Nuclear Operating Company
South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

August 9, 2022
NOC-AE-22003908
10 CFR 50.90
STI: 35348372

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

South Texas Project
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
Application to Revise Technical Specifications to Adopt
TSTF-554, "Revise Reactor Coolant Leakage Requirements"

Pursuant to 10 CFR 50.90, STP Nuclear Operating Company (STPNOC) is submitting a request for an amendment to the Technical Specifications (TS) for South Texas Project (STP), Units 1 and 2.

STPNOC requests adoption of TSTF-554, "Revise Reactor Coolant Leakage Requirements," which is an approved change to the Standard Technical Specifications (STS), into the STP, Units 1 and 2, TS. The proposed amendment revises the TS definitions of "leakage," clarifies the requirements when pressure boundary leakage is detected, and adds an Action when pressure boundary leakage is identified. The change is requested as part of the Consolidated Line Item Improvement Process (CLIIP).

The enclosure provides a description and assessment of the proposed changes. Attachment 1 provides the existing TS pages marked to show the proposed changes. Attachment 2 provides revised (clean) TS pages. Attachment 3 provides existing TS Bases pages marked to show the proposed changes for information only.

This letter contains no new regulatory commitments.

Approval of the proposed amendment is requested within 6 months of completion of the NRC's acceptance review. Once approved, the amendment shall be implemented within 90 days.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated State of Texas Official.

Should you have any questions regarding this submission, please contact Zachary Dibbern at (361) 972-4336 or me at (361) 972-4778.

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

Executed on August 9, 2022

Kimberly Harshaw
Executive VP and CNO

Enclosure: Application to Revise Technical Specifications to Adopt TSTF-554, "Revise Reactor Coolant Leakage Requirements"

cc:

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
1600 E. Lamar Boulevard
Arlington, TX 76011-4511

ENCLOSURE

Description and Assessment

Subject: Application to Revise Technical Specifications to Adopt TSTF-554, "Revise Reactor Coolant Leakage Requirements"

ATTACHMENTS:
1. Proposed Technical Specification Changes (Markup)
2. Revised Technical Specification Pages
3. Proposed Technical Specification Bases Changes (Markup) for Information Only

1 DESCRIPTION

STP Nuclear Operating Company (STPNOC) requests adoption of TSTF-554, "Revise Reactor Coolant Leakage Requirements," which is an approved change to the Standard Technical Specifications (STS), into the South Texas Project (STP), Units 1 and 2, Technical Specifications (TS). The proposed amendment revises the TS definitions of "leakage," clarifies the requirements when pressure boundary leakage is detected, and adds an Action when pressure boundary leakage is identified.

2 ASSESSMENT

2.1 Applicability of Safety Evaluation

STPNOC has reviewed the safety evaluation for TSTF-554 provided to the Technical Specifications Task Force in a letter dated December 18, 2020. This review included a review of the NRC staff's evaluation, as well as the information provided in TSTF-554. As described herein, STPNOC has concluded that the justifications presented in TSTF-554 and the safety evaluation prepared by the NRC staff are applicable to STP, Units 1 and 2, and justify this amendment for the incorporation of the changes into the STP TS.

2.2 Optional Changes and Variations.

STPNOC is proposing the following variations from the TS changes described in TSTF 554 or the applicable parts of the NRC staff's safety evaluation:

The STP TS contain requirements that differ from the STS on which TSTF-554 was based but are encompassed in the TSTF-554 justification.

- The STP TS are based on NUREG-0452, "Standard Technical Specifications, Westinghouse Plants." TSTF-554 is based on NUREG-1431 of the same title.
- In the STS on which TSTF-554 is based, all leakage definitions appear as part of a single defined term, "LEAKAGE." In the STP TS, the terms related to leakage are separate definitions. However, the changes to the affected defined terms "identified leakage" and "pressure boundary leakage" are the same as the changes made to those terms in TSTF-554.
- In the STP TS, the "Operational Leakage" specification is numbered 3.4.6.2. In the STS on which TSTF-554 is based, the TS is numbered 3.4.13.
- In TSTF-554, a new Action A is created that is applicable when there is pressure boundary leakage. In the STP TS, existing Action a applies when there is pressure boundary leakage or primary-to-secondary leakage not within limit. The STP TS are revised such that Action a is applicable to the existence of pressure boundary leakage and a new Action c is created that applies the existing actions when primary-to-secondary leakage is not within limit. This presentation and order are consistent with TSTF-554. The existing STP Action c is renamed Action d.
- In the STP TS, shutdown requirements are included in each Action. The TSTF 554 shutdown actions in STS Condition C are incorporated into STP Actions a and c. The requirements are consistent with TSTF-554.
- The STP TS Bases for the "Operational Leakage" specification are similar to the STS Bases. As a result, the TSTF-554 Bases changes are incorporated into the STP TS Bases without change.

3 REGULATORY ANALYSIS

3.1 No Significant Hazards Consideration Determination

STP Nuclear Operating Company (STPNOC) requests adoption of TSTF-554, "Revise Reactor Coolant Leakage Requirements," which is an approved change to the Standard Technical Specifications (STS), into the South Texas Project (STP), Units 1 and 2, Technical Specifications (TS). The proposed amendment revises the TS definitions of "leakage," clarifies the requirements when pressure boundary leakage is detected, and adds an Action when pressure boundary leakage is identified.

STPNOC has evaluated whether a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

(1) Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change revises the TS definitions of "leakage," clarifies the requirements when pressure boundary leakage is detected, and adds an Action when pressure boundary leakage is identified.

The proposed change revises the definition of pressure boundary leakage. Pressure boundary leakage is a precursor to some accidents previously evaluated. The proposed change expands the definition of pressure boundary leakage by eliminating the qualification that pressure boundary leakage must be from a "nonisolable" flaw. A new TS Action is created which requires isolation of the pressure boundary flaw from the Reactor Coolant System (RCS). This new action provides assurance that the flaw will not result in any accident previously evaluated.

Pressure boundary leakage, and the actions taken when pressure boundary leakage is detected, is not assumed in the mitigation of any accident previously evaluated. As a result, the consequences of any accident previously evaluated are unaffected.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change revises the TS definitions of "leakage," clarifies the requirements when pressure boundary leakage is detected, and adds an Action when pressure boundary leakage is identified. The proposed change does not alter the design function or operation of the RCS. The proposed change does not alter the ability of the RCS to perform its design function. Since pressure boundary leakage is an evaluated accident, the proposed change does not create any new failure

mechanisms, malfunctions, or accident initiators not considered in the design and licensing bases.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No

The proposed change revises the TS definitions of "leakage," clarifies the requirements when pressure boundary leakage is detected, and adds an Action when pressure boundary leakage is identified. The proposed change does not affect the initial assumptions, margins, or controlling values used in any accident analysis. The amount of leakage allowed from the RCS is not increased. The proposed change does not affect any design basis or safety limit or any Limiting Condition for Operation.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

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

3.2 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

4 ENVIRONMENTAL CONSIDERATION

The proposed change would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

Attachment 1

Proposed Technical Specification Changes (Markup)

Ē - AVERAGE DISINTEGRATION ENERGY

1.12 Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

FREQUENCY NOTATION

1.14 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS WASTE PROCESING SYSTEM

1.15 A GASEOUS WASTE PROCESSING SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known ~~either~~ **to** not ~~to~~ interfere with the operation of Leakage Detection Systems ~~or not to be PRESSURE BOUNDARY LEAKAGE~~, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

MASTER RELAY TEST

1.17 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

MEMBER OF THE PUBLIC

1.18 MEMBER(S) OF THE PUBLIC means an individual in a controlled area or UNRESTRICTED AREA. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

1.19 Not Used

OPERABLE - OPERABILITY

1.20 A system, subsystem, train, component or device shall be OPERABLE or have OPERABLITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.21 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.22 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.23 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a ~~nonisolable~~ fault in a Reactor Coolant System component body, pipe wall, or vessel wall. **Leakage past seals, packing, and gaskets is not PRESSURE BOUNDARY LEAKAGE.**

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 150 gallons per day of primary-to-secondary leakage through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE**, isolate the affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve within 4 hours or** ~~, or with primary-to-secondary leakage not within limit,~~ be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With Reactor Coolant System operational UNIDENTIFIED or IDENTIFIED LEAKAGE greater than the above limits, reduce leakage to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With primary-to-secondary leakage not within the limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours

~~c~~**d**. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

Attachment 2

Revised Technical Specification Pages

Ē - AVERAGE DISINTEGRATION ENERGY

1.12 Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURES RESPONSE TIME

1.13 The ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF Actuation Setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

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1.16 IDENTIFIED LEAKAGE shall be:

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b. Leakage into the containment atmosphere from sources that are both specifically located and known to not interfere with the operation of Leakage Detection Systems, or

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

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1.19 Not Used

OPERABLE - OPERABILITY

1.20 A system, subsystem, train, component or device shall be OPERABLE or have OPERABLITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE - MODE

1.21 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level, and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

1.22 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 14.0 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

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1.23 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a fault in a Reactor Coolant System component body, pipe wall, or vessel wall. Leakage past seals, packing, and gaskets is not PRESSURE BOUNDARY LEAKAGE.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 150 gallons per day of primary-to-secondary leakage through any one steam generator,

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

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

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, isolate the affected component, pipe, or vessel from the RCS by use of a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With Reactor Coolant System operational UNIDENTIFIED or IDENTIFIED LEAKAGE greater than the above limits, reduce leakage to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With primary to secondary leakage not within the limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours

d. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

Attachment 3

Proposed Technical Specification Bases Changes (Markup) for Information Only

There are no changes to this page.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

RCS Leakage Detection instrumentation consists of one Containment Atmosphere Radioactivity Monitor (particulate), and the Containment Normal Sump Level and Flow Monitoring System.

The Containment Normal Sump Level and Flow Monitoring System leakage detection method is accomplished by monitoring the containment sump using two independent methods. One method is the Flow Monitoring System. This method measures the volume of water pumped out of the sump over a period of time and calculates an average leak rate. The other volumetric method involves measuring a change in the Containment Normal Sump Level over time, which also provides a means of manually or automatically calculating an average leak rate. Since both of these methods provide a means to detect average leak rate, they are redundant. OPERABILITY of the Containment Normal Sump Level and Flow Monitoring System is dependent on the operability of LI-7812 "Containment Normal Sump Level" **or** FQI-7823 "Containment Normal Sump Discharge."

3/4.4.6.2 OPERATIONAL LEAKAGE

Background

Components that contain or transport the coolant to or from the reactor core make up the reactor coolant system (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.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, 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.

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).

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

Applicable Safety Analyses

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 a 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 is 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary-to-secondary leakage through any one steam generator to less than or equal to 150 gpd 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 UFSAR analysis for SGTR assumes the contaminated secondary fluid is only briefly released via the main steam safety valves and the majority is steamed to the condenser. The 1 gpm primary-to-secondary leakage safety analysis assumption is relatively inconsequential.

The SLB is more limiting for primary-to-secondary leakage. The safety analysis for the SLB assumes 500 gpd and 936 gpd primary-to-secondary leakage in the faulted and intact steam generators respectively as an initial condition. The dose consequences resulting from the SLB accident are bounded by a small fraction (i.e., 10%) of the limits defined in 10 CFR 100. The RCS specific activity assumed was 1.0 µCi/gm DOSE EQUIVALENT 1-131 at a conservatively high letdown flow of 250 gpm, with either a pre-existing or an accident initiated iodine spike. These values bound the Technical Specifications values.

The RCS operational leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

~~No~~ PRESSURE BOUNDARY LEAKAGE is **prohibited** ~~allowed, being indicative of material deterioration. Leakage of this type is unacceptable~~ as the leak itself could cause further **RCPB** deterioration, resulting in higher leakage. ~~Violation of this LCO could result in continued degradation of the Reactor Coolant Pressure Boundary. Leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.~~

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

b. UNIDENTIFIED LEAKAGE

One gallon per minute (gpm) of UNIDENTIFIED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. **Separating the sources of leakage (i.e., leakage from an identified source versus leakage from an unidentified source) is necessary for prompt identification of potentially adverse conditions, assessment of the safety significance, and corrective action.** ~~Violation of this LCO could result in continued degradation of the Reactor Coolant Pressure Boundary, if the leakage is from the pressure boundary.~~

Leakage from systems connected to the Reactor Coolant System will initially manifest itself as UNIDENTIFIED LEAKAGE until the source of the leak is identified. If the leakage exceeds the 1 gpm limit for UNIDENTIFIED LEAKAGE, then Action b is entered. When the source of the leakage is identified and UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are verified within limits, Action b can be exited.

c. Primary-to-Secondary Leakage Through Any One Steam Generator

The limit of 150 gpd per each steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 1). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, “The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the Reactor Coolant System Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include ~~PRESSURE BOUNDARY LEAKAGE or~~ controlled reactor coolant pump seal leakoff (a normal function not considered leakage). ~~Violation of this LCO could result in continued degradation of a component or system.~~

e. Reactor Coolant System 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 overpressurization 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.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

Applicability

In MODES 1, 2, 3, and 4, the potential for Reactor Coolant Pressure Boundary leakage is greatest when the Reactor Coolant System is pressurized.

In MODES 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage.

ACTIONS

a. If any PRESSURE BOUNDARY LEAKAGE exists, the affected component, pipe, or vessel must be isolated from the RCS by a closed manual valve, closed and de-activated automatic valve, blind flange, or check valve within 4 hours, or the reactor must be brought to MODE 3 within 6 hours and MODE 5 within the next 30 hours. While in this condition, structural integrity of the system should be considered because the structural integrity of the part of the system within the isolation boundary must be maintained under all licensing basis conditions, including consideration of the potential for further degradation of the isolated location. Normal leakage past the isolation device is acceptable as it will limit RCS LEAKAGE and is included in IDENTIFIED LEAKAGE or UNIDENTIFIED LEAKAGE. This action is necessary to prevent further deterioration of the RCPB. ~~or primary-to-secondary leakage is not within limit, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within the next 30 hours. This ACTION reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.~~

The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the Reactor Coolant Pressure Boundary are much lower, and further deterioration is much less likely.

b. UNIDENTIFIED LEAKAGE or IDENTIFIED LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. This ACTION is necessary to prevent further deterioration of the Reactor Coolant Pressure Boundary.

c. If any primary-to-secondary leakage is not within limit, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within the next 30 hours. This ACTION reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

Surveillance Requirements

4.4.6.2.1 Verifying Reactor Coolant System leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. ~~It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.~~ UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of a Reactor Coolant System water inventory balance.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. The Surveillance is modified by two Notes. Note 1 states that this Surveillance Requirement is not required to be performed in until 12 hours after establishment of steady state operation.

Steady state operation is required to perform a proper water inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows.

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity, containment normal sump inventory and discharge, and reactor head flange leakoff. ~~It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE.~~ These leakage detection systems are specified in LCO 3.4.6.1, “Reactor Coolant System Leakage Detection Systems.”

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

The Note in 4.4.6.2.1 states that this Surveillance Requirement is not applicable to primary-to-secondary leakage. This is because leakage of 150 gpd cannot be measured accurately by a RCS water inventory balance.

The frequency specified in the Surveillance Frequency Control Program is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

4.4.6.2.2 The Surveillance Requirements for Reactor Coolant System Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

4.4.6.2.3 This Surveillance Requirement verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one steam generator. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5 should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 1. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator.

The Surveillance Requirement is modified by a Note, which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For Reactor Coolant System primary-to-secondary leakage determination, steady state is defined as stable Reactor Coolant System pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows.

The frequency specified in the Surveillance Frequency Control Program is a reasonable interval to trend primary-to-secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. During normal operation the primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling. In MODES 3 and 4, the primary system radioactivity level may be very low, making it difficult to measure primary-to-secondary leakage. Leakage verification is provided by chemistry procedures that provide alternate means of calculating and confirming primary-to-secondary leakage is less than or equal to 150 gpd through any one SG (Ref. 2).

References

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

2. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines"

3/4.4.7 NOT USED