

# **WOLF CREEK**

NUCLEAR OPERATING CORPORATION

Terry J. Garrett  
Vice President Engineering

November 3, 2005  
ET 05-0021

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

**Subject: Docket No. 50-482: Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process**

Gentlemen:

Pursuant to 10 CFR 50.90, Wolf Creek Nuclear Operating Corporation (WCNOC) hereby requests an amendment to Facility Operating License No. NPF-42 for the Wolf Creek Generating Station (WCGS).

The proposed amendment would revise the Technical Specification (TS) requirements related to steam generator tube integrity. The change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity." The availability of this TS improvement was announced in the Federal Register on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process (CLIIP).

Attachments I through IV provide the evaluation, markup of TS pages, retyped TS pages, and proposed TS Bases changes respectively, in support of this amendment request. Attachment V contains a list of commitments. Attachment IV is provided for information only. Final TS Bases pages will be implemented pursuant to TS 5.5.14, "Technical Specifications (TS) Bases Control Program." A revision to the steam generator tube failure Emergency Action Level that reflects the approved TS 3.4.13, "RCS Operational LEAKAGE," limits will be implemented at the time the amendment is implemented.

It has been determined that this amendment application does not involve a significant hazard consideration as determined per 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

A001

This amendment application was reviewed by the Plant Safety Review Committee. In accordance with 10 CFR 50.91; a copy of this amendment application, with attachments, is being provided to the designated Kansas State official.

WCNOC requests approval of the proposed amendment by June 2006. It is anticipated that the license amendment, as approved, will be effective upon issuance, to be implemented prior to restart from Refueling Outage 15, scheduled to start in October 2006. Please contact me at (620) 364-4084 or Mr. Kevin Moles at (620) 364-4126 for any questions you may have regarding this application.

Very truly yours,



Terry J. Garrett

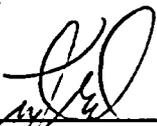
TJG/rlg

Attachments: I - Evaluation  
II - Markup of Technical Specification pages  
III - Retyped Technical Specification pages  
IV - Proposed TS Bases Changes (for information only)  
V - List of Commitments

cc: T. A. Conley (KDHE), w/a  
J. N. Donohew (NRC), w/a  
W. B Jones (NRC), w/a  
B. S. Mallett (NRC), w/a  
Senior Resident Inspector (NRC), w/a

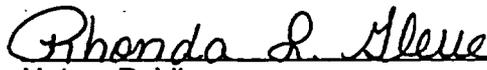
STATE OF KANSAS    )  
                                  ) SS  
COUNTY OF COFFEY )

Terry J. Garrett, of lawful age, being first duly sworn upon oath says that he is Vice President Engineering of Wolf Creek Nuclear Operating Corporation; that he has read the foregoing document and knows the contents thereof; that he has executed the same for and on behalf of said Corporation with full power and authority to do so; and that the facts therein stated are true and correct to the best of his knowledge, information and belief.

By   
Terry J. Garrett  
Vice President Engineering

SUBSCRIBED and sworn to before me this 3 day of Nov, 2005.

 NOTARY PUBLIC STATE OF KANSAS	<b>RHONDA L. GLEUE</b> My Appt. Exp. <u>5-11-2006</u>
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Notary Public

Expiration Date May 11, 2006

## EVALUATION

### 1.0 DESCRIPTION

The proposed license amendment revises the requirements in Technical Specifications (TS) related to steam generator tube integrity. The changes are consistent with NRC approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4. The availability of this technical specification improvement was announced in the Federal Register (Reference 1) on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process.

### 2.0 PROPOSED CHANGE

Consistent with the NRC-approved Revision 4 of TSTF-449, the proposed TS changes include:

- Revised TS definition of LEAKAGE
- Revised TS 3.4.13, "RCS Operational LEAKAGE"
- New TS 3.4.17, "Steam Generator (SG) Tube Integrity"
- Revised TS 5.5.9, "Steam Generator (SG) Program"
- Revised TS 5.6.9, "Steam Generator Tube Inspection Report"

Proposed revisions to the TS Bases are also included in this application. As discussed in the NRC's model safety evaluation, adoption of the revised TS Bases associated with TSTF-449, Revision 4, is an integral part of implementing this TS improvement. The changes to the affected TS Bases pages will be incorporated in accordance with TS 5.5.14, "Technical Specifications (TS) Bases Control Program."

### 3.0 BACKGROUND

The background for this application is adequately addressed by the NRC Notice of Availability published on May 6, 2005, (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

Wolf Creek Nuclear Operating Corporation (WCNOC) letter ET 05-0001 (Reference 2) dated April 18, 2005, submitted an amendment application requesting a one time change for Refueling Outage 14 and subsequent operating cycle to specify that the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded from steam generator tube inspections. Amendment No. 162 (Reference 3) dated April 28, 2005 approved the requested one time change. This amendment application incorporates in TS 5.5.9 the one time change approved by Amendment No. 162 as an alternate repair criteria. WCNOC intends to submit a permanent change to the steam generator tube inspection requirements subsequent to this application.

