3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 600 gallons per day total primary to secondary LEAKAGE through all steam generators (SGs);

e. 150 gallons per day primary to secondary LEAKAGE through any one SG; and

f. Four SGs shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Steam Generator Tube Surveillance Program not met.	B.1 Determine steam generator tube integrity is acceptable for continued operation	4 hours

(continued)

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.12 Steam Generator (SG) Tube Surveillance Program

Each steam generator (SG) shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the provisions for inservice inspection of ASME Code Class 1, 2, and 3 components which shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a. See Specification 5.7.2.11 for applicable inspection Frequencies.

a. SG Sample Selection and Inspection - Each SG shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of SG tubes specified in Table 5.7.2.12-1.

b. SG Tube Sample Selection and Inspection - The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.7.2.12-1. The inservice inspection of SG tubes shall be performed at the frequencies specified in Specification 5.7.2.12.f and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.7.2.12.g. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all SGs; the tubes selected for these inspections shall be selected on a random basis except:

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from the critical areas;

2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each SG shall include:

a) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

b) Tubes in those areas where experience has indicated potential problems,

c) A tube inspection (pursuant to Specification 5.7.2.12.g) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection, and

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

d) In addition to the samples required in 5.7.2.12.b.2.a) through c), all tubes which have had the F* criterion applied will be inspected in the tubesheet region. These F* tubes may be excluded from 5.7.2.12.b.2.a, provided the only previous wall penetration of greater than 20% was located below the F* distance of 1.40 inches (which includes NDE uncertainty) extending from either the bottom of the steam generator tube roll transition or the top of the tubesheet, whichever is lower in elevation.

e) Indications left in service at the flow distribution baffles and tube support plate elevations as a result of the application of the tube support plate voltage repair criteria shall be inspected by bobbin probe during all future refueling outages.

c. Examination Results - The results of each sample inspection shall be classified into one of the following three categories:

C-1 Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.

C-2 One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.

C-3 More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

-----------------------NOTE------------------------
In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.
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d. Supplemental Sampling Requirements - The tubes selected as the second and third samples (if required by Table 5.7.2.12-1) may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2. The inspections include those portions of the tubes where imperfections were previously found.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

e. Supplemental Inspection Requirements - Implementation of the steam generator tube to tube support plate repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections (including the flow distribution baffles) down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

f. Inspection frequency - The above required inservice inspections of the SG tubes shall be performed at the following frequencies:

1. The first inservice inspection shall be performed after 6 effective full power months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.7.2.12-1 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.7.2.12.f.1; the interval may then be extended to a maximum of once per 40 months; and

3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.7.2.12-1 during the shutdown subsequent to any of the following conditions:

   a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13, or

   b) A seismic occurrence greater than the Operating Basis Earthquake, or

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

d) A main steam line or feedwater line break.

g. Acceptance Criteria

1. Terms as used in this specification will be defined as follows:

a) Degradation - A service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;

b) Degraded Tube - A tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

c) % Degradation - The percentage of the tube wall thickness affected or removed by degradation;

d) Defect - An imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

e) Imperfection - An exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

f) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness. This definition does not apply to the portion of the tube in the tubesheet below the F* distance provided the tube is not degraded within the F* distance for F* tubes.

For tubes to which the F* criteria is applied, a minimum of 1.5 inches of the tube into the tubesheet from the top of the tubesheet or from the bottom of the roll transition, whichever is lower in elevation, shall be inspected using rotating pancake coil eddy current technique or an inspection method shown to give equivalent or better information on the orientation and length of cracking. A minimum of 1.40 inches (which includes NDE uncertainty) of continuous, sound expanded tube must be established, extending from either the bottom of the roll transition or the top of the tubesheet, whichever is lower in elevation, to the uppermost extent of the indication.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

This definition does not apply to flow distribution baffles and tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 5.7.2.12.g.1.1 for repair limit applicable to these intersections.

