NEXTera ENERGY SEABROOK

September 30, 2011

10 CFR 50.90

SBK-L-11181
Docket No. 50-443

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

Seabrook Station

License Amendment Request 11-05
License Amendment Request Regarding Cold Leg Injection Permissive

In accordance with the provisions of Section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), NextEra Energy Seabrook, LLC (NextEra) is submitting License Amendment Request (LAR) 11-05 for an amendment to the Technical Specifications (TS) for Seabrook Station. The proposed change modifies the circuitry that initiates high head safety injection (SI) by adding a new permissive, Cold Leg Injection Permissive (CLIP). This permissive prevents opening of the high head SI valves until reactor coolant system pressure decreases to the low pressure reactor trip set point.

Attachment 1 to this letter provides NextEra's evaluation of the proposed change, and Attachment 2 provides a markup of the TS showing the proposed change. New TS pages with the proposed change incorporated will be provided when requested by the NRC Project Manager. Associated TS Bases changes will be implemented in accordance with TS 6.7.6.j, TS Bases Control Program, upon implementation of the license amendment. As discussed in the evaluation, the proposed change does not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with the change.

No new commitments are made as a result of this change.

The Station Operation Review Committee has reviewed this LAR. A copy of this LAR has been forwarded to the New Hampshire State Liaison Officer pursuant to 10 CFR 50.91(b).

NextEra requests NRC review and approval of LAR 11-05 with issuance of a license amendment by September 30, 2012 to support proposed changes during the next scheduled refueling outage and implementation of the amendment within 30 days.

A001
NRR

Should you have any questions regarding this letter, please contact Mr. Michael O'Keefe, Licensing Manager, at (603) 773-7745.

Sincerely,

NextEra Energy Seabrook, LLC

Paul O. Freeman
Site Vice President

Attachments

1. NextEra Energy Seabrook's Evaluation of the Proposed Change
2. Markup of the Technical Specifications

cc: W.M. Dean, NRC Region I Administrator
G. E. Miller, NRC Project Manager
W. J. Raymond, NRC Senior Resident Inspector

Mr. Christopher M. Pope, Director Homeland Security and Emergency Management
New Hampshire Department of Safety
Division of Homeland Security and Emergency Management
Bureau of Emergency Management
33 Hazen Drive
Concord, NH 03305

Mr. John Giarrusso, Jr., Nuclear Preparedness Manager
The Commonwealth of Massachusetts
Emergency Management Agency
400 Worcester Road
Framingham, MA 01702-5399

NEXTera ENERGY SEABROOK

AFFIDAVIT

SEABROOK STATION UNIT 1

Facility Operating License NPF-86
Docket No. 50-443

License Amendment Request 11-05

License Amendment Request Regarding Cold Leg Injection Permissive

The following information is enclosed in support of this License Amendment Request:

- **NextEra Energy Seabrook's Evaluation of the Proposed Change**
- **Markup of the Technical Specifications**

I, Paul O. Freeman, Site Vice President of NextEra Energy Seabrook, LLC hereby affirm that the information and statements contained within this license amendment request are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Sworn and Subscribed before me this
30 day of September, 2011

Shirley Sweeney
Notary Public

SHIRLEY SWEENEY
MY COMMISSION EXPIRES NOV. 18, 2014
NOTARY PUBLIC
NEW HAMPSHIRE

Paul O. Freeman
Paul O. Freeman
Site Vice President

Attachment 1

NextEra Energy Seabrook's Evaluation of the Proposed Change

Subject: License Amendment Request Regarding Cold Leg Injection Permissive

1.0 SUMMARY DESCRIPTION

The proposed change revises the Seabrook Station Technical Specifications (TS) by adding a new permissive, Cold Leg Injection Permissive (CLIP) to provide additional time for the operator actions to mitigate an inadvertent operation of emergency core cooling system event. CLIP will permit automatic opening of the charging to cold leg injection valves only when required for high head safety injection as indicated by pressurizer pressure being below the Low Pressurizer Pressure Reactor Trip (LPPRT) setpoint. The valves will open when required for high head safety injection. CLIP is classified as an Engineered Safety Features Actuation System Interlock and will be added to the Technical Specifications.