### 4.0 TECHNICAL ANALYSIS

WCNOC has reviewed the safety evaluation (SE) published on March 2, 2005 (70 FR 10298) as part of the CLIP Notice for Comment. This included the NRC staff's SE, the supporting information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. WCNOC has concluded that the justifications presented in the TSTF proposal and

the SE prepared by the NRC staff are applicable to the Wolf Creek Generating Station (WCGS) and justify this amendment for the incorporation of the changes to the WCGS TSs.

**5.0 REGULATORY ANALYSIS**

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

**5.1 No Significant Hazards Consideration Determination**

WCNOC has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIP. WCNOC has concluded that the proposed determination presented in the notice is applicable to WCGS and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

**5.2 Applicable Regulatory Requirements/Criteria**

The applicable regulatory requirements and guidance associated with this application are adequately addressed by the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

**5.3 Verification and Commitments**

The following information is provided to support the NRC staff's review of this amendment application:

Plant Name, Unit No.	Wolf Creek Generating Station, Unit No. 1
Steam Generator (SG) Model(s)	Westinghouse Model F
Effective Full Power Years (EFPY) of service for currently installed SGs	16.85 EFPY as of September 20, 2005
Tubing Material	Inconel Alloy 600 Thermally Treated
Number of tubes per SG	5626
Number and percentage of tubes plugged in each SG	SG "A" – 29* tubes plugged (.52 percent) SG "B" – 35 tubes plugged (.62 percent) SG "C" – 20 tubes plugged (.36 percent) SG "D" – 97 tubes plugged (1.72 percent) *An additional 3 plugs are installed in SG "A" cold leg only, due to tubesheet drilling mistakes during manufacturing. No tubes are installed in those locations.

Number of tubes repaired in each SG	None
Degradation mechanism(s) identified	Anti- Vibration Bar Wear, Flow Distribution Baffle Plate Wear; Tube Support Plate Wear; Tube Free-span Wear; Wear due to presence of Foreign Objects
Current primary-to-secondary leakage limits:	per SG: 500 gallons per day Total: 1 gallon per minute Leakage is evaluated at room temperature.
Approved Alternate Tube Repair Criteria (ARC) 1. For Refueling Outage 14 and the subsequent operating cycle, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging.	-Approved by: Amendment No. 162 dated April 28, 2005 -Applicability: Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging
Approved SG Tube Repair Methods	None
Performance criteria for accident leakage	-1 gallon per minute total leakage through one SG and 1 gallon per minute combined total leakage through all four SGs -Leakage is evaluated at room temperature.

**6.0 ENVIRONMENTAL CONSIDERATION**

WCNOC has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298) as part of the CLIP. WCNOC has concluded that the staff's findings presented in that evaluation are applicable to WCGS and the evaluation is hereby incorporated by reference for this application.

**7.0 REFERENCES**

1. Federal Register Notice: Notice of Availability of Model Application Concerning Technical Specification; Improvement to Modify Requirements Regarding Steam Generator Tube Integrity; Using the Consolidate Line Item Improvement Process, published May 6, 2005 (70 FR 24126)
2. (WCNOC) letter ET 05-0001, "Exigent Request for Revision to Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program,"" from T. J. Garrett, WCNOC to USNRC, dated April 18, 2005.

3. Letter from J. N. Donohew, USNRC to R. A. Muench, WCNOG, "Wolf Creek Generating Station – Issuance of Exigent Amendment Re: Steam Generator (SG) Tube Surveillance Program (TAC NO. MC6757)," dated April 28, 2005

This application is being made in accordance with the CLIP. WCNOG is not proposing variations or deviations from the TS changes described in TSTF-449, Revision 4, or the NRC staff's model SE published on March 2, 2005 (70 FR 10298).

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## 1.1 Definitions (continued)

$\bar{E}$  - AVERAGE  
DISINTEGRATION ENERGY  
(continued)

gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

ENGINEERED SAFETY  
FEATURE (ESF) RESPONSE  
TIME

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

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or
3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. Unidentified LEAKAGE

(primary to secondary LEAKAGE)

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

(continued)

1.1 Definitions (continued)

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LEAKAGE  
(continued)

c. Pressure Boundary LEAKAGE

primary to secondary

LEAKAGE (except ~~SC~~ LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

OPERABLE--OPERABILITY

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

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:

- a. Described in Chapter 14, of the USAR;
- b. Authorized under the provisions of 10 CFR 50.59; or
- c. Otherwise approved by the Nuclear Regulatory Commission.

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(continued)

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. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and~~

~~d.~~ ~~150~~ ~~500~~ gallons per day primary to secondary LEAKAGE through any one ~~SG~~ steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p><i>operational</i></p> <p>A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.</p>	<p>A.1 Reduce LEAKAGE to within limits.</p> <p><i>or primary to secondary LEAKAGE</i></p>	4 hours
<p>B. Required Action and associated Completion Time of Condition A not met.</p> <p><u>OR</u></p> <p>Pressure boundary LEAKAGE exists.</p>	<p>B.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>B.2 Be in MODE 5.</p> <p><u>OR</u></p> <p><i>Primary to secondary LEAKAGE not within limit.</i></p>	<p>6 hours</p> <p>36 hours</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.13.1 -----NOTE-----<sup>(S)</sup></p> <p>1. Not required to be performed until 12 hours after establishment of steady state operation.</p> <p>2. Not applicable to primary to secondary LEAKAGE.</p> <p>Perform RCS water inventory balance.</p> <p>Verify RCS operational LEAKAGE is within limits by performance of</p>	<p>72 hours</p>
<p>SR 3.4.13.2 <del>Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.</del></p>	<p><del>In accordance with the Steam Generator Tube Surveillance Program</del></p>

