December 1, 2017

10 CFR 50.90

SBK-L-17185

Docket No. 50-443

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

Seabrook Station

License Amendment Request 17-04

Remove Limitation to Perform Certain Surveillance Requirements during Shutdown and Changes to Administrative Technical Specifications

Pursuant to 10 CFR 50.90, NextEra Energy Seabrook, LLC (NextEra) is submitting License Amendment Request (LAR) 17-04 to revise the Seabrook Station Technical Specifications (TS). The proposed change revises certain 18-month TS surveillance requirements to eliminate the condition that testing be conducted "during shutdown" and revises the administrative TS regarding plant staff and responsibilities.

The enclosure to this letter provides NextEra's evaluation of the proposed change. Attachment 1 to the enclosure provides markups of the TS showing the proposed changes, and Attachment 2 provides retyped TS pages containing the proposed changes.

As discussed in the evaluation, the proposed changes do not involve a significant hazards consideration pursuant to 10 CFR 50.92, and there are no significant environmental impacts associated with the change.

The Station Operation Review Committee has reviewed the proposed license amendment. In accordance with 10 CFR 50.91(b) (1), a copy of this letter is being forwarded to the designee of the State of New Hampshire.

There are no new or revised commitments made in this submittal.

NextEra requests NRC review and approval of this license amendment request by November 30, 2018 and implementation within 90 days.

Should you have any questions regarding this letter, please contact Mr. Ken Browne, Licensing Manager, at (603) 773-7932.

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

Executed on December 1, 2017

Sincerely,

Eric McCartney
Regional Vice President - Northern Region
NextEra Energy

Enclosure: Evaluation of the Proposed Change

cc: NRC Region I Administrator
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Enclosure

NextEra Energy Seabrook's Evaluation of the Proposed Change

Subject: Remove Limitation to Perform Certain Surveillance Requirements during Shutdown and Changes to Administrative Technical Specifications

Evaluation of the Proposed Change

1.0 SUMMARY DESCRIPTION

NextEra Energy Seabrook, LLC (NextEra) is submitting License Amendment Request (LAR) 17-04 to revise the Seabrook Station Technical Specification (TS). The proposed change revises certain 18-month TS surveillance requirements (SR) to eliminate the condition that testing be conducted "during shutdown" and revises the administrative TS regarding plant staff and responsibilities.

2.0 DETAILED DESCRIPTION

2.1 System Design and Operation

The proposed change involves SRs associated with the systems and components described below.

Emergency Core Cooling System (ECCS)

The ECCS consists of the centrifugal charging pumps, safety injection pumps, a refueling water storage tank (RWST), the residual heat removal pumps, the residual heat removal heat exchangers, the safety injection accumulators, and the associated valves and piping. The ECCS is comprised of two identical trains, each train independent of the other and fully redundant. The primary function of the ECCS following an accident is to remove the stored and fission product decay heat from the reactor core so that fuel rod damage, to the extent that it would impair effective cooling of the core, is prevented. The ECCS is designed to cool the reactor core as well as to provide additional shutdown capability following a loss of coolant accident, control rod ejection accident, steam or feedwater system break, or a steam generator tube rupture.

Containment Spray System (CBS)

The CBS system is designed to remove the energy discharged to the containment following a loss-of-coolant accident or main steam line break to prevent the containment pressure from exceeding design pressure and to reduce and maintain containment temperature and pressure within acceptable limits. The CBS system is actuated by high pressure in the containment. The CBS system is comprised of two identical trains, each train independent of the other and fully redundant.

Spray Additive Tank (SAT)

The spray additive tank (SAT) is mounted adjacent to the RWST, and drains by gravity into the RWST mixing chamber. The SAT provides the correct amount of sodium hydroxide solution to insure that the final containment recirculation sump pH after injection will be between 8.5 and 11.0 units for the various reactor coolant conditions.

Containment Isolation Valves

The Containment Isolation System is comprised of the valves, piping and actuators required to isolate the containment following a LOCA or steam line rupture. Containment isolation valve closure speeds and leak tightness will prevent radiological effects from exceeding the guidelines established by 10 CFR 100.

Emergency Feedwater (EFW) System

The EFW system provides the capability to remove heat from the reactor coolant system during emergency conditions when the main feedwater system is not available. The system is comprised of two full-sized pumps, one motor- driven and one turbine-driven.

Primary Component Cooling (PCCW) Water System

The PCCW system supplies flow to the safeguard components that are required for safe shutdown or to mitigate the consequences of an accident. The system consists of two independent and redundant flow loops.

Service Water (SW) / Ultimate Heat Sink (UHS)

The UHS employs two independent and redundant cooling loops. Each loop can be supplied by either of two full-capacity SW pumps (four pumps total) drawing water from the Atlantic Ocean or alternatively, each loop can be supplied by a full-capacity cooling tower pump (two pumps total) drawing water from a mechanical draft cooling tower.

Diesel Generators (DG)

Two redundant diesel generators are provided to automatically connect to the two trains of redundant emergency buses when a loss of all offsite power sources occurs. Each emergency bus and associated load group has sufficient redundancy to assure that the safety functions are performed.

2.2 Current TS Requirements

The SRs below verify that valves actuate to their correct position or that pumps start upon receipt of an actuation signal. Each SR contains a restriction that the test is performed "during shutdown."

