NEXTera® ENERGY SEABROOK

December 14, 2017
Docket No. 50-443
SBK-L-17208

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

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

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

REMOVE	**INSERT**
B 3/4 0-3c, 0-5, 0-8	B 3/4 0-3c, 0-5, 05a, 0-8
B 3/4 4-1	B 3/4 4-1
B 3/4 4-18, 4-19, 4-20, 4-23, 4-25, 4-26, 4- 27	B 3/4 4-18, 4-19, 4-20, 4-23, 4-25, 4-26, 4-26a, 4-27
B 3/4 7-19	B 3/4 7-19
B 3/4 8-20	B 3/4 8-20
B 3/4 9-4	B 3/4 9-4

Should you have any questions concerning this submittal, please contact me at (603) 773-7932.

Sincerely,

NextEra Energy Seabrook, LLC

Kenneth Browne
Licensing Manager

cc: D. Dorman, NRC Region I Administrator
J. Poole, NRC Project Manager, Project Directorate I-2
P. Cataldo, NRC Senior Resident Inspector

Enclosure to SBK-L-17208

a. The OPERABILITY of the equipment being returned to service; or

b. The OPERABILITY of other equipment.

The administrative controls ensure the time the equipment is returned to service in conflict with the requirements of the ACTIONS is limited to the time absolutely necessary to perform the allowed required testing. This Specification does not provide time to perform any other preventive or corrective maintenance.

An example of demonstrating the OPERABILITY of the equipment being returned to service is reopening a containment isolation valve that has been closed to comply with Required Actions and must be reopened to perform the required testing.

An example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to prevent the trip function from occurring during the performance of required testing on another channel in the other trip system. A similar example of demonstrating the OPERABILITY of other equipment is taking an inoperable channel or trip system out of the tripped condition to permit the logic to function and indicate the appropriate response during the performance of required testing on another channel in the same trip system.

<u>Specifications 4.0.1 through 4.0.5</u> establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration, or inspection 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 of operation will be met."

Specifications 4.0.2 and 4.0.3 apply in Chapter 6 only when invoked by a Chapter 6 Specification.

will allow operation to proceed to a MODE or other specified condition where other necessary post maintenance tests can be completed. An example of this process:

Emergency feedwater (EFW) pump turbine maintenance during refueling that requires testing at steam pressure > 500 psig. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

When a Section 6.7, "Procedures and Programs," specification states that the provisions of Specification 4.0.2 are applicable, a 25% extension of the testing interval, whether stated in the specification or incorporated by reference, is permitted. The exceptions to Specification 4.0.2 are those Surveillances for which the 25% extension of the interval specified in the frequency does not apply. These exceptions are stated in the individual specifications. The requirements of regulations take precedence over the TS. Examples of where Specification 4.0.2 does not apply are the Containment Leakage Rate Testing Program (Specification 6.15) required by 10 CFR 50, Appendix J, and the inservice testing of pumps and valves in accordance with applicable American Society of Mechanical Engineers Operation and Maintenance Code, as required by 10 CFR 50.55a. These programs establish testing requirements and frequencies in accordance with the requirements of regulations. The TS cannot, in and of themselves, extend a test interval specified in the regulations directly or by reference.

Specification 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside the specified limits when a Surveillance has not been completed within the specified surveillance interval. A delay period of up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater, applies from the point in time that it is discovered that the Surveillance has not been performed in accordance with Specification 4.0.2, and not at the time that the specified frequency was not met.

When a Section 6.7, "Procedures and Programs," specification states that the provisions of Specification 4.0.3 are applicable, it permits the flexibility to defer declaring the testing requirement not met in accordance with Specification 4.0.3 when the testing has not been completed within the testing interval (including the allowance of Specification 4.0.2 if invoked by the Section 6.7 specification).

This delay period provides adequate time to complete Surveillances that have been missed. This delay period permits the completion of a Surveillance before complying with ACTION requirements or other remedial measures that might preclude completion of the Surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the Surveillance, the safety significance of the delay in completing the required Surveillance, and the recognition that the most probable result of any particular Surveillance being performed is the verification of conformance with the requirements.

