation," is used to evaluate the conformance with 10 CFR 50.36.

- The NRC's acceptance criteria related to the quality of design of protection and control systems are based on 10 CFR 50.55a(a)(1), 10 CFR 50.55a(h), and General Design Criteria (GDCs) 1, 4, 13, 19, 20, 21, 22, 23, and 24. Specific review criteria are contained in SRP Sections 7.0, 7.2, 7.3, 7.4, 7.7, and 7.8.

3.1.2 Technical Evaluation

3.1.2.1 Suitability of Existing Instruments

The SS plant protection systems are the reactor protection system (RPS) and the ESFAS. The RPS is designed to trip the reactor by de-energizing the control element drive mechanisms whenever any monitored condition reaches a TSP. For each measured variable, the RPS uses a 2-out-of-4 channel logic arrangement, with each channel electrically and physically separated to ensure no loss of functionality with a single failure. The RPS is designed to automatically keeps the reactor operating within a safe region by shutting down the reactor whenever the limits of the region are approached. The reactor trip system keeps surveillance on process variables and whenever a direct process or calculated variable exceeds a setpoint, the reactor will be shut down to protect against either gross damage to fuel cladding or loss of system integrity which could lead to release of radioactive fission products into the containment. The ESFAS is designed to initiate safety features whenever any monitored condition reaches a TSP. The ESFAS also uses a 2-out-of-4 logic arrangement, with each channel electrically and physically separated to ensure no loss of functionality with a single failure. The ESFAS is designed to mitigate the consequences of certain design basis accidents (DBAs), particularly by protecting the integrity of the containment building.

The RPS is discussed in Section 7.2 of the SS updated final safety analyses report (UFSAR). The ESFAS is discussed in UFSAR Section 7.3. Systems required for safe shutdown are described in UFSAR Section 7.4. Instrumentation required to monitor, control, and provide interlocks for these systems is described in UFSAR Sections 7.2, 7.3, 7.6, and 7.7. The anticipated transient without scram (ATWS) mitigation system is described in UFSAR Section 7.6.

The change resulting from the SS SPU is that the full power T_{avg} window is ~~increases~~ from 571.0 °F to 589.1 °F. To address this change, the pressurizer level control program was revised. There were no other nuclear steam supply system (NSSS) control system setpoint changes required for the SPU. Therefore, the impact of this change on the NSSS control systems for the SPU was evaluated. The RPS and ESFAS setpoints for low-low steam generator (SG) level trip and high-high SG level ESFAS were revised. The revised setpoints were evaluated to determine the impact on the plant operating margin.

The licensee stated that the methodology used to calculate the values for these constants is consistent with the past practice and NRC-approved methods. In Section 3.2.2 of this SE, the NRC staff found this methodology acceptable because of previous NRC approval.

A) The Trip Setpoint is changed from ≥14.0% to ≥20.0% of narrow range instrument span.
B) The AV is changed from ≥12.6% to ≥19.5% of narrow range instrument span.
C) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

The basic function of the reactor protection circuits associated with low-low SG water level is to preserve the SG heat sink for removal of long-term residual heat. In the event of a complete loss of feedwater (FW), the reactor would be tripped on low-low SG water level and emergency feedwater pumps (FWPs) would provide FW to maintain residual heat removal (RHR) after trip. This reactor trip acts before the SGs are dry. Therefore, a low-low SG water level reactor trip circuit is provided for each SG to ensure that sufficient initial thermal capacity is available in the SG at the start of the transient. With the added power after the power uprate, the full power T_{avg} window is from 571.0 °F to 589.1 °F.

Enclosure 2 of the October 12, 2004 RAI response contains detailed process measurement errors and instrument uncertainties, as well as the calculated TSP and AV. The staff finds that the TSP calculation meets the requirements of 10 CFR 50.36 and RG 1.105 and, therefore, is acceptable.

To ensure acceptable plant response and adequate plant operating margins at the SPU power conditions, the American Nuclear Society (ANS) Condition I transients were analyzed using the LOFTRAN computer code and the following conditions:

- 50% load rejection
- 5%/minute ramp load changes
- 10% step load changes
- Turbine trip without reactor trip from P-9 setpoint

The major analysis assumptions are as follows.