2.0 DETAILED DESCRIPTION

The proposed change revises the Technical Specifications as follows:

1) Revise TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," to add low pressurizer pressure lead/lag time constants as a note
2) Revise TS Table 4.3-1 to add new note for low pressurizer pressure reactor trip lead/lag compensation surveillance requirements
3) Revise TS Tables 3.3-3, "Engineered Safety Features Actuation System Instrumentation," 3.3-4, "Engineered Safety Features Actuation System Instrumentation Trip Setpoints," and 4.3-2, "Engineered Safety Features Actuation System Instrumentation Surveillance Requirements," to add Functional Unit 10.d, "Engineered Safety Features Actuation System Interlock, Cold Leg Injection, P-15"
4) Revise TS Tables 3.3-4 and 4.3-2 to move the surveillance requirement for lead/lag time constants for low steam line pressure from a setpoint note to a surveillance requirement note
5) Editorial correction to the Table 4.3-2, Functional Unit, column heading

Attachment 2 provides the TS pages marked to show the proposed changes.

3.0 TECHNICAL EVALUATION

Background

Current Design

The charging to cold leg injection valves (1-SI-V-138 and 139) are provided as part of the Engineered Safety Features (ESF) to inject highly borated water from the Refueling Water Storage Tank (RWST) into the four Reactor Coolant System (RCS) cold legs. Flow is provided by the Centrifugal Charging Pumps (1-CS-P-2-A and P-2-B). A Safety Injection ("S") signal is provided by the ESF Actuation System (ESFAS) to open these valves when required to mitigate design bases events such as Loss-Of-Coolant Accidents (LOCAs) and Steam Line Breaks (SLBs).

Inadvertent generation of an "S" signal will result in a Condition II mass addition event when there is no loss of mass from the RCS. If the mass addition is not terminated by operator action, the pressurizer will be filled and the safety valves will open and discharge water (it is assumed that the valves will not reseat) thereby escalating the event to a Condition III Small Break LOCA. Condition II to III Event escalation is not permitted by the Seabrook Station commitment to ANSI N18.2.

Mitigation of the Inadvertent Operation of ECCS Event during Power Operation discussed in Updated Final Safety Analysis Report (UFSAR) Subsection 15.5.1 requires operator action to control the Atmospheric Steam Dump Valves (ASDVs) to cool the RCS down to and maintain 557°F and stop both Centrifugal Charging Pumps. These actions have to be accomplished in the time calculated to stop the mass addition prior to a water-solid pressurizer condition; therefore, they are Time Critical Actions (TCA). Since the margin between the expected and required performance times is small during this event, operator action requires periodic validation that the task can be completed within the required time.

New Design

CLIP will be designated ESFAS Interlock P-15 and will perform two nuclear safety-related functions:

1) Prevent the opening of either charging to cold leg injection valve (1-SI-V-138 and 139) when pressurizer pressure is above the LPPRT setpoint (1945 psig) in the presence of a credible single failure in the CLIP circuitry. The single failure requirement is met by providing coincidence logic in each Solid State Protection System (SSPS) train. The CLIP permissive signal consists of four pressurizer pressure instruments and their respective bistables which provide input to two independent redundant trains of logic circuitry, relays and motor control contacts. The "S" signal utilizes the same pressurizer pressure instruments but separate

parallel bistables and logic. Logic for both the CLIP and "S" signals utilize independent 2-out-of-4 low pressurizer pressure input logic, and CLIP uses 2-out-of-2 action logic, such that no single failure in the SSPS during an inadvertent "S" signal event will result in the presence of a CLIP signal when pressurizer pressure is above the LPPRT setpoint..

2) Permit the opening of at least one charging to cold leg injection valve when process conditions indicate a condition that requires high head safety injection in the presence of any credible single failure. The single failure requirement is met by the redundant trains of ESF and ESFAS.

CLIP functions in each train's SSPS are provided by adding two relays with contacts in series (designated P-15) with the existing "S" signal relay contacts in the circuitry that opens the charging to cold leg injection valve. The as-designed safety injection (SI) initiation circuits and actuation circuits for other functions will be maintained as is with no change in design. The proposed modifications are in addition to the as built circuits, and permit the cold leg injection valves to open when actual RCS pressure has degraded to the LPPRT setpoint. Since the LPPRT setpoint is higher than the low pressurizer pressure safety injection (LPPSI) setpoint, there will be no delay in the opening of the cold leg injection valves if the "S" signal is actuated by LPPSI. There will be a delay in the opening of the cold leg injection valves if the "S" signal is actuated by a signal other than LPPSI (i.e. main steam low pressure or containment high pressure). The valves' closing circuit interlock is adequately provided by the "S" signal and does not require a CLIP contact. Testing features are provided to ensure that testing will detect credible failures such as the failure of a tested contact to return to an open condition.

New Main Plant Computer System (MPCS) inputs and status monitoring lights on the Main Control Board will be provided. There is no change to the existing LPPRT bistable/input relay status monitor lights.

