VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

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VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
SURRY POWER STATION UNITS 1 AND 2
ANNUAL SUBMITTAL OF TECHNICAL SPECIFICATIONS BASES CHANGES
PURSUANT TO TECHNICAL SPECIFICATION 6.4.J

Pursuant to Technical Specification 6.4.J, "Technical Specifications (TS) Bases Control Program," Dominion hereby submits the changes to the Bases of the Surry TS implemented since September 30, 2006. A summary of these changes is provided in Attachment 1.

Each TS Bases change was reviewed and approved by the Station Nuclear Safety and Operating Committee. It was determined that the changes did not require a change to the TS or license, or involve a change to the UFSAR or Bases that required NRC prior approval pursuant to 10CFR50.59. These TS Bases changes were submitted to the NRC for information along with the associated License Amendment Request transmittals, submitted pursuant to 10CFR50.90, and have been implemented with the respective License Amendments.

Current TS Bases pages reflecting the changes discussed in Attachment 1 are provided in Attachment 2.

If you have any questions regarding this transmittal, please contact Mr. Gary D. Miller at (804) 273-2771.

Very truly yours,

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Attachments:
1. Summary of TS Bases Changes
2. Current TS Bases Pages

Commitments made in this letter: None.

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Attachment 1

Summary of TS Bases Changes

Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

SUMMARY OF TS BASES CHANGES

Proposed Technical Specifications Change and Supporting Safety Analysis Revisions to Address Generic Safety Issue (GSI) 191 (TS Bases Pages TS 3.4-3, TS 3.8-4, TS 3.8-5, and TS 3.19-2)

As part of the resolution to GSI-191, this change revised the method for starting the inside and outside recirculation spray (RS) pumps in response to a design basis accident. The RS pump start based on a time delay following a Consequence Limiting Safeguards (CLS) High High containment pressure setpoint was revised to a coincident CLS High High containment pressure and refueling water storage tank low level.

The associated Bases changes were included for information in a January 31, 2006 letter (Serial No. 06-014) and were incorporated into the Bases as part of the Unit 2 implementation of License Amendment --/249 issued on October 12, 2006. [Note that the Unit 1 License Amendment 250/-- will be implemented before the end of the Unit 1 refueling outage later this year.]

Steam Generator Tube Integrity (TS Bases Pages TS 3.1-14, TS 3.1-14a, TS 3.1-14b, TS 3.1-15, TS 3.1-26, TS 3.1-27, TS 3.1-28, TS 3.1-29, TS 3.1-30, TS 3.1-31, TS 4.13-1, TS 4.13-2, TS 4.19-1, and TS 4.19-2)

This change revised the TS requirements related to steam generator tube integrity and Reactor Coolant System leakage definitions and requirements. This change is consistent with the NRC-approved Revision 4 of Technical Specifications Task Force (TSTF) Standard Technical Specifications Change Traveler TSTF-449, "Steam Generator Tube Integrity."

The associated Bases changes were included for information in a January 19, 2007 letter (Serial No. 06-1021) and were incorporated into the Bases as part of the implementation of License Amendments 251/250 issued on March 29, 2007.

Revision of Main Control Room (MCR) and Emergency Switchgear Room (ESGR) Air Conditioning System Requirements (TS Bases Pages TS 3.10-6, TS 3.23-3, TS 3.23-4, and TS 3.23-5)

This change revised the MCR and ESGR air conditioning system TS requirements to reflect the completion of permanent modifications to the equipment and associated power supply configurations.

The associated Bases changes were included for information in a July 5, 2006 letter (Serial No. 06-387) and were incorporated into the Bases as part of the implementation of License Amendments 252/251 issued on April 2, 2007.

Missed Surveillance Requirements (TS Bases Pages TS 4.0-2, TS 4.0-3, TS 4.0-4, and TS 4.0-5)

This change revised the TS surveillance requirements for addressing a missed surveillance. This change is consistent with the NRC-approved Revision 6 of Technical Specifications Task Force (TSTF) Standard Technical Specifications Change Traveler TSTF-358, "Missed Surveillance Requirements."