- SR 4.5.2.e for ECCS automatic valves and ECCS pumps
- SR 4.6.2.1.c for containment spray pumps and automatic valves
- SR 4.6.2.2.c for spray additive tank automatic valves
- SR 4.6.3.2 for containment isolation valves
- SR 4.7.1.2.1.c for auxiliary feedwater pumps and automatic valves
- SR 4.7.3.b for primary component cooling water automatic valves
- SR 4.7.4.1.b and 4.7.4.2.b for automatic service water valves and cooling tower pumps and automatic valves
- SR 4.8.1.1.2.g for emergency diesel generators

The administrative controls in section 6.0 of the TS refer to the plant-specific titles of Station Director, Site Vice President, and SORC (Station Operation Review Committee). This section also includes (1) a requirement for management to issue an annual directive regarding the control room command function, (2) Table 6.2-1, Minimum Shift Crew Composition, and (3) Specification 6.2.4 for Shift Technical Advisor.

2.3 Reason for the Proposed Change

The proposed amendment will allow certain 18-month SRs previously performed while shutdown to be performed during power operation. This change will eliminate duplicate testing and reduce outage scope by removing testing from the refueling outages. The proposed change to the administrative controls removes requirements that unnecessarily duplicate regulatory requirements. The change also replaces obsolete plant-specific titles with generic titles, which will avoid the need for future license amendments when position titles change.

2.4 Description of the Proposed Change

Proposed change to SRs to remove the limitation "during shutdown:"

1. SR 4.5.2.e:

   Each ECCS subsystem shall be demonstrated OPERABLE:

   In accordance with the Surveillance Frequency Control Program~~, **during shutdown**,~~ by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on (Safety Injection actuation and Automatic Switchover to Containment Sump) test signals, and
   2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:
      a) Centrifugal charging pump,
      b) Safety Injection pump, and
      c) RHR pump.

2. SR 4.6.2.1.c:

   Each Containment Spray System shall be demonstrated OPERABLE:

   In accordance with the Surveillance Frequency Control Program ~~**during shutdown,**~~ by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal, and
   2) Verifying that each spray pump starts automatically on a Containment Pressure-Hi-3 test signal.

3. SR 4.6.2.2.c:

The Spray Additive System shall be demonstrated OPERABLE:

In accordance with the Surveillance Frequency Control Program~~, during shutdown,~~ by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal.

4. SR 4.6.3.2:

Each containment isolation valve shall be demonstrated OPERABLE ~~during shutdown~~ in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" Isolation valve actuates to its isolation position,

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" Isolation valve actuates to its isolation position, and

c. Verifying that on a Containment Purge and Exhaust Isolation test signal, each purge and exhaust valve actuates to its isolation position.

5. SR 4.7.1.2.1.c:

Each auxiliary feedwater pump shall be demonstrated OPERABLE:

In accordance with the Surveillance Frequency Control Program ~~during shutdown~~ by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Emergency Feedwater System Actuation test signal;

2) Verifying that each emergency feedwater pump starts as designed automatically upon receipt of an Emergency Feedwater Actuation System test signal;

3) Verifying that with all manual actions, including power source and valve alignment, the startup feedwater pump starts within the required elapsed time; and

4) Verifying that each emergency feedwater control valve closes on receipt of a high flow test signal.

6. SR 4.7.3.b:

At least two primary component cooling water loops shall be demonstrated OPERABLE:

In accordance with the Surveillance Frequency Control Program ~~during shutdown,~~ by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation signal.

7. SR 4.7.4.1.b:

Each service water loop shall be demonstrated OPERABLE:

In accordance with the Surveillance Frequency Control Program **~~during shutdown,~~** by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal.

8. SR 4.7.4.2.b:

Each service water cooling tower loop shall be demonstrated OPERABLE:

In accordance with the Surveillance Frequency Control Program **~~during shutdown,~~** by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal,

2) Each automatic valve in the flowpath actuates to its correct position on a Tower Actuation (TA) test signal and

3) Each service water cooling tower pump starts automatically on a TA signal.

9. SR 4.8.1.1.2.g:

In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously from standby condition**~~, during shutdown,~~** and verifying that both diesel generators achieve:

1) A generator voltage and frequency greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the start signal, and

2) A steady-state generator voltage and frequency of 4160 $\pm$ 420 volts and 60 $\pm$ 1.2 Hz.

Proposed changes to administrative controls:

Revise plant-specific titles:

6.1.1 The ~~Station Director~~ ***plant manager*** shall be responsible for overall station operation and shall delegate in writing the succession to this responsibility during his absence.

6.2.1.b. The ~~Station Director~~ ***plant manager*** shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

6.2.1.c. ~~The Site Vice President~~ ***A specified corporate officer*** shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

TS 6.12. PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

b. Shall become effective after review and acceptance by the ~~SORC~~ ***Onsite Review Group*** and approval of the ~~Station Director~~ ***plant manager.***

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

b. Shall become effective after review and acceptance by the ~~SORC~~ ***Onsite Review Group*** and the approval of the ~~Station Director~~ ***plant manager.***

6.14 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.14.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the ~~SORC~~ ***Onsite Review Group***. The discussion of each change shall contain:

8) Documentation of the fact that the change was reviewed and found acceptable by the ~~SORC~~ ***Onsite Review Group.***

b. Shall become effective upon review and acceptance by the ~~SORC~~ ***Onsite Review Group.***

Revise TS 6.1.2:

6.1.2 ~~The Shift Manager (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Site Vice President shall be reissued to all station personnel on an annual basis.~~ ***The Shift Manager (SM) shall be responsible for the control room command function. During any absence of the SM from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SM from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.***

Delete Table 6.2-1 and revise TS 6.2.2 accordingly to include the following:

a. ***A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.***

b. ***Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.2.a and 6.2.2.f for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members*** *provided immediate action is taken to restore the shift crew composition to within the minimum requirements.*

c. *A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.*

d. *An individual (Shift Technical Advisor (STA)) shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. The STA position shall be manned in MODES 1, 2, 3, and 4 unless the SM or the individual with a Senior Operator license meets the qualifications for the STA.*

e. *While the unit is in MODE 1, 2, 3 or 4, a licensed senior operator, either the SM or SRO, shall be on shift having had at least 6 months of hot operating*

f. The Operations Manager shall meet one of the following:

   1. Hold a senior operator license,

   2. Have held a senior operator license on a similar unit (PWR), or

   3. Have been certified for equivalent senior operator knowledge.

g. The Assistant Operations Manager shall hold a senior reactor operator license.

Delete TS 6.2.4:

~~6.2.4 SHIFT TECHNICAL ADVISOR~~

~~6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Control Room Commander in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the station.~~

3.0 TECHNICAL EVALUATION

Proposed change to SRs to remove the limitation "during shutdown"

The Seabrook TS contain a number of SRs that require performing the surveillance during shutdown. While the restriction is intended to ensure that the SRs are performed consistent with safe plant operation, many components affected by this restriction are safely tested at power. For example, the engineered safety features actuation system instrumentation required by TS 3.3.2 contains numerous SRs for slave relay tests. Many of the slave relay tests, which are performed during power operation, actuate ECCS pumps and valves. At the same time, many of these same ECCS pumps and valves have other SRs that require testing during shutdown to verify that valves actuate to their correct position or that pumps start upon receipt of an actuation signal. Because of the "during shutdown" limitation associated with these SRs, no credit is taken for the testing performed during plant operation. Although the slave relay testing performed during plant operation could satisfy the SRs required to be performed during shutdown, the testing performed during operation cannot be credited for these other tests because of the "during shutdown" restriction. Consequently, this unnecessary restriction results in duplicate testing of the affected components.

Generic Letter 91-04, Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle [Reference 1], specifically recommends the elimination of the condition "during shutdown" from SRs, stating:

> The staff concludes that the TS need not restrict surveillances as only being performed during shutdown. Nevertheless, safety dictates that when refueling interval surveillances are performed during power operation, licensees give proper regard for their effect on the safe operation of the plant. If the performance of a refueling interval surveillance during plant operation would adversely affect safety, the licensee should postpone the surveillance until the unit is shut down for refueling or is in a condition or mode that is consistent with the safe conduct of that surveillance.

NextEra manages the overall risk associated with on-line and outage work activities and establishes appropriate risk management plans. In the case that performing a SR would result in an unacceptable level of risk, NextEra would reschedule the activity. Similarly, if a SR that is normally performed during shutdown would adversely affect safety if performed during plant operation; NextEra would continue to perform the SR during shutdown.

The proposed change is consistent with NUREG-1431, Standard Technical Specifications - Westinghouse Plants [Reference 2]. The SRs in NUREG-1431 that correspond to the Seabrook SRs that verify automatic actuation of pumps and valves do not restrict performance of the SRs to shutdown conditions.

NextEra will continue to evaluate the risk impact of performing SRs as required by 10 CFR 50.65(a)(4), and SRs previously performed during shutdown will be performed during operation only when it is safe to do so. The proposed change will eliminate the need to perform duplicate testing and by reducing the testing required during shutdown periods, will improve the availability of systems important to maintaining the plant in a safe shutdown condition.

<u>Proposed changes to administrative controls</u>

<u>TS 6.1, Responsibility</u>

TS 6.1.1 assigns the Station Director responsibility for overall station operation. NextEra proposes to replace the plant-specific position title with the generic title "plant manager." The proposed change is consistent with NUREG-1431, which uses "plant manager" rather than a plant-specific position title, and it does not change or reassign the responsibility for overall operation of the station. Administrative in nature, the proposed change will reduce burden on NextEra and NRC resources by eliminating the need to process a license amendment as the result of an inconsequential change to the plant-specific title of the position that is assigned overall responsibility for operation of the station.

TS 6.1.2 assigns responsibility for the control room command function and directs issuance of an annual management directive regarding this responsibility. NextEra proposes to revise TS 6.1.2 consistent with Standard TS (STS) 5.1.2 in NUREG-1431 and with the requirement currently contained in Seabrook TS Table 6.2-1 regarding the control room command function. The change also deletes the requirement to issue a management directive regarding the control room command function because the directive is redundant to the requirement imposed by the proposed change to TS 6.1.2 and, therefore, is unnecessary.

The proposed change to TS 6.1.2 does not involve any technical changes and does not modify the responsibility for the control room command function. The change is only administrative in nature since it makes editorial changes consistent with NUREG-1431 and combines the current requirements in TS 6.1.2 and Table 6.2-1. Therefore, NextEra concludes that the proposed changes are acceptable.