When a Surveillance with a surveillance interval based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed

or outside specified limits. Considering the equipment OPERABLE or the variable within specified limits upon entering the MODE or condition specified in the Applicability statement with a surveillance requirement exempt from the provision of SR 4.0.4 ensures compliance with LCO 3.0.4. If the surveillance fails within the 24-hour period, then the equipment is inoperable or the variable is outside the specified limits and the applicable ACTIONS begin immediately upon the failure of the surveillance test. Similarly, if the testing is not completed within 24 hours, the equipment is inoperable or the variable is outside specified limits and the ACTION requirements are immediately applicable upon expiration of the 24 hours.

Specification 4.0.5 establishes the requirement that inservice inspection of ASME Code Class 1, 2, and 3 components shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code including applicable Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection required by Section XI of the ASME Boiler and Pressure Vessel Code including applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection activities.

The requirements of Specification 4.0.4 to perform surveillance activities before entry into an OPERATIONAL MODE or other specified condition takes precedence over the ASME OM Code provision which allows pumps that can only be tested during plant operation to be tested within 1 week following plant startup.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

An OPERABLE reactor coolant system loop consists of an OPERABLE reactor coolant pump and an OPERABLE steam generator.

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above the safety limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by placing the Control Rod Drive System in a condition incapable of rod withdrawal. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE. Managing of gas voids is important to RHR System OPERABILITY.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE. Managing of gas voids is important to RHR System OPERABILITY.

A Reactor Coolant "loops filled" condition is defined as follows: (1) Having pressurizer level greater than or equal to 55% if the pressurizer does not have a bubble, and greater than or equal to 17% when there is a bubble in the pressurizer. (2) Having the air and non-condensables evacuated from the Reactor Coolant System by either operating each reactor coolant pump for a short duration to sweep air from the Steam Generator U-tubes into the upper head area of the reactor vessel, or removing the air from the Reactor Coolant System via an RCS evacuation skid, and (3) Having vented the upper head area of the reactor vessel if the pressurizer does not have a bubble. (4) Having the Reactor Coolant System not vented, or if vented capable of isolating the vent paths within the time to boil.

Draining the RCS to a level that is lower than the stated limits (55% with no bubble or 17% with a bubble) and subsequently re-establishing the required levels does not preclude establishing the "loops filled" condition as long as the level is not dropped to the point at which additional air can be introduced into the steam generator tubes. If no additional air is introduced into the steam generator tubes, the refill of the RCS re-establishes the conditions that existed prior to the draining. Engineering Evaluation EE-08-012 demonstrates that, with the maximum amount of air/gas available from reactor coolant system sources in Mode 5 present in the steam generator tubes, any two steam generators provide adequate decay heat removal via natural circulation approximately 12 hours after shutdown.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, Reference (1):

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

    a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

    b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

Operation within the limits of the appropriate heatup and cooldown curves assures the integrity of the reactor vessel's ferritic material against fracture induced by combined thermal and pressure stresses. With the combination of pressure and temperature maintained below and to the right of the limit lines in TS Figures 3.4-2 and 3.4-3, plant Operation, including vacuum fill of the RCS, meets the LCO of TS 3.4.9.1. The limit curves are applicable during RCS vacuum fill. As the reactor vessel is subjected to increasing fluence, the toughness of the limiting beltline region material continues to diminish, and consequently, even more restrictive pressure/temperature (P/T) limits must be maintained. Each P/T limit curve defines an acceptable region for normal operation during heatup or cooldown maneuvering as pressure and temperature indications are monitored to ensure that operation is within the allowable region. A heatup or cooldown is defined as a temperature change of greater than or equal to 10°F in any one-hour period.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The P/T limits have been established in accordance with the requirements of ASME Boiler and Pressure Vessel Code Section XI, Appendix G, and the additional requirements of 10CFR50 Appendix G, Reference (4). The heatup and cooldown P/T limit curves for normal operation, Figures 3.4-2 and 3.4-3 respectively, are valid for a service period of 55 effective full power years (EFPY). The technical justification and methodologies utilized in their development are documented generically in WCAP-14040-A, Revision 4, Reference (3), and specifically for Seabrook Unit 1 in WCAP-17441-NP, Reference (5), and LTR-AMLRS-11-50, Reference (8). The P/T curves were generated based on the latest available reactor vessel information and latest calculated fluences.

Heatup and Cooldown limit curves are calculated using the adjusted RT_{NDT} (reference nil-ductility temperature) corresponding to the limiting beltline region material of the reactor vessel. The adjusted RT_{NDT} of the limiting material in the core region of the reactor vessel is determined by using the unirradiated reactor vessel material fracture toughness properties, estimating the radiation-induced ΔRT_{NDT}, and adding a margin. RT_{NDT} increases as the material is exposed to fast-neutron radiation. Therefore, to find the most limiting RT_{NDT} at any time period in the reactor's life, ΔRT_{NDT} due to the radiation exposure associated with that time period must be added to the unirradiated RT_{NDT} (IRT_{NDT}). The extent of the shift in RT_{NDT} is enhanced by certain chemical elements (such as copper and nickel) present in reactor vessel steels. The Nuclear Regulatory Commission (NRC) has published a method for predicting radiation embrittlement in Regulatory Guide 1.99, Revision 2, Reference (6). Regulatory Guide 1.99, Revision 2, is used for the calculation of Adjusted Reference Temperature (ART) values (IRT_{NDT} + ΔRT_{NDT} + margins for uncertainties) at the 1/4T and 3/4T locations, where T is the thickness of the vessel at the beltline region.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, best estimate copper and nickel content of the limiting beltline material, can be predicted using surveillance capsule data and the value of ΔRT_{NDT} computed by Regulatory Guide 1.99, Revision 2. Surveillance capsule data, documented in Reference (7), is available for three capsules (Capsules U, Y, and V) having already been removed from the reactor vessel. This surveillance capsule data was used to calculate chemistry factor (CF) values per Position 2.1 of Regulatory Guide 1.99, Revision 2. It also noted that Reference (7) concluded that all the surveillance data was credible as the beltline material was behaving as empirically predicted. The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} but does not include adjustments for possible errors in the pressure and temperature sensing instruments.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The results from the material surveillance program were evaluated according to ASTM E185. Capsules U, Y, and V were removed in accordance with the requirements of ASTM E185-82 and 10CFR50, Appendix H. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens were used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. The fluence values used to determine the CFs are the calculated fluence values at the surveillance capsule locations. The calculated fluence values were used for all cases. All calculated fluence values (capsule and projections) are documented in References (5) and (7). These fluences were calculated using the ENDF/B-VI scattering cross-section data set. The measured ΔRT_{NDT} values for the weld data were adjusted for chemistry using the ratio procedure given in Position 2.1 of Regulatory Guide 1.99, Revision 2. Since the ratio is equal to 1.0, the calculations are not affected by the ratio procedure.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{Ic}, for the metal temperature at that time. K_{Ic} is obtained from the reference fracture toughness curve, defined in ASME Boiler and Pressure Vessel Code Section XI, Appendix G, Reference (1). The K_{Ic} curve is given by the following equation:

$$K_{Ic} = 33.2 + 20.734 * e^{[0.02(T - RT_{NDT})]} \quad (1)$$

where,

K_{Ic} = reference stress intensity factor as a function of the metal temperature T and the metal reference nil-ductility temperature RT_{NDT}

This K_{Ic} curve is based on the lower bound of static critical K_I values measured as a function of temperature on specimens of SA-533 Grade B Class 1, SA-508-1, SA-508-2, and SA-508-3 steel.

The governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C^* K_{Im} + K_{It} < K_{Ic} \quad (2)$$

where,

K_{Im} = stress intensity factor caused by membrane (pressure) stress

K_{It} = stress intensity factor caused by the thermal gradients

K_{Ic} = function of temperature relative to the RT_{NDT} of the material

C = 2.0 for Level A and Level B service limits

C = 1.5 for hydrostatic and leak test conditions during which the reactor core is not critical

At any time during the heatup or cooldown transient, K_{Ic} is determined by the metal temperature at the tip of a postulated flaw at the 1/4T and 3/4T location, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from the temperature gradients through the vessel wall are calculated and then the corresponding (thermal) stress intensity factors, K_{It}, for the reference flaw are computed. From Equation 2, the pressure stress intensity factors are obtained and, from these, the allowable pressures are calculated.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

HEATUP (Continued)

During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and lower K_{Ic} values for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of pressure-temperature limitations for the case in which a 1/4T flaw located at the 1/4T location from the outside surface is assumed. Unlike the situation at the vessel inside surface, the thermal gradients established at the outside surface during heatup produce stresses which are tensile in nature and thus tend to reinforce any pressure stresses present. These thermal stresses are dependent on both the rate of heatup and the time (or coolant temperature) along the heatup ramp. Since the thermal stresses at the outside are tensile and increase with increasing heatup rate, a lower bound curve cannot be defined. Rather, each heatup rate of interest must be analyzed on an individual basis.

Following the generation of pressure-temperature curves for both the steady-state and finite heatup rate situations, the final limit curves are produced as follows. A composite curve is constructed based on a point-by-point comparison of the steady-state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the three values taken from the curves under consideration. The use of the composite curve is necessary to set conservative heatup limitations because it is possible for conditions to exist wherein, over the course of the heatup ramp, the controlling condition switches from the inside to the outside and the pressure limit must at all times be based on analysis of the most critical criterion.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

HEATUP (Continued)

10 CFR Part 50, Appendix G, Reference (4), addresses the metal temperature of the closure head flange and vessel flange regions. This rule states that the metal temperature of the closure flange regions must exceed the material unirradiated RT_{NDT} by at least 120°F for normal operation when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (3106 psi), which in this case is 621 psig. The limiting unirradiated RT_{NDT} of 30°F occurs in the vessel flange of the reactor vessel, consequently the minimum allowable temperature of this region is 150°F at pressures greater than 621 psig. However, per WCAP-17444-NP, Reference (9), Seabrook Unit 1 is justified for an exemption to these requirements. Therefore, these requirements are not contained in Figures 3.4-2 and 3.4-3.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

References

1. ASME Boiler and Pressure Vessel Code, Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure", dated 1998 through 2000 Addenda.

2. ASME Boiler and Pressure Vessel Code Case N-641, Section XI, Division 1, "Alternative Pressure-Temperature Relationship and Overpressure Protection System Requirements", dated January 17, 2000.

3. Westinghouse WCAP-14040-A, Revision 4, "Methodology Used to Develop Cold Overpressure Mitigating Setpoints and RCS Heatup and Cooldown Limit Curves", dated May 2004.

4. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements", U.S. Nuclear Regulatory Commission, Federal Register, Volume 60, No. 243, dated December 19, 1995.

5. Westinghouse WCAP-17441-NP, Revision 0, "Seabrook Unit 1 Heatup and Cooldown Limit Curves for Normal Operation", dated October 2011.

6. Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials", U. S. Nuclear Regulatory Commission, dated May 1988.

7. Westinghouse WCAP-16526-NP, Revision 0, "Analysis of Capsule V from FPL Energy-Seabrook Unit 1 Reactor Vessel Radiation Surveillance Program", dated March 2006.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

HEATUP (Continued)

References (Continued)

8. Westinghouse Letter LTR-AMLRS-11-50, Rev. 0, "Seabrook Unit 1 Heatup and Cooldown Limit Curves Applicability Evaluation", August 3, 2011.

9. Westinghouse WCAP-17444-NP, Revision 0, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Seabrook Unit 1", October 2011.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS (Continued)

COLD OVERPRESSURE PROTECTION (Continued)

The OPERABILITY of two PORVs, or two RHR suction relief valves, or a combination of a PORV and RHR suction relief valve, or an RCS vent opening of at least 1.58 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 225°F. Either PORV or either RHR suction relief valve has adequate relieving capability to protect the RCS from overpressurization during the following design basis transients: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures, or (2) the start of a centrifugal charging pump and its injection into a water-solid RCS.

The Maximum Allowed PORV Setpoint for the Cold Overpressure Mitigation System (COMS) is derived by analysis which models the performance of the COMS assuming various mass input and heat input transients. Operation with a PORV Setpoint less than or equal to the maximum Setpoint ensures that Appendix G criteria will not be violated with consideration for: (1) a maximum pressure overshoot beyond the PORV Setpoint which can occur as a result of time delays in signal processing and valve opening; (2) a 50°F heat transport effect made possible by the geometrical relationship of the RHR suction line and the RCS wide range temperature indicator used for COMS; (3) instrument uncertainties; and (4) single failure. To ensure mass and heat input transients more severe than those assumed cannot occur, Technical Specifications require both Safety injection pumps and all but one centrifugal charging pump to be made inoperable while in MODES 4, 5, and 6 with the reactor vessel head installed and not fully detensioned, and disallow start of an RCP if secondary coolant temperature is more than 50°F above reactor coolant temperature. Exceptions to these requirements are acceptable as described below.

Operation above 350°F but less than 375°F with only one centrifugal charging pump OPERABLE and no Safety Injection pumps OPERABLE is allowed for up to 4 hours. As shown by analysis, LOCAs occurring at low temperature, low pressure conditions can be successfully mitigated by the operation of a single centrifugal charging pump and a single RHR pump with no credit for accumulator injection. Given the short time duration and the condition of having only one centrifugal charging pump OPERABLE and the probability of a LOCA occurring during this time, the failure of the single centrifugal charging pump is not assumed.

Operation below 350°F but greater than 325°F with all centrifugal charging and Safety Injection pumps OPERABLE is allowed for up to 4 hours. During low pressure, low temperature operation all automatic Safety Injection actuation signals except Containment Pressure-High are blocked. In normal conditions, a single failure of the

3/4.7.6 CONTROL ROOM SUBSYSTEMS (Continued)

SURVEILLANCE REQUIREMENTS

SR 4.7.6.1

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train in accordance with the Surveillance Frequency Control Program provides an adequate check of this system. Systems with heaters must be operated for ≥15 continuous minutes with the heaters energized. The surveillance frequency is controlled under the Surveillance Frequency Control Program.

SRs also periodically test the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal.

The SRs verify that each CREMAFS train starts and operates on test actuation signals. The surveillance frequency is controlled under the Surveillance Frequency Control Program.

SR 4.7.6.2

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem TEDE and the CRE occupants are protected from smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, Action b. must be entered. Action b.3 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3 (Ref. 5), which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). These compensatory measures may also be used as mitigating actions as required by Action b.2. Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

3/4.8.3 ONSITE POWER DISTRIBUTION (continued)

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

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

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

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

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

APPLICABLE SAFETY ANALYSES

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

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

3/4.9.9 (THIS SPECIFICATION NUMBER IS NOT USED.)

3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis. Suspending fuel movement or crane operation does not preclude moving a component to a safe location.

3/4.9.12 FUEL STORAGE BUILDING EMERGENCY AIR CLEANING SYSTEM

The limitations on the Fuel Storage Building Emergency Air Cleaning System ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. Operation of the system with the heaters operating for at least 15 continuous minutes in accordance with the Surveillance Frequency Control Program demonstrates OPERABILITY of the system. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the safety analyses. ANSI N510-1980 will be used as a procedural guide for surveillance testing. Suspending fuel movement or crane operation does not preclude moving a component to a safe location.

One train of the Fuel Storage Building Emergency Air Cleaning System must be in operation during fuel movement. This requirement, however, does not apply to movement of a spent fuel cask containing irradiated fuel in preparation for transfer to dry storage. Movement of fuel after it has been inserted into a spent fuel cask and unlatched from the lifting tool is no longer a consideration with regard to this specification.

3/4.9.13 SPENT FUEL ASSEMBLY STORAGE

Restrictions on placement of fuel assemblies of certain enrichments within the Spent Fuel Pool is dictated by Specification 5.6.1.3. These restrictions ensure that the k_{eff} of the Spent Fuel Pool will always remain less than 1.0 assuming the pool to be flooded with unborated water and less than or equal to 0.95 when flooded with water borated to 500 ppm. The restrictions delineated in Specification 5.6.1.3 and the action statement are consistent with the criticality safety analysis performed for the Spent Fuel Pool as documented in the UFSAR.