The associated Bases changes were included for information in a January 31, 2007 letter (Serial No. 07-0069) and were incorporated into the Bases as part of the implementation of License Amendments 253/252 issued on May 3, 2007.

Attachment 2

Current TS Bases Pages

Surry Power Station Units 1 and 2
Virginia Electric and Power Company
(Dominion)

BASES

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 unit 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 limiting condition for operation (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, 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.

Leakage from the RCS is collected in the containment or by other systems. These systems are the Main Steam System, Condensate and Feedwater System, the Gaseous and Liquid Waste Disposal Systems, the Component Cooling System, and the Chemical and Volume Control System.

Detection of leaks from the RCS is by one or more of the following:

1. An increased amount of makeup water required to maintain normal level in the pressurizer.
2. A high temperature alarm in the leakoff piping provided to collect reactor head flange leakage.
3. Containment sump water level indication.
4. Containment pressure, temperature, and humidity indication.

If there is significant radioactive contamination of the reactor coolant, the radiation monitoring system provides a sensitive indication of primary system leakage. Radiation monitors which indicate primary system leakage include the containment gas and particulate monitors, the condenser air ejector monitor, the component cooling water monitor, and the steam generator blowdown monitor.

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).

<u>APPLICABLE SAFETY ANALYSES</u> - 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 1 gpm or increases to 1 gpm 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 main steam line break (MSLB) accident. 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 (Ref. 2) analysis for SGTR assumes the contaminated secondary fluid is released via power operated relief valves or safety valves. The source term in the primary system coolant is transported to the affected (ruptured) steam generator by the break flow. The affected steam generator discharges steam to the environment for 30 minutes until the generator is manually isolated. The 1 gpm primary to secondary LEAKAGE transports the source term to the unaffected steam generators. Releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service.

The MSLB is less limiting for site radiation releases than the SGTR. The safety analysis for the MSLB accident assumes 1 gpm total primary to secondary LEAKAGE, including 500 gpd leakage into the faulted generator. The dose consequences resulting from the MSLB and the SGTR accidents are within the limits defined in the plant licensing basis.

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

<u>LIMITING CONDITIONS FOR OPERATION</u> - RCS operational LEAKAGE shall be limited to:

a. <u>Pressure Boundary LEAKAGE</u>

   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.

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. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. 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 RCS 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 (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 3). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG 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.

APPLICABILITY - In REACTOR OPERATION conditions where T_{avg} exceeds 200°F, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In COLD SHUTDOWN and REFUELING SHUTDOWN, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

LCO 3.1.C.5 measures leakage through each individual pressure isolation valve (PIV) and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

3.1.C.2.a

Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This completion time 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 RCPB.

3.1.C.2.b and 3.1.C.3

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 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 unit conditions from full power conditions in an orderly manner and without challenging unit systems. In COLD SHUTDOWN, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

REFERENCES

1. UFSAR, Chapter 4, Surry Units 1 and 2.
2. UFSAR, Chapter 14, Surry Units 1 and 2.
3. NEI 97-06, "Steam Generator Program Guidelines."
4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

H. Steam Generator (SG) Tube Integrity

Applicability

The following specifications are applicable whenever T_{avg} (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Specifications

1. SG tube integrity shall be maintained, and all SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.
2. If the requirements of 3.1.H.1 are not met for one or more SG tubes, then perform the following:[1]
   a. Within 7 days, verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection; and
   b. Plug the affected tube(s) in accordance with the Steam Generator Program prior to T_{avg} exceeding 200°F following the next refueling outage or SG tube inspection.
3. If the required actions of Specification 3.1.H.2 are not completed within the specified completion time, or SG tube integrity is not maintained, the unit shall be brought to HOT SHUTDOWN within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

Note:

1. A separate TS action entry is allowed for each SG tube.

BASES

BACKGROUND - Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.1.A.2.

SG tube integrity means that the tubes are capable of performing their intended RCPB safety function consistent with the licensing basis, including applicable regulatory requirements.

Steam generator tubing is subject to a variety of degradation mechanisms. Steam generator tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.4.Q, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.4.Q, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 6.4.Q. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Ref. 1).