<p>-----NOTE-----</p> <p>Not required to be performed until 12 hours after establishment of steady state operation.</p> <p>Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.</p>	<p>72 hours</p>
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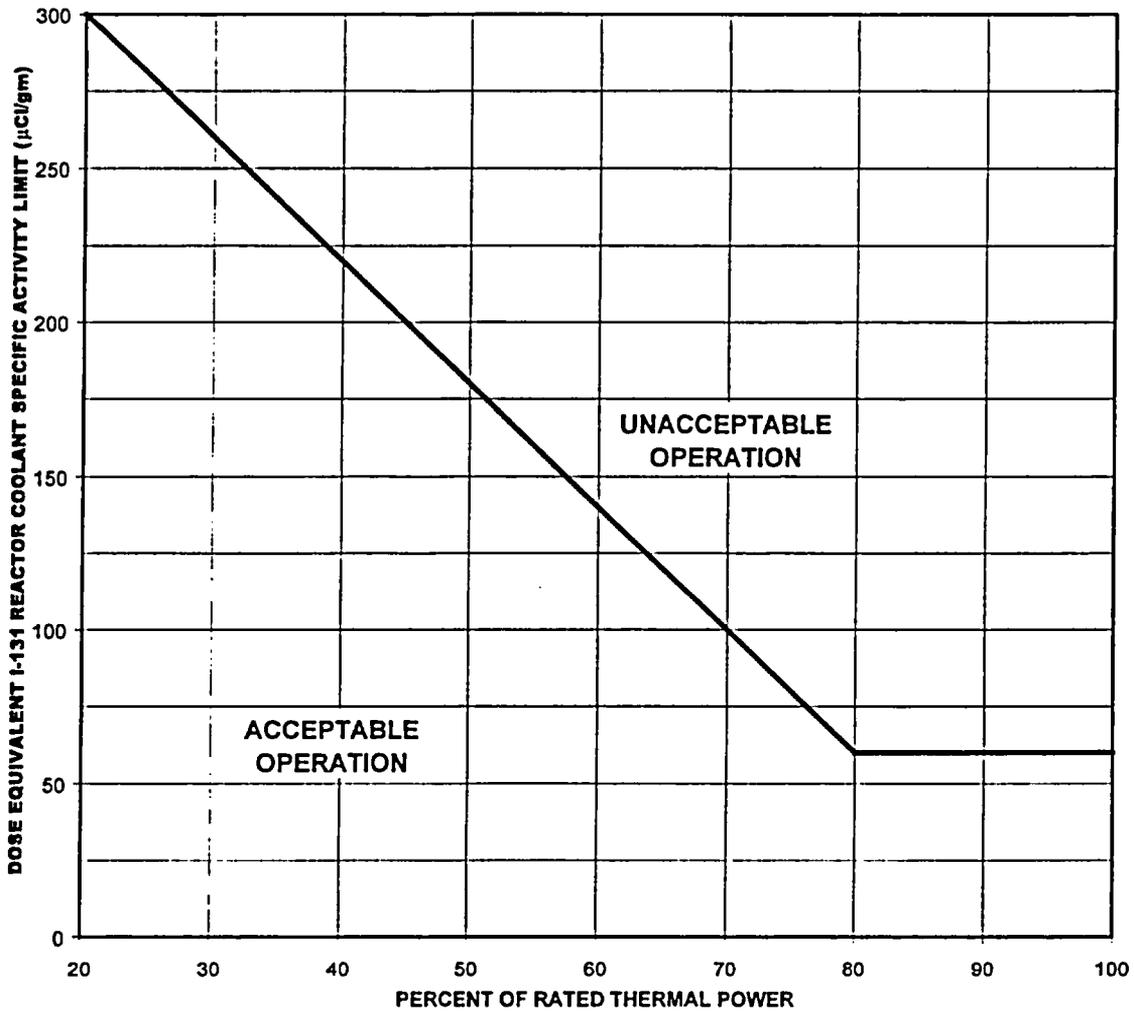


FIGURE 3.4.16-1 (page 1 of 1)  
Reactor Coolant DOSE EQUIVALENT I-131 Specific Activity  
Limit Versus Percent of RATED THERMAL POWER

INSERT NEW LCO 3.4.17 (next two pages)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 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.

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

ACTIONS

NOTE

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	<u>AND</u> A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met.  <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals

5.5.9

Steam Generator (SG) Tube Surveillance Program

INSERT 5.5.9

Steam generator tube integrity shall be demonstrated by performance of the following augmented inservice inspection program.

The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test frequencies.