TS 6.2.1, Offsite and Onsite Organizations

The proposed change revises TS 6.2.1.b to replace the plant-specific position title with the generic title "plant manager" as discussed above for TS 6.1.1. Similarly, the change replaces the plant-specific position title Site Vice President with "a specified corporate officer." Consistent with NUREG-1431, which uses "a specified corporate officer" rather than a plant-specific position title, the change does not alter or reassign the responsibility for overall plant nuclear safety. Administrative in nature, the proposed change will reduce burden on NextEra and NRC resources by eliminating the need to process a license amendment as the result of an inconsequential change to the plant-specific title of the position that is assigned responsibility for overall plant nuclear safety.

TS 6.2.2, Plant Staff and Table 6.2-1, Minimum Shift Crew Composition

NextEra proposes to delete TS Table 6.2-1, Minimum Shift Crew Composition, and revise TS 6.2.2, Plant Staff, accordingly. NRC regulation 10 CFR 50.54, Conditions of licenses, stipulates the requirements for staffing of licensed operators; therefore, NextEra proposes to eliminate TS Table 6.2-1, which unnecessarily duplicates the regulatory requirements. Current TS 6.2.2 is revised to remove the reference to Table 6.2-1, and reflect the contents of STS 5.2.2, Unit Staff, in NUREG-1431.

The proposed changes, which align the Seabrook TS with NUREG-1431, are consistent with 10 CFR 50.54(m). The changes are administrative in nature and have no technical implications with respect to the station organization, responsibilities, or unit staffing requirements. Therefore, NextEra concludes that the changes are acceptable.

TS 6.2.4, Shift Technical Advisor (STA) Function

The proposed amendment deletes TS 6.2.4 because the STA requirement is included as item d in the proposed change to TS 6.2.2. This change is administrative in nature because it consolidates in the revision to TS 6.2.2 the requirements for the STA currently provided in TS 6.2.4.1 and Table 6.2-1. The change does not alter any requirements or responsibilities for the STA position; therefore, the change is acceptable. .

TS 6.12, Process Control Program; TS 6.13, Offsite Dose Calculation Manual; and TS 6.14, Major Changes to Liquid, Gaseous, and Solid Radwaste Treatment Systems

The prosed change revises TS 6.12, TS 6.13, and TS 6.14 by replacing the plant-specific position title Station Director with the generic title "plant manager." The proposed change is consistent with NUREG-1431, which uses "plant manager" rather than a plant-specific position title. Administrative in nature, the proposed change will reduce burden on NextEra and NRC resources by eliminating the need to process a license amendment as the result of an inconsequential change to the plant-specific title of the position that is assigned overall responsibility for operation of the station.

These TS are also modified to replace the reference to SORC (Station Operation Review Committee) with the title Onsite Review Group. This change is the result of a title change to the review organization. The change is only administrative in nature and does not modify the composition or responsibilities of the review organization.

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements/Criteria

- 10 CFR 50.65, Requirements for monitoring the effectiveness of maintenance at nuclear power plants, requires in part that before performing maintenance activities (including but not limited to surveillance, post-maintenance testing, and corrective and preventive maintenance), the licensee shall assess and manage the increase in risk that may result from the proposed maintenance activities.
- 10 CFR 50.36, Technical specifications, requires that the technical specifications include surveillance requirements to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met.

The proposed change is consistent with the above regulatory requirements and criteria.

4.2 Precedent

The NRC has previously issued license amendments for the Shearon Harris Nuclear Power Plant, Unit No. 1 [Reference 3], Beaver Valley Power Station, Unit No. 2 [Reference 4], and D.C. Cook [Reference 5] to remove similar language on performing surveillance activities during shutdown. In each of these cases, the NRC staff concluded that removing the requirement to perform tests during shutdown was consistent with the STSs (in these cases, the STSs in NUREG-1431 for Westinghouse plants) and GL 91-04. During shutdown or at power, the licensees manage overall maintenance risk consistent with 10 CFR 50.65(a)(4).

The NRC issued Amendments 217 and 167 for St. Lucie Units 1 and 2, respectively [Reference 6]. The amendments revised the St. Lucie TS to align with NUREG-1432, Revision 4, Combustion Engineering Plants Standard TSs language describing the Administrative Controls requirements for Responsibility and Organization. The amendments deleted TS Table 6.2-1, Minimum Shift Crew Composition; and TS 6.2.3, Shift Technical Advisor Function; and revised TS 6.2.2, Plant Staff, to be consistent with NUREG-1432. The

NRC determined the proposed changes were acceptable because they were in accordance with 10 CFR 50.54(m) and made no changes to the station organization, responsibilities, or unit staffing requirements.

4.3 No Significant Hazards Consideration

The proposed change revises certain 18-month technical specifications (TS) surveillance requirements (SR) to eliminate the condition that testing be conducted "during shutdown" and revises the administrative TS regarding plant staff and responsibilities.

NextEra has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10CFR50.92, "Issuance of Amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The technical specification (TS) surveillance requirements and administrative controls associated with the proposed changes to the TS are not initiators of any accidents previously evaluated, so the probability of accidents previously evaluated is unaffected by the proposed changes. The proposed change does not alter the design, function, or operation of any plant structure, system, or component (SSC). The capability of any operable TS-required SSC to perform its specified safety function is not impacted by the proposed change. As a result, the outcomes of accidents previously evaluated are unaffected. Therefore, the proposed changes do not result in a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any previously evaluated?

   Response: No

   The proposed change does not challenge the integrity or performance of any safety-related systems. No plant equipment is installed or removed, and the changes do not alter the design, physical configuration, or method of operation of any plant SSC.