- Analyses for both low (571.0 °F) and high (589.1 °F) RCS full power T_{avg} and 0% and 10% average SG tube plugging conditions
- Credit for the NSSS control systems (rod control, steam dump control, and pressurizer pressure control) in automatic mode of control
- No credit for pressurizer and SG safety valves
- Best estimate fuel reactivity feedbacks [moderator temperature coefficient (MTC), Doppler power defect and control rod worth] at beginning of life (BOL) core conditions, which are limiting for the margin to trip assessment
- No credit for operator action

The licensee stated that analysis showed that the low-low SG level reactor TSP is adequate. The staff's evaluation of these transients and assumptions are included in Section 3.2 of this SE. The staff finds that the TSP calculation meets the requirements of 10 CFR 50.36 and RG 1.105 and, therefore, is acceptable.

3.1.2.3.2 ESFAS Instrumentation TSPs and AVs

In addition to the requirements for a reactor trip for anticipated abnormal transients, the ESFAS is provided with instrumentation and controls to sense accident situations and initiate the operation of necessary ESFs. The occurrence of a limiting fault, such as a LOCA or a steamline break, requires a reactor trip plus actuation of one or more of the ESFs to prevent or mitigate damage to the core and RCS components, and ensure containment integrity.

The specific functions that rely on the ESFAS for initiation and which are being modified as a result of this power uprate are the turbine trip, the start of motor-driven and turbine-driven emergency FWPs, and main FW line isolation (as required to prevent or mitigate the effect of excessive cooldown). The following TS changes were requested by the licensee:

(1) TS Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints, Functional Unit 5b, Turbine Trip, Steam Generator Water Level High-High (P14) is revised as follows:

A) The Trip Setpoint is changed from ≤86.0% to ≤90.8% of narrow range instrument span.
B) The Allowable Value is changed from ≤87.7% to ≤91.3% of narrow range instrument span.
C) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

(2) TS Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints, Functional Unit 6a, Feedwater Isolation, Steam Generator Water Level Hi-Hi (P14) is revised as follows:

A) The Trip Setpoint is changed from ≤86.0% to ≤90.8% of narrow range instrument span.
B) The Allowable Value is changed from ≤87.7% to ≤91.3% of narrow range instrument span.
C) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

(3) TS Table 3.3-4, Engineered Safety Features Actuation System Instrumentation Trip Setpoints, Functional Unit 7c, Emergency Feedwater, Steam Generator Water Level Low-Low, Start Motor-Driven Pump and Start Turbine-Driven Pump is revised as follows:

A) The Trip Setpoint is changed from ≥14.0% to ≥20.0% of narrow range instrument span.
B) The Allowable Value is changed from ≥12.6% to ≥19.5% of narrow range instrument span.
C) The Total Allowance (TA), Z, and Sensor Error (S) are changed from values to N.A.

equivalent to or a justified extrapolation from proven designs; (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs; and (4) is not susceptible to thermal-hydraulic instability. The NRC's acceptance criteria are based on (1) GDC-10, insofar as it requires that the reactor core be designed with appropriate margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs; and (2) GDC-12, insofar as it requires that the reactor core and associated coolant, control, and protection systems be designed to assure that power oscillations, which can result in conditions exceeding SAFDLs, are not possible or can reliably and readily be detected and suppressed. Specific review criteria are contained in SRP Section 4.4.

The staff reviewed the thermal and hydraulics safety analysis and evaluations performed to support operation of the SS core consisting of 17x17 robust fuel assemblies with intermediate flow mixer grids [Reference 19] at the analyzed SPU core power level of 3659 MWt.

The thermal and hydraulic, departure from nucleate boiling (DNB), safety analyses performed for the SPU, were based upon the Revised Thermal Design Procedure (RTDP) methodology [Reference 17] and the WRB-2M DNB correlation [Reference 18]. The RCS methodology applies statistical calculations to obtain DNB sensitivity factors for uncertainties (random portions only) in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters. The Standard Thermal Design Procedure (STDP) methodology and the WRB-2 [Reference 19] or W-3 DNB correlation were used when RCS and WRB-2M were not applicable. The analysis results indicate that the 95/95 DNB design basis is met for operation and AOOs under SPU conditions.

The W version of the VIPRE-01 Code was used to perform ~~NDBR~~ DNBR calculations with the WRB-2, WRB-2M and the W-3 DNB correlations. The application of VIPRE for these SPU analyses was within the conditions specified in the staff's Safety Evaluation Report (SER) contained in WCAP-14565-P-A [Reference 20].

The NRC staff has reviewed the licensee's analyses related to the effects of the proposed SPU on the thermal and hydraulic design of the core and the RCS. The NRC staff concludes that the licensee has adequately accounted for the effects of the proposed SPU on the thermal and hydraulic design and demonstrated that the design: (1) has been accomplished using acceptable analytical methods; (2) is equivalent to proven designs; (3) provides acceptable margins of safety from conditions that would lead to fuel damage during normal reactor operation and AOOs; and (4) is not susceptible to thermal-hydraulic instability. Based on this, the NRC staff concludes that the thermal and hydraulic design will continue to meet the requirements of GDCs 10 and 12 following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to thermal and hydraulic design.

