10 CFR 50.90

Progress Energy

Serial: RNP-RA/06-0028

MAY 3 0 2006

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23

REQUEST FOR TECHNICAL SPECIFICATIONS
CHANGE REGARDING STEAM GENERATOR TUBE INTEGRITY
USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

Ladies and Gentlemen:

In accordance with the provisions of 10 CFR 50.90, Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc., is submitting a request for an amendment to the Technical Specifications (TS) for H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2. The proposed amendment would modify the TS requirements related to steam generator tube integrity. The change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Improved Standard Technical Specifications 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).

Attachment I provides an Affirmation as required by 10 CFR 50.30(b).

Attachment II provides a description of the proposed change, the requested confirmation of applicability, and plant-specific verifications.

Attachment III provides the existing TS pages marked-up to show the proposed changes.

Attachment IV provides the revised and retyped TS pages.

Attachment V is provided for information only and includes the existing TS Bases pages marked-up to show the proposed changes.

Approval of the proposed license amendment is requested by November 3, 2006, with the amendment being implemented within 120 days of issuance.

In accordance with 10 CFR 50.91, a copy of this application is being provided to the State of South Carolina.

Progress Energy Carolinas, Inc.
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville, SC 29550

A001

If you should have any questions regarding this submittal, please contact Mr. C. T. Baucom at (843) 857-1253.

Sincerely,

J. F. Lucas
Manager - Support Services - Nuclear

JFL/cac

Attachments:
I. Affirmation
II. Request for Technical Specifications Change Regarding Steam Generator Tube Integrity Using the Consolidated Line Item Improvement Process
III. Proposed Technical Specifications Changes (Mark-Up)
IV. Revised and Retyped Technical Specifications Pages
V. Proposed Changes to Technical Specifications Bases Pages

c: Mr. T. P. O'Kelley, Director, Bureau of Radiological Health (SC)
Mr. H. J. Porter, Director, Division of Radioactive Waste Management (SC)
Dr. W. D. Travers, NRC, Region II
Mr. C. P. Patel, NRC, NRR
NRC Resident Inspector, HBRSEP
Attorney General (SC)

AFFIRMATION

The information contained in letter RNP-RA/06-0028 is true and correct to the best of my information, knowledge, and belief; and the sources of my information are officers, employees, contractors, and agents of Carolina Power and Light Company, also known as Progress Energy Carolinas, Inc. I declare under penalty of perjury that the foregoing is true and correct.

Executed On: 5/30/06

T. D. Walt
Vice President, HBRSEP, Unit No. 2

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REQUEST FOR TECHNICAL SPECIFICATIONS CHANGE REGARDING STEAM GENERATOR TUBE INTEGRITY USING THE CONSOLIDATED LINE ITEM IMPROVEMENT PROCESS

1.0 INTRODUCTION

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

2.0 DESCRIPTION OF PROPOSED AMENDMENT

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

- Revised TS 1.1, "Definitions"
- Revised TS 3.4.13, "RCS (Reactor Coolant System) Operational LEAKAGE"
- New TS 3.4.18, "Steam Generator (SG) Tube Integrity"
- Revised TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program"
- Revised TS 5.6.8, "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 change. The revision to the affected TS Bases pages will be incorporated in accordance with the 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.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

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.

5.0 TECHNICAL ANALYSIS

Carolina Power and Light Company, now doing business as Progress Energy Carolinas, Inc., has reviewed the safety evaluation (SE) published on March 2, 2005 (70 FR 10298) as part of the CLIIP Notice for Comment. This included the NRC staff's SE, the information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. Carolina Power and Light Company has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to H. B. Robinson Steam Electric Plant (HBRSEP), Unit No. 2, and justify this amendment for the incorporation of the changes to the HBRSEP, Unit No. 2, TS.

6.0 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.

