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SCE&G®
A SCANA COMPANY

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RC-06-0084

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

ATTN: R. E. Martin

Dear Sir / Madam:

Subject: VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)
DOCKET NO. 50/395
OPERATING LICENSE NO. NPF-12
LICENSE AMENDMENT REQUEST - LAR 05-3594
STEAM GENERATOR TUBE INTEGRITY

In accordance with provisions of section 50.90 of Title 10 of the Code of Federal Regulations (10 CFR), South Carolina Electric & Gas Company (SCE&G), acting for itself and as agent for South Carolina Public Service Authority, hereby requests an amendment to the VCSNS Technical Specifications (TS).

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

Attachment I provides a description of the proposed change and confirmation of applicability.
Attachment II provides the existing TS pages marked up to show the proposed change.
Attachment III provides retyped TS pages.
Attachment IV provides the existing TS Bases pages marked up to show the proposed change.
Attachment V provides retyped TS Bases pages.
Attachment VI discusses any new Regulatory Commitments.

SCE&G requests approval of the proposed license amendment as soon as possible, with the amendment being implemented within 90 days.

The VCSNS Plant Safety Review Committee and the Nuclear Safety Review Committee have reviewed and approved the proposed change. SCE&G has notified the State of South Carolina in accordance with 10CFR50.91 (b).

Another License Amendment Request (LAR), LAR 05-0677 - On-Line Monitoring of Instrument Channel Performance, affecting TS page 1-4 in this LAR was submitted on February 6, 2006. A revised page will be resubmitted should LAR 05-0677 be approved first.

A001

If you have any questions or require additional information regarding this proposed amendment, please contact Mr. Rob Sweet at (803) 345-4080.

I certify under penalty of perjury that the foregoing is true and correct.

5/24/06
Executed on

Jeffrey B. Archie

JT/JBA/dr

Attachment(s): 6

I. Description of Proposed Change and Confirmation of Applicability
II. Proposed Technical Specification Change - Mark-up
III. Proposed Technical Specification Change - Retyped
IV. Proposed Technical Specification Bases Change - Mark-up
V. Proposed Technical Specification Bases Change - Retyped
VI. List of Regulatory Commitments

c: N. O. Lorick
S. A. Byrne
N. S. Carns
J. H. Hamilton (w/o Attachments)
R. J. White
W. D. Travers
NRC Resident Inspector
P. Ledbetter
K. M. Sutton
T. P. O'Kelley
RTS (LAR 05-3594)
File (813.20)
DMS (RC-06-0084)

VIRGIL C. SUMMER NUCLEAR STATION (VCSNS)
LICENSE AMENDMENT REQUEST - LAR 05-3594
STEAM GENERATOR TUBE INTEGRITY-DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

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

2.0 DESCRIPTION OF PROPOSED AMMENDMENT

Consistent with the NRC-approved Revision 4 of TSTF-449, South Carolina Electric & Gas Company (SCE&G) proposes an amendment to revise the VCSNS TS. The proposed TS changes include:

- Revised TS definition of IDENTIFIED LEAKAGE
- Revised TS definition of PRESSURE BOUNDARY LEAKAGE
- New TS 3/4.4.5, "Steam Generator Tube Integrity"
- Revised TS 3/4.4.6.2, Reactor Coolant System (RCS) Operational Leakage
- New TS 6.8.4.k, Steam Generator Program
- New TS 6.9.1.12, Steam Generator Tube Inspection Report

Proposed revisions to the TS Bases are also included in this application. As discussed in the NRC's model safety evaluation, adoption of the revised TS Bases associated with TSTF-449, Revision 4, is an integral part of implementing this TS improvement. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program (TS 6.8.4.i).

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 10298), and TSTF-449, Revision 4.

5.0 TECHNICAL ANALYSIS

SCE&G 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 supporting information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. SCE&G has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to VCSNS and justify this amendment for the incorporation of the changes to the VCSNS 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.:	V. C. Summer Nuclear Station (Single Unit 1)
Steam Generator Model:	Westinghouse D75
Effective Full Power Years of Service for Currently Installed SGs:	9.2
Tubing Material:	Inconel 690TT
Number of Tubes per SG:	6307
Number and Percentage of Tubes Plugged in each SG:	A (3, 0.05%) B (1, 0.02%) C (4, 0.06%)
Number of Tubes Repaired in each SG:	No tubes repaired
Degradation Mechanisms Identified:	Wear at Tube Support Plates (not classified active)
Current Primary-to-Secondary Leakage Limits:	150 gpd through any one steam generator 450 gpd total from all SGs Leakage is calculated at room temperature
Approved Alternate Tube Repair Criteria (ARC):	No approved ARC
Approved SG Tube Repair Methods:	No approved SG Tube Repair Methods
Performance Criteria for Accident Leakage:	1 gpm primary-to-secondary leakage is assumed in the licensing basis accident analysis. The conversion to mass flow is based on room temperature assuming a reference density of 62.4 lb/ft^3.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

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

8.0 ENVIRONMENTAL EVALUATION

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

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. SCE&G 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).