   No physical changes are made to the plant, so no new causal mechanisms are introduced. Therefore, the proposed changes to the TS do not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not challenge the integrity or performance of any safety-related systems. No plant equipment is installed or removed, and the changes do not alter the design, physical configuration, or method of operation of any plant SSC. No physical changes are made to the plant, so no new causal mechanisms are introduced. Therefore, the proposed changes to the TS do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in the margin of safety?

   Response: No.

   The ability of any operable SSC to perform its designated safety function is unaffected by the proposed changes. The proposed changes do not alter any safety analyses assumptions, safety limits, limiting safety system settings, or method of operating the plant. The changes do not adversely affect plant operating margins or the reliability of equipment credited in the safety analyses. With the proposed change, each DC electrical train remains fully capable of performing its safety function. Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based on the above, NextEra concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c) and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

In conclusion, based on the considerations 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 general 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 forth 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. Generic Letter 91-04, Changes in Technical Specifications Surveillance Intervals to Accommodate a 24-Month Fuel Cycle, April 2, 1991
2. NUREG-1431, Standard Technical Specifications Westinghouse Plants, Revision 4.0, April 2012
3. NRC letter 'Issuance of Amendment No. 77 to Facility Operating License No. NPF-63 Regarding Deletion of Shutdown Requirement from Selected Surveillances for Shearon Harris Nuclear Power Plant, Unit 1 (TAC No. M98271)," April 14, 1998 (ML020590045)
4. NRC letter "Beaver Valley Power Station Unit 2 - Issuance of Amendment Re: Revision of 18 Months Surveillance Criteria for Containment Relay Testing (TAC No. MA9865)," October 13, 2000 (ML003750330)
5. NRC letter "Donald C. Cook Nuclear Power Plant, Units 1 and 2 - Issuance of Amendments (TAC Nos. MB5695 and MB5696)," April 22, 2003 (ML030500450)
6. NRC letter "St. Lucie Plant, Unit Nos. 1 and 2 - Issuance of Amendments Regarding Standardization of Administrative Controls - Responsibility and Organization (TAC Nos. MF2496 and MF2497)," February 7, 2014 (ML14016A248)

Attachment 1

Markup of the Technical Specifications

SURVEILLANCE REQUIREMENTS

4.5.2 (Continued)

d. In accordance with the Surveillance Frequency Control Program by:

1) Verifying automatic interlock action of the RHR system from the Reactor Coolant System to ensure that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 440 psig, the interlocks prevent the valves from being opened.

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. In accordance with the Surveillance Frequency Control Program~~, during shutdown,~~ by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on (Safety Injection actuation and Automatic Switchover to Containment Sump) test signals, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying OPERABILITY of each pump when tested in accordance with the INSERVICE TESTING PROGRAM:

1) Centrifugal charging pump;

2) Safety Injection pump; and

3) RHR pump.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST* and automatically transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:
   1) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position**, and
   2) Verifying Containment Spray locations susceptible to gas accumulation are sufficiently filled with water.

b. By verifying OPERABILITY of each pump when tested in accordance with the INSERVICE TESTING PROGRAM;

c. In accordance with the Surveillance Frequency Control Program during ~~shutdown,~~ by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal, and
   2) Verifying that each spray pump starts automatically on a Containment Pressure-Hi-3 test signal.

d. By verifying each spray nozzle is unobstructed following activities that could result in nozzle blockage.

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*In MODE 4, when the Residual Heat Removal System is in operation, an OPERABLE flow path is one that is capable of taking suction from the refueling water storage tank upon being manually realigned.

**Not required to be met for system vent flow paths opened under administrative control.

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank containing a volume of between 9420 and 9650 gallons of between 19 and 21% by weight NaOH solution, and

b. Two gravity feed paths each capable of adding NaOH solution from the chemical additive tank to the Refueling Water Storage Tank.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;

b. In accordance with the Surveillance Frequency Control Program by:

1) Verifying the contained solution volume in the tank, and

2) Verifying the concentration of the NaOH solution by chemical analysis.

c. In accordance with the Surveillance Frequency Control Program ~~during shutdown,~~ by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE*.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Not used

4.6.3.2 Each containment isolation valve shall be demonstrated OPERABLE ~~during shutdown~~ in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" Isolation valve actuates to its isolation position,

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" Isolation valve actuates to its isolation position, and

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

1) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;

2) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and

3) Verifying that valves FW-156 and FW-163 are OPERABLE for alignment of the startup feedwater pump to the emergency feedwater header.

b. In accordance with the Surveillance Frequency Control Program by verifying the following pumps develop the required discharge pressure and flow as specified in the Technical Requirements Manual:

1) The motor-driven emergency feedwater pump;

2) The steam turbine-driven emergency feedwater pump when the secondary steam supply pressure is greater than 500 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

3) The startup feedwater pump.

c. In accordance with the Surveillance Frequency Control Program ~~during shutdown~~ by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Emergency Feedwater System Actuation test signal;

2) Verifying that each emergency feedwater pump starts as designed automatically upon receipt of an Emergency Feedwater Actuation System test signal;

3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent primary component cooling water loops shall be OPERABLE, including one OPERABLE pump in each loop.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one primary component cooling water (PCCW) loop inoperable, restore the required primary component cooling water loop to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two primary component cooling water loops shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. In accordance with the Surveillance Frequency Control Program ~~during shutdown,~~ by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation signal.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. In accordance with the Surveillance Frequency Control Program ~~during shutdown,~~ by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal.

