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U. S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION UNIT 1
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
TECHNICAL SPECIFICATION BASES REVISION
UPDATED THROUGH JANUARY 2009

In accordance with Technical Specification 6.8.4.i., South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, submits revisions to the Technical Specification (TS) Bases in accordance with The Technical Specification Bases Control Program.

This TS Bases update includes changes to the TS Bases since the previous submittal in October 2005. These changes were made under the provisions of 10CFR50.59. The timeliness associated with the submittal of Technical Specification Bases revisions has been entered in the station's corrective action program. Technical changes are annotated by vertical revision bars and the Revision Notice number at the bottom of the page.

If you have any questions or require additional information, please contact Bruce Thompson at (803) 931-5042.

I certify under penalty of perjury that the information contained herein is true and correct.

6-9-2010
Executed on

Thomas D. Gatlin

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TECHNICAL SPECIFICATION BASES REVISIONS
UPDATED THROUGH JANUARY 2009

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INSTRUCTION SHEET
V.C. SUMMER NUCLEAR STATION UNIT 1
TECHNICAL SPECIFICATION

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3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

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. SG 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 3.4.1.1, "Reactor Coolant System, Reactor Coolant Loops and Coolant Circulation, Startup and Power Operation," LCO 3.4.1.2, "Reactor Coolant System, Hot Standby," LCO 3.4.1.3, "Reactor Coolant System, Hot Shutdown," and LCO 3.4.1.4.1, "Reactor Coolant System, Cold Shutdown-Loops Filled."

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.

SG tubing is subject to a variety of degradation mechanisms. SG tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanical 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 6.8.4.k, "Steam Generator Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.4.k, 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 6.8.4.k. 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 (Reference 1).

STEAM GENERATOR TUBE INTEGRITY (Continued)

Applicable Safety Analyses

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding a SGTR is the basis for this Specification. The accident analysis for a SGTR event accounts for a bounding primary-to-secondary leakage rate equal to 1 gpm and the leakage rate associated with a double-ended rupture of a single tube. Contaminated fluid in a ruptured steam generator is only briefly released to the atmosphere as steam via the main steam safety valves. To maximize its contribution to the dose releases, the entire 1 gpm primary-to-secondary leakage is assumed to occur in the intact steam generators where it can be released during the subsequent cooldown of the plant.

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 gpm, or is assumed to increase to 1 gpm 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 greater than or equal to the limits in LCO 3.4.8, "Reactor Coolant System, Specific Activity." 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 (Reference 2), 10 CFR 100 (Reference 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

Limiting Condition for Operation (LCO)

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

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

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 6.8.4.k and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

STEAM GENERATOR TUBE INTEGRITY (Continued)

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 verses displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" 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 (Reference 4) and Draft Regulatory Guide 1.121 (Reference 5).

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 total from all SGs. 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 LCO 3.4.6.2 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.

STEAM GENERATOR TUBE INTEGRITY (Continued)

Applicability

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 in MODE 1, 2, 3, or 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In Modes 5 and 6, primary-to-secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.

Actions

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.

a. The Condition 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 Surveillance Requirement 4.4.5.2. 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, LCO 3.4.5 Action b. applies.

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

   If the evaluation determines that the affected tube(s) have tube integrity, the ACTION statement allows plant 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 entering MODE 4 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.

STEAM GENERATOR TUBE INTEGRITY (Continued)

ACTIONS (Continued)

b. If the required actions and associated completion times of LCO 3.4.5 Action a. are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within the next 30 hours.

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

Surveillance Requirements (SR)

4.4.5.1 During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Reference 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.

A condition monitoring assessment of the SG tubes is performed during SG inspections. 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 method used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.4.5.1. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Reference 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.4.k contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

STEAM GENERATOR TUBE INTEGRITY (Continued)

Surveillance Requirements (Continued)

4.4.5.2 During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.8.4.k are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

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

References

1. NEI 97-06, "Steam Generator Program Guidelines"
2. 10 CFR 50, Appendix A, GDC 19, "Control Room"
3. 10 CFR 100, "Reactor Site Criteria"
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976
6. EPRI TR-107569, "Pressurized Water Reactor Steam Generator Examination Guidelines"

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.