Evaluation

Conditions the Proposed Change is intended to Resolve

Mitigation of the inadvertent operation of emergency core cooling system (ECCS) during power operation event requires that operators perform certain TCAs in the calculated time to terminate water mass addition prior to creating a water-solid pressurizer condition. Licensed operators are challenged to meet the TCA time requirement for terminating high head SI following inadvertent operation of ECCS during power operation event. The proposed plant modification would significantly increase the available operator time to terminate an inadvertent SI event. However, the cold leg injection valves will open when required for high head safety injection.

Following implementation of CLIP, the mass addition for the inadvertent operation of ECCS during power operation event will be limited to reactor coolant pump seal injection (RCPSI). This will increase the time to approach water-solid pressurizer conditions, which will increase the performances times for the operator actions required to terminate an inadvertent SI event.

Functional Limitations of the CLIP Design

The pressurizer pressure instrumentation has four independent sensors; however, two of those sensors share a common sensing line. It can be postulated that a failure of the common sensing line could cause the output of these two pressurizer pressure channels to go low enough to satisfy the CLIP and to generate a low-low pressurizer pressure "S" signal to open the cold leg injection valves without a significant loss of coolant that would require high head SI flow. Thus, the failure of the common sensing line could initiate an inadvertent ECCS actuation event for which CLIP would not provide margin for operator action to mitigate the event. The probability of a failure in the common sensing line is low. The common pressurizer instrument tap is the standard Westinghouse design, and is listed in UFSAR Subsection 7.1.2.12 (5) as an exception to the guidance of Regulatory Guide 1.151, 7/1983 for the independence of sensing lines. Millstone Unit 3, which has already received NRC approval of the CLIP modification, also has this sensing line configuration. General Design Criterion 21 requires that the protection system be designed for high functional reliability commensurate with the safety functions to be performed. Based on the low probability of this failure, previous NRC acceptance of the CLIP design at Millstone 3 with the same instrument sensing line configuration, and the existing exception to sensing line independence in the current licensing bases, NextEra concludes that excluding this event from the design basis for CLIP is acceptable. NextEra further concludes that the reliability of the CLIP permissive is commensurate with the safety function performed.

Mass and Energy Release – Main Steamline Break

An evaluation was performed to address the impact of the CLIP modification on the steamline break (SLB) mass and energy (M&E) release Stretch Power Uprate (SPU) analyses. For the SLB M&E analyses, the CLIP modification has the potential to delay initiation of ECCS injection by inhibiting auto-open of the cold leg injection valves until both an "S" signal and low-pressurizer pressure (LPP) reactor trip signal are present. There are three parts to the evaluation: part 1 addresses the licensing-basis cases for SLB M&E release inside containment, part 2 addresses the licensing-basis cases for SLB M&E release outside containment, and part 3 addresses SLBs smaller than those analyzed for the updated final safety analysis report (UFSAR) for which there may be an "S" signal but no signal associated with the CLIP.

For the SLB inside containment analyses, two different break types are analyzed: double-ended ruptures (DERs) and split breaks. All cases from the SPU analysis were reviewed with respect to the timing of SI flow actuation and when the LPP reactor trip signal was received.

In the SPU analysis for DERs, the first signal is low steam pressure for all cases. Using the output results, the "S" signal actuation is compared to the LPP reactor trip signal for each case. For all of the DER cases in the SPU analyses, the difference in time is greater than the CLIP required delay (CLIP signal delay plus cold leg injection valve opening time). These cases, therefore, remain bounded by the later "S" Signal actuation.

In the SPU analysis for split breaks, the first signal is the time of the first high containment pressure setpoint. With the dynamic compensation of the LPP reactor trip signal, all of the split break cases in the SPU analysis have a difference in time greater than the CLIP required delay. These cases, therefore, remain bounded by the later "S" signal actuation.

The SPU analysis for the SLB M&E release outside containment was also evaluated for the CLIP modification. Each SLB case actuated ECCS flow on a low-low pressurizer pressure "S" signal. The CLIP modification requires an "S" Signal and the LPP reactor trip signal. The results show that the credited "S" Signal is much later than the LPP reactor trip signal. The results from the SPU analysis remain valid and bounding for the CLIP modification.