4.7.4.2 Each service water cooling tower loop shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. In accordance with the Surveillance Frequency Control Program ~~during shutdown,~~ by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal,

2) Each automatic valve in the flowpath actuates to its correct position on a Tower Actuation (TA) test signal and

3) Each service water cooling tower pump starts automatically on a TA signal.

4.7.4.3 The service water pumphouse shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the water level to be at or above 25.1' (-15.9' Mean Sea Level).

4.7.4.4 The mechanical draft cooling tower shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying the water in the mechanical draft cooling tower basin to be at a level of greater than or equal to 42.15* feet.

b. In accordance with the Surveillance Frequency Control Program by verifying that the water in the cooling tower basin to be at a bulk average temperature of less than or equal to 70°F.

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*With the cooling tower in operation with valves aligned for tunnel heat treatment, the tower basin level shall be maintained at greater than or equal to 40.55 feet.

4.8.1.1.2 (Continued)

14) Simulating a Tower Actuation (TA) signal while the diesel generator is loaded with the permanently connected loads and auto-connected emergency (accident) loads, and verifying that the service water pump automatically trips, and that the cooling tower pump automatically starts. After energization the steady state voltage and frequency of the emergency buses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz; and

15) While diesel generator 1A is loaded with the permanently connected loads and auto-connected emergency (accident) loads, manually connect the 1500 hp startup feedwater pump to 4160-volt bus E5. After energization the steady-state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz.

g. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously from standby condition~~, during shutdown,~~ and verifying that both diesel generators achieve:

1) A generator voltage and frequency greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the start signal, and

2) A steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.

INSERT 6.1.2

The Shift Manager (SM) shall be responsible for the control room command function. During any absence of the SM from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SM from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

INSERT 6.2.2

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.2.a and 6.2.2.d for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. An individual (Shift Technical Advisor (STA)) shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. The STA position shall be manned in MODES 1, 2, 3, and 4 unless the SM or the individual with a Senior Operator license meets the qualifications for the STA.

e. While the unit is in MODE 1, 2, 3 or 4, a licensed senior operator, either the SM or SRO, shall be on shift having had at least 6 months of hot operating experience.

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

plant manager

6.1.1 The ~~Station Director~~ shall be responsible for overall station operation and shall delegate in writing the succession to this responsibility during his absence.

INSERT 6.1.2

6.1.2 ~~The Shift Manager (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Site Vice President shall be reissued to all station personnel on an annual basis.~~

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions for departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR and updated in accordance with the requirements of 10 CFR 50.71.

plant manager

b. The ~~Station Director~~ shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

A specified corporate officer

c. ~~The Site Vice President~~ shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

INSERT 6.2.2

6.2.2 STATION STAFF

~~a.~~ ~~Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1;~~

~~b.~~ ~~At least one licensed Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;~~

~~c.~~ ~~A Health Physics Technician* shall be on site when fuel is in the reactor;~~

~~d.~~ ~~All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation; and~~

~~e.~~ ~~Deleted~~

f. The Operations Manager shall meet one of the following:

1. Hold a senior operator license,
2. Have held a senior operator license on a similar unit (PWR), or
3. Have been certified for equivalent senior operator knowledge.

g. The Assistant Operations Manager shall hold a senior reactor operator license.

~~*The Health Physics Technician may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.~~

DELETED

TABLE 6.2-1

~~MINIMUM SHIFT CREW COMPOSITION~~[~~(1)~~]

~~POSITION~~	~~NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION~~	
	~~MODE 1, 2, 3, or 4~~	~~MODE 5 or 6~~
~~SM~~[~~(2,4)~~]	~~1~~	~~1~~
~~SRO~~[~~(4)~~]	~~1~~	~~None~~
~~RO~~	~~2~~	~~1~~
~~NSO~~	~~2~~	~~1~~
~~STA~~	~~1~~[~~(3)~~]	~~None~~

~~SM — Shift Manager with a Senior Reactor Operator license on Unit 1~~
~~SRO — Individual with a Senior Reactor Operator license on Unit 1~~
~~RO — Individual with an Operator license on Unit 1~~
~~NSO — Nuclear Systems Operator~~
~~STA — Shift Technical Advisor~~

~~TABLE NOTATIONS~~

~~(1)~~ ~~The shift crew composition may be one less than the minimum requirements of Table 6.2 1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2 1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewperson being late or absent.~~

~~(2)~~ ~~During any absence of the Shift Manager from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Shift Manager from the control room while the unit is in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.~~

~~(3)~~ ~~The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Shift Manager or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.~~

~~(4)~~ ~~While the unit is in MODE 1, 2, 3 or 4, a licensed senior operator, either the SM or SRO, shall be on shift having had at least 6 months of hot operating experience.~~

6.2.3 (THIS SPECIFICATION NUMBER IS NOT USED)

6.2.4 ~~SHIFT TECHNICAL ADVISOR~~ THIS SPECIFICATION NUMBER IS NOT USED

~~6.2.4.1 The Shift Technical Advisor shall provide advisory technical support to the Control Room Commander in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the station.~~

6.3 (THIS SPECIFICATION NUMBER IS NOT USED)

6.4 (THIS SPECIFICATION NUMBER IS NOT USED)

6.5 (THIS SPECIFICATION NUMBER IS NOT USED)

6.6 (THIS SPECIFICATION NUMBER IS NOT USED)

HIGH RADIATION AREA

6.11.1 (Continued)

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.12 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Assurance Program (OQAP). This documentation shall contain:

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

Onsite Review Group

b. Shall become effective after review and acceptance by the ~~SORC~~ and approval of the ~~Station Director~~.

plant manager

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Control Program (OQAP). This documentation shall contain:

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the ~~SORC~~ and the approval of the ~~Station Director.~~

Onsite Review Group

plant manager

OFFSITE DOSE CALCULATION MANUAL (ODCM)

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and each affected page shall indicate the revision number the change was implemented.