3/4.4.6.2 OPERATIONAL LEAKAGE

Background

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

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

10 CFR 50, Appendix A, GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage. Regulatory Guide 1.45 describes acceptable methods for selecting leakage detection systems.

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

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leak tight. 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).

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 or increases to 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 gallons per day is significantly less than the conditions assumed in the safety analysis.

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

The FSAR analysis for SGTR accounts for a bounding primary-to-secondary leakage rate equal to 1 gpm and the leakage rate associated with a double-ended rupture of a single tube. Leakage through the ruptured tube is the dominate contributor to dose releases. Since contaminated fluid in the ruptured steam generator is only briefly released to the atmosphere as steam via the main steam safety valves, the entire 1 gpm primary-to-secondary leakage is assumed to occur in the intact steam generators where it can be released during the subsequent cooldown of the plant. Overall, this pathway is a small contributor to dose releases.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes the entire 1 gpm primary-to-secondary leakage is through the effected steam generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

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

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

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

OPERATIONAL LEAKAGE (Continued)

b. UNIDENTIFED LEAKAGE

One gallon per minute (gpm) of UNIDENTIFED 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 Reactor Coolant Pressure Boundary, if the leakage is from the pressure boundary.

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

The limit of 150 gallons per day (gpd) per steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Reference 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 gpd." 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 UNIDENTIFED LEAKAGE and is well with 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 LEAKAGE. Violation of this LCO could result in continued degradation of a component or system.

e. CONTROLLED LEAKAGE

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 33 gpm with the modulating valve in the supply line fully open at a nominal RCS reassure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analysis.

f. Reactor Coolant System Pressure Isolation Valve Leakage

10CFR50.2, 10CFR50.55a(c), and GDC 55 of 10CFR50, Appendix A define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB) which separate the high pressure RCS from an attached low pressure system. During their service lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV leakage LCO allows leakage through these valves in amounts that do not compromise safety.

OPERATIONAL LEAKAGE (Continued)

The Reactor Coolant System Pressure Isolation Valve (PIV) Leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of IDENTIFIED LEAKAGE governed by LCO 3.4.6.2, "Reactor Coolant System Operational Leakage." This is true during operation only when the loss of RCS mass through two series valves is determined by water inventory balance (SR 4.4.6.2.1.d). A known component of the identified leakage before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing. Leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other PIV is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low-pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting system are degraded or degrading. Excessive PIV leakage could lead to overpressure of the low-pressure piping or components, potentially resulting in a loss of coolant accident (LOCA) outside of containment.

The PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The NRC, through NUREG-1431, has endorsed this PIV leakage rate limit.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

Leakage from the RCS Pressure Isolation Valves may be identified by surveillance testing performed during plant heatup or cooldown above 2000 psig and may be adjusted to obtain the leakage value at 2235 ± 20 psig using calculation guidance provided by the ASME OM Code.

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.

OPERATIONAL LEAKAGE (Continued)

Actions

a. If any PRESSURE BOUNDARY LEAKAGE exists, 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. Any operational leakage, excluding PRESSURE BOUNDARY LEAKAGE and primary-to-secondary leakage, in excess of the LCO limits must be reduced to within the 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. With PIV leakage in excess of the limit, the high pressure portion of the affected system must be isolated within 4 hours, or be in at least hot standby within the next 6 hours, and cold shutdown within the following 30 hours. This ACTION is necessary to prevent over pressurization of low pressure systems, and the potential for intersystem LOCA.

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.

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

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.

OPERATIONAL LEAKAGE (Continued)

Surveillance Requirements (Continued)

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity and 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 LCO 3.4.6.1, "Reactor Coolant System, Leakage Detection Systems."

Part (d) notes 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 72-hour frequency 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 This Surveillance Requirement verifies RCS Pressure Isolation Valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

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 2. 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 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 (Reference 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 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the reactor building spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 7.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 REACTOR BUILDING COOLING SYSTEM

The OPERABILITY of the reactor building cooling system ensures that 1) the reactor building air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the reactor building spray systems during post-LOCA conditions.