6.1 Verification and Commitments

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

Plant Name, Unit No.	HBRSEP, Unit No. 2
Steam Generator Model	Westinghouse Model 44F
Effective Full Power Years (EFPY) of service for currently installed SGs	16.7 EFPY (through March 5, 2006)
Tubing Material	600TT
Number of tubes per SG	3214
Number and percentage of tubes plugged in each SG	A: 7 (0.22%) B: 9 (0.28%) C: 10 (0.31%)
Number of tubes repaired in each SG	A: None B: None C: None
Degradation mechanism(s) identified	None
Current primary-to-secondary leakage limits	Per SG: 150 gallons per day (gpd) Total : 0.3 gallons per minute (gpm) Leakage is evaluated at room temperature
Approved Alternate Tube Repair Criteria (ARC)	None
Approved SG Tube Repair Methods	None
Performance criteria for accident leakage	• Steam Line Break: Faulted SG 0.11 gpm, Unaffected SGs 0.19 gpm • SG Tube Rupture: Faulted SG 0.08 gpm, Intact SGs 0.22 gpm • RCP Locked Rotor: 0.3 gpm total • Rod Withdrawal: 0.3 gpm total

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

Carolina Power and Light Company has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIIP. Carolina Power and Light Company has concluded that the proposed determination presented in the notice is applicable to HBRSEP, Unit No. 2, and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a). As shown in the preceding table, the typical value for assumed accident leakage rate for HBRSEP, Unit No. 2, is 0.3 gpm through the three SGs and 150 gpd through any single SG, as opposed to the 1.0 gpm and 500 gpd leakage rates listed in the generic no significant hazards consideration. The proposed TS changes include a reduction from 150 gpd to 75 gpd to accommodate this difference.

8.0 ENVIRONMENTAL EVALUATION

Carolina Power and Light Company has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298) as part of the CLIIP. Carolina Power and Light Company has concluded that the NRC staff's findings presented in that evaluation are applicable to HBRSEP, Unit No. 2, and the evaluation is hereby incorporated by reference for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. Carolina Power and Light Company 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).

United States Nuclear Regulatory Commission
Attachment III to Serial: RNP-RA/06-0028
17 Pages (including cover page)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PROPOSED TECHNICAL SPECIFICATIONS CHANGES (MARK-UP)

1.1 Definitions

Ē - AVERAGE DISINTEGRATION ENERGY (continued)

iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

LEAKAGE

LEAKAGE shall be:

a. <u>Identified LEAKAGE</u>

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or return), 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 (primary to secondary LEAKAGE);

b. <u>Unidentified LEAKAGE</u>

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

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

LEAKAGE (except ~~SG~~ 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 each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

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

(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. ~~0.3 gpm total primary to secondary LEAKAGE through all steam generators (SGs); and~~

~~e.~~ 75 ~~150~~ gallons per day primary to secondary LEAKAGE through any one steam generator (SG) ~~SG~~.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance. -------------------NOTES--------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. ---	~~Once within 12 hours after reaching steady state operation conditions~~ ~~AND~~ 72 hours ~~thereafter during steady state operation~~
SR 3.4.13.2	~~Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.~~ -------------------NOTE---------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is ≤ 75 gallons per day through any one SG.	~~In accordance with the Steam Generator Tube Surveillance Program~~ 72 hours

Insert New TS Section 3.4.18

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 Steam Generator (SG) Tube Integrity

LCO 3.4.18 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
	AND	
	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. OR SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

Insert New TS Section 3.4.18

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.18.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.18.2	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

Replace with Insert:
Revised TS Section 5.5.9

5.5.9 Steam Generator (SG) Tube Surveillance Program

This program provides controls for the inservice inspection of SG tubes to assure the continued integrity of the Reactor Coolant System pressure boundary and shall include the following:

a. Tube Inspection

Entry from either the hot-leg side or cold-leg side with examination encompassing the area from the hot-leg tube end completely around the U-bend to the top support of the cold leg is considered a tube inspection.

b. Sample Selection and Testing

Selection and testing of steam generator tubes shall be made on the following basis:

1. One steam generator shall be inspected during inservice inspection in accordance with the following requirements:

   (a) The inservice inspection may be limited to one steam generator on a rotating sequence basis. This examination shall include at least 9% of the tubes if the results of the first or a prior inspection indicate that all three generators are performing in a comparable manner.

   (b) When other steam generators are required to be examined by Table 5.5-1 and if the condition of the tubes in one or more generators is found to be more severe than in the other steam generators, the steam generator sampling sequence at the subsequent inservice inspection shall be modified to examine the steam generator or generators with the more severe condition.