3.2.10 Functional Design of Control Rod Drive System (CRDS)

The NRC staff's review covered the functional performance of the CRDS to confirm that the system can effect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents. The review also covered the CRDS

3.2.13 Transient and Accident Analyses

The licensee re-analyzed the UFSAR Chapter 15 LOCA and non-LOCA transients and accidents in support of the SS SPU. These analyses were performed at a rated core power of 3659 ~~3587~~ MWt using plant parameter values for those operating conditions. The initial condition uncertainties were recalculated at power uprate conditions for use in the SS SPU program. These uncertainty calculations were performed for the uprate operating conditions based on the plant-specific instrumentation and plant calibration and calorimetric procedures. The staff reviewed the licensee's transient and accident analyses at the ~~3587~~ 3659 MWt SPU conditions to verify the acceptance criteria are still met under these conditions. The staff's review of the LOCA and non-LOCA transients and accidents is discussed in the following sections.

3.2.13.1 LOCA Analyses

The licensee described the SS LBLOCA and small-break LOCA (SBLOCA) analyses performed at the uprated power for 17x17 W robust fuel (with ZIRLO cladding) assemblies [References 1, 2, 3, 4, and 5]. The LBLOCA analyses were performed with the W LBLOCA methodology described in WCAP-12945-P-A [Reference 22]. The SBLOCA analyses' results were recalculated using the W SBLOCA methodology described in the W NOTRUMP (COSI) SBLOCA methodology. The NRC staff reviewed these analyses to assure that the licensee met the requirements of 10 CFR 50.46.

The licensee provided the LOCA plant-specific analyses results for the W fuel. The following table provides the licensee's LBLOCA analysis results.

*TABLE 1 - LOCA ANALYSIS RESULTS

Limiting Break Type/Size/location	LBLOCA/ Pump Discharge	SBLOCA/ 4-Inch Pump Discharge
Fuel Type	W 17 x 17 ZIRLO fuel	W 17 x 17 ZIRLO fuel
Peak Cladding Temperature (PCT)	1784 °F	1373 °F
Maximum Local Oxidation	3.53% *	0.20% *
Maximum Total Core-wide Hydrogen Generation (All Fuel)	(0.3%)*	(<<1.0%)*

*These LOCA local oxidation and core-wide hydrogen generator values [Reference 5] are bounding LOCA values for the fuel. The licensee states that operational controls are such that the total oxidation (including LOCA and pre-LOCA) will always be below 16%. The values for core-wide hydrogen generation do not include a pre-LOCA amount. This is reasonable because normal operational monitoring and procedures maintain operational (pre-LOCA) core-wide hydrogen at a very low level.

The calculated values given in the table above are less than the limits specified in 10 CFR 50.46(b) (1)-(3), which requires the PCT to be less than 2200 °F, the maximum

Therefore, while the NRC staff cannot endorse the licensee's evaluation as a valid mechanistic model of the phenomena, the staff believes, on an interim basis, that there is sufficient basis to approve the license amendment with respect to LTC and potential H_3BO_3 precipitation concerns.

This NRC staff conditional acceptance will remain effective until generic concerns associated with LTC are rectified, at which time the licensee will have to establish that it is in compliance with the resolution of the generic concerns.

3.2.13.2 Non-LOCA Transients and Accidents

The licensee re-analyzed SS's UFSAR Chapter 15 non-LOCA events at the SPU conditions of 3659 ~~3587~~ MWt. The licensee used the NRC previously-approved computer codes and methodologies for each of the non-LOCA transient analyses at SS. The licensee used the RETRAN computer code in the SS non-LOCA SPU safety analyses, simulating a W 4-loop plant design, applicable to SS, as described and presented in WCAP-14882-P-A [Reference 32]. The licensee used RETRAN in combination with VIPRE-01 for reactor core subchannel thermal-hydraulic calculations, a neutronic code such as ANC, and a fuel performance code such as FACTRAN in core design, as described in [References 20, 15, 16, and 30], respectively. The licensee used TWINKLE [Reference 29], a multidimensional neutron computer code, in conjunction with FACTRAN [Reference 30], a code for thermal transients in a UO_2 fuel rod, to perform the rod cluster control assembly (RCCA) ejection and uncontrolled RCCA withdrawal from a subcritical or low power startup condition analyses. The licensee met the conditions and restrictions set on the specific codes. Where applicable, the licensee used the previously-approved RTDP methodology discussed in WCAP-11397-P-A [Reference 17] in performing the non-LOCA safety analyses. The staff finds the codes and methodologies used by the licensee to perform the safety analyses under SPU conditions acceptable since the licensee satisfies the conditions and restrictions set on the specific codes for application at SS. Table 6.3.1-1 and Table 6.3.1-2 in the licensee's application [Reference 1] provide non-LOCA selected analyses results and non-LOCA plant initial conditions, respectively, for the SS SPU conditions [Reference 1].