The reactor building cooling system and the reactor building spray system provide post accident cooling of the reactor building atmosphere. These two independent systems incorporate different principles of heat removal, with RB Spray being more effective in the short term in limiting peak pressure and temperature conditions within the RB. Since RB Spray operation maximizes margin to the RB design limits for maximum pressure and temperature, the allowable out of service time requirements for the reactor building cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the reactor building spray system have been maintained consistent with that assigned other inoperable ESF equipment since the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere.

Valves XVB-3107A(B)-SW and XVB-3106A(B)-SW have been designed and interlocked to other equipment controls to mitigate two scenarios in which a pipe water hammer could occur. The first water hammer scenario was postulated to occur when the RBCUs are operating in their normal lineup where they are being cooled by the non-safety Industrial Cooling Water System and the Service Water booster pumps (SWBP) are started during normal swap over to the SW system for or after a Loss Of Offsite Power (LOOP). The second water hammer scenario is postulated to occur when the SW system is aligned to provide cooling for the RBCUs and a LOOP occurs.

To minimize the affects of the first water hammer scenario vacuum relief valves XVV-13143A(B)-SW downstream of valve XVB-3107A(B)-SW will replace, with air, any vacuum void downstream of closed valves XVB-3107A(B)-SW that may be formed due to gravity drain down of water to the SW pond. Upon the start of the SWBPs and opening of valves XVB-3107A(B)-SW, the air in the piping will act as a cushion to minimize any water hammer affects that could occur downstream of XVB-3107A(B)-SW. The opening logic of valves XVB-3107A(B)-SW has a delayed opening after valve 3106A(B)-SW begins to open. The delay allows fluid flow momentum to build to assure that additional void formation in the RBCU piping inside containment will not occur during swap over to the SW system.

To minimize the effects of the second water hammer scenario XVB-3107A(B)-SW, fast closing air operated butterfly valves, close in seven seconds upon de-energizing of the SWBPs. During times that the RBCUs are aligned with the SW system, if a LOOP were to occur, the fast valve closure will trap water in the high points above the valve and prevent void formation due to gravity drain down of the water to the SW pond. Interface logic is provided to equipment controls that tie the start of the respective SWBP to the closed position of the respective valve XVB-3107A(B)-SW. The controls prevent a SWBP start if the respective valve XVB-3107A(B)-SW failed to fully close allowing drain down of the water to the SW Pond.

The accident analysis requires the service water booster pump to be passing 4,000 gpm to both RBCUs within 86.5 seconds. This time encompasses the driving of all necessary service water valves to the correct positions, i.e., fully opened or fully closed. Reference Technical Specification Bases B 3/4.3.1 and B 3/4.3.2 for additional details.

3/4.6.3 PARTICULATE IODINE CLEANUP SYSTEM

The OPERABILITY of the containment filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses.

3/4.6.4 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the reactor building atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the reactor building atmosphere or pressurization of the reactor building and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits required by the safety analysis for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

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

3/4.6.5 COMBUSTIBLE GAS CONTROL

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

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

ULTIMATE HEAT SINK (Continued)

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

3/4.7.6 CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS)

BACKGROUND

The CREFS provides a protected environment from which occupants can control the unit following an uncontrolled release of radioactivity, hazardous chemicals, or smoke.

The CREFS consists of two independent, redundant trains that recirculate and filter the air in the control room envelope (CRE) and a CRE boundary that limits the inleakage of unfiltered air. Each CREFS train consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and a fan. Ductwork, valves or dampers, doors, barriers, and instrumentation also form part of the system. A second bank of HEPA filters follows the adsorber section to collect carbon fines and provides backup in case of failure of the main HEPA filter bank.