2. The minimum sample size, inspection result classification and the associated required action shall be in conformance with the requirements specified in Table 5.5-1. The results of each sampling examination of a steam generator shall be classified into the following three categories:

(continued)

Replace with Insert:
Revised TS Section 5.5.9

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

Category C-1: less than 5% of the total number of tubes examined are degraded but none are defective.

Category C-2: Between 5% and 10% of the total number of tubes examined are degraded, but none are defective or one tube to not more than 1% of the sample is defective.

Category C-3: More than 10% of the total number of tubes examined are degraded, but none are defective or more than 1% of the sample is defective.

In the first sample of a given steam generator during any inservice inspection, degraded tubes not beyond the plugging limit detected by the prior examinations in that steam generator shall be included in the above percentage calculations, only if these tubes are demonstrated to have a further wall penetration of greater than 10% of the nominal tube wall thickness.

3. Tubes shall be selected for examination primarily from those areas of the tube bundle where service experience has shown the most severe tube degradation.

4. The tubes examined in a given steam generator during the first examination of any inservice inspection shall include all non-plugged tubes in that steam generator that from prior examination were degraded, plus additional tubes are required to satisfy the minimum sample size specified in Table 5.5-1. If any selected tube does not permit passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection. This information shall be included in the report required by Specification 5.6.8.b.

5. During the second and third sample examinations of any inservice inspection, the tube inspection may be limited to those sections of the tube lengths where

(continued)

Replace with Insert:
Revised TS Section 5.5.9

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

imperfections were detected during the prior examination.

6. During subsequent inservice inspections, the tube inspection may be limited to certain areas of the tube sheet array and those sections of the tube lengths where imperfections were detected during previous inservice inspections.

c. Examination Method and Requirements

Steam generator tubes shall be examined in accordance with the method prescribed in Appendix IV, "Eddy Current Examination of Non-Ferromagnetic Steam Generator Heat Exchanger Tubes," as contained in ASME Boiler and Pressure Vessel Code, Section XI, "Inservice Inspection of Nuclear Power Plant Components."

d. Inspection Intervals

1. Inservice inspections shall not be more than 24 calendar months apart, except that reduced or tightened inspection intervals shall be governed as specified in 5.5.9.d.3 and d.4.

2. The inservice inspections may be scheduled to be coincident with refueling outages or any plant shutdown, provided the inspection intervals of 5.5.9.d.1, d.3, or d.4, as applicable, are not exceeded.

3. If two consecutive inservice inspections covering a time span of at least 12 months yield results that fall in C-1 category, the inspection frequency may be extended to 40 month intervals between inspections.

4. If the results of the inservice inspection of steam generator tubing conducted in accordance with Table 5.5-1 at 40 month intervals fall in category C-3, the inspection frequency shall be reduced to at least once per 20 months. The increase in inspection frequency shall apply until a subsequent inspection meets the

(continued)

Replace with Insert:
Revised TS Section 5.5.9

5.5 Programs and Manuals

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

conditions specified in 5.5.9.d.3 and the interval can be extended to a 40 month period.

5. Unscheduled inspections shall be conducted in accordance with Specification 5.5.9.b on any steam generator with primary-to-secondary tube leakage (not including leaks originating from tube-to-tube sheet welds) exceeding Specification 3.4.13.

All steam generators shall be inspected before returning to power in the event of a seismic occurrence greater than an operating basis earthquake, a LOCA requiring actuation of engineering safeguards, or a main steam line or feedwater line break.

e. Acceptance Limits

Definitions:

Imperfection is an exception to the dimension, 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.

Degradation means a service induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

Degraded Tube is a tube that contains imperfections caused by degradation equal to or greater than 20% of the nominal tube wall thickness.

Defect is an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.

Plugging Limit is the imperfection depth beyond which a degraded tube must be removed from service by plugging, because the tube may become defective prior to the next scheduled inspection of that tube. The plugging limit is 47% of the nominal tube wall thickness if the next inspection interval of that tube is 12 months, and a 2%

(continued)

Replace with Insert:
Revised TS Section 5.5.9

5.5 Programs and Manuals

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

reduction in the plugging limit for each 12 month period until the next inspection of the inspected steam generator.

f. Corrective Measures

All tubes that leak or are determined to have degradation exceeding the plugging limit shall be plugged prior to return to power.