g) Preservice Inspection - An inspection of the full length of each tube in each SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial MODE 1 operation using the equipment and techniques expected to be used during subsequent inservice inspections.

h) Tube Inspection - An inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

i) Unserviceable - The condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operational Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break accident as specified in Specification 5.7.2.12.f.

j) F* Distance is the distance into the tubesheet from the bottom of the steam generator tube roll transition or the top of the tubesheet, whichever is lower in elevation (further into the tubesheet), that has been conservatively chosen to be 1.40 inches (which includes NDE uncertainty).

k) F* Tube is the tube with degradation equal to or greater than 40%, below the F* distance and not degraded (i.e., no indications of degradation) within the F* distance.

l) The Tube Support Plate Repair Limit - The Tube Support Plate Repair Limit is used for the disposition of Alloy 600 steam generator tubes for continued service that are experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates and flow distribution baffle (FDB). At tube support plate intersections (and FDB), the repair limit is based on maintaining steam generator tube serviceability as described below:

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

1. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the flow distribution baffles and tube support plates with bobbin voltages less than or equal to the lower voltage repair limit of 1.0 volt will be allowed to remain in service.

2. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the flow distribution baffles and tube support plates with the bobbin voltage greater than the lower voltage repair limit of 1.0 volt, will be repaired, except as noted in Specification 5.7.2.12.g.1.1.3 below.

3. Steam generator tubes with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the flow distribution baffles and tube support plates with a bobbin voltage greater than the lower voltage repair limit of 1.0 volt but less than or equal to the upper voltage limit*, may remain inservice if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit* will be plugged or repaired.

4. Certain intersections as identified in Attachment 2 of WAT-D-10709 ("Tennessee Valley Authority, Watts Bar Nuclear Power Plant Unit 1, Application for Implementation of Voltage Based Repair Criteria, Westinghouse Steam Generator Tubes Affected by ODSCC at TSPs," J. W. Irons, Revision 0, 1/12/00) will be excluded from application of the voltage-based repair criteria as it is determined that these intersection may collapse or deform following a postulated LOCA + SSE event.

5. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.7.2.12.g.1.1.1, 5.7.2.12.g.1.1.2, and 5.7.2.12.g.1.1.3.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left[\frac{CL - \Delta t}{CL}\right]}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left[\frac{CL - \Delta t}{CL}\right]$$

where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

CL = cycle length (the time between two scheduled steam generator inspections)

V_{SL} = structural limit voltage

Gr = average growth rate per cycle length

NDE = 95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e. a value of 20-percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach used in specifications 5.7.2.12.g.1.1.1, 5.7.2.12.g.1.1.2, and 5.7.2.12.g.1.1.3.

* The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented. V_{URL} will differ at the tube support plates and flow distribution baffle.

2. The SG shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.7.2.12-1.

h. Reports - The content and frequency of written reports shall be in accordance with Specification 5.9.9.

5.9 Reporting Requirements (continued)

5.9.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to the initial and each reload cycle, or prior to any remaining portion of a cycle, and shall be documented in the COLR for the following:

LCO 3.1.4	Moderator Temperature Coefficient
LCO 3.1.6	Shutdown Bank Insertion Limit
LCO 3.1.7	Control Bank Insertion Limits
LCO 3.2.1	Heat Flux Hot Channel Factor
LCO 3.2.2	Nuclear Enthalpy Rise Hot Channel Factor
LCO 3.2.3	Axial Flux Difference
LCO 3.9.1	Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. When an initial assumed power level of 102 percent of rated thermal power is specified in a previously approved method, 100.6 percent of rated thermal power may be used only when feedwater flow measurement (used as input for reactor thermal power measurement) is provided by the leading edge flowmeter (LEFM) as described in document number 6 listed below. When feedwater flow measurements from the LEFM are unavailable, the originally approved initial power level of 102 percent of rated thermal power (3411 MWt) shall be used.