The CRE is the area within the confines of the CRE boundary that contains the spaces that control room occupants inhabit to control the unit during normal and accident conditions. This area encompasses the control room, and may encompass other non-critical areas to which frequent personnel access or continuous occupancy is not necessary in the event of an accident. The CRE is protected during normal operation, natural events, and accident conditions. The CRE boundary is the combination of walls, floor, roof, ducting, doors, penetrations and equipment that physically form the CRE. The OPERABILITY of the CRE boundary must be maintained to ensure that the inleakage of unfiltered air into the CRE will not exceed the inleakage assumed in the licensing basis analysis of design basis accident (DBA) consequences to CRE occupants. The CRE and its boundary are defined in the Control Room Envelope Habitability Program.

The CREFS is an emergency system, parts of which may also operate during normal unit operations in the standby mode of operation. Upon receipt of the actuating signal(s), normal air supply to the CRE is isolated, and the stream of ventilation air is recirculated through the system filter trains. The prefilters remove any large particles in the air to prevent excessive loading of the HEPA filters and charcoal adsorbers.

Actuation of the CREFS to the emergency mode of operation aligns the system for recirculation air within the CRE through the redundant trains of HEPA and the charcoal filters. The emergency mode of operation maintains pressurization and filtered ventilation of the air supply to the CRE.

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) (Continued)

BACKGROUND (Continued)

Outside air is filtered and added to the air being recirculated from the CRE. Pressurization of the CRE minimizes infiltration of unfiltered air through the CRE boundary from all the surrounding areas adjacent to the CRE boundary.

The air entering the CRE is continuously monitored by a radiation detector. Detector output above the setpoint will cause actuation of the emergency mode of operation.

A single CREFS train operating at a nominal flow rate of < 21,270 scfm will pressurize the CRE to about 0.125 inches water gauge relative to external areas adjacent to the CRE boundary. The CREFS operation in maintaining the CRE habitable is discussed in the FSAR, Section 9.4 (Ref. 1).

Redundant supply and recirculation trains provide the required filtration should an excessive pressure drop develop across the other filter train. Normally open isolation dampers are arranged in series pairs so that the failure of one damper to shut will not result in a breach of isolation. The CREFS is designed in accordance with Seismic Category I requirements.

The CREFS is designed to maintain a habitable environment in the CRE for 30 days of continuous occupancy after a Design Basis Accident (DBA) without exceeding a 5 rem whole body dose or its equivalent to any part of the body.

APPLICABLE SAFETY ANALYSES

The CREFS components are arranged in redundant, safety related ventilation trains. The location of components and ducting within the CRE ensures an adequate supply of filtered air to all areas requiring access. The CREFS provides airborne radiological protection for the CRE occupants as demonstrated by the CRE occupant dose analyses for the most limiting design basis loss of coolant accident, fission product release presented in the FSAR, Chapter 15 (Ref. 2).

The CREFS provides protection from smoke and hazardous chemicals to the CRE occupants. The analysis of hazardous chemical releases demonstrates that the toxicity limits are not exceeded in the CRE following a hazardous chemical release (Ref. 3). The evaluation of a smoke challenge demonstrates that it will not result in the inability of the CRE occupants to control the reactor either from the control room or from the remote shutdown panels (Ref. 4).

The worst case single active failure of a component of the CREFS, assuming a loss of offsite power, does not impair the ability of the system to perform its design function.

The CREFS satisfies Criterion 3 of 10 CFR 50.36(d)(2)(ii).

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) (Continued)

LCO

Two independent and redundant CREFS trains are required to be OPERABLE to ensure that at least one is available if a single active failure disables the other train. Total system failure, such as from a loss of both ventilation trains or from an inoperable CRE boundary, could result in exceeding a dose of 5 rem whole body or its equivalent to any part of the body to the CRE occupants in the event of a large radioactive release.

Each CREFS train is considered OPERABLE when the individual components necessary to limit CRE occupant exposure are OPERABLE. A CREFS train is OPERABLE when the associated:

a. Fan is OPERABLE,

b. HEPA filters and charcoal adsorbers are not excessively restricting flow, and are capable of performing their filtration functions, and

c. Ductwork, valves, and dampers are OPERABLE, and air circulation can be maintained.

In order for the CREFS trains to be considered OPERABLE, the CRE boundary must be maintained such that the CRE occupant dose from a large radioactive release does not exceed the calculated dose in the licensing basis consequence analyses for DBAs, and that CRE occupants are protected from hazardous chemicals and smoke.

