NEXTera ENERGY SEABROOK

December 19, 2012

Docket No. 50-443
SBK-L-12265

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 with the enclosed pages as follows:

REMOVE	**INSERT**
Page B 3/4 4-5	Page B 3/4 4-5
Page B 3/4 4-8	Page B 3/4 4-8
Page B 3/4 4-10	Page B 3/4 4-10
----------	Pages B 3/4 4-10a through B 3/4 4-10e
Page B 3/4 6-3	Page B 3/4 6-3
Page B 3/4 6-3a	Page B 3/4 6-3a
Page B 3/4 6-3b	Page B 3/4 6-3b
Page B 3/4 6-4	Page B 3/4 6-4
Page B 3/4 7-4	Page B 3/4 7-4

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Should you have any questions concerning this submittal, please contact me at (603) 773-7745.

Sincerely,

NextEra Energy Seabrook, LLC

Michael O'Keefe
Licensing Manager

cc: NRC Region I Administrator
NRC Project Manager, Project Directorate I-2
NRC Senior Resident Inspector

Enclosure to SBK-L-12265

3/4.4.5 STEAM GENERATORS (SG) TUBE INTEGRITY (Continued)

rupture of a SG tube that relieves to the lower pressure secondary system. The analysis assumes that contaminated fluid is released to the atmosphere through the main steam safety valves or the atmospheric steam dump valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary-to secondary leakage from all SGs of 1 gallon per minute and 500 gallons per day from any one SG or is assumed to increase to these values as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.8, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 50.67 (Ref. 3), or the NRC approved licensing basis (e.g., a small fraction of these limits).

Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.7.6.k, "Steam Generator (SG) Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria. There are three SG performance criteria: structural integrity, accident-induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

3/4.4.5 STEAM GENERATORS (SG) TUBE INTEGRITY (Continued)

If SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The shutdown times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

4.4.5.1

During shutdown periods, the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

During SG inspections, a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations. The portion of the SG tubes below 15.21 inches from the top of the tubesheet is excluded from periodic inspections and plugging.

The Steam Generator Program defines the Frequency of SR 4.4.5.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.7.6.k contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

BACKGROUND

General Design Criteria (GDC) 30 of Appendix A to 10 CFR 50 (Ref. 1) requires means for detecting and, to the extent practical, identifying the location of the source of RCS leakage. Regulatory Guide 1.45, Revision 0, (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication or warning signal is necessary to permit proper evaluation of all unidentified leakage. In addition to meeting the operability requirements, the monitors are typically set to provide the most sensitive response without causing an excessive number of spurious alarms. The containment sumps used to collect unidentified leakage are instrumented to alarm for increases above the normal flow rates.

The reactor coolant contains radioactivity that, when released to the containment, may be detected by radiation monitoring instrumentation. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS leakage.

Air temperature and pressure monitoring methods may also be used to infer unidentified leakage to the containment. Containment temperature and pressure fluctuate slightly during plant operation, but a rise above the normally indicated range of values may indicate RCS leakage into the containment. The relevance of temperature and pressure measurements is affected by containment free volume and, for temperature, detector location. Alarm signals from these instruments can be valuable in recognizing rapid and sizable leakage to the containment. Temperature and pressure monitors are not required by this LCO.

The above-mentioned leakage detection methods or systems differ in sensitivity and response time. Some of these systems could serve as early alarm systems signaling the operators that closer examination of other detection systems is necessary to determine the extent of any corrective action that may be required.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

LCO

This LCO requires instruments of diverse monitoring principles to be operable to provide confidence that small amounts of unidentified leakage are detected in time to allow actions to place the plant in a safe condition when RCS leakage indicates possible RCPB degradation. The LCO requires two instruments to be operable.

The containment sumps are used to collect unidentified leakage. There are two sumps in the containment building, one on the (-)26'-0" level, and the other on the (-)53'-4" level in the reactor instrument pit. Under normal conditions, the lower sump will always be dry as there are no drains directed to it. The LCO requirements apply to the total amount of unidentified leakage collected in both sumps. The RCS leakage monitor runs every 30 minutes and determines the amount of leakage going into the containment drain sumps. This monitor uses raw level indication, converts this to volume, and determines the difference in volume since the last execution. The monitor initiates an alarm when there is leakage of one gpm. The identification of unidentified leakage will be delayed by the time required for the unidentified leakage to travel to the containment sump, and it may take longer than one hour to detect a one gpm increase in unidentified leakage, depending on the origin and magnitude of the leakage. This sensitivity is acceptable for containment sump monitor operability.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

LCO (Continued)

The reactor coolant contains radioactivity that, when released to the containment, can be detected by the gaseous and particulate containment atmosphere radioactivity monitor. Only one of the two monitors is required to be operable. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS leakage, but have recognized limitations. Reactor coolant radioactivity levels will be low during initial reactor startup and for a few weeks thereafter, until activated corrosion products have been formed and fission products appear from fuel element cladding contamination or cladding defects. If there are few fuel element cladding defects and low levels of activation products, it may not be possible for the gaseous or particulate containment atmosphere radioactivity monitors to detect a one gpm increase within one hour during normal operation. However, the gaseous and particulate containment atmosphere radioactivity monitors are operable when they are capable of detecting a one gpm increase in unidentified leakage within one hour given an RCS activity equivalent to that assumed in the design calculations for the monitors (Reference 3).

The LCO is satisfied when monitors of diverse measurement means are available. Thus, the containment sump monitor, in combination with the gaseous or particulate radioactivity monitor, provides an acceptable minimum.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be operable.

In MODE 5 or 6, the temperature is to be $\leq$ 200°F and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation are much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

a. With the required containment sump monitor inoperable, no other form of sampling can provide the equivalent information; however, the containment atmosphere radioactivity monitor will provide indication of changes in leakage. Together with the containment atmosphere radioactivity monitor, the periodic surveillance for RCS water inventory balance, surveillance requirement (SR) 4.4.6.2.1.d, must be performed at an increased frequency of 24 hours to provide information that is adequate to detect leakage.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

ACTIONS (a) (Continued)

A footnote is added allowing that SR 4.4.6.2.1.d is not required to be performed until 12 hours after establishing steady state operation (stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Restoration of the required sump monitor to operable status within 30 days is required to regain the function after the monitor's failure. This time is acceptable, considering the frequency and adequacy of the RCS water inventory balance required by Action a.1.

b. With both the gaseous and particulate containment atmosphere radioactivity monitoring instrumentation channels inoperable, alternative action is required. Either grab samples of the containment atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 4.4.6.2.1.d, must be performed to provide alternate periodic information.

With a sample obtained and analyzed or water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the containment atmosphere radioactivity monitors.

The 24 hour interval provides periodic information that is adequate to detect leakage. A footnote is added allowing that SR 4.4.6.2.1.d is not required to be performed until 12 hours after establishing steady state operation (stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The 30 day restoration time recognizes at least one other form of leakage detection is available.

c. With the containment sump monitor and the containment atmosphere particulate radioactivity monitor inoperable, the only means of detecting leakage is the containment gaseous monitor. The containment atmosphere gaseous radioactivity monitor typically cannot detect a one gpm leak within one hour when RCS activity is low. In addition, this configuration does not provide the required diverse means of leakage detection. Indirect methods of monitoring RCS leakage must be implemented. Grab samples of the containment atmosphere must be taken to provide alternate periodic information.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

ACTIONS (c) (Continued)

The 12 hour interval is sufficient to detect increasing RCS leakage. The Action provides 7 days to restore another RCS leakage monitor to operable status to regain the intended leakage detection diversity. The 7 day restoration time ensures that the plant will not be operated in a degraded configuration for a lengthy time period. Two leakage detections systems must be restored to operable status within 30 days to meet the LCO or the plant must shutdown.

SURVEILLANCE REQUIREMENTS

SR 4.4.6.1.a.1

SR 4.4.6.1.a.1 requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 4.4.6.1.a.2

SR 4.4.6.1.a.2 requires the performance of a digital channel operational test on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The frequency of 92 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 4.4.6.1.a.3 and 4.4.6.1.b

These SRs require the performance of a channel calibration for each of the RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, Section IV, GDC 30.

2. Regulatory Guide 1.45, Revision 0, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3. FSAR, Section 5.2.5.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.2 OPERATIONAL-LEAKAGE

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational Leakage LCO is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the identified leakage from the unidentified leakage is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

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.

The Containment Building Spray System suction and discharge piping must be maintained full of water to ensure system operability. The piping may be considered full of water, even with some gas voids present, if an evaluation concludes that the system remains capable of performing its specified safety function.

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).

Surveillance requirement (SR) 4.6.2.1.d requires verification that each spray nozzle is unobstructed following activities that could cause nozzle blockage. An air or smoke flow test is used to ensure that each spray nozzle is unobstructed and that spray coverage of the containment during an accident is not degraded. Normal plant activities are not expected to initiate this SR. However, activities such as inadvertent spray actuation that causes fluid flow through the spray nozzles or a loss of foreign material control when working on the system may require performing the surveillance.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.2 SPRAY ADDITIVE SYSTEM (Continued)

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.

The Spray Additive Tank System piping must be maintained full of water to ensure system operability. The piping may be considered full of water, even with some gas voids present, if an evaluation concludes that the system remains capable of performing its specified safety function.

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 (CIV) 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

3/4.6.3 CONTAINMENT ISOLATION VALVES (Continued)

containment and is consistent with the requirements of General Design Criteria 54 through 57 of Appendix A to 10 CFR Part 50. 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.

The containment isolation system is designed to isolate penetrations that are not required for operation of the engineered safety features systems in the event of a LOCA or main steam line break. The containment isolation devices are either passive or active. Automatic containment isolation valves are active devices that are designed to close without operator action within time limits on a containment isolation signal following an accident. Passive devices are normally closed barriers that require no mechanical movement to perform their isolation function. Passive containment isolation barriers are operable when their applicable surveillance requirements are met and:

1. Manual valves are locked in the closed position,
2. Automatic valves are de-activated and locked in the closed position,
3. Blind flanges are in place, and
4. Closed systems are intact.

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.

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.

3/4.6.3 CONTAINMENT ISOLATION VALVES (Continued)

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 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.

3/4.7.1 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 remain open to remove condensate from the EFW steam supply lines.
- 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.