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 critical parameters, their sampling frequency, sampling points, and control band limits;

b. Procedures used to measure the critical parameters;

c. Requirements for the documentation and review of sample results;

d. Procedures which identify the administrative events and corrective actions required to return the secondary chemistry to its normal control band following an out of control band condition; and

e. Identification of the authority responsible for the interpretation of the sample results.

(continued)

Insert:
Revised TS Section 5.5.9

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.

Insert:
Revised TS Section 5.5.9
(continued)

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 75 gallons per day 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 the following criteria shall be plugged: 47% of the nominal tube wall thickness if the next inspection interval of that tube is 12 months, and a 2% reduction in the plugging limit for each 12 month period until the next inspection of the inspected SG.

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 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

Insert:
Revised TS Section 5.5.9
(continued)

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 (continued)

TABLE 5.5-1

STEAM GENERATOR TUBE INSPECTION

1st SAMPLE EXAMINATION			2nd SAMPLE EXAMINATION		3rd SAMPLE EXAMINATION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per Steam Generator (SG) S=3(N/n)% where: N is the number of steam generators in the plant = 3 n is the number of steam generators inspected during an examination	C-1	Acceptable for Continued Service	N/A	N/A	N/A	N/A
	C-2	Plug tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator	C-1	Acceptable for Continued Service	N/A	N/A
			C-2	Plug tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator	C-1	Acceptable for Continued Service
					C-2	Plug tubes exc. plug limit. Acceptable for continued service
					C-3	Perform action required under C-3 of 1st sample examination
			C-3	Perform action required under C-3 of 1st sample examination	N/A	N/A
	C-3	Inspect all tubes in this SG, plug tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in each other steam generator not included in the inservice inspection program. Report results to NRC.	All other SGs are C-1	Acceptable for Continued Service	N/A	N/A
			Some SGs are C-2 but no additional SGs are C-3	Perform action required under C-2 of 2nd sample examination above	N/A	N/A
			Additional SG is C-3	Inspect all tubes in this SG and plug tubes exceeding the plugging limit. Report results to NRC	N/A	N/A

5.6 Reporting Requirements (continued)

5.6.7 Tendon Surveillance Report

a. Notification of a pending sample tendon test, along with detailed acceptance criteria, shall be submitted to the NRC at least two months prior to the actual test.

b. A report containing the sample tendon test evaluation shall be submitted to the NRC within six months of conducting the test.

5.6.8 Steam Generator Tube Inspection Report

Replace with Insert: Revised TS Section 5.6.8

~~a. A report of the number of tubes plugged in each steam generator shall be submitted to the NRC within 14 days after completion of the tube plugging.~~

~~b. A report of the results of the steam generator tube inspection shall be submitted to the NRC within 60 days after completion of the final inspection.~~

~~Reports shall include:~~

~~1. Number and extent of tubes inspected~~
~~2. Location and percent of wall thickness penetration for each eddy current indication and any leaks.~~
~~3. Identification of tubes plugged.~~

~~c. A report of examination results falling in Category C-3 of Table 5.5-1 shall be submitted to the NRC within 30 days, and prior to resumption of plant operation.~~

~~A report of investigations conducted to determine cause(s) of the tube degradation and corrective measures taken to prevent recurrence shall be submitted within 90 days following completion of the startup test program.~~

Insert:
Revised TS
Section
5.6.8

5.6.8 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.

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

United States Nuclear Regulatory Commission
Attachment IV to Serial: RNP-RA/06-0028
27 Pages (including cover page)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

REVISED AND RETYPED TECHNICAL SPECIFICATIONS PAGES

(continued)

1.1 Definitions

Ē-AVERAGE DISINTEGRATION ENERGY (continued)	iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.
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 return), 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 return) that is not identified LEAKAGE; c. Pressure Boundary LEAKAGE 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 each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.
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

(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. 75 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
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.13.1	------------------NOTES------------------ 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.13.2	------------------NOTE------------------ Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is ≤ 75 gallons per day through any one SG.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.18 Steam Generator (SG) Tube Integrity

LCO 3.4.18 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
	AND	
	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. OR SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.18.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.18.2	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 (continued)

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.