<u>APPLICABLE SAFETY ANALYSES</u> - The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of an SGTR event assumes a bounding primary to secondary LEAKAGE rate of 1 gpm, which is conservative with respect to the operational LEAKAGE rate limits in Specification 3.1.C, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The UFSAR analysis for SGTR assumes the contaminated secondary fluid is released via power operated relief valves or safety valves. The source term in the primary system coolant is transported to the affected (ruptured) steam generator by the break flow. The affected steam generator discharges steam to the environment for 30 minutes until the generator is manually isolated. The 1 gpm primary to secondary LEAKAGE transports the source term to the unaffected steam generators. Releases continue through the unaffected steam generators until the Residual Heat Removal System is placed in service.

The analyses 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 or is assumed to increase to 1 gallon per minute 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 within Specification 3.1.D, "Maximum Reactor Coolant 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 Regulatory Guide 1.183 (Ref. 4), as appropriate.

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

LIMITING CONDITIONS FOR OPERATION - 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 an 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 6.4.Q, "Steam Generator 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.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that significantly affect burst or collapse. In that context, the term "significantly" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 5) and Draft Regulatory Guide 1.121 (Ref. 6).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in Specification 3.1.C, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

<u>APPLICABILITY</u> - Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced when T_{avg} exceeds 200°F.

RCS conditions are far less challenging in COLD SHUTDOWN and REFUELING SHUTDOWN than during INTERMEDIATE SHUTDOWN, HOT SHUTDOWN, REACTOR CRITICAL and POWER OPERATION. In COLD SHUTDOWN and REFUELING SHUTDOWN, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.

<u>ACTIONS</u> - The actions are modified by a Note clarifying that the conditions may be entered independently for each SG tube. This is acceptable because the required actions provide appropriate compensatory actions for each affected SG tube. Complying with the required actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent condition entry and application of associated required actions.

3.1.H.2.a and b

Specification 3.1.H.2 applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 4.19. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Specification 3.1.H.3 applies.

A completion time of 7 days is sufficient to complete the evaluation while minimizing the risk of unit operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, required action 3.1.H.2.b allows unit operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to T_{avg} exceeding 200°F following the next refueling outage or SG inspection. This completion time is acceptable since operation until the next inspection is supported by the operational assessment.

3.1.H.3

If the required actions and associated completion times of Specification 3.1.H.2 are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT SHUTDOWN within 6 hours and COLD SHUTDOWN within the following 30 hours.

The allowed completion times are reasonable, based on operating experience, to reach the desired unit conditions from full power conditions in an orderly manner and without challenging plant systems.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 50.67.
4. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.
5. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
6. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
7. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

Basis

The spray systems in each reactor unit consist of two separate parallel Containment Spray Subsystems, each of 100 percent capacity, and four separate parallel Recirculation Spray Subsystems, each of 50 percent capacity.

Each Containment Spray Subsystem draws water independently from the refueling water storage tank (RWST). The water in the tank is cooled to 45°F or below by circulating the water through one of the two RWST coolers with one of the two recirculating pumps. The water temperature is maintained by two mechanical refrigerating units as required. In each Containment Spray Subsystem, the water flows from the tank through an electric motor driven containment spray pump and is sprayed into the containment atmosphere through two separate sets of spray nozzles. The capacity of the spray systems to depressurize the containment in the event of a Design Basis Accident is a function of the pressure and temperature of the containment atmosphere, the service water temperature, and the temperature in the refueling water storage tank as discussed in the Basis of Specification 3.8.

Each Recirculation Spray Subsystem draws water from the common containment sump. In each subsystem the water flows through a recirculation spray pump and recirculation spray cooler, and is sprayed into the containment atmosphere through a separate set of spray nozzles. Two of the recirculation spray pumps are located inside the containment and two outside the containment in the containment auxiliary structure.

