FPL Energy
Seabrook Station

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SBK-L-08216

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 Specification Bases in accordance with Enclosure 1.

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

Sincerely,

FPL Energy Seabrook, LLC

Michael O'Keefe

Michael O'Keefe
Licensing Manager

cc: S. J. Collins, NRC Region I Administrator
G. E. Miller, NRC Project Manager, Project Directorate I-2
W. J. Raymond, NRC Senior Resident Inspector

A001
NRR

Enclosure 1 to SBK-L-08216

Change Instructions for Seabrook Station Technical Specification Bases

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Enclosure 2 to SBK-L-08216

2.1.2 Reactor Coolant System (RCS) Pressure SL

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 3), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with GDC 28, "Reactivity Limits" (Ref. 3), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 4). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 5).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 50.67, "Accident source term" (Ref. 6).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 3). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

The Reactor Trip System setpoints (Ref. 2), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 2). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

2.1.2 Reactor Coolant System (RCS) Pressure SL (continued)

More specifically, no credit is taken for operation of any of the following:

a. Pressurizer power operated relief valves (PORVs),

b. Steam line relief valve,

c. Steam Dump System,

d. Reactor Control System,

e. Pressurizer Level Control System, or

f. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 7) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES due to overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized.

2.1.3 SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable:

If the reactor core SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable. The Allowed Outage Time (Completion Time) of 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour. Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 50.67, "Accident Source Term," limits (Ref. 6). The Allowed Outage Time (Completion Time) of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

SAFETY LIMIT VIOLATIONS (continued)

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, or 5 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 10.
2. UFSAR, Chapters 7 and 15.
3. 10 CFR 50, Appendix A, GDC 14, GDC 15, and GDC 28.
4. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
5. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
6. 10 CFR 50.67, Accident Source Term.
7. USBS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide core protection against DNB as a result of complete loss of forced coolant flow. The specified Setpoints assure a Reactor trip signal is generated before the Low Flow Trip Setpoint is reached. Time delays are incorporated in the Underfrequency and Undervoltage trips to prevent spurious Reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the Reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.5 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the Reactor trip breakers after the Underfrequency Trip Setpoint is reached shall not exceed 0.6 second. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power, the Reactor trip from the Turbine trip is automatically blocked by P-9 (a power level of approximately 45% of RATED THERMAL POWER); and on increasing power, the Reactor trip from the Turbine trip is reinstated automatically by P-9.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels that initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

P-6 On increasing power, P-6 allows the manual block of the Source Range trip (i.e., prevents premature block of Source Range trip). On decreasing power, Source Range Level trips are automatically reactivated and high voltage is restored.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

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

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 6 hours.

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

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

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

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

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

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION (Continued)

If the RCS is drained to the point where additional air is available to enter the steam generators, i.e., to a reduced inventory condition [El.(-)36"], then the air/gas must be removed from the steam generator tubes prior to the steam generators being available as a heat sink. This will require either the removal of the air from the Reactor Coolant System via the RCS evacuation skid or operating each reactor coolant pump for a short duration to sweep air from the Steam Generator U-tubes (only required for those generators to be credited for decay heat removal). Operating the reactor coolant pumps to sweep the loops re-establishes the conditions that existed prior to draining the RCS. Using the evacuation skid results in a larger volume of air/gas contained in the steam generator u-tubes than exists under the initial shutdown conditions, however Engineering Evaluations EE-08-012 demonstrates the natural circulation conditions will be established for this circumstance.

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP in MODES 4 and 5 are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold-leg temperatures.

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)

SR 4.4.5.2

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.7.6.k are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 50.67

4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.

5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

APPLICABLE SAFETY ANALYSIS

Except for primary to secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary leakage from all steam generators (SGs) is one gallon per minute and 500 gallons per day from any one SG or increases to these values as a result of accident-induced conditions. The LCO requirement to limit primary to secondary leakage through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analyses for SLB and SGTR assume one gallon per minute primary to secondary leakage. For the SLB, the tube leakage is conservatively apportioned as 500 gpd to the faulted SG and 940 gpd total to the other three SGs in order to maximize dose consequences. Similarly, the SGTR analysis assumes the tube leakage is 313.33 gpd to the faulted SG and 1126.67 gpd total to the other three SGs in order to maximize dose consequences. The dose consequences resulting from these accidents are within the limits defined in 10 CFR 50.67 or the staff approved licensing basis (i.e., a small fraction of these limits). The RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational leakage shall be limited to:

Pressure Boundary Leakage

No pressure boundary leakage is allowed, being indicative of material deterioration. Leakage of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher leakage. Violation of this LCO could result in continued degradation of the RCPB. Leakage past seals and gaskets is not pressure boundary leakage.

3/4.4.7 (THIS SPECIFICATION NUMBER IS NOT USED)

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR 50.67 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Seabrook site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guidelines of 10 CFR 50.67 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

Primary containment OPERABILITY is maintained by limiting leakage to $\leq 1.0\ L_a$, except prior to the first startup after performing a required Primary Containment Leakage Rate Testing Program leakage test. At this time, applicable leakage limits must be met. The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to $0.75\ L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is in accordance with the Containment Leakage Rate Testing Program.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 3.5 psi and (2) the containment peak pressure does not exceed the design pressure of 52 psig during LOCA conditions.

The maximum peak pressure expected to be obtained from a LOCA event is 49.6 psig. The limit of 16.2 psia for initial positive containment pressure will limit the total pressure to 49.6 psig which is less than the design pressure and is consistent with the safety analyses.

3/4.6.1 PRIMARY CONTAINMENT (Continued)

3/4.6.1.5 AIR TEMPERATURE

The limitation in containment average air temperature ensures that the containment average air temperature does not exceed the initial temperature condition assumed in the overall safety analysis for a steam line break accident. Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 52 psig in the event of a LOCA. A visual inspection in accordance with the Containment Leakage Rate Testing Program demonstrates this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 36-inch containment shutdown purge supply and exhaust isolation valves are not utilized during plant operation in MODES 1, 2, 3, and 4. A blind flange is installed establishing a Type "B" penetration. The penetration is surveilled in accordance with Surveillance Requirement 4.6.1.1a in MODES 1, 2, 3, and 4.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 36-inch valves, the 8-inch valves are capable of closing during a LOCA or steam line break accident. Therefore, the SITE BOUNDARY dose guideline values of 10 CFR 50.67 would not be exceeded in the event of an accident during containment PURGING operation. The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is determined by the actual need for opening the valves for safety-related reasons; e.g., containment pressure control or the reduction of airborne radioactivity to facilitate personnel access for surveillance and maintenance activities.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The containment purge supply and exhaust isolation valves are leakage rate tested in accordance with the Containment Leakage Rate Testing Program.

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 (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 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.

Normally closed containment isolation barriers are considered 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.

When these containment isolation barriers are administratively placed in a condition in which no mechanical movement needs to occur for the components to perform their intended function, the barriers are considered passive components. This provision does not apply to power-operated valves that are de-activated for maintenance or tagging purposes; those conditions are addressed below.

3/4.6.3 CONTAINMENT ISOLATION VALVES (Continued)

Closing, deactivating, and securing an operable, fail closed, automatic CIV to isolate a containment penetration to comply with the action of TS 3.6.3 because the redundant CIV is inoperable does not necessitate declaring the valve inoperable provided:

a. The CIV is operable and the valve will revert to an operable active isolation device upon restoration of power or the opening air supply, and

b. No maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit that would require a surveillance test to demonstrate operability of the CIV, and

c. If the CIV is a dual function valve that renders a TS-required system or component inoperable while deactivated and closed, entry into and compliance with the actions for the inoperable TS-required system or component is necessary.

Closing, deactivating, and securing an operable, fail closed, automatic CIV so that it may serve as an isolation boundary for a clearance order does not necessitate declaring the valve inoperable provided:

a. The CIV is operable and the valve will revert to an operable active isolation device upon restoration of power or the opening air supply, and

b. No maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit that would require a surveillance test to demonstrate operability of the CIV, and

c. If the CIV is a dual function valve that renders a TS-required system or component inoperable while deactivated and closed, entry into and compliance with the actions for the inoperable TS-required system or component is necessary.

In addition, a fail closed containment isolation valve that is closed, deactivated, and secured for purposes other than those discussed above may also be considered operable provided the stipulations in items a, b, and c above are met.

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.

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 50.67 during accident conditions.

Verifying that the enclosure boundary is intact, or has integrity, involves confirming that the doors are closed except during normal transit entry and exit. Additionally, pressure boundary seals must also be intact to maintain the integrity of the containment enclosure.

3/4.7.1 TURBINE CYCLE (Continued)

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.

3/4.7.1.4 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 50.67 (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 50.67 (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

3/4.7.1 TURBINE CYCLE (Continued)

3/4.7.1.4 SPECIFIC ACTIVITY (Continued)

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).