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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 75 gallons per day 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 the following criteria shall be plugged: 47% of the nominal tube wall thickness if the next inspection interval of that tube is 12 months, and a 2% reduction in the plugging limit for each 12 month period until the next inspection of the inspected SG.

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 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

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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.

5.5.10 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 critical parameters, their sampling frequency, sampling points, and control band limits;

b. Procedures used to measure the critical parameters;

c. Requirements for the documentation and review of sample results;

d. Procedures which identify the administrative events and corrective actions required to return the secondary chemistry to its normal control band following an out of control band condition; and

e. Identification of the authority responsible for the interpretation of the sample results.

(continued)

5.5 Programs and Manuals (continued)

5.5.11 Ventilation Filter Testing Program (VFTP)

This program provides controls for implementation of the following required testing of Engineered Safety Feature (ESF) ventilation filter systems at the frequencies specified in Positions C.5 and C.6 of Regulatory Guide 1.52, Revision 2, March 1978, and conducted in general conformance with ANSI N510-1975 or N510-1980.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows the specified penetration and system bypass leakage when tested in accordance with the referenced standard at the system flowrate specified below.

ESF Ventilation System	Penetration /Bypass	Flowrate	Reference Std
Control Room Emergency	<0.05%	3300 - 4150 ACFM	Regulatory Guide 1.52, Revision 2, March 1978, C.5.a, C.5.c, C.5.d (using ANSI N510-1980)
Spent Fuel Building	≤1%	11070- 13530 CFM	ANSI N510-1975
Containment Purge	≤1%	31500- 38500 CFM	ANSI N510-1975

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows the specified penetration and system bypass leakage when tested in accordance with the referenced standard at the system flowrate specified below.

ESF Ventilation System	Penetration /Bypass	Flowrate	Reference Std
Control Room Emergency	<0.05%	3300 - 4150 ACFM	Regulatory Guide 1.52, Revision 2, March 1978, C.5.a, C.5.c, C.5.d (using ANSI N510-1980)
Spent Fuel Building	≤1%	11070- 13530 CFM	ANSI N510-1975
Containment Purge	≤1%	31500- 38500 CFM	ANSI N510-1975

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, 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 (86°) and the relative humidity specified below.

ESF Filter System	Penetration	RH
Control Room Emergency	≤2.5%	70%
Spent Fuel Building	≤10%	70%
Containment Purge	≤10%	95%

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters, and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Filter System	Delta P	Flowrate
Control Room Emergency	≤3.4 inches water gauge	3300 - 4150 ACFM
Spent Fuel Building	<6 inches water gauge	12300 CFM ±10%
Containment Purge	<6 inches water gauge	35000 CFM ±10%

e. Demonstrate that the heaters for the Spent Fuel Building ventilation filter system maintains the filter inlet air at ≤ 70% relative humidity when tested in accordance with ASME N510-1975.

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 Decay Tanks, the quantity of radioactivity contained in The Waste Gas Decay Tanks and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas Decay Tanks and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate

(continued)

5.5 Programs and Manuals

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

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 Waste Gas Decay Tank is less than the amount that would result in a whole body exposure of ≥ 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 each outdoor liquid radwaste tank that is not surrounded by liners, dikes, or walls, capable of holding the tank's contents and that does not have tank overflows and surrounding area drains connected to the Liquid Waste Disposal System is less than or equal to ten (10) Curies, excluding tritium and dissolved or entrained noble gases.

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 shall be established requiring testing of both new fuel oil and stored fuel oil. The program shall include sampling and testing requirements, and acceptance criteria. The testing methods shall be in accordance with applicable ASTM Standards. The acceptance criteria shall be in accordance with the diesel engine manufacturer specifications. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has not become contaminated with other products during transit, thus altering the quality of the fuel oil.

(continued)

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program (continued)

b. Acceptability of fuel oil for use by testing the following parameters at a 31 day frequency:

API or specific gravity, viscosity, water and sediment, and cloud point.

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

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides controls 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 involve either of the following:

1. a change in the TS incorporated in the license; or

2. a change to the updated FSAR 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 UFSAR.