For the condition involving an "S" signal actuation with pressurizer pressure above the LPPRT setpoint, sensitivity cases varying the start time for ECCS injection, including no ECCS injection have concluded that the instantaneous and integrated M&E releases are insensitive to the injection start time. These results were expected as the ECCS injection occurs at relatively low flow rates due to high reactor coolant system pressure, and boron injection occurs long after the return to power has been mitigated by increasing reactor coolant system temperature. Any delay in initiation of ECCS injection has a negligible effect on core cooling throughout the event and core reactivity during the initial return to power. As discussed above for double-ended ruptures and split breaks, the break spectrum analyzed for the Seabrook SPU remains conservative for the CLIP modification.

The main steamline break evaluation concludes that the cases analyzed in the SPU analyses would not be impacted by the CLIP modification. The non-bounding small SLBs not receiving SI remain bounded by the peak DER cases for pressure and temperature. This evaluation shows that the installation of a CLIP would not impact the Seabrook SLB licensing-basis.

Fluid Systems

The proposed CLIP modification adds no new unanalyzed events relative to the centrifugal charging pumps' performance or ability to deliver the safety analysis credited flow. Upon initiation of an inadvertent "S" Signal with pressurizer pressure greater than the LPPRT setpoint, the cold leg injection valves will remain closed. The pumps' minimum recirculation flow isolation valves will open to provide the required minimum recirculation flow to each pump. In the event of a single active failure of one of the pump's minimum recirculation flow isolation valves to open, the unaffected pump remains capable of delivering the safety analysis credited flow, if and when required. The consequence of this event is equivalent to any pre-modification event that generates an "S" signal and where the RCS subsequently returns to pressure (e.g., a secondary side high energy break) in conjunction with a single active failure to open one of the minimum recirculation flow isolation valves.

Non-LOCA Analysis

The CLIP impacts only non-LOCA events that model high-head SI, including: Hot Zero-Power Steamline Break (HZP-SLB), UFSAR Section 15.1.5; Feedline Break (FLB), UFSAR Section 15.2.8; and Inadvertent ECCS, UFSAR Section 15.5.1. Additionally, the Chemical and Volume Control System (CVCS) Malfunction in UFSAR Section 15.5.2 is indirectly impacted by this modification. Each of these events has been reanalyzed or evaluated considering the CLIP modification. All other non-LOCA events are not impacted.

Other related issues

Dynamic Compensation of Low Pressurizer Pressure Reactor Trip

The mass and energy release analyses take credit for the dynamic compensation of the low pressurizer pressure reactor trip. Since this feature is credited in the Seabrook design basis analysis, the lead/lag time constants for the dynamic compensation will be added to the technical specifications.

CVCS Malfunction

Prior to CLIP implementation, the CVCS malfunction event was bounded by the inadvertent ECCS actuation at power. With the addition of CLIP, the inadvertent ECCS actuation at power is no-longer the limiting event. As part of this effort, the CVCS malfunction described in section 15.5.2 of the UFSAR has been reevaluated. In the past, the evaluation described in the UFSAR indicated that the operator had sufficient time to take corrective action. In the revised evaluation, an assumption is made of a time critical operator action of 10 minutes (600 sec) to terminate charging.

The result is that pressurizer pressure will remain below the pressurizer safety valve setpoint for at least 45 minutes to provide time for operator action to stop the mass addition from RCPSI.

<u>Conclusion</u>

The CLIP installation would not impact the licensing-basis for SLB and M & E Releases. The HZP SLB event remains bounding for operation at the uprate conditions. For the FLB, the Emergency Feedwater System capacity is adequate to remove decay heat, to prevent overpressurizing the RCS, and to prevent uncovering the core. For the inadvertent ECCS event, there is no hazard to the integrity of the RCS. Following implementation of CLIP, the mass addition for the Inadvertent Operation of ECCS Event during Power Operation will be limited to RCPSI. This will increase the time to approach water-solid pressurizer conditions, which will increase the time allowed to terminate an inadvertent SI event. For the CVCS malfunction sequence of events, the operator has sufficient time to take corrective action to prevent pressurizer filling.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

- 10 CFR 50.36, Technical Specifications, states (c) Technical specifications will include items in the following categories:

  (1) *Safety limits, limiting safety system settings, and limiting control settings*. (i)(A) Safety limits for nuclear reactors are limits upon important process variables that are found to be necessary to reasonably protect the integrity of certain of the physical barriers that guard against the uncontrolled release of radioactivity. If any safety limit is exceeded, the reactor must be shut down. The licensee shall notify the Commission, review the matter, and record the results of the review, including the cause of the condition and the basis for corrective action taken to preclude recurrence. Operation must not be resumed until authorized by the Commission. The licensee shall retain the record of the results of each review until the Commission terminates the license for the reactor, except for nuclear power reactors licensed under § 50.21(b) or § 50.22 of this part. For these reactors, the licensee shall notify the Commission as required by § 50.72 and submit a Licensee Event Report to the Commission as required by § 50.73. Licensees in these cases shall retain the records of the review for a period of three years following issuance of a Licensee Event Report.