With one Containment Spray Subsystem and two Recirculation Spray Subsystems operating together, the spray systems are capable of cooling and depressurizing the containment to 0.5 psig (Unit 1), 1.0 psig (Unit 2) in less than 60 minutes and to subatmospheric pressure within 4 hours following the Design Basis Accident. The Recirculation Spray Subsystems are capable of maintaining subatmospheric pressure in the containment indefinitely following the Design Basis Accident when used in conjunction with the Containment Vacuum System to remove any long term air inleakage. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig (Unit 1), 1.0 psig (Unit 2) (from 1 hour to 4 hours) and is maintained less than 0.0 psig (after 4 hours).

(3) assuring that environmental conditions will not preclude access to close the valves and 4) that this administrative or manual action will prevent the release of radioactivity outside the containment.

The Reactor Coolant System temperature and pressure being below 350°F and 450 psig, respectively, ensures that no significant amount of flashing steam will be formed and hence that there would be no significant pressure buildup in the containment if there is a loss-of-coolant accident. Therefore, the containment internal pressure is not required to be subatmospheric prior to exceeding 350°F and 450 psig.

The allowable value for the containment air partial pressure is presented in TS Figure 3.8-1 for service water temperatures from 25 to 95°F. The RWST water shall have a maximum temperature of 45°F.

(Unit 1) The horizontal limit line in TS Figure 3.8-1 is based on LOCA peak calculated pressure criteria, and the sloped line is based on LOCA subatmospheric peak pressure criteria.

(Unit 2) The horizontal upper limit line in TS Figure 3.8-1 is based on MSLB peak calculated pressure criteria, and the sloped line from 70°F to 95°F service water temperatures is based on LOCA depressurization criteria.

If the containment air partial pressure rises to a point above the allowable value the reactor shall be brought to the HOT SHUTDOWN condition. If a LOCA occurs at the time the containment air partial pressure is at the maximum allowable value, the maximum containment pressure will be less than design pressure (45 psig), the containment will depressurize to 0.5 psig (Unit 1), 1.0 psig (Unit 2) within 1 hour and less than 0.0 psig within 4 hours. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig (Unit 1), 1.0 psig (Unit 2) for the interval from 1 to 4 hours following the Design Basis Accident.

If the containment air partial pressure cannot be maintained greater than or equal to 9.0 psia (Unit 1), the minimum pressure in Figure 3.8-1 (Unit 2), the reactor shall be brought to the HOT SHUTDOWN condition. The shell and dome plate liner of the containment are capable of withstanding an internal pressure as low as 3 psia, and the bottom mat liner is capable of withstanding an internal pressure as low as 8 psia.

References

UFSAR Section 4.2.2.4	Reactor Coolant Pump
UFSAR Section 5.2	Containment Isolation
UFSAR Section 5.2.1	Design Bases
UFSAR Section 5.2.2	Isolation Design
UFSAR Section 5.3.4	Containment Vacuum System

Containment penetrations that terminate in the Auxiliary Building or Safeguards and provide direct access from containment atmosphere to outside atmosphere must be isolated or capable of being closed by at least one barrier during REFUELING OPERATIONS. The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere must be isolated by at least one barrier during REFUELING OPERATIONS. Isolation may be achieved by an OPERABLE isolation valve, a closed valve, a blind flange, or by an equivalent isolation method. Equivalent isolation methods must be evaluated and may include use of a material that can provide a temporary, atmospheric pressure ventilation barrier.

For the personnel airlock, equipment access hatch, and other penetrations, 'capable of being closed' means the openings are able to be closed; they do not have to be sealed or meet the leakage criteria of TS 4.4. Station procedures exist that ensure in the event of a fuel handling accident, that the open personnel airlock and other penetrations can and will be closed. Closure of the equipment hatch will be accomplished in accordance with station procedures and as allowed by dose rates in containment. The radiological analysis of the fuel handling accident does not take credit for closure of the personnel airlock, equipment access hatch or other penetrations.

The fuel building ventilation exhaust and containment ventilation purge exhaust may be diverted through charcoal filters whenever refueling is in progress. However, there is no requirement for filtration since the Fuel Handling Accident analysis takes no credit for these filters. At least one flow path is required for cooling and mixing the coolant contained in the reactor vessel so as to maintain a uniform boron concentration and to remove residual heat.

The requirements in this specification for the control and relay room emergency ventilation system, control room bottled air system, and the main control room and emergency switchgear room air conditioning system (chillers and air handling units) apply to the shutdown unit. If any of the specified limiting conditions is not met, the requirements appropriately suspend activities that could result in a release of radioactivity that might require isolation of the control room envelope and place irradiated fuel in a safe position without delay and in a controlled manner. The requirements applicable to the operating unit are contained in Specifications 3.19 and 3.23.

If the requirements of Specification 3.19.B.1, 3.19.B.2, or 3.19.B.3 are not met within 48 hours after achieving HOT SHUTDOWN, both units shall be placed in COLD SHUTDOWN within the next 30 hours.

Basis

Following a design basis accident, the containment will be depressurized to 0.5 psig (Unit 1), 1.0 psig (Unit 2) in less than 1 hour and to subatmospheric pressure within 4 hours. The radiological consequences analysis demonstrates acceptable results provided the containment pressure does not exceed 0.5 psig (Unit 1), 1.0 psig (Unit 2) for the interval from 1 to 4 hours following the Design Basis Accident. Beyond 4 hours, containment pressure is assumed to be less than 0.0 psig, terminating leakage from containment. The main control room is maintained at a positive differential pressure using bottled air during the first hour, when the containment leakrate is greatest.

The main control room is contained in the control room pressure boundary or envelope, which is defined in the Technical Specification 3.23 Basis.

The control room pressure boundary is permitted to be opened intermittently under administrative control without declaring the boundary inoperable. The administrative control must provide the capability to re-establish the control room pressure boundary. For normal ingress into and egress from the pressure boundary, the individual entering or exiting the area has control of the door.

2. If two Unit 2 AHUs on different chilled water loops and in different air conditioning zones (2-VS-AC-7 and 2-VS-AC-8 or 2-VS-AC-6 and 2-VS-AC-9) become inoperable, restore operability of the two inoperable AHUs within seven (7) days or bring Unit 2 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

3. If two Unit 2 AHUs in the same air conditioning zone (2-VS-AC-8 and 2-VS-AC-9 or 2-VS-AC-6 and 2-VS-AC-7) become inoperable, restore operability of at least one Unit 2 AHU in each air conditioning zone (2-VS-AC-8 or 2-VS-AC-9 and 2-VS-AC-6 or 2-VS-AC-7) within one (1) hour or bring Unit 2 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

4. If more than two Unit 2 AHUs become inoperable, restore operability of at least one Unit 2 AHU in each air conditioning zone (2-VS-AC-8 or 2-VS-AC-9 and 2-VS-AC-6 or 2-VS-AC-7) within one (1) hour or bring Unit 2 to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the following 30 hours.

c. Both Unit 1 AHUs or both Unit 2 AHUs powered from the respective H buses (1-VS-AC-1 and 1-VS-AC-7 or 2-VS-AC-6 and 2-VS-AC-8) must be OPERABLE whenever both units are above COLD SHUTDOWN.

   1. If one or two AHUs on each unit powered from an H bus is inoperable, restore operability of the inoperable AHU(s) on one unit within one (1) hour or bring both units to HOT SHUTDOWN within the next six (6) hours and be in COLD SHUTDOWN within the next 30 hours.

Basis

When the supply of compressed bottled air is depleted, the Main Control Room (MCR) and Emergency Switchgear Room (ESGR) Emergency Ventilation System is manually started to continue to maintain the control room pressure at the design positive pressure so that leakage is outleakage. One train of the main control room emergency ventilation consists of one fan powered from an independent emergency power source.

The MCR and ESGR Emergency Ventilation System is designed to filter the intake air to the control room pressure envelope during a loss of coolant accident. The control room pressure envelope consists of the control room complex (including the control room, control room annex area, Units 1 and 2 air conditioning equipment rooms, and Units 1 and 2 computer rooms), Units 1 and 2 emergency switchgear rooms, Unit 1 relay room, and Unit 2 relay room (including Mechanical Equipment Room 3 (MER-3)).

High efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers. The charcoal adsorbers are installed to reduce the potential intake of radio-iodine to the control room.

If the system is found to be inoperable, there is no immediate threat to the control room, and reactor operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within the specified time, procedures are initiated to establish conditions for which the filter system is not required.

The MCR and ESGR Air Conditioning System (ACS) cools the control room pressure envelope. From an ACS perspective, the envelope consists of four zones: 1) the Unit 1 side of the control room (including the Unit 1 air conditioning equipment and computer rooms), 2) the Unit 2 side of the control room (including the annex area, the Unit 2 air conditioning equipment and computer rooms), 3) the Unit 1 ESGR and relay room (referred to as the Unit 1 ESGR), and 4) the Unit 2 ESGR and relay room (including MER-3), referred to as the Unit 2 ESGR. The design basis of the MCR and ESGR ACS is to maintain the control room pressure envelope temperature within the equipment design limits for 30 days of continuous occupancy after a design basis accident (DBA). The ACS includes five chillers (1-VS-E-4A, 4B, 4C, 4D, and 4E). Chillers 4A, 4B, and 4C are located in MER-3, in the Unit 2 ESGR. Chillers 4D and 4E are located in MER-5, in the Unit 2 Turbine Building. The chillers supply chilled water to eight air handling units (AHUs), arranged in two independent and redundant chilled water loops. Each chilled water loop provides redundant 100% heat removal capacity per unit. Each loop contains four AHUs (one AHU in each unit's air conditioning zones), the necessary power supplies, the associated valves, piping (from the supply header to return header), instrumentation, and controls. Each AHU has 100% capacity for cooling its zone.

The combination of five chillers and two chilled water loops affords considerable flexibility in meeting the cooling requirements. Two chillers are powered from single emergency buses (1-VS-E-4C from 2H, 1-VS-E-4E from 1H). The remaining three chillers can be powered from either of two emergency buses (1-VS-E-4A from 1J or 2J, 1-VS-E-4B from 1J or 2H, and 1-VS-E-4D from 1H or 2J). The AHUs are powered from the four emergency buses in pairs. For example, the Unit 1 MCR and ESGR AHUs 1-VS-AC-1 and 1-VS-AC-7 are powered from the 1H bus; the redundant Unit 1 MCR and ESGR AHUs 1-VS-AC-2 and 1-VS-AC-6 are powered from the 1J bus. Control of the ACS is by manual action.

The chillers are procedurally aligned by power supply to meet TS 3.23.C.1.b, and the AHU pairs are normally aligned to match the power supplies of the OPERABLE chillers. For example, chiller 1-VS-E-4E and AHUs 1-VS-AC-1 and 1-VS-AC-7 are powered from the 1H emergency bus. However, due to the number of emergency diesel generators (EDGs) and the chiller/AHU piping layout, only one chiller and AHU pair can be powered from each emergency bus at a time. Also, if chilled water is needed in both chilled water loops, two chillers must be operated. Only one chiller can be operated on each chilled water loop at a time, and the 4D and 4E chillers cannot be operated simultaneously. The combinations of OPERABLE chillers/AHUs allowed by procedure ensure that sufficient cooling capacity is available during a DBA with a coincident loss of offsite power (LOOP) and single failure of an EDG, a chiller, or an AHU.

Acceptable operating alignments include one chiller supplying one chilled water loop with four operating AHUs, or two chillers supplying two chilled water loops with two AHUs operating on each loop. In either case, one AHU must be operated in the MCR and ESGR air conditioning zones of each unit. During normal operation, and accident scenarios with a LOOP and single failure of an EDG, one chiller providing chilled water to one chilled water loop with four operating AHUs is sufficient to maintain the MCR and ESGR air temperature within normal limits. In the event of a DBA with all mitigation equipment operating (i.e., higher heat loads due to offsite power available and no single failures), two chillers and two chilled water loops, with one operating AHU in each unit's MCR and ESGR, are necessary to maintain temperatures within normal limits; with one chiller, one chilled water loop, and four operating AHUs, temperatures will be maintained within the equipment design limits.

The MCR and ESGR ACS is considered to be OPERABLE when the individual components necessary to cool the MCR and ESGR envelope are OPERABLE. The operability requirements for the chillers and AHUs are separate but interdependent. The required chillers are considered OPERABLE when required chilled water and service water flowpaths, required power supplies, and controls are OPERABLE. A chiller does not have to be in operation to be considered OPERABLE. An AHU is OPERABLE when the associated chilled water flowpath, fan, motor, dampers, as well as associated ductwork and controls, are OPERABLE.

The Technical Specifications require the operability of the ACS components. Due to the redundancy and diversity of components, the inoperability of one active component does not render the ACS incapable of performing its function. This allows increased flexibility in unit operations under circumstances when more than one ACS component is inoperable. Similarly, the inoperability of two different components, each in a different loop or powered from a different power supply, does not necessarily result in a loss of function for the ACS. However, due to the emergency power design (three EDGs and four emergency buses), realignment of the swing or shared EDG is required in certain instances of inoperable AHUs and is directed by procedure.

The requirements and action statements for the AHUs powered from an H emergency bus eliminate the potential for complex operator actions in certain instances of two inoperable AHUs. The swing EDG can supply either J bus, but not both. With an AHU powered from the H bus inoperable on each unit, a DBA with a LOOP and no single failure would result in one air conditioning zone with no AHU available. In this case, in order to ensure power is available to an AHU in each air conditioning zone, operators would have to procedurally realign the swing diesel and cross-connect emergency buses. By prohibiting the simultaneous inoperability of an H-bus powered AHU on each unit, cross-connect of the emergency buses will not be necessary. Realignment of the swing diesel is still required, and procedures direct the operators to realign the swing EDG (from the MCR) as necessary to ensure that there is an operating AHU in the MCR and ESGR air conditioning zones of each unit.

BASES

4.0.1 Surveillance Requirement (SR) 4.0.1 establishes the requirement that SRs must be met during the REACTOR OPERATION conditions or other specified conditions in the individual Limiting Conditions for Operation (LCO) that apply, unless otherwise specified in the individual SRs. This Specification is to ensure that Surveillances are performed to verify the operability of systems and components, and that variables are within specified limits. Failure to meet a Surveillance within the specified frequency, in accordance with SR 4.0.2, constitutes a failure to meet an LCO. Surveillances may be performed by means of any series of sequential, overlapping, or total steps provided the entire Surveillance is performed within the specified frequency.

Systems and components are assumed to be OPERABLE when the associated SRs have been met. Nothing in this Specification, however, is to be construed as implying that systems or components are OPERABLE when:

a. The systems or components are known to be inoperable, although still meeting the SRs; or

b. The requirements of the Surveillance(s) are known not to be met between required Surveillance performances.

Surveillances do not have to be performed when the unit is in a REACTOR OPERATION condition or other specified condition for which the requirements of the associated LCO are not applicable, unless otherwise specified. The SRs associated with a test exception are only applicable when the test exception is used as an allowable exception to the requirements of a Specification.

Unplanned events may satisfy the requirements (including applicable acceptance criteria) for a given SR. In this case, the unplanned event may be credited as fulfilling the performance of the SR. This allowance includes those SRs whose performance is normally precluded in a given REACTOR OPERATION condition or other specified condition.

Surveillances, including Surveillances invoked by Action Statements, do not have to be performed on inoperable equipment because the Action Statements define the remedial measures that apply. Surveillances have to be met and performed in accordance with SR 4.0.2, prior to returning equipment to OPERABLE status.

Upon completion of maintenance, appropriate post maintenance testing is required to declare equipment OPERABLE. This includes ensuring applicable Surveillances are not failed and their most recent performance is in accordance with SR 4.0.2. Post maintenance testing may not be possible in the current REACTOR OPERATION condition or other specified conditions in the individual LCO due to the necessary unit parameters not having been established. In these situations, the equipment may be considered OPERABLE provided testing has been satisfactorily completed to the extent possible and the equipment is not otherwise believed to be incapable of performing its function. This will allow operation to proceed to a REACTOR OPERATION condition or other specified condition where other necessary post maintenance tests can be completed.

An example of this process is Auxiliary Feedwater (AFW) pump turbine maintenance during refueling that requires testing at steam pressures that cannot be obtained until the unit is at HOT SHUTDOWN conditions. However, if other appropriate testing is satisfactorily completed, the AFW System can be considered OPERABLE. This allows startup and other necessary testing to proceed until the plant reaches the steam pressure required to perform the testing.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

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

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

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

When a Surveillance with a frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering POWER OPERATION after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 4.0.3 allows for the full delay period of up to the specified frequency to perform the Surveillance. However, since there is not a time interval specified, the missed Surveillance should be performed at the first reasonable opportunity.

SR 4.0.3 provides a time limit for, and allowances for the performance of, Surveillances that become applicable as a consequence of REACTOR OPERATION condition changes imposed by Action Statements.

Failure to comply with the specified frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 4.0.3 is a flexibility which is not intended to be used as an operational convenience to extend Surveillance intervals. While up to 24 hours or the limit of the specified frequency is provided to perform the missed Surveillance, it is expected that the missed Surveillance will be performed at the first reasonable opportunity. The determination of the first reasonable opportunity should include consideration of the impact on plant risk (from delaying the Surveillance as well as any plant configuration changes required or shutting the plant down to perform the Surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the Surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementation guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed Surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed Surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed Surveillances will be placed in the licensee's Corrective Action Program.

If a Surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Allowed Outage Time(s) of the Action Statement(s) for the applicable LCO conditions begin immediately upon expiration of the delay period. If a Surveillance is failed within the delay period, then the equipment is inoperable, or the variable is outside the specified limits and the Allowed Outage Time(s) of the Action Statement(s) for the applicable LCO conditions begin immediately upon the failure of the Surveillance.

Completion of the Surveillance within the delay period allowed by this Specification, or within the Allowed Outage Time(s) of the Action Statement(s), restores compliance with SR 4.0.1.

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

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

Exceptions to Specification 4.0.4 allow performance of surveillance requirements associated with a Limiting Condition for Operation after entry into the applicable operational condition.

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

4.13 RCS OPERATIONAL LEAKAGE

Applicability

The following specifications are applicable to RCS operational LEAKAGE whenever T_{avg} (average RCS temperature) exceeds 200°F (200 degrees Fahrenheit).

Objective

To verify that RCS operational LEAKAGE is maintained within the allowable limits.

Specifications

A. Verify RCS operational LEAKAGE is within the limits specified in TS 3.1.C by performance of RCS water inventory balance once every 24 hours.[1, 2]

B. Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG once every 72 hours.[1]

Notes:

1. Not required to be completed until 12 hours after establishment of steady state operation.
2. Not applicable to primary to secondary LEAKAGE.

BASES

SURVEILLANCE REQUIREMENTS (SR)

SR 4.13.A

Verifying RCS LEAKAGE to be within the Limiting Condition for Operation (LCO) limits ensures the integrity of the reactor coolant pressure boundary (RCPB) 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 an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The surveillance is modified by two notes. Note 1 states that this SR is not required to be completed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. 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 RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in the TS 3.1.C Bases.

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 24 hour frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 4.13.B

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.1.H, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 4. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG.

If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG. The surveillance 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 RCS primary to secondary LEAKAGE determination, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

The surveillance frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 4).

REFERENCES

1. UFSAR, Chapter 4, Surry Units 1 and 2.
2. UFSAR, Chapter 14, Surry Units 1 and 2.
3. NEI 97-06, "Steam Generator Program Guidelines."
4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

4.19 STEAM GENERATOR (SG) TUBE INTEGRITY

Applicability

Applies to the verification of SG tube integrity in accordance with the Steam Generator Program.

Objective

To provide assurance of SG tube integrity.

Specifications

A. Verify SG tube integrity in accordance with the Steam Generator Program.

B. Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program prior to T_{avg} exceeding 200°F following a SG tube inspection.

BASES

SURVEILLANCE REQUIREMENTS (SR)

SR 4.19.A

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, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.19.A. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 7). 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 6.4.Q contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 4.19.B

During an 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 6.4.Q 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 and Reference 7 provide 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 T_{avg} exceeding 200°F 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. Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," July 2000.

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

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

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