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 provides controls to ensure 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

(continued)

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

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.

a. The SFDP shall contain the following:

   1. 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;

   2. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

   3. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

   4. Other appropriate limitations and remedial or compensatory actions.

b. 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:

   1. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

   2. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

   3. A required system redundant to the support system(s) for the supported systems described in b.1 and b.2 above is also inoperable.

c. 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.

(continued)

5.5 Programs and Manuals (continued)

5.5.16 Containment Leakage Rate Testing Program

This program provides controls for implementation of 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 for Type A testing. 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 te following exception:

a. NEI 94-01 - 1995, Section 9.3.2: The first Type A test performed after the April 9, 1992, Type A test shall be performed no later than April 9, 2007.

Type B and C testing shall be implemented in the program in accordance with the requirements of 10 CFR 50, Appendix J, Option A.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 40.5 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.1% of the containment air weight per day.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criteria is $\leq$ 1.0 L_a. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq$ 0.60 L_a for the Type B and Type C tests, and $\leq$ 0.75 L_a for Type A tests.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

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

5.6.1 DELETED

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 15 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 the format of Table 3 in the Radiological Assessment Branch Technical Position, Revision 1, November 1979.

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

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 shall be submitted 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 DELETED

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. Shutdown Margin (SDM) for Specification 3.1.1;

2. Moderator Temperature Coefficient limits for Specification 3.1.3;

3. Shutdown Bank Insertion Limits for Specification 3.1.5;

4. Control Bank Insertion Limits for Specification 3.1.6;

5. Heat Flux Hot Channel Factor ($F_Q(Z)$) limit for Specification 3.2.1;

6. Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$) limit for Specification 3.2.2;

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

7. Axial Flux Difference (AFD) limits for Specification 3.2.3; and

8. Boron Concentration limit for Specification 3.9.1.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC. The approved version shall be identified in the COLR. These methods are those specifically described in the following documents:

1. XN-75-27(A), "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors," approved version as specified in the COLR.

2. XN-NF-84-73(P), "Exxon Nuclear Methodology for Pressurized Water Reactors: Analysis of Chapter 15 Events," approved version as specified in the COLR.

3. XN-NF-82-21(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," approved version as specified in the COLR.

4. Steam Line Break Methodology as defined by:

   ANF-84-093(P)(A), "Steamline Break Methodology for PWRs," approved version as specified in the COLR.

   EMF-84-093(P)(A), "Steam Line Break Methodology for PWRs," approved version as specified in the COLR.

5. XN-75-32(A), "Computational Procedure for Evaluating Rod Bow," approved version as specified in the COLR.

6. XN-NF-82-49(A), "Exxon Nuclear Corporation Evaluation Model EXEM PWR Small Break Model," approved version as specified in the COLR.

7. EMF-2087 (P)(A), "SEM/PWR-98: ECCS Evaluation Model for PWR LBLOCA Applications," approved version as specified in the COLR.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

8. XN-NF-78-44(A), "Generic Control Rod Ejection Analysis," approved version as specified in the COLR.

9. XN-NF-621(A), "XNB Critical Heat Flux Correlation," approved version as specified in the COLR.

10. ANF-1224(A), "Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

11. XN-NF-82-06(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

12. WCAP-10080-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code," approved version as specified in the COLR.

13. WCAP-10081-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP code," approved version as specified in the COLR.

14. WCAP-8301 (Proprietary) and WCAP-8305 (Nonproprietary), "LOCTA-IV Program: Loss of Coolant Transient Analysis," approved version as specified in the COLR.

15. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 87 to Facility Operating License No. DPR-23, Carolina Power & Light Co., H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261," USNRC, Washington, DC 20555, 7 Nov. 84.

16. ANF-88-054(P), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," approved version as specified in the COLR.

17. ANF-88-133 (P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 Gwd/MTU," approved version as specified in the COLR.

(continued)

5.6 Reporting Requirements (continued)

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

18. ANF-89-151(A), "ANF-RELAP Methodology for Pressurized Water Reactors: Analysis of Non-LOCA Chapter 15 Events," approved version as specified in the COLR.

