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Docket No. 50-443
SBK-L-21135

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

Seabrook Station
Submittal of Changes to the Seabrook Station Technical Specification Bases

NextEra Energy Seabrook, LLC submits the enclosed changes to the Seabrook Station Technical Specification Bases. The changes were made in accordance with Technical Specification 6.7.6.j., "Technical Specification (TS) Bases Control Program." Please update the Technical Specification Bases as follows:

REMOVE	INSERT
B 3/4 2-2	B 3/4 2-2 B 3/4 2-2a
B 3/4 2-3	B 3/4 2-3
B 3/4 3-3	B 3/4 3-3 B 3/4 3-3a B 3/4 3-3b
B 3/4 8-20	B 3/4 8-20

Should you have any questions concerning this submittal, please contact me at (603) 773-7631.

Sincerely,

NextEra Energy Seabrook, LLC

Matthew Levander
Licensing Manager

cc: D. Lew, NRC Region I Administrator
J. Poole, NRC Project Manager, Project Directorate I-2
C. Newport, NRC Senior Resident Inspector

Enclosure to SBK-L-21135

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA, the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance in accordance with the Surveillance Frequency Control Program is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

The heat flux hot channel factor, $F_Q(Z)$, is a measure of the peak fuel pellet power within the reactor core. $F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. $F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution. Violating the LCO limits for $F_Q(Z)$ could result in unacceptable consequences if a design basis event were to occur while $F_Q(Z)$ exceeds its specified limit.

Using the incore detector system, $F_Q(Z)$ is measured during the initial power ascension following a refueling and then periodically in accordance with the Surveillance Frequency Control Program (SFCP). However, the measurements are generally taken with the core at or near equilibrium conditions and do not compensate for variations that may be present during non-equilibrium situations such as load following or power ascension.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

To assure $F_Q(Z)$ does not exceed the value assumed in the safety analyses, an allowance is made for both measurement uncertainties and manufacturing tolerance before $F_Q(Z)$ is adjusted to account for the calculated worst-case transient conditions. This non-equilibrium component of $F_Q(Z)$, referred to as $F_Q^W(Z)$, is thus adjusted by W(Z), a cycle-specific, elevation dependent function that accounts for the effects of normal operational transients based on the expected power maneuvers over the full range of core burnup conditions. $F_Q(Z)$ is also adjusted by R_j, a cycle-specific, burnup dependent penalty factor that accounts for potential decreases in the $F_Q^W(Z)$ margin between surveillances. When the $F_Q^W(Z)$ margin is predicted to decrease, the COLR will specify an appropriate R_j factor based on the predicted trend. When the margin is predicted to increase, R_j unity (1.0) thereby imposing no penalty on the $F_Q^W(Z)$ measurement. The COLR applies the varying W(Z) and Rj functions to establish a selection of pre-analyzed, successively more restrictive Relaxed Axial Offset Control (RAOC) operating spaces. A RAOC operating space is the combination of AFD and Control Bank Insertion Limits, based on the methodology described in WCAP-10217-A, Revision 1A. A more restrictive RAOC operating space limits the range of allowable non-equilibrium power shapes through a smaller AFD band and/or shallower control rod insertion limits. The smaller operating space results in more transient F_Q margin.

If $F_Q^W(Z)$ exceeds its limit, a more restrictive RAOC operating space may be implemented within 4 hours in accordance with Required Action 1. Required Action 1 also requires SR 4.2.2.2.a and SR 4.2.2.2.b performance within 72 hours if control rod motion was necessary to implement the new operating space. The SR performance is appropriate since the control rod motion could potentially change the measured power distribution, which could impact the transient F_Q margin.

If none of the COLR specified operating spaces provides the required F_Q margin, Required Action 2 requires a reduction in THERMAL POWER and the RPS power range neutron flux-high and overpower ΔT trip setpoints by a percentage amount equal to the percentage that $F_Q^W(Z)$ exceeds its limit. The magnitude of the THERMAL POWER reduction is specified in the COLR. SR 4.2.2.2.a and SR 4.2.2.2.b performance is required prior to increasing THERMAL POWER above the limit imposed by Required Action 2.1, thereby assuring adequate $F_Q^W(Z)$ margin beforehand.

SR 4.2.2.2.d performance is required upon achieving equilibrium conditions after exceeding the core power at which $F_Q^W(Z)$ was last determined by ≥20% RTP, or in accordance with the SFCP, whichever occurs first. Following a refueling, SR 4.2.2.2.d need not be completed until a power level for extended operation has been achieved, thus assuring the core is sufficiently stable at the intended operating conditions required to perform the surveillance.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. There is additional margin available to offset any other DNBR penalties and for plant design flexibility.