3.2.13.2.1 Excessive Heat Removal Due to FW System Malfunctions and Increase in FW Flow or Decrease in FW Temperature

A change in SG FW conditions that results in an increase in FW flow or a decrease in FW temperature could result in excessive heat removal from the RCS. Such changes in FW flow or FW temperature are a result of a failure of a FW control valve or FW bypass valve, failure in the FW control system, or operator error. Excessive heat removal causes a decrease in moderator temperature which increases core reactivity and can lead to a power level increase and a decrease in shutdown margin. Any unplanned power level increase may result in fuel damage or excessive reactor system pressure. Reactor protection and safety systems are actuated to mitigate the transient. The acceptance criteria are based on critical heat flux not being exceeded, pressure in the RCS and main steam system (MSS) being maintained below 110% of the design pressures, and the peak linear heat generation rate not exceeding a value that would cause fuel centerline melt. Specific review criteria are contained in SRP Section 15.1.1-4.

- Evaluation of irradiated material properties.
- Void swelling assessment including available data and effects on RVIs.
- Development of a long-term RVIs aging management strategy.

evaluate the results of the EPRI MRP

The licensee's commitment to ~~participate in the industry's research~~ program of degradation of PWR RV internal components and to develop an inspection program for the RV internals that is based on the recommendations of the industry initiatives as appropriate are consistent with Table Matrix-1 of NRC RS-001, Revision 0 and are, therefore, acceptable. Based on this assessment, the staff concludes that FPLE has established an acceptable course of action for managing age-related degradation in the SS RV internals under the SPU conditions for the unit.

The NRC staff finds that reasonable controls for the implementation and for subsequent evaluation of proposed changes pertaining to the above commitment are provided by the licensee's administrative processes, including its commitment management program. Should the licensee choose to incorporate a regulatory commitment into the UFSAR or other document with established regulatory controls, the associated regulations would define the appropriate change-control and reporting requirements. The staff has determined that the commitment does not warrant the creation of regulatory requirements which would require prior NRC approval of subsequent changes. The NRC staff has agreed that NEI 99-04, Revision 0, "Guidelines for Managing NRC Commitment Changes," provides reasonable guidance for the control of regulatory commitments made to the NRC staff. (See Regulatory Issue Summary 2000-17, "Managing Regulatory Commitments Made by Power Reactor Licensees to the NRC Staff," dated September 21, 2000.) The commitment should be controlled in accordance with the industry guidance or comparable criteria employed by a specific licensee. The staff may choose to verify the implementation and maintenance of this commitment in a future inspection or audit.

3.6.5.3 Summary

The NRC staff has reviewed the licensee's evaluation of the effects of the proposed SPU on the susceptibility of reactor internal and core support materials to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of reactor internal and core support materials. The NRC staff further concludes that the licensee has demonstrated that the reactor internal and core support materials will continue to be acceptable and will continue to meet the requirements of GDC-1 and 10 CFR 50.55a following implementation of the proposed SPU. Therefore, the NRC staff finds the proposed SPU acceptable with respect to reactor internal and core support materials.

3.6.5.4 Conclusion

The staff has reviewed the licensee's proposed LAR to increase the rated core thermal power for SS by 5.2% and has evaluated the impact that the SPU conditions will have on the structural integrity assessments for the RV and RV internals. The staff has determined that the changes identified in the proposed LAR will not significantly impact the remaining safety margins required for following RCS-related structural integrity assessments: (1) RV surveillance program for SS; (2) USE assessment for the RV; (3) PT limits for the SS RV; (4) PTS assessment for the SS RV beltline materials; and (5) structural integrity assessment of the SS

The staff finds that the licensee's response is satisfactory because no new procedures will be required, and the licensee has adequately identified and committed to implementing the necessary changes to EOPs/AOPs that will be affected by the uprate. Furthermore, the response indicated that operators will be trained on the changes before they are implemented.