The LCO is modified by a Note allowing the CRE boundary to be opened intermittently under administrative controls. This Note only applies to openings in the CRE boundary that can be rapidly restored to the design condition, such as doors, hatches, floor plugs, and access panels. For entry and exit through doors, the administrative control of the opening is performed by the person(s) entering or exiting the area. For other openings, these controls should be proceduralized and consist of stationing a dedicated individual at the opening who is in continuous communication with the operators in the CRE. This individual will have a method to rapidly close the opening and to restore the CRE boundary to a condition equivalent to the design condition when a need for CRE isolation is indicated.

APPLICABILITY

In MODES 1, 2, 3, and 4, and during movement of irradiated fuel assemblies, the CREFS must be OPERABLE to ensure that the CRE will remain habitable during and following a DBA.

In MODES 5 and 6, the CREFS is required to cope with the release from the rupture of an outside waste gas tank.

During movement of irradiated fuel assemblies, the CREFS must be OPERABLE to cope with the release from a fuel handling accident.

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) (Continued)

ACTIONS

3.7.6.a.1

When one CREFS train is inoperable, for reasons other than an inoperable CRE boundary, action must be taken to restore OPERABLE status within 7 days. In this LCO, the remaining OPERABLE CREFS train is adequate to perform the CRE occupant protection function. However, the overall reliability is reduced because a failure in the OPERABLE CREFS train could result in loss of CREFS function. The 7 day Allowed Outage Time (AOT) is based on the low probability of a DBA occurring during this time period, and ability of the remaining train to provide the required capability.

In MODE 1, 2, 3, or 4, if the inoperable CREFS train or the CRE cannot be restored to OPERABLE status within the required AOT, the unit must be placed in a MODE that minimizes accident risk. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 5 within 36 hours. The AOTs 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.

3.7.6.a.2

If the unfiltered inleakage of potentially contaminated air past the CRE boundary and into the CRE can result in CRE occupant radiological dose greater than the calculated dose of the licensing basis analyses of DBA consequences (allowed to be up to 5 rem whole body or its equivalent to any part of the body), or inadequate protection of CRE occupants from hazardous chemicals or smoke, the CRE boundary is inoperable. Actions must be taken to restore an OPERABLE CRE boundary within 90 days.

During the period that the CRE boundary is considered inoperable, action must be initiated to implement mitigating actions to lessen the effect on CRE occupants from the potential hazards of a radiological or chemical event or a challenge from smoke. Actions must be taken within 24 hours to verify that in the event of a DBA, the mitigating actions will ensure that CRE occupant radiological exposures will not exceed the calculated dose of the licensing basis analyses of DBA consequences, and that CRE occupants are protected from hazardous chemicals and smoke. These mitigating actions (i.e., actions that are taken to offset the consequences of the inoperable CRE boundary) should be preplanned for implementation upon entry into the LCO, regardless of whether entry is intentional or unintentional. The 24 hour AOT is reasonable based on the low probability of a DBA occurring during this time period, and the use of mitigating actions. The 90 day AOT is reasonable based on the determination that the mitigating actions will ensure protection of CRE occupants within analyzed limits while limiting the probability that CRE occupants will have to implement protective measures that may adversely affect their ability to control the reactor and maintain it in a safe shutdown condition in the event of a DBA. In addition, the 90 day AOT is a reasonable time to diagnose, plan and possibly repair, and test most problems with the CRE boundary.

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) (Continued)

ACTIONS (Continued)

3.7.6.a.3

If both CREFS trains are inoperable in MODE 1, 2, 3, or 4 for reasons other than an inoperable CRE boundary (i.e., LCO ACTION 3.7.6.a.2), the CREFS may not be capable of performing the intended function and the unit is in a condition outside the accident analyses. Therefore, LCO 3.0.3 must be entered immediately.

3.7.6.b.1

In MODE 5 or 6, or during movement of irradiated fuel assemblies, if the inoperable CREFS train cannot be restored to OPERABLE status within the required AOT, action must be taken to immediately place the OPERABLE CREFS train in the emergency mode. This action ensures that the remaining train is OPERABLE, that no failures preventing automatic actuation will occur, and that any active failure would be readily detected.