- General Design Criterion 13—Instrumentation and control. Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

- General Design Criterion 15—Reactor coolant system design. The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

- General Design Criterion 20—Protection system functions. The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

- General Design Criterion 21—Protection system reliability and testability. The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

- General Design Criterion 22—Protection system independence. The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

- General Design Criterion 23—Protection system failure modes. The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.
- General Design Criterion 24—Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

The changes proposed in this request will continue to meet the above regulatory requirements.

4.2 Precedent

The NRC staff has approved similar license for a Cold Leg Injection Permissive design for the Millstone 3 Power Station.

- Safety Evaluation By The Office Of Nuclear Reactor Regulation Related To Amendment No. 242 To Renewed Facility Operating License No. NPF-49 Dominion Nuclear Connecticut, Incorporated Millstone Power Station, Unit 3 Docket No. 50-423 (August 12, 2008; Adams Accession No. ML081640535) [Reference 2]

4.3 Significant Hazards Consideration

No Significant Hazards Consideration

The proposed change modifies the circuitry that initiates high head safety injection (SI) by adding a new permissive, Cold Leg Injection Permissive (CLIP). This permissive prevents opening of the high head SI valves until reactor coolant system pressure decreases to the low pressure reactor trip setpoint.

In accordance with 10 CFR 50.92, NextEra Energy Seabrook has concluded that the proposed change does not involve a significant hazards consideration (SHC). The basis for the conclusion that the proposed change does not involve a SHC is as follows:

1. *The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.*

The proposed change adds an additional permissive before high head safety injection is initiated to assist the operators in mitigating the consequences of an inadvertent initiation of the emergency core cooling system (ECCS). This change in the ECCS actuation circuitry does not increase the probability of any accident previously evaluated because:

- there is no effect on any of the systems structures or components that are used for normal operation of the plant,
- there is no effect on any of the fission product barriers,
- this change will not affect the normal operating procedures,
- the change does not affect the sensing instrumentation used to initiate the protective functions.

The revised circuitry will delay the initiation of high head SI until reactor coolant pressure is below the low pressurizer pressure reactor trip setpoint; however, the proposed changes does not significantly increase the consequences of accidents previously evaluated. The proposed change does not alter ECCS flow or SI actuation delay times. The delayed opening of the high head SI valves has been evaluated for the effect on the consequences of the following:

- Mass and energy release for steam line break accidents,
- Steam line break – UFSAR section 15.1.5 (specifically Hot Zero-Power conditions)
- Feedwater line break – UFSAR section 15.2.8
- Inadvertent Operation of Emergency Core Cooling System During Power Operation – UFSAR section 15.5.1
- Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory – UFSAR section 15.5.2

For all of the above evaluated accidents, the consequences remain bounded by the analyses of record. For the inadvertent initiation of ECCS event, the proposed change assists the operators in mitigating the event by significantly extending the time for the pressurizer to fill. Additional evaluations of Small Break LOCA, Best Estimate Large Break LOCA, Long term cooling, LOCA forces, Cold Overpressure Mitigation/Low Temperature Over Pressure Protection, Steam Generator Tube Rupture, and LOCA Mass and Energy Release were performed and it was concluded that they were not affected by this change.

In addition evaluations were performed for the centrifugal charging pumps and reactor vessel internals; and for the NSSS design transients to determine if the change in the timing of the high head injection would have an effect and it was concluded that these components and transients are not adversely affected

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. *The proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.*

   The proposed change adds new components to the solid state protection system similar to the components and configurations that are already installed. The sequence of operation of equipment used to mitigate the consequences of an accident is changed, however, it does not add any different types of equipment. The proposed change is a change to the protection circuitry for the plant and not to the system or equipment used for normal operation of the plant. It does not alter any fluid flow paths or fission product barriers and does not change the method of control of any plant systems. The proposed change does not alter or prevent the ability of the ECCS to perform its specified function to mitigate the consequences of an initiating event within assumed acceptance limits. The evaluation of the centrifugal charging pumps, reactor internals, control systems and NSSS design transients confirmed that new failure modes were not created

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. *The proposed changes do not involve a significant reduction in the margin of safety.*

   Margin of safety is associated with confidence in the ability of the fission product barriers (i.e., fuel cladding, reactor coolant system pressure boundary, and containment structure) to limit the level of radiation dose to the public. The proposed changes will not relax any criteria used to establish safety limits and will not relax any safety system settings. The safety analysis acceptance criteria are not affected by this change. The proposed change will not result in plant operation in a configuration outside the design basis.