The approved analytical methods are specifically those described in the following documents:

1. WCAP-9272-P-A, WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary). (Methodology for Specifications 3.1.4 - Moderator Temperature Coefficient, 3.1.6 - Shutdown Bank Insertion Limit, 3.1.7 - Control Bank Insertion Limits, 3.2.1 - Heat Flux Hot Channel Factor, 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor, 3.2.3 - Axial Flux Difference, and 3.9.1 - Boron Concentration.

2a. WCAP-12945-P-A, Volume I (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best-Estimate Loss of Coolant Analysis," March 1998 (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

b. WCAP-10054-P-A, "Small Break ECCS Evaluation Model Using NOTRUMP Code," August 1985. Addendum 2, Rev. 1: "Addendum to the Westinghouse Small Break ECCS Evaluation Model using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997. (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor, and 3.2.2 - Nuclear Enthalpy Rise Hot Channel Factor).

(continued)

5.9 Reporting Requirements

5.9.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

3. WCAP-10216-P-A, Revision 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F(Q) SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary). (Methodology for Specifications 3.2.1 - Heat Flux Hot Channel Factor (W(Z) Surveillance Requirements For F(Q) Methodology) and 3.2.3 - Axial Flux Difference (Relaxed Axial Offset Control).)

4. WCAP-12610-P-A, "VANTAGE + FUEL ASSEMBLY REFERENCE CORE REPORT," April 1995. (W Proprietary). (Methodology for Specification 3.2.1 - Heat Flux Hot Channel Factor).

5. WCAP-15088-P, Rev. 1, "Safety Evaluation Supporting A More Negative EOL Moderator Temperature Coefficient Technical Specification for the Watts Bar Nuclear Plant," July 1999, (W Proprietary), as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. 20 (Methodology for Specification 3.1.4 - Moderator Temperature Coefficient.).

6. Caldon, Inc. Engineering Report-80P, "Improving Thermal Accuracy and Plant Safety While Increasing Operating Power Level Using the LEFM✓™ System," Revision 0, March 1997; and Caldon, Inc. Engineering Report-160P, "Supplement to Topical Report ER-80P: Basis for a Power Uprate With the LEFM✓™," Revision 0, May 2000; as approved by the NRC staff's Safety Evaluation accompanying the issuance of Amendment No. 31.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

(continued)

5.9 Reporting Requirements (continued)

5.9.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

LCO 3.4.3 RCS Pressure and Temperature (P/T) Limits
LCO 3.4.12 Cold Overpressure Mitigation System (COMS)

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC. The acceptability of the analytical methods is documented in NRC letter, "WATTS BAR UNIT 1 - ACCEPTANCE FOR REFERENCING OF PRESSURE TEMPERATURE LIMITS METHODOLOGY AND PRESSURE TEMPERATURE LIMITS REPORT (TAC M89048)", September 22, 1995 and "EXEMPTION FROM THE REQUIREMENTS OF 10 CFR Part 50.60, ACCEPTANCE CRITERIA FOR FRACTURE PREVENTION MEASURES FOR LIGHTWATER NUCLEAR POWER REACTORS FOR NORMAL OPERATION - WATTS BAR NUCLEAR PLANT (TAC NO. M99063)." September 29, 1997. Specifically, the analytical methods are described in the following references:

1. Letter, W. J. Museler to NRC, regarding request for exemption from 10 CFR 50.60, March 10, 1994.

2. Letter, D. E. Nunn to NRC, regarding heatup and cooldown curves for normal operation (submitting WCAP-14176 and WCAP-14040, Rev. 1), December 23, 1994.

3. Letter, R. R. Baron to NRC, responding to NRC July 11, 1995, request for additional information, July 31, 1995.

4. Letter, R. R. Baron to NRC providing more information regarding cold overpressure mitigating system setpoints, September 8, 1995.

5. Letter, J. A. Scalice to NRC, regarding request for exemption from 10 CFR 50.60, concerning use of Code Case N-514 to determine LTOP setpoints, dated June 20, 1997.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluency period and for any revision or supplement thereto.

(continued)

5.9 Reporting Requirements (continued)

5.9.7 EDG Failures Report

If an individual emergency diesel generator (EDG) experiences four or more valid failures in the last 25 demands, these failures and any nonvalid failures experienced by that EDG in that time period shall be reported within 30 days. Reports on EDG failures shall include the information recommended in Regulatory Guide 1.9, Revision 3, Regulatory Position C.4, or existing Regulatory Guide 1.108 reporting requirement.

5.9.8 PAMS Report

When a Report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.9.9 SG Tube Inspection Report

Following each inservice inspection of steam generator (SG) tubes, in accordance with the SG Tube Surveillance Program, the number of tubes plugged and tubes sleeved in each SG shall be reported to the NRC within 15 days.

The complete results of the SG tube inservice inspection shall be submitted to the NRC within 12 months following the completion of the inspection. The report shall include:

1. Number and extent of tubes inspected,
2. Location and percent of wall-thickness penetration for each indication of an imperfection, and
3. Identification of tubes plugged.

Results of SG tube inspections that fall into Category C-3 shall be reported to the NRC in accordance with 10 CFR 50.72. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

The results of the inspection of F* tubes shall be reported to the Commission in accordance with 10 CFR 50.4, prior to the restart of the unit. This report shall include:

1. Identification of F* tubes.
2. Uppermost elevation of the degradation and extent of the degradation.

NRC approval of this report is not required prior to restart.

For implementation of the voltage based repair criteria to tube support plate (and flow distribution baffle) intersections, notify the NRC prior to returning the steam generators to service should any of the following conditions arise:

(continued)

5.9 Reporting Requirements

5.9.9 SG Tube Inspection Report (continued)

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leakage limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

2. If circumferential crack-like indications are detected at the tube support plate intersection and flow distribution baffles.

3. If indications are identified that extend beyond the confines of the tube support plate and flow distribution baffles.

4. If indications are identified at the tube support plate and flow distribution baffle elevations that are attributable to primary water stress corrosion cracking.

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

BASES

BACKGROUND (continued)

The voltage based repair limit of Specification 5.7.2.12.g.1.1 implements the guidance of Generic Letter (GL) 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections and flow distribution baffles. The voltage-based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate and flow distribution baffles. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of Specification 5.7.2.12.g.1.1 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this Specification).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation at the 95-percent prediction interval curve reduced to account for the lower 95/95-percent tolerance bound for tubing material properties at 650°F (i.e. the 95-percent lower tolerance limit (LTL) curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit, V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{GR} - V_{NDE}$$

where V_{GR} represents the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

The mid-cycle equation in Specification 5.7.2.12.g.1.1 should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates and flow distribution baffles.

(continued)

BASES

BACKGROUND (continued)

Specification 5.9.9 implements several reporting requirements recommended by GL 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL Section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL Section 6.b(c) criteria.

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment pocket 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 All Steam Generators (SGs)

The LEAKAGE limits incorporated into LCOs 3.4.13.d and 3.4.13.e are more restrictive than the standard operating LEAKAGE limits and are intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shut down in a timely manner.

The 600 gallons per day total primary to secondary LEAKAGE through all SGs ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 limits in the event of either a steam generator tube rupture or steam line break. The limit is consistent with, or conservative to, the assumptions used in the analysis of these accidents.

(continued)

BASES

LCO (continued)

e. Primary to Secondary LEAKAGE through Any One SG

The 150 gallons per day primary to secondary LEAKAGE through any one SG ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions. The limit is consistent with, or conservative to, the assumptions used in the analysis of these accidents.

f. Steam Generator OPERABILITY

Four SGs are also required to be OPERABLE. This requirement is met by satisfying the augmented inservice inspection requirements of the Steam Generator Tube Surveillance Program (Specification 5.7.2.12).

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

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

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary 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.

(continued)