6.14 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.14.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the ~~SORC~~. The discussion of each change shall contain:

Onsite Review Group

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

*Licensees may choose to submit the information called for in this Specification as part of the FSAR update, pursuant to 10 CFR 50.71.

6.14.1 (Continued)

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the ~~SORC~~. Onsite Review Group

b. Shall become effective upon review and acceptance by the ~~SORC~~.

6.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.15% of primary containment air weight per day.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

Attachment 2

Retyped Technical Specifications Pages

SURVEILLANCE REQUIREMENTS

4.5.2 (Continued)

d. In accordance with the Surveillance Frequency Control Program by:

1) Verifying automatic interlock action of the RHR system from the Reactor Coolant System to ensure that with a simulated or actual Reactor Coolant System pressure signal greater than or equal to 440 psig, the interlocks prevent the valves from being opened.

2) A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or abnormal corrosion.

e. In accordance with the Surveillance Frequency Control Program by:

1) Verifying that each automatic valve in the flow path actuates to its correct position on (Safety Injection actuation and Automatic Switchover to Containment Sump) test signals, and

2) Verifying that each of the following pumps start automatically upon receipt of a Safety Injection actuation test signal:

a) Centrifugal charging pump,

b) Safety Injection pump, and

c) RHR pump.

f. By verifying OPERABILITY of each pump when tested in accordance with the INSERVICE TESTING PROGRAM:

1) Centrifugal charging pump;

2) Safety Injection pump; and

3) RHR pump.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST* and automatically transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one Containment Spray System inoperable, restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable Spray System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:
   1) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position**, and
   2) Verifying Containment Spray locations susceptible to gas accumulation are sufficiently filled with water.

b. By verifying OPERABILITY of each pump when tested in accordance with the INSERVICE TESTING PROGRAM;

c. In accordance with the Surveillance Frequency Control Program by:

   1) Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal, and
   2) Verifying that each spray pump starts automatically on a Containment Pressure-Hi-3 test signal.

d. By verifying each spray nozzle is unobstructed following activities that could result in nozzle blockage.

*In MODE 4, when the Residual Heat Removal System is in operation, an OPERABLE flow path is one that is capable of taking suction from the refueling water storage tank upon being manually realigned.

**Not required to be met for system vent flow paths opened under administrative control.

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Spray Additive System shall be OPERABLE with:

a. A spray additive tank containing a volume of between 9420 and 9650 gallons of between 19 and 21% by weight NaOH solution, and

b. Two gravity feed paths each capable of adding NaOH solution from the chemical additive tank to the Refueling Water Storage Tank.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With the Spray Additive System inoperable, restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the Spray Additive System to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 The Spray Additive System shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;

b. In accordance with the Surveillance Frequency Control Program by:

1) Verifying the contained solution volume in the tank, and

2) Verifying the concentration of the NaOH solution by chemical analysis.

c. In accordance with the Surveillance Frequency Control Program by verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure-Hi-3 test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 Each containment isolation valve shall be OPERABLE*.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one or more of the isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 Not used

4.6.3.2 Each containment isolation valve shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by:

a. Verifying that on a Phase "A" Isolation test signal, each Phase "A" Isolation valve actuates to its isolation position,

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" Isolation valve actuates to its isolation position, and

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by:

1) Verifying that each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in its correct position;

2) Verifying that each automatic valve in the flow path is in the fully open position whenever the Auxiliary Feedwater System is placed in automatic control or when above 10% RATED THERMAL POWER; and

3) Verifying that valves FW-156 and FW-163 are OPERABLE for alignment of the startup feedwater pump to the emergency feedwater header.

b. In accordance with the Surveillance Frequency Control Program by verifying the following pumps develop the required discharge pressure and flow as specified in the Technical Requirements Manual:

1) The motor-driven emergency feedwater pump;

2) The steam turbine-driven emergency feedwater pump when the secondary steam supply pressure is greater than 500 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3;

3) The startup feedwater pump.

c. In accordance with the Surveillance Frequency Control Program by:

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of an Emergency Feedwater System Actuation test signal;

2) Verifying that each emergency feedwater pump starts as designed automatically upon receipt of an Emergency Feedwater Actuation System test signal;

3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent primary component cooling water loops shall be OPERABLE, including one OPERABLE pump in each loop.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

With one primary component cooling water (PCCW) loop inoperable, restore the required primary component cooling water loop to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two primary component cooling water loops shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. In accordance with the Surveillance Frequency Control Program by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation signal.

SURVEILLANCE REQUIREMENTS

4.7.4.1 Each service water loop shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. In accordance with the Surveillance Frequency Control Program by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal.