- a. Steam Generator Sample Selection and Inspection - Steam generator tube integrity shall be determined during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 5.5.9-1.
- b. Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.9.c and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9.d. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; 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 these critical areas;
  2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator 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, and
    - c) A tube inspection (pursuant to Specification 5.5.9.d.1.f) 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

(continued)

INSERT 5.5.9 (page 1 of 2)

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

- a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the “as found” condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The “as found” condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.
- b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.
  1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.
  2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.
  3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, “RCS Operational LEAKAGE.”
- c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following alternate tube repair criteria may be applied as an alternative to the 40% depth-based criteria:

INSERT 5.5.9 (page 2 of 2)

1. For Refueling Outage 14 and the subsequent operating cycle, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging.
- d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.
1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.
  2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.
  3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.
- e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

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

adjacent tube shall be selected and subjected to a tube inspection.

3. The tubes selected as the second and third samples (if required by Table 5.5.9-2 during each inservice inspection may be subjected to a partial tube inspection provided:
  - a) 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
  - b) The inspections include those portions of the tubes where imperfections were previously found.
4. For Refueling Outage 14, a sample of the SG B and C inservice tubes from the top of the hot leg tubesheet to 10 inches below the top of the tubesheet shall be inspected by rotating probe. This sample shall include a 20% minimum sample of the total population of bulges and overexpansions within the SG from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet.

The results of each sample inspection shall be classified into one of the following three categories:

<u>Category</u>	<u>Inspection Results</u>
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.

- c. Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

(continued)

5.5 Programs and Manuals

5.5.9

Steam Generator (SG) Tube Surveillance Program (continued)

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 steam generator conducted in accordance with Table 5.5.9-2 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 5.5.9.c.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 steam generator in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:
  - a) Reactor-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 Double Design Earthquake, or
  - c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or
  - d) A main steam line or feedwater line break.

d. Acceptance Criteria

1. As used in this Specification:

(continued)

5.5 Programs and Manuals

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

- a) Imperfection means 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;
- b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube;
- c) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;
- d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;
- e) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;
- 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. During Refueling Outage 14 and the subsequent operating cycle, this criterion does not apply to degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet. Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging. During Refueling Outage 14 and the subsequent operating cycle, all tubes with degradation identified in the portion of the tube within the region from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be removed from service;
- g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of a Double Design Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.c.3.c, above;

(continued)

5.5 Programs and Manuals

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

h) Tube Inspection means an inspection of the steam generator tube from the tube end (hot leg side) completely around the U-bend to the top support of the cold leg. During Refueling Outage 14 and the subsequent operating cycle, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded;

i) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial Power Operation using the equipment and techniques expected to be used during subsequent inservice inspections; and

j) During Refueling Outage 14 and the subsequent operating cycle:

Bulge refers to a tube diameter deviation within the tubesheet of 18 volts or greater as measured by bobbin probe; and

Overexpansion refers to a tube diameter deviation within the tubesheet of 1.5 mils or greater as measured by bobbin probe.

2. Steam generator tube integrity shall be determined after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.5.9-2.

e. Reports

The contents and frequency of reports concerning the steam generator tube surveillance program shall be in accordance with Specification 5.6.10.

(continued)

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

TABLE 5.5.9-1  
MINIMUM NUMBER OF STEAM GENERATORS TO BE  
INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
	Two	Three	Four	Two	Three	Four
No. of Steam Generators per Unit						
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One <sup>2</sup>			One <sup>1</sup>	One <sup>2</sup>	One <sup>3</sup>

TABLE NOTATIONS

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.
2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.
3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

(continued)

5.5 Programs and Manuals

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

TABLE 5.5.9-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
			C-3	Perform action for C-3 result of first sample	C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G.	All other S.G.s are C-1	None	N.A.	N.A.
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G.	Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
Additional S.G. is C-3			Inspect all tubes in each S.G. and plug defective tubes.	N.A.	N.A.	

$$S = 3 \cdot \frac{N}{n} \%$$

Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection.

(continued)

5.6 Reporting Requirements

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5.6.10 Steam Generator Tube Inspection Report

INSERT 5.6.10

- a. ~~Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission.~~
- b. ~~The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a report within 12 months following completion of the inspection. This Special 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.~~~~
- c. ~~Results of steam generator tube inspections, which fall into Category C-3, shall be reported in a Special Report to the Commission within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

INSERT 5.6.10

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

- a. The scope of inspections performed on each SG;
- b. Active degradation mechanisms found;
- c. Nondestructive examination techniques utilized for each degradation mechanism;
- d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;
- e. Number of tubes plugged during the inspection outage for each active degradation mechanism;
- f. Total number and percentage of tubes plugged to date; and
- g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

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1.1 Definitions (continued)

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**E̅ - AVERAGE  
DISINTEGRATION ENERGY  
(continued)**

gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

**ENGINEERED SAFETY  
FEATURE (ESF) RESPONSE  
TIME**

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

**LEAKAGE**

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or
3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE;

(continued)

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1.1 Definitions (continued)

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LEAKAGE (continued)	c. <u>Pressure Boundary LEAKAGE</u>  LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for channel OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total steps.
MODE	A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE--OPERABILITY	A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are:  a. Described in Chapter 14, of the USAR;  b. Authorized under the provisions of 10 CFR 50.59; or  c. Otherwise approved by the Nuclear Regulatory Commission.

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(continued)

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; and
- d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<p>A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.</p>	<p>A.1 Reduce LEAKAGE to within limits.</p>	<p>4 hours</p>
<p>B. Required Action and associated Completion Time of Condition A not met.</p> <p><u>OR</u></p> <p>Pressure boundary LEAKAGE exists.</p> <p><u>OR</u></p> <p>Primary to secondary LEAKAGE not within limit.</p>	<p>B.1 Be in MODE 3.</p> <p><u>AND</u></p> <p>B.2 Be in MODE 5.</p>	<p>6 hours</p> <p>36 hours</p>

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE	FREQUENCY
<p>SR 3.4.13.1</p> <p>-----NOTES-----</p> <ol style="list-style-type: none"> <li>1. Not required to be performed until 12 hours after establishment of steady state operation.</li> <li>2. Not applicable to primary to secondary LEAKAGE.</li> </ol> <p>-----</p> <p>Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.</p>	<p>72 hours</p>
<p>SR 3.4.13.2</p> <p>-----NOTE-----</p> <p>Not required to be performed until 12 hours after establishment of steady state operation.</p> <p>-----</p> <p>Verify primary to secondary LEAKAGE is <math>\leq</math> 150 gallons per day through any one SG.</p>	<p>72 hours</p>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 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.

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

ACTIONS

NOTE

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.	7 days
	<u>AND</u> A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met.  <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	<u>AND</u> B.2 Be in MODE 5.	36 hours

**SURVEILLANCE REQUIREMENTS**

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5 Programs and Manuals

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5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

- a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.
- b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.
  1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.
  2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm per SG.

(continued)

5.5 Programs and Manuals

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5.5.9 Steam Generator (SG) Program (continued)

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

The following alternate tube repair criteria may be applied as an alternative to the 40% depth-based criteria:

1. For Refueling Outage 14 and the subsequent operating cycle, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging.

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

(continued)

5.5 Programs and Manuals

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5.5.9 Steam Generator (SG) Program (continued)

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.
- e. Provisions for monitoring operational primary to secondary LEAKAGE.

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(continued)

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5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2, and in accordance with the guidance specified below.

- a. Demonstrate for each of the ESF systems that an in-place test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1% when tested in accordance with Regulatory Guide 1.52, Revision 2 at the system flowrate specified below  $\pm 10\%$ .

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation System-Filtration	2000 cfm
Control Room Emergency Ventilation System-Pressurization	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	6500 cfm

(continued)

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5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

- b. Demonstrate for each of the ESF systems that an in-place test of the charcoal absorber shows a penetration and system bypass < 0.05% when tested in accordance with Regulatory Guide 1.52, Revision 2 at the system flowrate specified below  $\pm$  10%.

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation System – Filtration	2000 cfm
Control Room Emergency Ventilation System-Pressurization	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	6500 cfm

- c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal absorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and greater than or equal to the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Emergency Ventilation System (Filtration/Pressurization)	2.5%	70%
Auxiliary/Fuel Building Emergency Exhaust	5%	70%

- d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters, the prefilters, and the charcoal absorbers is less than the value specified below when tested in accordance with Regulatory Guide 1.52, Revision 2, at the system flowrate specified below  $\pm$  10%.

ESF Ventilation System	Delta P	Flowrate
Control Room Emergency Ventilation System - Filtration	6.6 in. W.G.	2000 cfm
Control Room Emergency Ventilation System - Pressurization	3.6 in. W.G.	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	4.7 in. W.G.	6500 cfm

(continued)

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5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

- e. Demonstrate at least once per 18 months that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI N510-1975.

ESF Ventilation System	Wattage
Control Room Emergency Ventilation System - Pressurization	5 ± 1 kW
Auxiliary/Fuel Building Emergency Exhaust	37 ± 3 kW

- f. Demonstrate at least once per 18 months for each of the ESF systems that following the creation of an artificial Delta P across the combined HEPA filters, the prefilters, and the charcoal absorbers of not less than the value specified below (dirty filter conditions), that the flowrate through these flow paths is with ± 10% of the value specified below when tested in accordance with ANSI N510-1980.

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration System	6.6 in. W.G.	2000 cfm
Control Room Pressurization System	3.6 in. W.G.	750 cfm
Auxiliary/Fuel Building Emergency Exhaust	4.7 in. W.G.	6500 cfm

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas Holdup System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, Revision 0, July 1981, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Revision 2, July 1981, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures."

(continued)

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5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

The program shall include:

- a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Holdup System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);
- b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of  $\geq 0.5$  rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and
- c. A surveillance program to ensure that the quantity of radioactivity contained in the following outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.
  - a. Reactor Makeup Water Storage Tank
  - b. Refueling Water Storage Tank
  - c. Condensate Storage Tank, and
  - d. Outside Temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

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(continued)

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5.5.13 Diesel Fuel Oil Testing Program (continued)

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
  - 1. an API gravity or an absolute specific gravity within limits,
  - 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
  - 3. water and sediment content within the limits for ASTM 2D fuel oil;
- b. Other properties for ASTM 2D fuel oil are analyzed within 31 days following sampling and addition to storage tanks; and
- c. Total particulate concentration of the fuel oil is  $\leq 10$  mg/l when tested every 31 days in accordance with ASTM D-2276, Method A.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
  - 1. a change in the TS incorporated in the license; or
  - 2. a change to the USAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the USAR.

(continued)

5.5 Programs and Manuals

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5.5.14 Technical Specifications (TS) Bases Control Program (continued)

- d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

- a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
- b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;
- c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
- d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

- a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or
- b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
- c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

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(continued)

## 5.5 Programs and Manuals

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### 5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

### 5.5.16 Containment Leakage Rate Testing Program

- a. A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exceptions:
  1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.
  2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.
- b. The peak calculated containment internal pressure for the design basis loss of coolant accident,  $P_a$ , is 48 psig.
- c. The maximum allowable containment leakage rate,  $L_a$ , at  $P_a$ , shall be 0.20% of containment air weight per day.
- d. Leakage rate acceptance criteria are:
  1. Containment leakage rate acceptance criterion is  $\leq 1.0 L_a$ . During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are  $< 0.60 L_a$  for the Type B and Type C tests and  $\leq 0.75 L_a$  for Type A tests;

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(continued)

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5.5.16 Containment Leakage Rate Testing Program (continued)

2. Air lock testing acceptance criteria are:
  - a) Overall air lock leakage rate is  $\leq 0.05 L_a$  when tested at  $\geq P_a$ .
  - b) For each door, leakage rate is  $\leq 0.005 L_a$  when pressurized to  $\geq 10$  psig.
- e. The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.
- f. The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.17 Reactor Vessel Head Closure Bolt Integrity

This program provides the requirements to support normal plant operation with one reactor vessel head closure bolt less than fully tensioned for one operating cycle. The provisions of this program shall be implemented when a head closure bolt becomes stuck in a partially inserted position such that the amount of thread engagement is not sufficient to take the tensioning loads without damage to the vessel threads or a bolt is not capable of being inserted into the bolt hole.

Prior to operation with one reactor vessel head closure bolt less than fully tensioned, the following conditions shall apply:

- a. The circumstances associated with the less than fully tensioned closure bolt will be verified to be bounded by the analysis that was referenced in the letter dated September 15, 2000 (WO 00-0036).
- b. A review of the results of the visual examinations performed on the closure bolts shall be performed to ensure that there is no indication of sufficient degradation of closure bolts that could affect the conclusions of Specification 5.5.17a. above.

Within 30 days following startup of the plant, a report shall be submitted to the Commission identifying the circumstances for operation with one reactor vessel head closure bolt less than fully tensioned.

Operation with the same reactor vessel head closure bolt less than fully tensioned shall be limited to one operating cycle (i.e., until the next refueling outage).

## 5.0 ADMINISTRATIVE CONTROLS

### 5.6 Reporting Requirements

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The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Not Used.

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Not Used.

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(continued)

5.6 Reporting Requirements

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5.6.5 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:
1. Specification 3.1.3: Moderator Temperature Coefficient (MTC),
  2. Specification 3.1.5: Shutdown Bank Insertion Limits,
  3. Specification 3.1.6: Control Bank Insertion Limits,
  4. Specification 3.2.3: Axial Flux Difference,
  5. Specification 3.2.1: Heat Flux Hot Channel Factor,  $F_Q(Z)$ ,
  6. Specification 3.2.2: Nuclear Enthalpy Rise Hot Channel Factor ( $F_{\Delta H}^N$ ),
  7. Specification 3.9.1: Boron Concentration,
  8. SHUTDOWN MARGIN for Specification 3.1.1 and 3.1.4, 3.1.5, 3.1.6, and 3.1.8,
  9. Specification 3.3.1: Overtemperature  $\Delta T$  and Overpower  $\Delta T$  Trip Setpoints,
  10. Specification 3.4.1: Reactor Coolant System pressure, temperature, and flow DNB limits, and
  11. Specification 2.1.1: Reactor Core Safety Limits.
- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
1. WCNOC Topical Report TR 90-0025.W01, "Core Thermal Hydraulic Analysis Methodology for the Wolf Creek Generating Station."
  2. WCAP-11397-P-A, "Revised Thermal Design Procedure."
  3. WCNOC Topical Report NSAG-006, "Transient Analysis Methodology for the Wolf Creek Generating Station."

(continued)

5.6 Reporting Requirements

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5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

4. WCAP-10216-P-A, "Relaxation of Constant Axial Offset Control -  $F_Q$  Surveillance Technical Specification."
  5. WCNOC Topical Report NSAG-007, "Reload Safety Evaluation Methodology for the Wolf Creek Generating Station."
  6. NRC Safety Evaluation Report dated March 30, 1993, for the "Revision to Technical Specification for Cycle 7."
  7. WCAP-10266-P-A, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code."
  8. WCAP-11596-P-A, "Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores."
  9. WCAP 10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code."
  10. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report."
  11. WCAP-8745-P-A, "Design Bases for the Thermal Power  $\Delta T$  and Thermal Overtemperature  $\Delta T$  Trip Functions."
- 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.

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(continued)

5.6 Reporting Requirements

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

- a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:
  - 1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and
  - 2. Specification 3.4.12, "Low Temperature Overpressure Protection System."
- b. The analytical methods used to determine the RCS pressure and temperature and Cold Overpressure Mitigation System limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
  - 1. NRC letter dated December 2, 1999, "Wolf Creek Generating Station, Acceptance for Referencing of Pressure Temperature Limits Report (TAC No. MA4572)," and
  - 2. WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January, 1996.
- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not Used.

5.6.8 PAM Report

When a report is required by Condition B or F 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.

5.6.9 Not Used.

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(continued)

## 5.6 Reporting Requirements

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### 5.6.10 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

- a. The scope of inspections performed on each SG;
  - b. Active degradation mechanisms found;
  - c. Nondestructive examination techniques utilized for each degradation mechanism;
  - d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;
  - e. Number of tubes plugged during the inspection outage for each active degradation mechanism;
  - f. Total number and percentage of tubes plugged to date; and
  - g. The results of condition monitoring, including the results of tube pulls and in-situ testing.
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## 5.0 ADMINISTRATIVE CONTROLS

### 5.7 High Radiation Area

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As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601(a) and (b) of 10 CFR Part 20:

- 5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation:
- a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment.
  - b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
  - c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.
  - d. Each individual or group entering such an area shall possess:
    1. A radiation monitoring device that continuously displays radiation dose rates in the area; or
    2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
    3. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or
    4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,

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(continued)

5.7 High Radiation Area

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5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

- (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
  - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, or personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.
- e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation:

- a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:
  - 1. All such door and gate keys shall be maintained under the administrative control of the Shift Manager/Control Room Supervisor or health physics supervision, or his or her designee.
  - 2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.
- b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

(continued)

5.7 High Radiation Area

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5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

- c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual or group entering such an area shall possess:
  - 1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
  - 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or
  - 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and,
    - (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
    - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area, or
  - 3. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

(continued)

5.7 High Radiation Area

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5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

- e. Except for individuals qualified in radiation protection procedures or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.
  - f. Such individual areas that are within a larger area, such as PWR containment, where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.
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**ATTACHMENT IV  
PROPOSED TS BASES CHANGES (for information only)**

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## BASES

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**APPLICABLE SAFETY ANALYSES (continued)** Some accident/safety analyses have been performed at zero power conditions assuming only two RCS loops are in operation to conservatively bound lower modes of operation. The events which assume only two RCPs in operation include the uncontrolled RCCA Bank withdrawal from subcritical event and the hot zero power rod ejection events. While all accident/safety analyses performed at full rated thermal power assume that all the RCS loops are in operation, selected events examine the effects resulting from a loss of RCP operation. These include the complete and partial loss of forced RCS flow, RCP locked rotor, and RCP shaft break events. For each of these events, it is demonstrated that all the applicable safety criteria are satisfied. For the remaining accident/safety analyses, operation of all four RCS loops during the transient up to the time of reactor trip is assumed thereby ensuring that all the applicable acceptance criteria are satisfied. Those transients analyzed beyond the time of reactor trip were examined assuming that a loss of offsite power occurs which results in the RCPs coasting down.

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the limit values, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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**LCO** The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power.

An OPERABLE RCS loop consists of an OPERABLE RCP and an OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow.

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**APPLICABILITY** In MODES 1 and 2, the reactor when critical has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

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**BASES**

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LCO  
(continued)

- a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that the secondary side water temperature of each SG be  $\leq 50^\circ\text{F}$  above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature  $\leq 368^\circ\text{F}$ . This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

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**APPLICABILITY**

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODES 1 and 2";
- LCO 3.4.6, "RCS Loops - MODE 4";
- LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
- LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
- LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
- LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

## BASES

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### LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be removed from operation for  $\leq 1$  hour per 8 hour period. The purpose of the Note is to permit tests that are required to be performed without flow or pump noise. The 1 hour time period is adequate to perform the necessary testing, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, thereby maintaining the margin to criticality. Boron reduction with coolant at boron concentrations less than required to assure the SDM is maintained is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that the secondary side water temperature of each SG be  $\leq 50^\circ\text{F}$  above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature  $\leq 368^\circ\text{F}$ . This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

An OPERABLE RCS loop is comprised of an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program which has the minimum water level specified in SR 3.4.6.2.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

**BASES**

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LCO  
(continued)

- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be  $\leq 50^\circ\text{F}$  above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with any RCS cold leg temperature  $\leq 368^\circ\text{F}$ . This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required. A ~~6~~ OPERABLE SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

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**APPLICABILITY**

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side wide range water level of at least two SGs is required to be  $\geq 66\%$ .

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops - MODES 1 and 2";
- LCO 3.4.5, "RCS Loops - MODE 3";
- LCO 3.4.6, "RCS Loops - MODE 4";
- LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
- LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and

BASES

APPLICABLE  
SAFETY ANALYSES

RCS

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, ~~the~~ operational LEAKAGE related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analyses for events resulting in steam discharge to the atmosphere assume a 1 gpm primary to secondary LEAKAGE as an initial condition.

the other forms of

are

INSERT B 3.4.13A

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. Other accidents or transients involving secondary steam release to the atmosphere, include the steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The USAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via atmospheric relief valves.

the entire

is through the affected

The safety analysis for the SLB accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB and SGTR accidents are well within the limits defined in 10 CFR 100 (Ref. 5) (i.e., a small fraction of these limits).