19. EMF-92-081(A), "Statistical Setpoint/Transient Methodology for Westinghouse Type Reactors," approved version as specified in the COLR.

20. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel," approved version as specified in the COLR.

21. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

22. EMF-96-029(P)(A), "Reactor Analysis System for PWRs," approved version as specified in the COLR.

23. EMF-92-116, "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

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

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

(continued)

5.6 Reporting Requirements (continued)

5.6.6 Post Accident Monitoring (PAM) Instrumentation Report

When a report is required by Condition B or H 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.

5.6.7 Tendon Surveillance Report

a. Notification of a pending sample tendon test, along with detailed acceptance criteria, shall be submitted to the NRC at least two months prior to the actual test.

b. A report containing the sample tendon test evaluation shall be submitted to the NRC within six months of conducting the test.

5.6.8 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.

g. The results of condition monitoring, including the results of tube pulls and in-situ testing.

5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

5.7.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) of 10 CFR 20, each High Radiation Area in which the intensity of radiation is 1000 mRem/hour or less shall be barricaded and conspicuously posted as a High Radiation Area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP).

Radiation control personnel or personnel escorted by radiation control personnel shall be exempt from the RWP issuance requirements during the performance of their assigned duties within the RCA, provided they comply with approved radiation protection procedures for entry into High Radiation Areas.

Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device that continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device provided for each individual that continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate levels in the area have been established and personnel are aware of them.

c. An individual qualified as a radiation control technician with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the radiation control supervisor in the RWP.

(continued)

5.7 High Radiation Area (continued)

5.7.2 The requirements of 5.7.1 shall apply to each High Radiation Area in which the intensity of radiation is greater than 1000 mRem/hour at 30 centimeters (12 inches) 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. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the SS on duty and/or the radiation control supervisor. Entrance thereto shall also be controlled by requiring issuance of an RWP. The exemption from RWP issuance requirements discussed in 5.7.1 is not applicable for any High Radiation Area in which the intensity of radiation is greater than 1000 mRem/hour.

United States Nuclear Regulatory Commission
Attachment V to Serial: RNP-RA/06-0028
18 Pages (including cover page)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2

PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS BASES PAGES

BASES

LCO (continued)

OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus 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.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops – MODE 3";
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 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

(continued)

BASES (continued)

LCO (continued)

will prevent the occurrence of an inadvertent control rod withdrawal transient. An alternate condition, described in item c.4 of the Note, is to maintain SDM within the MODE 3 limit for no RCS loops in operation as specified in the COLR. This SDM limit is sufficient to prevent a return to criticality in the event of simultaneous withdrawal of the two most reactive control rod banks as assumed in the inadvertent control rod transient analysis.

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.

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 RTBs in the closed position. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the RTBs open.

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 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6).

(continued)

BASES

LCO (continued)	Note 2 requires that there be a steam bubble in the pressurizer or the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. An OPERABLE RCS loop comprises 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 train 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.
APPLICABILITY	In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop or train of either RCS or RHR provides sufficient circulation for these purposes. However, two circuits consisting of any combination of RCS loops and RHR trains are required to be OPERABLE to meet single failure considerations. 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 3.4.7, "RCS Loops—MODE 5, Loops Filled"; LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled"; LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation—High Water Level" (MODE 6); and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation—Low Water Level" (MODE 6).
ACTIONS	A.1 If one required RCS loop or RHR train is inoperable and only one required RCS loop remains OPERABLE, the intended redundancy for heat removal is lost. Action must be initiated to restore a second RCS loop or RHR train to

(continued)

BASES

LCO (continued)

experience has shown that boron stratification is not likely 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 initial startup 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, therefore 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 allows one RHR train to be inoperable and de-energized for a period of up to 2 hours, provided that the other RHR train is OPERABLE. This permits periodic surveillance tests to be performed on the inoperable train during the only time when such testing is safe and possible.

Note 3 requires that there be a steam bubble in the pressurizer or the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP). 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 trains 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 trains.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An ~~OPERABLE~~ SG can perform as a heat sink when it has an adequate water level, the RCS is not vented, and is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program~~.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes ~~a 0.3 gpm primary to secondary LEAKAGE as the initial condition~~.

that primary to secondary LEAKAGE from all steam generators (SGs) is 0.3 gpm or increases to 0.3 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 75 gallons per day is significantly less than the conditions assumed in the safety analyses.