3.7.1.2 Changes to Operator Actions Sensitive to the SPU

The licensee stated that there will be no new operator actions required by the SPU. The licensee described one action in the EOPs that will be affected by the SPU, which is a reduction in maximum hot-leg recirculation switchover time from 9 hours to ~~7~~ hours. A minimum time of ~~5.8~~ hours for this action has been established. There will be no changes to operator actions in the AOPs. The licensee stated [Reference 28]: 6

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> Assurance is provided that hot leg injection emergency operating procedure actions will occur consistent with the calculated times because of two reasons: (1) there will be a margin of 1 hour (5 hours in the emergency operating procedures minus 4.0 hours calculated) for minimum time to hot leg injection, and a margin of 1.46 hours (7.46 hours calculated minus 6 hours in the emergency operating procedures) for maximum time to hot leg injection, and (2) based on discussions with Seabrook Station operations staff, the actions described in the hot leg injection emergency operating procedure can be performed in about 10 minutes.

In Section 3.2.13.1.4, the NRC staff documents the results of their review of the reduced time for switchover (with the times provided by the licensee in letter dated December 28, 2004, [Reference 28]) and considered it conditionally acceptable.

The staff finds that the licensee's response is satisfactory because the licensee has adequately described the effect of the power uprate on operator actions and the time limit estimates involved with the switch-over action are conservative relative to the time needed to successfully complete this action.

3.7.2.3 Changes to Control Room Controls, Displays and Alarms

In its March 17, 2004, letter, the licensee stated that:

> The SPU will have a limited impact on the operator interfaces for control room displays, controls, and alarms. The plant modification process will implement the required changes through normal quality program controls and the design control program. Control room indications have "band markings," that are controlled by the Operation's Department administrative control procedures and plant design control procedures. The band markings, which are mimicked in the Simulator, indicate normal operating limits, abnormal operating limits or Engineered Safety Feature actuation setpoints. The plant change packages for the uprate that identify control room changes will be processed in accordance with approved plant procedures. Training on TS changes affecting control room indications and alarms will be completed prior to implementation of the uprate.

The staff finds that the licensee's response is satisfactory because the licensee has adequately identified the changes that will occur to alarms, displays, and controls as a result of the power

3.8.6.3 Summary

Based on a review of the information that was provided, the staff finds that the licensee has adequately considered and addressed the effects of the proposed power uprate on the capability of the SFP cooling system to perform its design-basis cooling function, and on the capability to provide makeup following a loss of SFP cooling. The staff concludes that the SFP cooling system will maintain its ability to function and that the capability to provide SFP makeup will continue to be assured following implementation of the proposed power uprate and, consequently, SS will continue to satisfy GDC-61, and reactor safety will not be degraded. Therefore, the NRC staff finds that the proposed SPU is acceptable with respect to SFP cooling.

3.8.7 Main Turbine

3.8.7.1 Regulatory Evaluation

main turbine

The NRC acceptance criteria for the ~~condensate and FW systems~~ are based on GDC-4. GDC-4 is related to the safety-related portions of the system being capable of withstanding the environmental conditions associated to normal operations, maintenance, testing, and postulated accidents.

3.8.7.2 Technical Evaluation

As discussed in Reference [1], Attachment 1 (Section 8.3.1), and supplemented by the licensee's response to Observation No. 9 in Enclosure 1 of the submittal dated May 26, 2004, the licensee (in conjunction with General Electric Energy Services) evaluated the impact of increased throttle flow on the SS turbine-generator and related SSCs at the proposed SPU conditions. The evaluation was based on heat balance input parameters, such as SG outlet conditions, FW heater performance, and condenser pressure, as well as transient loading considerations. The licensee concluded that the existing HP turbine steam path would need to be modified to accommodate the increased steam flow that is necessary for SPU operation. The modified steam path will be designed to achieve a full throttle capability of 3719.6 MWt which will provide additional operating flexibility and design margin to ensure high standards of reliability and output at the uprated operating conditions. The licensee also plans to replace the HP turbine stage 1 nozzle plates and HP diaphragms and buckets with new, high efficiency components. Because the planned modifications will not result in any significant change in the inertia of the turbine rotor assembly, the tendency of the turbine speed to exceed design specifications following a load rejection event should not be affected by the proposed power uprate. Consequently, the current turbine overspeed trip settings will continue to be adequate for SPU operation and the vulnerability of SSCs to turbine missiles will not be affected.

3.8.7.3 Summary

Based on a review of the information that was provided, the staff finds that the licensee has adequately considered and addressed the effects of the proposed power uprate on the main turbine. The staff concludes that the main turbine will continue to be protected from overspeed conditions and the vulnerability of SSCs to turbine missiles will not be affected by the proposed power uprate and, consequently, SS will continue to satisfy GDC-4, and reactor safety will not