The safety analysis for RCS main loop piping for GDC-4 (Ref. 1) assumes 1 gpm unidentified leakage and monitoring per Regulatory Guide 1.45 (Ref. 2) are maintained (Ref. 4).

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

LCO

RCS 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 RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

INSERT B 3.4.13 A

that primary to secondary LEAKAGE from all steam generators (SGs) is one gallon per minute or increases to one gallon per minute 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.

BASES

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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 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)

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the accident analyses involving secondary steam discharge to the atmosphere. Violation of this LCO could exceed the offsite dose limits for these accidents. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary to Secondary LEAKAGE through Any One SG

The 500 gallons per day limit on one SG 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 rupture. If leakage is through many cracks, then the cracks are very small, and the above assumption is conservative.

INSERT B 3.4.13 B

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APPLICABILITY

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

RCS operational

INSERT B 3.4.13 B

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. 6). 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.

RCS operational

BASES

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APPLICABILITY  
(continued)

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.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual 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 leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

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ACTIONS

A.1

Unidentified LEAKAGE <sup>or</sup> identified LEAKAGE <sup>or primary to secondary</sup> 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.

B.1 and B.2

If any pressure boundary LEAKAGE exists, <sup>or primary to secondary LEAKAGE is not within limit,</sup> or if unidentified <sup>or</sup> identified LEAKAGE <sup>or primary to secondary LEAKAGE</sup> 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 MODE 3 within 6 hours and MODE 5 within 36 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 RCPB are much lower, and further deterioration is much less likely.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first

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INSERT B 3.4.13 C

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.

INSERT B 3.4.13 D

This SR verifies that primary to secondary LEAKAGE is less 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 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 6. 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. 7).

BASES

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.13.2 (continued)

integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions. This surveillance does not tie directly to any of the leakage criteria in the LCO or of the CONDITIONS; therefore failure to meet this surveillance is considered failure to meet the integrity goals of the LCO and LCO 3.0.3 applies.

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REFERENCES

1. 10 CFR 50, Appendix A, GDC 4 and 30.
  2. Regulatory Guide 1.45, May 1973.
  3. USAR, Section 15.6.3.
  4. NUREG-1061, Volume 3, November 1984.
  5. 10 CFR 100.
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6. NEI 97-06, "Steam Generator Program Guidelines."
7. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

BASES

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- REFERENCES
1. 10 CFR 100.11, 1973.
  2. USAR, Section 15.6.3.
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INSERT NEW LCD 3.4.17 BASES

## B 3.4 REACTOR COOLANT SYSTEM (RCS)

### B 3.4.17 Steam Generator (SG) Tube Integrity

#### BASES

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##### 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 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, 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.

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 5.5.9, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, 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 5.5.9. 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).

## BASES

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### APPLICABLE SAFETY ANALYSES

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 equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for an SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via atmospheric relief valves.

The analysis for design basis accidents and transients other than an 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 equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

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### LCO

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

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

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. For Refueling Outage 14 and the subsequent operating cycle, degradation found in the portion of the tube below 17 inches from the top of the hot leg tube sheet does not require plugging. The portion of the tubes below 17 inches from the top of the hot leg tube sheet is excluded from tube inspections (Ref. 7) The tube-to-tubesheet weld is not considered part of the tube.

## BASES

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### LCO (continued)

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, "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 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 (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

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**BASES**

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**LCO**  
(continued)

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 per SG. 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 3.4.13, "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 an 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.

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**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.

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**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.1 and A.2

Condition A 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 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is

BASES

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ACTIONS

A.1 and A.2 (continued)

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, Condition B applies.

A Completion Time of 7 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, Required Action A.2 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.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition 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 36 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.

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SURVEILLANCE  
REQUIREMENTS

SR 3.4.17.1

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

**BASES**

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**SURVEILLANCE  
REQUIREMENTS**

**SR 3.4.17.1** (continued)

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 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 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 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

**SR 3.4.17.2**

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 5.5.9 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.

BASES

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- REFERENCES
1. NEI 97-06, "Steam Generator Program Guidelines."
  2. 10 CFR 50 Appendix A, GDC 19.
  3. 10 CFR 100.
  4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
  5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
  6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."
  7. License Amendment No. 162, "Wolf Creek Generating Station – Issuance of Exigent Amendment RE: Steam Generator (SG) Tube Surveillance Program (TAC NO. MC6757)," April 28, 2005.
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### LIST OF COMMITMENTS

The following table identifies those actions committed to by WCNOG in this document. Any other statements in this submittal are provided for information purposes and are not considered to be commitments. Please direct questions regarding these commitments to Mr. Kevin Moles at (620) 364-4126.

<b>COMMITMENT</b>	<b>Due Date/Event</b>
The proposed changes to the WCGS Technical Specifications will be implemented prior to startup from Refueling Outage 15.	Prior to startup from Refueling Outage 15