4.7.4.2 Each service water cooling tower loop shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that each valve (manual, power-operated, or automatic) servicing safety related equipment that is not locked, sealed, or otherwise secured in position is in its correct position; and

b. In accordance with the Surveillance Frequency Control Program by verifying that:

1) Each automatic valve servicing safety-related equipment actuates to its correct position on its associated Engineered Safety Feature actuation test signal,

2) Each automatic valve in the flowpath actuates to its correct position on a Tower Actuation (TA) test signal and

3) Each service water cooling tower pump starts automatically on a TA signal.

4.7.4.3 The service water pumphouse shall be demonstrated OPERABLE in accordance with the Surveillance Frequency Control Program by verifying the water level to be at or above 25.1' (-15.9' Mean Sea Level).

4.7.4.4 The mechanical draft cooling tower shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying the water in the mechanical draft cooling tower basin to be at a level of greater than or equal to 42.15* feet.

b. In accordance with the Surveillance Frequency Control Program by verifying that the water in the cooling tower basin to be at a bulk average temperature of less than or equal to 70°F.

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*With the cooling tower in operation with valves aligned for tunnel heat treatment, the tower basin level shall be maintained at greater than or equal to 40.55 feet.

4.8.1.1.2 (Continued)

14) Simulating a Tower Actuation (TA) signal while the diesel generator is loaded with the permanently connected loads and auto-connected emergency (accident) loads, and verifying that the service water pump automatically trips, and that the cooling tower pump automatically starts. After energization the steady state voltage and frequency of the emergency buses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz; and

15) While diesel generator 1A is loaded with the permanently connected loads and auto-connected emergency (accident) loads, manually connect the 1500 hp startup feedwater pump to 4160-volt bus E5. After energization the steady-state voltage and frequency of the emergency bus shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz.

g. In accordance with the Surveillance Frequency Control Program or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously from standby condition and verifying that both diesel generators achieve:

1) A generator voltage and frequency greater than or equal to 3740 volts and 58.8 Hz within 10 seconds after the start signal, and

2) A steady-state generator voltage and frequency of 4160 ± 420 volts and 60 ± 1.2 Hz.

6.1 RESPONSIBILITY

6.1.1 The plant manager shall be responsible for overall station operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Manager (SM) shall be responsible for the control room command function. During any absence of the SM from the control room while the unit is in MODE 1, 2, 3, or 4, an individual with an active Senior Reactor Operator (SRO) license shall be designated to assume the control room command function. During any absence of the SM from the control room while the unit is in MODE 5 or 6, an individual with an active SRO license or Reactor Operator license shall be designated to assume the control room command function.

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be established and defined for the highest management levels through intermediate levels to and including all operating organization positions. These relationships shall be documented and updated, as appropriate, in the form of organization charts, functional descriptions for departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the FSAR and updated in accordance with the requirements of 10 CFR 50.71.

b. The plant manager shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

d. The individuals who train the operating staff and those who carry out health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 STATION STAFF

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator shall be assigned for each control room from which a reactor is operating in MODES 1, 2, 3, or 4.

b. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.2.a and 6.2.2.d for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements.

c. A radiation protection technician shall be on site when fuel is in the reactor. The position may be vacant for not more than 2 hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

d. An individual (Shift Technical Advisor (STA)) shall provide advisory technical support to the unit operations shift crew in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. This individual shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. The STA position shall be manned in MODES 1, 2, 3, and 4 unless the SM or the individual with a Senior Operator license meets the qualifications for the STA.

e. While the unit is in MODE 1, 2, 3 or 4, a licensed senior operator, either the SM or SRO, shall be on shift having had at least 6 months of hot operating experience.

f. The Operations Manager shall meet one of the following:

   1. Hold a senior operator license,
   2. Have held a senior operator license on a similar unit (PWR), or
   3. Have been certified for equivalent senior operator knowledge.

g. The Assistant Operations Manager shall hold a senior reactor operator license.

TABLE 6.2-1

DELETED

6.2.3 (THIS SPECIFICATION NUMBER IS NOT USED)

6.2.4 (THIS SPECIFICATION NUMBER IS NOT USED)

6.3 (THIS SPECIFICATION NUMBER IS NOT USED)

6.4 (THIS SPECIFICATION NUMBER IS NOT USED)

6.5 (THIS SPECIFICATION NUMBER IS NOT USED)

6.6 (THIS SPECIFICATION NUMBER IS NOT USED)

HIGH RADIATION AREA

6.11.1 (Continued)

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.12 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Assurance Program (OQAP). This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the Onsite Review Group and approval of the plant manager.

6.13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by the Operational Quality Control Program (OQAP). This documentation shall contain:

   1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b. Shall become effective after review and acceptance by the Onsite Review Group and the approval of the plant manager.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

c. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and each affected page shall indicate the revision number the change was implemented.

6.14 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*

6.14.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):

a. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Onsite Review Group. The discussion of each change shall contain:

1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;

4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the License application and amendments thereto;

5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;

*Licensees may choose to submit the information called for in this Specification as part of the FSAR update, pursuant to 10 CFR 50.71.

6.14.1 (Continued)

6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made;

7) An estimate of the exposure to plant operating personnel as a result of the change; and

8) Documentation of the fact that the change was reviewed and found acceptable by the Onsite Review Group.

b. Shall become effective upon review and acceptance by the Onsite Review Group.

6.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with Nuclear Energy Institute (NEI) topical report NEI 94-01, Revision 3-A, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," and conditions and limitations specified in NEI 94-01, Revision 2-A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 49.6 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.15% of primary containment air weight per day.

The provisions of SR 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.