When RCS $F^N_{\Delta H}$ is measured, no additional allowances are necessary prior to comparison with the established limit. Appropriate $F^N_{\Delta H}$ measurement uncertainties are already incorporated into the limits $F^N_{\Delta H}$ established in the CORE OPERATING LIMITS REPORT for each measurement system, and a bounding $F^N_{\Delta H}$ measurement uncertainty has been applied in determination of the design DNBR value. The appropriate $F^N_{\Delta H}$ measurement uncertainty is 4%.

3/4.2.4 QUADRANT POWER TILT RATIO

The purpose of this specification is to detect gross changes in core power distribution between monthly Incore Detector System surveillances. During normal operation the QUADRANT POWER TILT RATIO is set equal to 1.0 once acceptability of core peaking factors has been established by review of incore surveillances. The limit of 1.02 is established as an indication that the power distribution has changed enough to warrant further investigation.

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

Injection pumps start and automatic valves position, (2) Reactor trip, (3) feedwater isolation, (4) startup of the emergency diesel generators, (5) containment spray pumps start and automatic valves position, (6) containment isolation, (7) steam line isolation, (8) turbine trip, (9) emergency feedwater pumps start and automatic valves position, (10) containment cooling fans start and automatic valves position, and (11) automatic service water valves position.

The Engineered Safety Features Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates Turbine trip, closes main feedwater valves on T_{avg} below Setpoint, prevents the opening of the main feedwater valves which were closed by a Safety Injection or High Steam Generator Water Level signal, allows Safety Injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of Safety Injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates Safety Injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of Safety Injection actuation on low pressurizer pressure, and the manual block of SI and steamline isolation on steamline low pressure. On the manual block of steamline low pressure, manual block of steamline low pressure automatically initiates steamline isolation on steam generator pressure negative rate - high.

P-14 On increasing steam generator water level, P-14 automatically trips the turbine and all feedwater isolation valves; inhibits feedwater control valve modulation; and blocks the start of the startup feedwater pump.

WCAP-14333-P-A, Revision 1, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times" provides the justification for increasing the bypass times for testing and the Allowed Outage Times (AOTs) in the Reactor Trip System instrumentation and Engineered Safety Features Actuation System instrumentation Technical Specifications. WCAP-14333 justifies the following AOTs and Bypass test times:

- Inoperable analog channel AOT of 72 hours,
- Analog channel testing in bypass time of 12 hours, and
- Inoperable logic train AOT of 24 hours.

WCAP-15376-P-A, Revision 1, "Risk-Informed Assessment of the RTS and ESFAS Surveillance Test Intervals and Reactor Trip Breaker Test and Completion Times," provides the justification for changes to increase the AOT and the bypass test time for the reactor trip breakers. WCAP-15376-P-A justifies the following for the Reactor Trip System instrumentation:

- Inoperable reactor trip breaker AOT of 24 hours, and
- Reactor trip breaker testing in bypass time of 4 hours.

Planned maintenance and Tier 2 Restrictions

Consistent with the NRC Safety Evaluation (SE) requirements for WCAP-14333-P-A and WCAP-15376-P-A, Tier 2 insights must be included in the decision making process before removing an RTS or ESFAS logic train from service (WCAP-14333-P-A) or an RTB train (WCAP-15376-P-A) and implementing the associated extended (risk-informed) AOT. These "Tier 2 restrictions" are considered to be necessary to avoid risk significant plant configurations during the time an RTS or ESFAS logic train, or an RTB train is inoperable.

Entry into an AOT for an inoperable RTS or ESFAS logic train or an RTB train is not a typical pre-planned evolution during the MODES of Applicability for this equipment, other than when necessary for surveillance testing. Since the AOT may be entered due to equipment failure, some of the Tier 2 restrictions discussed below may not be met at the time of AOT entry. In addition, it is possible that equipment failure may occur after an RTS or ESFAS logic train or an RTB train is removed from service for surveillance testing or planned maintenance, such that one or more of the required Tier 2 restrictions are no longer met. In cases of equipment failure the programs and procedures in place to address the requirements of 10CFR 50.65(a)(4) require assessment of the emergent condition with appropriate actions taken to manage risk. Depending on the specific situation, these actions could include activities to restore the inoperable logic train or RTB train and exit the AOT, or to fully implement the Tier 2 restrictions, or to perform a unit shutdown, as appropriate from a risk management perspective.