An alternative to Required Action 3.7.6.b.1 is to immediately suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

3.7.6.b.2

In MODE 5 or 6, or during movement of irradiated fuel assemblies, with two CREFS trains inoperable or with one or more CREFS trains inoperable due to an inoperable CRE boundary, action must be taken immediately to suspend activities that could result in a release of radioactivity that might require isolation of the CRE. This places the unit in a condition that minimizes the accident risk. This does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 4.7.6.a

The control room temperature should be checked periodically to ensure that the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by the CREFS and that the control room will remain habitable for operations personnel during and following all credible accident conditions.

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

SR 4.7.6.b

Standby systems should be checked periodically to ensure that they function properly. As the environment and normal operating conditions on this system are not too severe, testing each train once every month provides an adequate check of this system. The VCSNS CREFS does not have heaters and each train need only be operated for a minimum of 15 minutes to demonstrate the function of the system. The 31 day Frequency is based on the reliability of the equipment and the two train redundancy.

SR 4.7.6.c

This SR verifies that the required CREFS testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing the performance of the HEPA filter, charcoal adsorber efficiency, minimum flow rate, and the physical properties of the activated charcoal. Specific test Frequencies and additional information are discussed in detail in the VFTP implementing procedures.

SR 4.7.6.d

This SR verifies that each CREFS train starts and operates on an actual or simulated actuation signal. The Frequency of 18 months is based on industry operating experience and is consistent with the typical refueling cycle.

SR 4.7.6.e

This SR verifies the OPERABILITY of the CRE boundary by testing for unfiltered air inleakage past the CRE boundary and into the CRE. The details of the testing are specified in the Control Room Envelope Habitability Program.

The CRE is considered habitable when the radiological dose to CRE occupants calculated in the licensing basis analyses of DBA consequences is no more than 5 rem whole body or its equivalent to any part of the body and the CRE occupants are protected from hazardous chemicals and smoke. This SR verifies that the unfiltered air inleakage into the CRE is no greater than the flow rate assumed in the licensing basis analyses of DBA consequences. When unfiltered air inleakage is greater than the assumed flow rate, ACTION 3.7.6.a.2 must be entered. Action 3.7.6.a.2 allows time to restore the CRE boundary to OPERABLE status provided mitigating actions can ensure that the CRE remains within the licensing basis habitability limits for the occupants following an accident. Compensatory measures are discussed in Regulatory Guide 1.196, Section C.2.7.3, (Ref. 5)

CONTROL ROOM EMERGENCY FILTRATION SYSTEM (CREFS) (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

which endorses, with exceptions, NEI 99-03, Section 8.4 and Appendix F (Ref. 6). These compensatory measures may also be used as mitigating actions as required by Action 3.7.6.a.2. Temporary analytical methods may also be used as compensatory measures to restore OPERABILITY (Ref. 7). Options for restoring the CRE boundary to OPERABLE status include changing the licensing basis DBA consequence analysis, repairing the CRE boundary, or a combination of these actions. Depending upon the nature of the problem and the corrective action, a full scope inleakage test may not be necessary to establish that the CRE boundary has been restored to OPERABLE status.

REFERENCES

1. FSAR, Section 9.4.
2. FSAR, Chapter 15.
3. FSAR, Section 6.4.
4. FSAR, Section 9.5.
5. Regulatory Guide 1.196.
6. NEI 99-03, "Control Room Habitability Assessment," March 2003.
7. Letter from Eric J. Leeds (NRC) to James W. Davis (NEI) dated January 30, 2004, "NEI Draft White Paper, Use of Generic Letter 91-18 Process and Alternative Source Terms in the Context of Control Room Habitability." (ADAMS Accession No. ML040300694).

3/4.7.7 SNUBBERS

All snubbers on systems required for safe shutdown/accident mitigation shall be OPERABLE. This includes safety and non-safety related snubbers on systems used to protect the code boundary and to ensure the structural integrity of these systems under dynamic loads.