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

For the SGTR, the activity released due to the 0.3 gpm primary to secondary LEAKAGE is relatively insignificant compared to the activity released via the ruptured tube. The safety analysis for the SGTR accident assumes 0.3 gpm total primary to secondary LEAKAGE in all generators as an initial condition. After mixing in the secondary side, the activity is then released via the SG PORVs or safeties. This release pathway continues until the SGs are isolated, which is relatively soon for the affected SG compared to the intact SGs. The dose consequences resulting from the SGTR accident are within the limits defined in 10 CFR 50.67.

The RCS operational LEAKAGE satisfies Criterion 2 of the NRC Policy Statement.

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.

(continued)

BASES (continued)

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment atmosphere radiation monitoring systems, condensate measuring system, dewpoint monitoring equipment, 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 identified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 75 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 3). The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational LEAKAGE criterion of 75 gallons per day in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures and maintaining primary to secondary LEAKAGE within the applicable accident analysis assumptions.

~~d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)~~

~~Total primary to secondary LEAKAGE amounting to 0.3 gpm through all SGs produces acceptable offsite doses in the SGTR accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. 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 150 gallons per day limit on one SG produces acceptable dose consequences in the SGTR accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.~~

(continued)

BASES (continued)

APPLICABILITY

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

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

LCO 3.4.14, "RCS Pressure Isolation Valves (PIVs)," 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.

ACTIONS

A.1

Unidentified LEAKAGE~~,~~ or identified LEAKAGE~~, or primary to secondary LEAKAGE~~ in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, primary to secondary is not within limit, or Required Action A.1 is not met. ~~or if unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours~~, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to 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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

acting on the RCPB are much lower, and further deterioration is much less likely.

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 appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. ~~Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.~~

The surveillance is modified by two notes. Note 1 states that

The RCS water inventory balance must be met with the reactor at steady state operating conditions. ~~Therefore, the initial performance of~~ this SR is required within 12 hours after reaching continuous steady state operation~~, and is performed when steady state operation is achieved at ≥ 500 psig~~.

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

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE because LEAKAGE of 75 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72 hour Frequency during steady state operation is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less than or equal to 75 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 3.4.18, "Steam Generator Tube Integrity," should be evaluated. The 75 gallons per day limit is measured at room temperature as described in Reference 4. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, the entire primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

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

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

~~This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube 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.~~

REFERENCES

1. UFSAR, Section 3.1.
2. UFSAR, Chapter 15.
3. NEI 97-06, "Steam Generator Program Guidelines."
4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

Insert New TS Bases Section 3.4.18

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.18 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "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

(continued)

Insert New TS Bases Section 3.4.18

BASES (Continued)

BACKGROUND (continued)

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).

APPLICABLE SAFETY ANALYSES

greater than

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 a 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 a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is discharged to the main condenser~~.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 0.3 gallon per minute or is assumed to increase to 0.3 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 100.11~~ and 10 CFR 50.67 (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).

(continued)

Insert New TS Bases Section 3.4.18

BASES (Continued)

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

In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 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

(continued)

Insert New TS Bases Section 3.4.18

LCO (continued)

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).

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 150 gpm per SG, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage. 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 75 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

(continued)

Insert New TS Bases Section 3.4.18

BASES (Continued)

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4.

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

ACTIONS

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequently 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 3.4.18.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Condition A does not apply to the occurrence of primary to secondary LEAKAGE, which is monitored and maintained in accordance with LCO 3.4.13. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

(continued)

Insert New TS Bases Section 3.4.18

BASES (Continued)

ACTIONS (continued)

A.1 and A.2 (continued)

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with an 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.

SURVEILLANCE REQUIREMENTS

SR 3.4.18.1

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.

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring

(continued)

Insert New TS Bases Section 3.4.18

BASES (Continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.18.1 (continued)

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, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.18.1. 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 3.4.18.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 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.

(continued)

Insert New TS Bases Section 3.4.18

BASES (Continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.18.2 (continued)

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

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 50.67 ~~and 10 CFR 100.11~~.
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."