   The proposed change does involve a change in the timing of the mitigation of inadvertent ECCS actuation and steam line break.

   This change provides additional time for mitigating the Inadvertent Operation of Emergency Core Cooling System during Power Operation event prior to filling the pressurizer water solid, by preventing the injection of high head safety injection when it is not required. .

The change potentially delays the injection of high head safety injection on a steam line break. For double ended ruptures the existing mass and energy release inside containment analyses of record assumes Safety Injection delivery based on Low steam line pressure within 25 seconds. The analysis of this change demonstrates that for all cases the CLIP permissive and the associated delay will be satisfied prior to the assumed 25 second delay. For split breaks, the mass and energy release inside containment analysis of record assumes Safety Injection delivery within approximately 38 seconds. The analysis of this change concludes that with dynamic compensation of the Low Pressurizer Pressure Reactor trip (LPPRT) setpoint, the delivery of high head safety injection will continue to meet this assumption. Dynamic compensation of the LPPRT, which is credited in the Seabrook design basis analyses, is being added to the technical specification description of the function and surveillance criteria are added. The mass and energy release for steam line breaks outside containment rely on the low-low pressurizer pressure safety injection as the actuation signal and this timing of safety injection from this actuation signal is not affected by this change. The mass and energy release for small steam line breaks have been reviewed. Sensitivity studies conclude that the mass and energy releases are insensitive to injection because ECCS injection occurs at relatively low flow rates due to high reactor coolant system pressure and boron injection occurs long after return to power has been mitigated by increasing RCS temperature. The overall steam line break analyses results are not affected because SI delivery times with the CLIP modification will remain bounded by the SI delivery times in the current design basis steam line break analyses.

The feed line break accident conservatively has been evaluated with no safety injection so there is no effect on margin from the delay in high head safety injection. The assumption for operator action to mitigate the consequences of a chemical and volume control malfunction are not changed by this modification so there is no change in the margin for that event. As discussed above the consequences of the other accidents evaluated remain bounded by the analyses of record. The results of analyses and evaluations supporting the proposed change demonstrate acceptance criteria continue to be met.

Therefore, these proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, NextEra concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(b), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusions

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

NextEra has evaluated the proposed amendment for environmental considerations. The review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set for in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. Seabrook Station UFSAR, Revision 14, sections 7.1.2.12, 15.1.5, 15.2.5, 15.2.8, 15.5.1, 15.5.2
2. NRC letter Safety Evaluation By The Office Of Nuclear Reactor Regulation Related To Amendment No. 242 To Renewed Facility Operating License No. NPF-49 Dominion Nuclear Connecticut, Incorporated Millstone Power Station, Unit 3 Docket No. 50-423 (August 12, 2008; Adams Accession No. ML081640535)

Attachment 2

Mark-up of the Technical Specifications (TS)

The attached markup reflects the currently issued version of the TS and Facility Operating License. At the time of submittal, the Facility Operating License was revised through Amendment No. 125.

Listed below are the license amendment requests that are awaiting NRC approval and may impact the currently issued version of the Facility Operating License affected by this LAR.

LAR	Title	NextEra Energy Seabrook Letter	Date Submitted
LAR 10-02	Application for Change to the Technical Specifications for the Containment Enclosure Emergency Air Cleanup System	SBK-L-10074	05/14/2010
LAR 11-01	Application to Revise the Technical Specifications for Reactor Coolant Leakage Detection Instrumentation	SBK-L-11066	04/21/2011
LAR 11-03	License Amendment Request Regarding Containment Spray Nozzle Surveillance Requirement	SBK-L-11130	07/14/2011

The following TS pages are included in the attached markup:

Technical Specification	Title	Page
-------	Table 2.2-1 Reactor Trip System Instrumentation Trip Setpoints	2-4 2-10
3/4.3.1	Table 4.3-1 Reactor Trip System Instrumentation Surveillance Requirements	3/4 3-9 3/4 3-13
3/4.3.2	Table 3.3-3 Engineered Safety Features Actuation System Instrumentation	3/4 3-21
3/4.3.2	Table 3.3-4 Engineered Safety Features Actuation System Instrumentation Trip Setpoints	3/4 3-28 3/4 3-29
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TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	N.A.	N.A.	N.A.	N.A.
2. Power Range, Neutron Flux					
a. High Setpoint	7.5	4.56	1.42	≤109% of RTP*	≤111.1% of RTP*
b. Low Setpoint	8.3	4.56	1.42	≤25% of RTP*	≤27.1% of RTP*
3. Power Range, Neutron Flux, High Positive Rate	1.6	0.5	0	≤5% of RTP* with a time constant ≥2 seconds	≤6.3% of RTP* with a time constant ≥2 seconds
4. (NOT USED)					
5. Intermediate Range, Neutron Flux	17.0	8.41	0	≤25% of RTP*	≤31.1% of RTP*
6. Source Range, Neutron Flux	17.0	10.01	0	≤10^5 cps	≤1.6×10^5 cps
7. Overtemperature ΔT	N.A.	N.A.	N.A.	See Note 1	See Note 2
8. Overpower ΔT	N.A.	N.A.	N.A.	See Note 3	See Note 4
9. Pressurizer Pressure - Low	N.A.	N.A.	N.A.	≥1945 psig	≥1,933 psig SEE NOTE 5
10. Pressurizer Pressure - High	N.A.	N.A.	N.A.	≤2385 psig	≤2,397 psig

*RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	Value specified in COLR,
T	=	As defined in Note 1,
T"	=	Indicated T_{avg} at RATED THERMAL POWER, °F, (Calibration temperature for ΔT instrumentation, value specified in the COLR),
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	A function of the indicated difference between the top and bottom detectors of the power-range neutron ion chambers as specified in the COLR.

NOTE 4: Cycle dependent values for the channel's Allowable Value are specified in the COLR.

NOTE 5: Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are $\tau_1 \geq 10$ seconds and $\tau_2 \leq 1$ seconds

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(13)	N.A.	1,2,3*,4*,5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4, 5)	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Power Range, Neutron Flux, High Positive Rate	N.A.	R(4)	Q	N.A.	N.A.	1, 2
4. (NOT USED)						
5. Intermediate Range, Neutron Flux	S	R(4, 5)	S/U(1)	N.A.	N.A.	1***, 2
6. Source Range, Neutron Flux	S	R(4, 5)	S/U(8),Q(9)	N.A.	N.A.	2**, 3, 4, 5
7. Overtemperature ΔT	S	R	Q	N.A.	N.A.	1, 2
8. Overpower ΔT	S	R	Q	N.A.	N.A.	1, 2
9. Pressurizer Pressure--Low	S	R (16)	Q	N.A.	N.A.	1
10. Pressurizer Pressure--High	S	R	Q	N.A.	N.A.	1, 2
11. Pressurizer Water Level--High	S	R	Q	N.A.	N.A.	1
12. Reactor Coolant Flow--Low	S	R	Q	N.A.	N.A.	1

TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(12) Number not used.

(13) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

(14) Local manual shunt trip prior to placing breaker in service.

(15) Automatic undervoltage trip

(16) CHANNEL CALIBRATION shall include verification that the time constants are adjusted to the prescribed values

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. RWST Level--Low-Low	4	2	3	1, 2, 3, 4	15
Coincident With: Safety Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
9. Loss of Power (Start Emergency Feedwater)					
a. 4.16 kV Bus E5 and E6-Loss of Voltage	2/bus	2/bus	1/bus	1, 2, 3, 4	14
b. 4.16 kV Bus E5 and E6-Degraded Voltage Coincident with SI	2/bus	2/bus	1/bus	1, 2, 3, 4	14
	See Item 1. above for all Safety Injection initiating functions and requirements.				
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	19
b. Reactor Trip, P-4	2	2	2	1, 2, 3	21
c. Steam Generator Water Level, P-14	4/stm. gen.	2/stm. gen.	3/stm. gen.	1, 2, 3	18
d. Cold LEG Injection, P-15	4	2	3	1,2,3	18

TABLE 3.3-4 (Continued)

TABLE NOTATIONS

*Time constants utilized in the lead-lag controller for Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. ~~CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.~~

**The time constant utilized in the rate-lag controller for Steam Line Pressure-Negative Rate-High ~~is greater than or equal to~~ $\geq$ 50 seconds. ~~CHANNEL CALIBRATION shall ensure that this time constant is adjusted to this value.~~