The following WCAP-14333-P-A Tier 2 restrictions on concurrent removal of certain equipment will be implemented as described above when entering the AOT for an inoperable RTS or ESFAS logic train:

- To preserve ATWS mitigation capability, activities that degrade the availability of the auxiliary feedwater system, RCS pressure relief system (pressurizer PORVs and safety valves), AMSAC, or turbine trip should not be scheduled when a logic train is inoperable.

- To preserve LOCA mitigation capability, one complete ECCS train that can be actuated automatically must be maintained OPERABLE. Note that TS 3.5.2, "ECCS Subsystems -T_{avg} Greater Than or Equal To 350°F", ensures that this restriction is met. Therefore, this restriction does not have to be implemented by a separate procedure or program.

- To preserve reactor trip and safeguards actuation capability, activities that cause master relays or slave relays in the available train and activities that cause analog channels to be unavailable should not be scheduled when a logic train is inoperable.

- Activities on electrical systems (AC and DC power) and cooling systems (service water and component cooling water) that support the systems or functions listed in the first three bullets should not be scheduled when a logic train is inoperable. That is, one complete train of a function that supports a complete train of a function noted above must be available.

The following WCAP-15376-P-A Tier 2 restrictions on concurrent removal of certain equipment will be implemented as described above when entering the AOT for an inoperable RTB:

- The probability of failing to trip the reactor on demand will increase when a RTB is removed from service, therefore, systems designed for mitigating an ATWS event should be maintained available. RCS pressure relief (pressurizer PORVs and safety valves), auxiliary feedwater flow (for RCS heat removal), AMSAC, and turbine trip are important to ATWS mitigation. Therefore, activities that degrade the availability of the auxiliary feedwater system, RCS pressure relief system (pressurizer PORVs and safety valves), AMSAC, or turbine trip should not be scheduled when an RTB is inoperable.

- Due to the increased dependence on the available reactor trip train when one logic train is unavailable, activities that degrade other components of the RTS, including master relays or slave relays, and activities that cause analog channels to be unavailable, should not be scheduled when a logic train is inoperable.

- Activities on electrical systems (AC and DC power) that support the systems or functions listed in the first two bullets should not be scheduled when a RTB is inoperable.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that: (1) the associated action will be initiated when the radiation level monitored by each channel or combination thereof reaches its Setpoint, (2) the specified coincidence logic is maintained, and (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance. The radiation monitors for plant operations sense radiation levels in selected plant systems and locations and determine whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents

3/4.8.3 ONSITE POWER DISTRIBUTION (continued)

emergency DG supplies power to the 4.16 kV emergency buses. Control power for the 4.16 kV breakers is supplied from the Class 1E batteries.

Although not explicitly contained in TS 3.8.3.1 and 3.8.3.2, the MCCs that support the design function of the on-site AC power system must be energized to permit the functioning of structures, systems, and components important to safety under all normal and accident conditions. The AC distribution system ensures the safety functions of the Reactor Coolant Makeup, Residual Heat Removal, Emergency Core Cooling, Containment Heat Removal, Containment Atmosphere Cleanup, and the Cooling Water Systems can be accomplished. The accident analyses assume that the ESF systems are operable, which includes the availability of necessary power. Consequently, the MCCS that support these functions are required to be energized to maintain operability of the associated ESF systems and components.

No bus ties exist between redundant buses; however, manual bus tie breakers provide the capability to interconnect load center buses within a single train. Bus ties may be used when a unit substation transformer is out of service for maintenance or repair. Bus ties are provided only for operational flexibility. The unit substations are not designed to supply the total load of both buses when bus ties are used. When a bus tie breaker is used, loading on each unit substation will be administratively controlled to be within the rating of the unit substation transformer.

The 120V Vital Instrumentation and Control Power System consists of the uninterruptible power supply (UPS) units and the 120-volt vital instrument panels arranged in two trains. One vital UPS unit that provides power to one NSSS instrumentation channel (Channel IV) is powered from either the 480V system or 125V DC system depending on the available 480V bus voltage. Two vital UPS units that provide redundant power supplies to the balance-of-plant train A and train B vital instrument panels and three Vital UPS units that provide power to Channels I, II and III NSSS instrumentation are normally powered from the 480V system and can also convert 125V DC power from the station batteries to 120V AC power. These UPS units feed six electrically independent 120-volt AC vital instrument panels which serve as instrument and control power supplies.

The DC electrical power distribution system for each train consists of two 125-volt DC buses.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient SAFETY analyses in the UFSAR assume Engineered Safety Features (ESF) systems are OPERABLE. The AC, DC, and DC vital bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC, DC, and AC vital bus electrical power distribution systems in MODES 1 through 4 is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining power distribution systems OPERABLE during accident conditions in the event of: