FPL Energy
Seabrook Station

FPL Energy Seabrook Station
P.O. Box 300
Seabrook, NH 03874
(603) 773-7000

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Docket No. 50-443
SBK-L-04050

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

FPL 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 Specifications Bases according to the instructions in Enclosure 1.

Should you have any questions concerning this matter, please contact me at (603) 773-7194.

Very truly yours,

FPL Energy Seabrook, LLC

Mark E. Warner
Site Vice President

cc: S. J. Collins, NRC Region I Administrator
S. P. Wall, NRC Project Manager, Project Directorate I-2
G.T. Dentel, NRC Senior Resident Inspector

A001

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2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

The Reactor Trip Setpoint Limits specified in Table 2.2-1 are the nominal values at which the Reactor trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the core and Reactor Coolant System are prevented from exceeding their safety limits during normal operation and design-basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. The Setpoint for a Reactor Trip System or interlock function is considered to be adjusted consistent with the nominal value when the "as-measured" Setpoint is within the band allowed for rack calibration accuracy and bistable setting accuracy. This setpoint methodology may result in the "as measured" trip setpoint exceeding the trip setpoint listed in the Technical Specifications. For example, a bistable with a trip setpoint of ≤ 109% has a span of 120%, a rack calibration accuracy of ± 0.50%, and a bistable setting accuracy of ± 0.25%. The bistable is considered to be adjusted to the trip setpoint, consistent with the Technical Specifications, as long as the "as measured" value for the bistable is ≤ 109% + [(0.5 + 0.25) (120%) / 100)], or ≤ 109.9%.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable, since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as-measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other uncertainties of the instrumentation to measure the process variable and the uncertainties in calibrating the instrumentation. In Equation 2.2-1, $Z + R + S \leq TA$, the interactive effects of the errors in the rack and the sensor, and the "as measured" values of the errors are considered. Z, as specified in Table 2.2-1, in percent span, is the statistical summation of errors assumed in the analysis excluding those associated with the sensor and rack drift and the accuracy of their measurement. TA or Total Allowance is the difference, in percent span, between the Trip Setpoint and the value used in the analysis for Reactor trip. R or Rack Error is the "as-measured" deviation, in percent span, for the affected channel from the specified Trip Setpoint. S or Sensor Error is either the "as-measured" deviation of the sensor from its calibration point or the value specified in Table 2.2-1, in percent span, from the analysis assumptions. Use of Equation 2.2-1 allows for a sensor drift factor, an increased rack drift factor, and provides a threshold value for REPORTABLE EVENTS.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The methodology to derive the Trip Setpoints is based on combining all of the uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

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Specification 4.0.4 establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

Certain Specifications allow an exception to the requirements of Specification 4.0.4 for individual Surveillance Requirements when the surveillance can only be performed after entering the MODE or condition specified in the Applicability statement. When surveillance requirements become applicable as a consequence of an exception to Specification 4.0.4, the Action requirements of the Specification may be delayed for up to 24 hours to permit completion of the surveillance when the allowed outage time of the Action requirement is less than 24 hours.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.5 establishes the requirement that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and the ASME OM 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 and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and the ASME OM 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 and testing activities.

Specification 4.0.5 (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and the ASME OM Code including applicable Addenda. 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.

3/4.1.2 BORATION SYSTEMS

The limitations on OPERABILITY of isolation provisions for the Boron Thermal Regeneration System and the Reactor Water Makeup System in Modes 4, 5, and 6 ensure that the boron dilution flow rates cannot exceed the value assumed in the transient analysis.

The "equivalent to" statement in the Action for LCO 3.1.2.7 is a provision providing an alternate method of emergency boration via the RWST at an increased flow rate to account for the lower boron concentration within the RWST.

Upon a failure to meet the LCO, the action statement requires immediate suspension of core alternations and positive reactivity changes. Operations that individually add limited, positive reactivity are acceptable when, combined with other actions that add negative reactivity, the overall net reactivity addition is zero or negative. For example, a positive reactivity addition caused by temperature fluctuations from inventory addition or temperature control fluctuations is acceptable if it is combined with a negative reactivity addition such that the overall, net reactivity addition is zero or negative. Refer to TS Bases 3/4.9.1, Boron Concentration, for limits on boron concentration and water temperature for MODE 6 action statements involving suspension of positive reactivity changes. In addition, if the shutdown margin is not verified within one hour, boration must commence and continue until shutdown margin is adequate and compliance with the LCO is restored.

A resin bed is considered saturated with boron when the effluent boron concentration is within 5% or 5 ppm, whichever is greater, of the Reactor Coolant System boron concentration at the time the resin bed was saturated. Saturation ensures that no further boron may be removed by the resin bed regardless of the current boron concentration.

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

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

Table 3.3-1 contains the action statements for inoperable Reactor Trip System Instrumentation. Actions 4 and 5, associated with the source range neutron flux instruments, each include a requirement to suspend operations involving positive reactivity changes. When complying with this action, operations that individually add limited, positive reactivity are acceptable when, combined with other actions that add negative reactivity, the overall net reactivity addition is zero or negative. For example, a positive reactivity addition caused by temperature fluctuations from inventory addition or temperature control fluctuations is acceptable if it is combined with a negative reactivity addition such that the overall, net reactivity addition is zero or negative.

Specified surveillance intervals and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation. The NRC Safety Evaluation Reports for WCAP-10271 and its supplements and revisions were provided on February 21, 1985, February 22, 1989 and April 30, 1990.

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

The Engineered Safety Features Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A Setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for rack calibration accuracy and bistable setting accuracy. This setpoint methodology may result in the "as measured" trip setpoint exceeding the trip setpoint listed in the Technical Specifications. For example, a bistable with a trip setpoint of $\leq$ 109% has a span of 120%, a rack calibration accuracy of $\pm$ 0.50%, and a bistable setting accuracy of $\pm$ 0.25%. The bistable is considered to be adjusted to the trip setpoint, consistent with the Technical Specifications, as long as the "as measured" value for the bistable is $\leq$ 109% + [(0.5 + 0.25) (120%) / 100)], or $\leq$ 109.9%.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for the Setpoints have been specified in Table 3.3-4. Operation with Setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error. An optional provision has been included for determining the OPERABILITY of a channel when its Trip Setpoint is found to exceed the Allowable Value. The methodology of this option utilizes the "as measured" deviation from the specified calibration point for rack and sensor components in conjunction with a statistical combination of the other

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown.

In the event that no safety valves are OPERABLE in MODE 4 or 5, the action statement requires immediately suspending positive reactivity changes and placing an RHR loop in operation in the shutdown cooling mode. An operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures. Operations that individually add limited, positive reactivity are acceptable when, combined with other actions that add negative reactivity, the overall net reactivity addition is zero or negative. For example, a positive reactivity addition caused by temperature fluctuations from inventory addition or temperature control fluctuations is acceptable if it is combined with a negative reactivity addition such that the overall, net reactivity addition is zero or negative.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

The safety significance of RCS leakage varies depending on the source, rate, and duration of the leak; therefore, detection and monitoring of RCS leakage are necessary. In addition, a means of separating the identified from the unidentified leakage is necessary to permit the operators to take prompt corrective action in the event of a leak that is detrimental to the safety of the facility.

Unidentified Leakage

Uncollected leakage to the containment atmosphere, which is ultimately collected in the containment drainage sumps where the leak rate can be established and monitored, is unidentified leakage. Unidentified leakage to the containment atmosphere is kept to a minimum (normal leakage is estimated to range from 20 to 40 gallons per day) to permit the leakage detection system to detect positively and rapidly a small increase in leakage. Identified leakage and unidentified leakage are separated so that a small unidentified leak will not be masked by larger, acceptable identified leakage. The one-gallon per minute limit on unidentified leakage is a reasonable, minimum detectable amount that the leakage detection system can detect in a reasonable time period.

Identified Leakage

A limited amount of leakage is expected from auxiliary systems inside containment that cannot practically be made 100% leak tight. Identified leakage, which consists of collectable, detectable, leakage from specifically known and located sources, does not interfere with the ability of the leakage detection system to detect unidentified leakage. Identified leakage is monitored separately from unidentified leakage. Up to 10 gpm of identified leakage is acceptable because the leakage is from known sources that will not mask a small, unidentified leak and is well within the capability of the RCS make up system.

Primary to Secondary Leakage

The total steam generator tube leakage limit of 1 gpm for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

Controlled Leakage

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

Pressure Isolation Valve Leakage

The specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing over-pressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. RCS Pressure Isolation Valve (PIV) Leakage measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS leakage when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code.

As stated in Appendix H of WCAP-14535A (November 1996), Appendix VIII of Section XI of the ASME Boiler and Pressure Vessel Code is not applicable when examining the reactor coolant pump flywheels.

The requirements of this LCO apply only to the Reactor Coolant System pressure boundary as defined in 10 CFR 50.2. The RCS pressure boundary includes all those pressure-containing components such as pressure vessels, piping, pumps, and valves which are:

1) Part of the RCS, or
2) Connected to the RCS up to and including any and all of the following:

   (i) The outermost containment isolation valve in system piping which penetrates primary reactor containment,
   (ii) The second of two valves normally closed during normal reactor operation in system piping which does not penetrate primary reactor containment,
   (iii) The RCS safety and relief valves.

The ASME code exempts certain small bore piping and components which are part of the RCS pressure boundary provided that the flow resulting from a postulated failure under normal operating conditions is within the capacity of makeup systems operable from an on-site emergency power source. As a result, lines with a diameter of 3/8 inch or less and lines containing a 3/8-inch flow restrictor are not included within the scope of this Technical Specification.

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on accumulator volume, boron concentration, and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

In MODES 1 and 2, the accumulator power-operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In MODES 1, 2, 3, and in MODE 4 within 12 hours of entry into MODE 3 from 4, the accumulator isolation valves are open with their power removed whenever pressurizer pressure is greater than 1000 psig. In addition, as these accumulator isolation valves fail to meet single-failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single-failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS cold-leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

Operability of the ECCS flow paths is contingent on the ability of the encapsulations surrounding the containment sump isolation valves (CBS-V8 and CBS-V14) to perform their design functions. During the recirculation phase of an accident, any postulated leakage resulting from the failure of the valves or piping will be contained within the encapsulations, preserving the water inventory needed to support ECCS operation during recirculation. Consequently, maintaining the encapsulations intact with leakage within allowable limits is necessary to ensure operability of the ECCS flow paths. Although designed to withstand containment pressure, the encapsulations do not function as a containment boundary, but rather the release of radioactive fluid and gasses to the environment.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (Continued)

Each operable RHR subsystem must remain aligned to provide injection into all four RCS cold legs to meet the assumptions in the ECCS analysis. Isolating RHR flow to any RCS cold leg in MODES 1, 2, or 3 would render both trains of ECCS inoperable, placing the plant in a condition outside design bases.

With the RCS temperature below 350°F, the ECCS operational requirements are reduced. Only one OPERABLE ECCS subsystem is acceptable without single failure consideration during MODE 4 operation on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements, as well as the reduced probability of occurrence of a Design Basis Accident (DBA). It is understood in these reductions in operational requirements that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA. LCO Condition d. requires that an OPERABLE flow path must be capable of taking suction from the refueling water storage tank upon being manually realigned and transferring suction to the containment sump during the recirculation phase of operation. Thus, LCO Condition d. allows for the manual realignment of the OPERABLE ECCS subsystem to support the ECCS mode of operation.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (Continued)

This allowance recognizes that components which comprise the OPERABLE ECCS subsystem such as RHR pumps and heat exchangers may be aligned in other modes of operation to support plant evolutions, e.g., decay heat removal operation. Therefore, in this case, the RHR train is considered OPERABLE during alignment and operation for decay heat removal, if capable of manually being realigned (remote of local) to the ECCS mode of operation and not otherwise inoperable.

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps and safety injection pumps except the required OPERABLE charging pump to be inoperable in MODES 4 and 5 and in MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV or RHR suction relief valve.

When the RCS has a vent area equal to or greater than 18 square inches, or the RCS is in a reduced inventory condition, i.e., whenever reactor vessel water level is lower than 36 inches below the reactor vessel flange, one Safety Injection pump may be made OPERABLE when in MODE 5 or MODE 6 with the reactor vessel head on and the vessel head closure bolts not fully detensioned. When operating in this configuration, cold overpressure protection is provided by either the mechanical vent opening in the RCS boundary, equal to or greater than 18 square inches, or the additional void volume existing when operating in a reduced inventory condition. Either configuration is required to be present prior to making the SI pump OPERABLE. This required RCS vent area or reduced inventory condition and the cold overpressure protection surveillance requirements to verify the presence of the RCS vent area or verify that the reactor vessel water level is lower than 36 inches below the reactor vessel flange provides assurance that a mass addition transient can be mitigated and that adequate cold overpressure protection is provided.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Verifying the correct alignment of manual, power-operated, and automatic valves provides assurance that the proper flow paths exist for operation of ECCS under accident conditions. This verification includes only those valves in the direct flow paths through safety-related equipment whose position is critical to the proper functioning of the safety-related equipment. Vents, drains, sampling connections, instrument taps, etc., that are not directly in the flow path and are not critical to proper functioning of the safety-related equipment are excluded from this surveillance requirement. This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position because these valves are verified in their correct position prior to locking, sealing, or securing. Also, this requirement does not apply to valves that cannot be inadvertently misaligned, such as check valves.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS (Continued)

An automatic valve may be aligned in other than its accident position provided (1) the valve receives an automatic signal to re-position to its required position in the event of an accident, and (2) the valve is otherwise operable (stroke time within limits, motive force available to re-position the valve, control circuitry energized, and mechanically capable of re-positioning).

With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, non-operating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the refueling water storage tank (RWST) to the RCS full of water (by verifying at the accessible ECCS piping high points and pump casings, excluding the operating centrifugal charging pump) ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of non-condensable gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following a safety injection (SI) signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.

It should be noted that Surveillance Requirement 4.5.2b.1 Bases also

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The two independent Containment Spray Systems provide post-accident cooling of the containment atmosphere. The Containment Spray Systems also provide a mechanism for removing iodine from the containment atmosphere, and, therefore, the time requirements for restoring an inoperable Spray System to OPERABLE status have been maintained consistent with those assigned other inoperable ESF equipment.

Verifying the correct alignment of manual, power-operated, and automatic valves provides assurance that the proper flow paths exist for operation of the Containment Spray System under accident conditions. This verification includes only those valves in the direct flow paths through safety-related equipment whose position is critical to the proper functioning of the safety-related equipment. Vents, drains, sampling connections, instrument taps, etc., that are not directly in the flow path and are not critical to proper functioning of the safety-related equipment are excluded from this surveillance requirement. This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position because these valves are verified in their correct position prior to locking, sealing, or securing. Also, this requirement does not apply to valves that cannot be inadvertently misaligned, such as check valves.

An automatic valve may be aligned in other than its accident position provided (1) the valve receives an automatic signal to re-position to its required position in the event of an accident, and (2) the valve is otherwise operable (stroke time within limits, motive force available to re-position the valve, control circuitry energized, and mechanically capable of re-positioning).

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the Spray Additive System ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.2 SPRAY ADDITIVE SYSTEM (Continued)

Verifying the correct alignment of manual, power-operated, and automatic valves provides assurance that the proper flow paths exist for operation of the Spray Additive System under accident conditions. This verification includes only those valves in the direct flow paths through safety-related equipment whose position is critical to the proper functioning of the safety-related equipment. Vents, drains, sampling connections, instrument taps, etc., that are not directly in the flow path and are not critical to proper functioning of the safety-related equipment are excluded from this surveillance requirement. This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position because these valves are verified in their correct position prior to locking, sealing, or securing. Also, this requirement does not apply to valves that cannot be inadvertently misaligned, such as check valves.

An automatic valve may be aligned in other than its accident position provided (1) the valve receives an automatic signal to re-position to its required position in the event of an accident, and (2) the valve is otherwise operable (stroke time within limits, motive force available to re-position the valve, control circuitry energized, and mechanically capable of re-positioning).

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment.

Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.3 CONTAINMENT ISOLATION VALVES (continued)

In the event that one containment isolation valve becomes inoperable, the valve must be restored to an operable status within four hours or the affected penetration must be isolated. Additionally, if the penetration is open, the second isolation barrier in the penetration (either another containment isolation valve or the associated closed system within containment) must remain operable. The operability of the closed system is established by its governing Technical Specification. For example, the SG U-tubes would comprise an operable closed system functioning as a containment barrier if tube leakage was within the leakage limitations of T.S. 3.4.6.2. For the hydrogen analyzer portion of the Combustible Gas Control system, the system outside of containment is qualified as an additional containment isolation barrier.

The method of isolating a penetration with an inoperable containment isolation valve must include the use of an isolation barrier that cannot be adversely affected by a single active failure. Barriers that meet this criterion include: (1) a deactivated automatic valves secured in the isolation position, (2) a closed manual valve, and (3) a blind flange. Closed systems within containment do not meet the isolation criterion because they are vulnerable to failures. Isolating a penetration with a deactivated automatic valve may be accomplished using either the inoperable valve, if it can be verified to be fully closed, or the operable automatic valve. Manual valves and blind flanges used to isolate a penetration must be within the penetration's ASME class boundary and qualified to ASME Class 2.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with: (1) zirconium-water reactions, (2) radiolytic decomposition of water, and (3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The Hydrogen Mixing Systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

3/4.6.5 CONTAINMENT ENCLOSURE BUILDING

3/4.6.5.1 CONTAINMENT ENCLOSURE EMERGENCY AIR CLEANUP SYSTEM

The OPERABILITY of the Containment Enclosure Emergency Air Cleanup System ensures that during LOCA conditions containment vessel leakage into the annulus, and radioactive materials leaking from engineered safety features equipment, from the electrical penetration areas, and from the mechanical penetration tunnel, will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere.

The EAH system components associated with this Technical Specification include those dampers, fans, filters, etc., and required ductwork and instrumentation that evacuate or isolate areas, route air, and filter the exhaust prior to discharge to the environment. Included among these components are:

- Containment enclosure cooling fans (EAH-FN-5A and 5B)
- Containment enclosure ventilation area return fans (EAH-FN-31A and 31B)
- Containment enclosure emergency exhaust fans (EAH-FN-4A and 4B)
- Charging pump room return air fans (EAH-FN-180A and 180B)
- Containment enclosure emergency clean up filters (EAH-F-9 and F-69)
- PAB / CEVA isolation dampers (PAH-DP-35A, 36A, 35B, and 36B)

The EAH system also provides cooling to the following areas and equipment during normal and emergency operation: containment enclosure ventilation equipment area, the charging pumps, safety injection pumps, residual heat removal pumps, containment spray pumps, and the mechanical penetration area. However, the EAH cooling function is not associated with this Technical Specification, but rather is controlled under Technical Requirement 24, Area Temperature Monitoring.

3/4.6.5.2 CONTAINMENT ENCLOSURE BUILDING INTEGRITY

CONTAINMENT ENCLOSURE BUILDING INTEGRITY ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the Containment Enclosure Emergency Air Cleanup System, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 100 during accident conditions.

3/4.6.5.3 CONTAINMENT ENCLOSURE BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment enclosure building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide: (1) protection for the steel vessel from external missiles, (2) radiation shielding in the event of a LOCA, and (3) an annulus surrounding the primary containment that can be maintained at a negative pressure during accident conditions. A visual inspection in accordance with the Containment Leakage Rate Testing Program, is sufficient to demonstrate this capability.

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

BACKGROUND

The auxiliary feedwater system (AFW) consists of the emergency feedwater system (EFW) and the startup feedwater pump (SUFP). The EFW system provides RCS heat removal during emergency conditions, including small break LOCAs, when the main feedwater system is not available. The non safety-related SUFP serves as a back up to the EFW pumps.

Upon loss of normal feedwater flow, the reactor is tripped, and the decay and sensible heat is transferred to the steam generators by the RCS via the reactor coolant pumps or by natural circulation when the pumps are not operational. Heat is removed from the steam generators via the main condensers or the main steam safety or atmospheric relief valves. Steam generator water inventory is maintained by water makeup from the EFW system. The system will supply feed water to the steam generators to remove sufficient heat to prevent the over-pressurization of the RCS and to allow for eventual system cool down.

The EFW system is comprised of two 100% capacity pumps (one motor- and one turbine-driven) whose water source is the Condensate Storage Tank (CST). Either EFW pump provides the required 650-gpm flow at a steam generator pressure of 1236 psia. Suction lines are individually run from the CST to each pump. A common EFW pump recirculation line discharges back to the CST. This return line supports testing the pump on recirculation and ensures minimum flow to prevent pump damage for any system low-flow operating condition. Both pumps feed a common discharge header, which in turn supplies the four emergency feed lines. Additional, redundant pumping capability is provided by the SUFP, which is capable of being powered from emergency bus E5. The flow paths associated with the SUFP are (1) the main flow path which flows through the main feedwater isolation valves, and (2) the emergency flow path through the EFW system flow control valves.

SAFETY ANALYSES

The EFW system mitigates the consequences of any event involving a loss of normal feedwater. The design basis of the EFW system is to deliver water to the steam generators to remove heat at the minimum required flow at a steam generator pressure corresponding to the lowest set safety valve setting plus 3% (1236 psia).

The limiting design basis accidents for the EFW system include feedwater line break and loss of normal feedwater. The system is designed to provide the required flow following a single active failure and a loss of off-site power and has been evaluated for a station blackout event.

LIMITING CONDITION FOR OPERATION (LCO)

The LCO ensures that the EFW system will perform its design safety function to mitigate the consequences of loss of feed accidents that could otherwise result in over-pressurization of the RCS.

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM (Continued)

LCO (Continued)

The system is considered OPERABLE when the components and flow paths required to provide feedwater flow to the steam generators are OPERABLE. This requires operability of the three AFW pumps and the required piping, valves, instrumentation and controls:

- The EFW flow control valves and discharge header stop check valves must be fully open to meet the assumptions in the EFW flow analysis while the AFW system is required to be operable (MODES 1, 2, and 3). Isolating an EFW header renders the system inoperable and action d. becomes applicable.
- The EFW pump recirculation valves must remain closed to meet the 650 gpm flow requirement for ANS Condition II events. Opening a recirculation valve renders the associated EFW pump inoperable.
- The main steam upstream drain valves must normally remain open; however, brief closure for periods of less than 15 minutes will not render the turbine-driven pump inoperable.
- Operation of the EFW system to control steam generator levels following a reactor trip does not result in any inoperability that requires entry into the action statements of this TS.
- The EFW system is used only during emergency conditions when the main feedwater system is not available. During all other modes of plant operation, including startup, hot standby, and normal power operation, the system is de-pressurized and has zero flow.
- Operability of the turbine-driven EFW pump requires two operable steam supplies. The steam supply valves, MS-V-393 and V-394 are dual function valves that must open on an EFW initiation signal and close to provide containment isolation per TS 3.6.3. The valves are provided with a backup nitrogen supply that supports only the containment isolation function. With the backup nitrogen isolated or the accumlator pressure less than 500 psig (equivalent to 1530 psig for a single bottle configuration or 772 psig for a dual bottle configuration), the valves are inoperable per TS 3.6.3. However, isolating the penetration in accordance with the action of TS 3.6.3 will render the turbine-driven EFW pump inoperable.

APPLICABILITY

In MODEs 1, 2, and 3, the AFW system is required to be operable in the event of a loss of feedwater event. In MODEs 4 and below, the steam generators are not normally used for heat removal and the AFW system is not required.

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM (Continued)

ACTIONS

With one AFW pump inoperable, the action provides a 72-hour AOT for restoring the pump to an operable status before requiring a plant shutdown. This time is reasonable based on the availablility of redundant equipment and the low probability of an accident occurring during this time. Additional actions with more limiting AOTs apply to conditions involving more than one inoperable AFW pump. In the event that all AFW pumps are inoperable, the plant is in a seriously degraded condition. Consequently, the plant should not be perturbed by any action, including a power change, that might result in a plant trip and demand on the EFW system. The seriousness of this condition requires immediately initiating corrective action to restore at least one AFW pump to operable status as soon as possible.

SURVEILLANCES

Various surveillance requirements, with frequencies ranging from 31 days to eighteen months, demonstrate the operability of the AFW system. Every 31 days, each non-automatic valve in the flow path that is not locked, sealed, or otherwise secured, is verified in its correct position. This verification includes only those valves in the direct flow path through safety-related equipment whose position is critical to the proper functioning of the safety-related equipment. Vents, drains, sampling connections, instrument taps, etc., that are not directly in the flow path and are not critical to proper functioning of the safety-related equipment are excluded from this surveillance requirement.

Testing of the steam-driven EFW pump is exempt from the provisions of TS 4.0.4 for entry into MODE 3. This allowance is necessary because the surveillance testing, which requires a minimum steam pressure of 500 psig, cannot be performed until the plant reaches MODE 3. Once steam pressure reaches 500 psig, administrative controls establish a 24-hour time limit for completing the testing.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the indicated minimum water volume ensures that sufficient water is available to cool the RCS to a temperature of 350°F. The OPERABILITY of the concrete enclosure ensures this availability of water following rupture of the condensate storage tank by a tornado generated missile. The contained water volume limit includes an allowance for water not usable because of instrument uncertainty, tank discharge line location, or other physical characteristics.

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.4 SPECIFIC ACTIVITY

BACKGROUND

Activity in the secondary coolant results from Reactor Coolant System leakage through the steam generator tube(s). Under steady state conditions, the activity is primarily iodines with relatively short half-lives and, thus, indicates current conditions. During transients, I-131 spikes have been observed as well as increased releases of some noble gases. Other fission product isotopes, as well as activated corrosion products in lesser amounts, may also be found in the secondary coolant.

A limit on secondary coolant specific activity during power operation minimizes releases to the environment because of normal operation, anticipated operational occurrences, and accidents.

This limit is lower than the activity value that might be expected from a 1 gpm tube leak (LCO 3.4.6.2, "Reactor Coolant System Leakage - Operational Leakage") of primary coolant at the limit of 1.0 μCi/gm (LCO 3.4.8, "Reactor Coolant System Specific Activity"). The steam line failure is assumed to result in the release of the noble gas and iodine activity contained in the steam generator inventory, the feedwater, and the reactor coolant LEAKAGE. Most of the iodine isotopes have short half-lives (i.e., <20 hours).

With the specified activity limit, the resultant 2-hour thyroid dose to a person at the SITE BOUNDARY would be a small fraction of the 10 CFR 100 (Ref. 1) limits if the main steam safety valves (MSSVs) were open for 2 hours following a trip from full power.

Operating a unit at the allowable limits could result in a 2 hour SITE BOUNDARY exposure of a small fraction of the 10 CFR 100 (Ref. 1) limits, or the limits established as the NRC staff approved licensing basis.

APPLICABLE SAFETY ANALYSES

The accident analysis of the main steam line break (MSLB), as discussed in the UFSAR, Chapter 15 (Ref. 2) assumes the initial secondary coolant specific activity to have a radioactive isotope concentration of 0.10 μCi/gm DOSE EQUIVALENT I-131. This assumption is used in the analysis for determining the radiological consequences of the postulated accident. The accident analysis, based on this and other assumptions, shows that the radiological consequences of an MSLB do not exceed a small fraction of the unit SITE BOUNDARY limits (Ref. 1) for whole body and thyroid dose rates.

With the loss of offsite power, the remaining steam generators are available for core decay heat dissipation by venting steam to the atmosphere through the MSSVs and steam generator atmospheric dump valves (ADVs). The Emergency Feedwater System supplies the necessary makeup to the steam generators. Venting continues until the reactor coolant temperature and pressure have decreased sufficiently for the Residual Heat Removal System to complete the cooldown.

In the evaluation of the radiological consequences of this accident, the activity released from the steam generator connected to the failed steam line is assumed to be released directly to the environment. The unaffected steam generator is assumed to discharge steam and any entrained activity through the MSSVs and ADVs during the event. Since no credit is taken in the analysis for activity plateout or retention, the resultant radiological consequences represent a conservative estimate of the potential integrated dose due to the postulated steam line failure.

Secondary specific activity limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.4 SPECIFIC ACTIVITY (Continued)

LIMITING CONDITION FOR OPERATION (LCO)

As indicated in the Applicable Safety Analyses, the specific activity of the secondary coolant is required to be ≤ 0.10 μCi/gm DOSE EQUIVALENT I-131 to limit the radiological consequences of a Design Basis Accident (DBA) to a small fraction of the required limit (Ref. 1).

Monitoring the specific activity of the secondary coolant ensures that when secondary specific activity limits are exceeded, appropriate actions are taken in a timely manner to place the unit in an operational MODE that would minimize the radiological consequences of a DBA.

APPLICABILITY

In MODES 1, 2, 3, and 4, the limits on secondary specific activity apply due to the potential for secondary steam releases to the atmosphere.

MODE 4 is conditioned by a footnote to recognize that sampling in MODE 4 is limited by the steam generator conditions necessary to obtain a sample. Upon entering MODE 4 from MODE 5, there is not enough steam pressure in the steam generator to provide a sample through the normal sample point. Due to plant limitations, a representative sample can be obtained with greater than 100 psig steam pressure in the steam generator. By requiring the sample to be taken within 12 hours of achieving the 100 psig steam pressure, adequate time for obtaining and analyzing a sample are ensured.

In MODES 5 and 6, the steam generators are not being used for heat removal. Both the RCS and steam generators are depressurized, and primary to secondary LEAKAGE is minimal. Therefore, monitoring of secondary specific activity is not required.

ACTIONS

DOSE EQUIVALENT I-131 exceeding the allowable value in the secondary coolant, is an indication of a problem in the RCS and contributes to increased post accident doses. If the secondary specific activity cannot be restored to within limits within the associated completion time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in a least HOT STANDBY within 6 hours, and in COLD SHUTDOWN within the following 30 hours. The ACTION completion times are reasonable, based on operating experience, to reach the required unit shutdown conditions from full power in an orderly manner and without challenging unit systems.

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.4 SPECIFIC ACTIVITY (Continued)

SURVEILLANCE REQUIREMENTS (SR)

SR 4.7.1.4

This SR verifies that the secondary specific activity is within the limits of the accident analysis. A gamma isotopic analysis of the secondary coolant, which determines DOSE EQUIVALENT I-131, confirms the validity of the safety analysis assumptions as to the source terms in post accident releases. It also serves to identify and trend any unusual isotopic concentrations that might indicate changes in reactor coolant activity or LEAKAGE. The 31-day frequency is based on the detection of increasing trends of the level of DOSE EQUIVALENT I-131, and allows for appropriate action to be taken to maintain levels below the LCO limit.

REFERENCES

1. 10 CFR 100.11.
2. UFSAR, Chapter 15.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses.

3/4.7.1.6 ATMOSPHERIC RELIEF VALVES

The OPERABILITY of the Atmospheric Relief Valves (ARVs) ensures the controlled removal of reactor decay heat during reactor cooldown, plant startup, and after a turbine trip, when the condenser and/or the turbine bypass system are not available. When available, the ARVs can be used to reduce main steam pressure for both hot shutdown and cold shutdown conditions. The ARVs provide a method for cooling the plant to residual heat removal entry conditions should the turbine bypass system to the condenser be unavailable. This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST).

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.6 ATMOSPHERIC RELIEF VALVES (Continued)

One ARV line for each of the four steam generators is provided. Each ARV line consists of one ARV and an associated block valve. The ARVs are provided with upstream block valves to provide an alternate means of isolation.

The ARVs are equipped with pneumatic controllers to permit control of the cooldown rate. The ARVs are provided with a pressurized gas supply of bottled nitrogen that, on a loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ARVs. The nitrogen supply is sized to provide sufficient pressurized gas to operate the ARVs for the time required for Reactor Coolant System cooldown to RHR entry conditions. The ARVs are OPERABLE with only a DC power source available. In addition, handwheels are provided for local manual operation.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on a steam generator RT_{NDT} of 60°F and are sufficient to prevent brittle fracture.

3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Primary Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. Portions of the PCCW system automatically isolate on low and low-low head tank level signals. This feature, which is necessary for operability of the PCCW system, isolates the non-seismic portions of the system so that the safety function of PCCW is not compromised in the event of a failure in the non-seismic part of the system. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

The system design also includes an automatic trip of the PCCW pumps on high system temperature. The purpose of this function is to protect the PCCW system components from further degradation by preventing recycling of hot PCCW fluid through the process heat exchangers. Tripping a PCCW pump on high temperature requires that a single failure has already occurred; therefore, the high temperature pump trip function is not a requirement for PCCW system operability.

Verifying the correct alignment of manual, power-operated, and automatic valves provides assurance that the proper flow paths exist for operation of the Primary Component Cooling Water System under accident conditions. This verification includes only those valves in the direct flow paths through safety-related equipment whose position is critical to the proper functioning of the safety-related equipment. Vents, drains, sampling connections, instrument taps, etc., that are not directly in the flow path and are not critical to proper functioning of the safety-related equipment are excluded from this surveillance requirement. This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position because these valves are verified in their correct position prior to locking, sealing, or securing. Also, this requirement does not apply to valves that cannot be inadvertently misaligned, such as check valves.

3/4.7.3 PRIMARY COMPONENT COOLING WATER SYSTEM (Continued)

An automatic valve may be aligned in other than its accident position provided (1) the valve receives an automatic signal to re-position to its required position in the event of an accident, and (2) the valve is otherwise operable (stroke time within limits, motive force available to re-position the valve, control circuitry energized, and mechanically capable of re-positioning).

3/4.7.4 SERVICE WATER SYSTEM/ULTIMATE HEAT SINK

The Service Water System consists of two independent loops, each of which can operate with either a service water pump train or a cooling tower pump train. Each service water loop consists of a service water pump and the piping, valves, and other components necessary to provide the flowpath required for heat removal. Each service water cooling tower loop consists of a service water cooling tower pump and the necessary piping, valves and other components required to provide its flowpath. The OPERABILITY of the Service Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses, which also assumes loss of either the cooling tower or ocean cooling.

Cooling is normally provided by the Atlantic Ocean via the service water pumphouse. A seismically qualified mechanical draft cooling tower is provided as a backup to the ocean cooling water source because the supply from the circulating water tunnels is not seismically qualified. The mechanical draft cooling tower was designed to use three cells to support two units. Unit 1 utilizes two train-related cells; cell 1 serves Train A and has a single fan, the common cell serves Train B and has two fans. The cooling tower design basis is to provide the necessary ultimate heat sink in the event of a loss of ocean tunnel water flow; however, this source may be used during normal operations subject to the level and temperature limitations of this specification.

Switchover from the service water pumphouse to the mechanical draft cooling tower is accomplished either automatically (Tower Actuation (TA) signal) or manually. Manual action is required to realign the system from the cooling tower to the service water pumphouse. While a cooling tower pump is operating, interlocks prevent the train associated service water pumps from starting. To provide additional protection, during operation while aligned to the cooling tower, the service water pump control switches may be maintained in the pull-to-lock position to prevent inadvertent pump operation. As previously discussed, realignment to the service water pumphouse requires manual action; maintaining the control switches in the pull-to-lock position does not change this required action sequence. Pump operation is not affected by maintaining the control switches in the pull-to-lock position during this period; therefore, OPERABILITY of the service water pumps is not compromised.

3/4.7.4 SERVICE WATER SYSTEM/ULTIMATE HEAT SINK (Continued)

The limitations on service water pumphouse minimum water level and the requirements for cooling tower OPERABILITY are based on providing a 30-day cooling water supply to safety-related equipment without exceeding the safety related equipment design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

The Cooling Tower is normally aligned to allow return flow to bypass the tower sprays and return to the basin. In addition, the control switches for the cooling tower fans are normally maintained in the "pull-to-lock" position. Upon receipt of a Tower Actuation Signal, the fans and sprays are manually operated as required. This manual operation, which is governed by procedures, ensures that ice does not buildup on the cooling tower tile fill and fans. The cooling tower basin temperature limit of 70°F provides sufficient time for manual initiation of the cooling tower sprays and fans following the design basis seismic event with a concurrent LOCA, during the design extreme ambient temperature conditions. Under this scenario, manual action is sufficient to maintain the cooling tower basin at a temperature which precludes equipment damage during the postulated design basis event.

Verifying the correct alignment of manual, power-operated, and automatic valves provides assurance that the proper flow paths exist for operation of the Service Water System under accident conditions. This verification includes only those valves in the direct flow paths through safety-related equipment whose position is critical to the proper functioning of the safety-related equipment. Vents, drains, sampling connections, instrument taps, etc., that are not directly in the flow path and are not critical to proper functioning of the safety-related equipment are excluded from this surveillance requirement. This surveillance does not apply to valves that are locked, sealed, or otherwise secured in position because these valves are verified in their correct position prior to locking, sealing, or securing. Also, this requirement does not apply to valves that cannot be inadvertently misaligned, such as check valves.

An automatic valve may be aligned in other than its accident position provided (1) the valve receives an automatic signal to re-position to its required position in the event of an accident, and (2) the valve is otherwise operable (stroke time within limits, motive force available to re-position the valve, control circuitry energized, and mechanically capable of re-positioning).

3/4.7.5 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.7.6 CONTROL ROOM SUBSYSTEMS

The OPERABILITY of the Control Room Emergency Makeup Air and Filtration Subsystem ensures that the control room will remain habitable for operations personnel during and following credible accident conditions. Cumulative operation of the system with the heaters on for 10 hours over a 31-day period is sufficient to reduce the buildup of moisture on the absorbers and HEPA filters. Heaters cycle on and off to maintain the relative humidity below 70%. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50. ANSI N510-1980 will be used as a procedural guide for surveillance testing.

The OPERABILITY of the Control Room Emergency Makeup Air and Filtration Subsystem is also contingent on maintaining the integrity of the Control Room complex envelope. Envelope integrity is maintained by controlling activities that could introduce sources of makeup air or infiltration of unfiltered air other than that assumed in the UFSAR. Examples of activities that could render either or both subsystem trains inoperable: (1) removal of penetration seals; (2) blocking open or removing either Control Room door (C312, C325); (3) open access doors to filter units 1-CBA-F-38, 8038; (4) repositioning of remote intake manual isolation valves 1, 2-CBA-V9; (5) any activity which allows makeup air to be drawn into the system from locations other than the remote intakes (e.g., removal of an opacity detector or radiation monitor in the DG Building, cutting of either makeup air line, etc.). Breaches to the envelope shall be controlled by station programs and may require an engineering evaluation to ensure UFSAR assumptions remain valid. Refer to Engineering Evaluation 91-39, Rev. 1 and CR 02-16293 for specific information and compensatory measures.

The OPERABILITY of the safety-related Control Room Air Conditioning Subsystem ensures that the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system is not exceeded. The safety-related Control Room Air Conditioning Subsystem consists of two independent and redundant trains that provide cooling of recirculated control room air. The design basis of the safety-related Control Room Air Conditioning Subsystem is to maintain the control room temperature for 30 days of continued occupancy. The safety-related chillers are designed to operate in conditions down to the design basis winter temperature. When the chiller units unload due to insufficient heat load on the system, each Control Room air Conditioning Subsystem remains operable. Surveillance to demonstrate OPERABILITY will verify each subsystem has the capability to maintain the control room area temperature less than the limiting equipment qualification temperature. The operational surveillance will be performed on a quarterly basis, requiring each safety-related Control Room Air Conditioning Subsystem to operate over a twenty-four hour period. This will ensure the safety related subsystem can remove the heat load based on daily cyclic outdoor air temperature.

The Control Room Air Conditioning fans are necessary to support both the operation of the Control Room Emergency Makeup Air and Filtration and the Control Room Air Conditioning Subsystems.

3/4.7.7 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety-related systems is maintained during and following a seismic or other event initiating dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2-kip, 10-kip and 100-kip capacity manufactured by Company "A" are of the same type. The same design mechanical snubbers manufactured by Company "B" for the purposes of this Technical Specification would be of a different type, as would hydraulic snubbers from either manufacturer.

A list of individual snubbers with detailed information of snubber location and size and of system affected shall be available at the plant in accordance with Section 50.71(c) of 10 CFR Part 50. The accessibility of each snubber shall be determined and approved by the Station Operation Review Committee (SORC). The determination shall be based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operations (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10. The addition or deletion of any hydraulic or mechanical snubber shall be made in accordance with Section 50.59 of 10 CFR Part 50.

Surveillance to demonstrate OPERABILITY is by performance of the requirements of an approved inservice inspection program.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubbers for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) limits for plutonium. This limitation will ensure that leakage from Byproduct, Source, and Special Nuclear Material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with Surveillance Requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e., sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 (THIS SPECIFICATION NUMBER IS NOT USED.)

3/4.7.10 (THIS SPECIFICATION NUMBER IS NOT USED.)

3/4.8.1 AC SOURCES (Continued)

SURVEILLANCE REQUIREMENTS (SR) (continued)

regulator performance.

The SR also requires that the EDGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations at keep-warm values.

The 10-year frequency is consistent with the recommendations of RG 1.108 (Ref. 9).

MODES 5 AND 6

During operation in MODEs 5 and 6, the required AC sources include one off-site circuit capable of supplying the on-site Class 1E distribution system and an operable emergency diesel generator. These minimum AC sources ensure that (1) the unit can be maintained in the shutdown condition, (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit, and (3) adequate AC power is available to mitigate an event postulated to occur during shutdown.

If the minimum required AC sources are not operable, the action statement requires immediately suspending core alternation, positive reactivity changes, movement of irradiated fuel, and crane operation with loads over the fuel pool. With respect to suspending positive reactivity changes, operations that individually add limited, positive reactivity are acceptable when, combined with other actions that add negative reactivity, the overall net reactivity addition is zero or negative. For example, a positive reactivity addition caused by temperature fluctuations from inventory addition or temperature control fluctuations is acceptable if it is combined with a negative reactivity addition such that the overall, net reactivity addition is zero or negative. Refer to TS Bases 3/4.9.1, Boron Concentration, for limits on boron concentration and water temperature for MODE 6 action statements involving suspension of positive reactivity changes.

3/4.8.1 AC SOURCES (Continued)

REFERENCES

1. 10 CFR 50, Appendix A, GDC 17.
2. UFSAR, Chapter 8.
3. Regulatory Guide 1.9, Rev. 3. *
4. UFSAR, Chapter 6.
5. UFSAR, Chapter 15.
6. Regulatory Guide 1.93, Rev. 0, December 1974.
7. Generic Letter 84-15, "Proposed Staff ACTIONs to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
8. 10 CFR 50, Appendix A, GDC 18.
9. Regulatory Guide 1.108, Rev. 1, August 1977.*
10. Regulatory Guide 1.137, Rev. 1, October 1979.*
11. ANSI Std. C84.1
12. IEEE Std. 387-1984**
13. Generic Letter 91-04, April 1991.

3/4.8.2 and 3/4.8.3 DC SOURCES and ONSITE POWER DISTRIBUTION

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that: (1) the facility can be maintained in the shutdown or refueling condition for extended time periods and (2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

With less than the minimum required on-site power distribution systems or DC power sources, the action statement requires immediately suspending core alternations, positive reactivity changes, or movement of irradiated fuel. With respect to suspending positive reactivity changes, operations that individually add limited, positive reactivity are acceptable when, combined with other actions that add negative reactivity, the overall net reactivity addition is zero or negative. For example, a positive reactivity addition caused by temperature fluctuations from inventory addition or temperature control fluctuations is acceptable if it is combined with a negative reactivity addition such that the overall, net reactivity addition is zero or negative. Refer to TS Bases 3/4.9.1, Boron Concentration, for limits on boron concentration and water temperature for MODE 6 action statements involving suspension of positive reactivity changes.

* Seabrook Station is only committed to demonstrating the OPERABILITY of the diesel generators in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," Revision 2, December 1979; 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, Errata September 1977; and 1.137, "Fuel-Oil Systems for Standby Generators." Revision 1, October 1979. Exceptions to these Regulatory Guides are noted in the UFSAR.

** Seabrook Station is only committed to IEEE Std. 387-1972 and 1977.

3/4.8.2 and 3/4.8.3 DC SOURCES and ONSITE POWER DISTRIBUTION (continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std. 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values, and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates, and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage, and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and 0.015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than 0.020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than 0.010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7-day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than 0.020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than 0.040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The Surveillance Requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's air circuit breakers, molded case circuit breakers, and overload devices are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The OPERABILITY of the motor-operated valves thermal overload protection ensures that the thermal overload protection will not prevent safety-related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of the thermal overload protection are in accordance with Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor Operated Valves," Revision 1, March 1977.

3/4.9.5 (THIS SPECIFICATION NUMBER IS NOT USED.)

3/4.9.6 (THIS SPECIFICATION NUMBER IS NOT USED.)

3/4.9.7 (THIS SPECIFICATION NUMBER IS NOT USED.)

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal (RHR) loop be in operation ensures that: (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. However, only one RHR loop is required for decay heat removal with water level at least 23 feet above the reactor vessel flange and the upper internals removed from the reactor vessel. The large volume of water above the flange provides backup decay heat removal capability.

Administrative controls ensure two RHR loops are maintained operable with water level ≥ 23 feet above the flange when the upper internals are installed in the reactor vessel. The upper internals provide a flow restriction between the core region and the cavity such that the water volume above the flange may not support effective core cooling following a loss of the operating RHR loop with the internals installed. Following a loss of RHR with the upper internals installed, boiling in the core and core uncovery are possible while the cavity water remains relatively cool. As a result, maintaining two RHR loops operable provides back up cooling capability in the event of a failure of the operating RHR loop.

Closure of the Equipment Hatch containment penetration using the Containment Outage Door may satisfy the containment closure requirement of the action statements for Technical Specifications 3.9.8.1 and 3.9.8.2, when the Containment Outage Door is being used during the movement of non-recently irradiated fuel assemblies within containment in lieu of the Containment Equipment Hatch.