***Value specified applies when "as measured" Trip Setpoint is greater than the specified Trip Setpoint.

****Value specified applies when "as measured" Trip Setpoint is less than the specified Trip Setpoint.

*****Time constants utilized in the lead-lag controller for Pressurizer Pressure-Low are $\tau_1 \geq 10$ seconds and $\tau_2 \leq 1$ seconds.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TOTAL ALLOWANCE (TA)	Z	SENSOR ERROR (S)	TRIP SETPOINT	ALLOWABLE VALUE
9. Loss of Power (Start Emergency Feedwater)					
a. 4.16 kV Bus E5 and E6 Loss of Voltage	N.A.	N.A.	N.A.	≥ 2975 volts with a ≤ 1.20 second time delay.	≥ 2908 volts with a ≤ 1.315 second time delay.
b. 4.16 kV Bus E5 and E6 Degraded Voltage	N.A.	N.A.	N.A.	≥ 3933 volts with a ≤ 10 second time delay.	≥ 3902 volts with a ≤ 10.96 second time delay.
Coincident with: Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints and Allowable Values.				
10. Engineered Safety Features Actuation System Interlocks					
a. Pressurizer Pressure, P-11	N.A.	N.A.	N.A.	≤ 1950 psig	≤ 1962 psig
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	N.A.
c. Steam Generator Water Level, P-14	See Item 5. above for all Steam Generator Water Level Trip Setpoints and Allowable Values.				

d. Cold LEG Injection, P-15 N.A. N.A N.A ≥ 1945 psig ≥ 1933 psig ***

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

~~CHANNEL~~ FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Safety Injection (Reactor Trip, Feedwater Isolation, Start Diesel Generator, Phase "A" Isolation, Containment Ventilation Isolation, and Emergency Feedwater, Service Water to Secondary Component Cooling Water Isolation, CBA Emergency Fan/Filter Actuation, and Latching Relay).								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
c. Containment Pressure-Hi-1	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Pressurizer Pressure Low	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Low	S	R(4)	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
2. Containment Spray								
a. Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
C. Containment Pressure-Hi-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

~~CHANNEL~~ FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Containment Pressure-Hi-3	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
c. Containment Ventilation Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
4) Containment On Line Purge Radioactivity- High	S	R	Q(2)	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

~~CHANNEL~~ FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steam Line Isolation								
a. Manual Initiation (System)	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Containment Pressure-Hi-2	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Steam Line Pressure-Low	S	R (4)	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
e. Steam Line Pressure-Negative Rate-High	S	R (4)	Q	N.A.	N.A.	N.A.	N.A.	3
5. Turbine Trip								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2
b. Steam Generator Water Level-High-High (P-14)	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2
6. Feedwater Isolation								
a. Steam Generator Water Level--High-High (P-14)	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
7. Emergency Feedwater								
a. Manual Initiation								
1) Motor-driven pump	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3
2) Turbine-driven pump	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

~~CHANNEL~~ FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. Emergency Feedwater (Continued)								
b. Automatic Actuation and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3
c. Steam Generator Water Level-Low-Low, Start Motor-Driven Pump and Turbine-Driven Pump	S	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
d. Safety Injection, Start Motor-Driven Pump and Turbine-Driven Pump	See Item 1. above for all Safety Injection Surveillance Requirements.							
e. Loss-of-Offsite Power Start Motor-Driven Pump and Turbine-Driven Pump	See Item 9. for all Loss-of-Offsite Power Surveillance Requirements.							
8. Automatic Switchover to Containment Sump								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	M(1)	Q	1, 2, 3, 4
b. RWST Level Low-Low Coincident With	N.A.	R	Q	Q(3)	N.A.	N.A.	N.A.	1, 2, 3, 4
Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

~~CHANNEL~~ FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MASTER RELAY TEST	SLAVE RELAY TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. Loss of Power (Start) Emergency Feedwater)								
a. 4.16 kV Bus E5 and E6 Loss of Voltage	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
b. 4.16 kV Bus E5 and E6 Degraded Voltage Coincident With Safety Injection	N.A.	R	N.A.	M	N.A.	N.A.	N.A.	1, 2, 3, 4
	See Item 1. above for all Safety Injection Surveillance Requirements							
10. Engineered Safety Features Actuation System Interlocks								
a. Pressurizer Pressure, P-11	N.A.	R	Q	N.A.	N.A.	N.A.	N.A.	1, 2, 3
b. Reactor Trip, P-4	N.A.	N.A.	N.A.	N.A.	R	N.A.	N.A.	1, 2, 3
c. Steam Generator Water Level, P-14	S	R	Q	N.A.	M(1)	M(1)	Q	1, 2, 3
d. Cold Leg Injection, P-15	S	R(4)	Q	N.A.	M(1)	M(1)	Q	1, 2, 3

TABLE NOTATION

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) A DIGITAL CHANNEL OPERATIONAL TEST will be performed on this instrumentation.

(3) Setpoint verification is not applicable.

(4) CHANNEL CALIBRATION shall include verification that the time constants are adjusted to the prescribed values.