10.0 ATTACHMENT I REFERENCES

1. Notice for Comment published on March 2, 2005 (70 FR 10298)
2. Notice of Availability published on May 6, 2005 (70 FR 24126)

ATTACHMENT II

PROPOSED TECHNICAL SPECIFICATION CHANGES (MARK-UP)

Attachment to License Amendment No. XXX
To Facility Operating License No. NPF-12
Docket No. 50-395

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
Index V	Index V
Index XIII	Index XIII
1-3	1-3
1-4	1-4
3/4 4-11	3/4 4-11
3/4 4-12	3/4 4-12
3/4 4-13	--
3/4 4-14	--
3/4 4-15	--
3/4 4-16	--
3/4 4-17	--
3/4 4-19	3/4 4-19
3/4 4-20	3/4 4-20
6-12d	6-12d
--	6-12e
--	6-12f
--	6-16b
B 3/4 4-3	B 3/4 4-3
--	B 3/4 4-3a
--	B 3/4 4-3b
--	B 3/4 4-3c
--	B 3/4 4-3d
--	B 3/4 4-3e
B 3/4 4-4	B 3/4 4-4
--	B 3/4 4-4a
--	B 3/4 4-4b
--	B 3/4 4-4c
--	B 3/4 4-4d
--	B 3/4 4-4e
B 3/4 4-5	B 3/4 4-5

SCE&G -- EXPLANATION OF CHANGES

Page	Affected Section	Bar #	Description of Change	Reason for Change
Index V	3/4.4.5	1	Changed Title Section.	Administrative.
Index XIII	3/4.4.5	1	Changed Title Section.	Administrative.
Index XIII	3/4.4.8	2	Changed Page #.	TS Amendment 154 failed to revise the Index Page.
1-3	1.15.c	1	Add (primary-to-secondary leakage).	To define leakage as primary-to-secondary (TSTF-449 Rev. 4).
1-4	1.21	1	Change "steam generator tube" leakage to "primary-to-secondary" leakage.	To apply correct definition per change to 1.15.c (TSTF-449 Rev. 4).
3/4 4-11	3/4.4.4	1	Replace Steam Generator section with new Steam Generator Tube Integrity section.	Steam Generator program moved to TS 6.8.4.k. New section addresses SG tube integrity. (TSTF-449 Rev. 4).
3/4 4-12 through 3/4 4-17	3/4.4.4	1	Section was replaced.	Pages 3/4 4-12 through 3/4 4-17 were deleted.
3/4 4-19	3.4.6.2	1	Editorial revision to 3.4.6.2.	To address RCS operational leakage.
	3.4.6.2.c	2	Editorial revision to 3.4.6.2.c.	Revised limiting condition to address primary-to-secondary leakage (TSTF-449 Rev. 4).
	Action a	3	Apply primary-to-secondary leakage to action a.	Revised to address primary-to-secondary leakage (TSTF-449 Rev. 4).
	Action b	4	Apply primary-to-secondary leakage to action b.	Revised to address primary-to-secondary leakage (TSTF-449 Rev. 4).
	4.4.6.2.1	5	Editorial revision to SR 4.4.6.2.1.	To address RCS operational leakage.
3/4 4-20	4.4.6.2.1.d	1	Identify footnote (1) and apply primary-to-secondary leakage to SR 4.4.6.2.1.d.	Clarify scope of leakage surveillances and when they will be performed.
	4.4.6.2.3	2	Add 4.4.6.2.3.	To verify primary-to-secondary leakage.

Page	Affected Section	Bar #	Description of Change	Reason for Change
	footnote	3	Add footnote (1).	Clarify scope of leakage surveillances and when they will be performed.
6-12d New 6-12e 6-12f	6.8.4.k	1	Insert new section 6.8.4.k, "Steam Generator Program".	Add new section to describe steam generator program requirements (TSTF-449 Rev. 4).
New 6-16b	6.9.1.12	1	Insert new "Steam Generator Inspection Report".	Describe reporting requirements for steam generator inspections (TSTF-449 Rev. 4).
B 3/4 4-3 New B 3/4 4-3a B 3/4 4-3b B 3/4 4-3c B 3/4 4-3d B 3/4 4-3e	3/4.4.5		Insert Bases for Steam Generator Tube Integrity.	Describe conservative bases for steam generator tube integrity (TSTF-449 Rev. 4).
B 3/4 4-4 New B 3/4 4-4a B 3/4 4-4b B 3/4 4-4c B 3/4 4-4d B 3/4 4-4e	3/4.4.6.2	1	Revise Bases for Operational Leakage.	To reflect the conservative limits for operational leakage (TSTF-449 Rev. 4).
B 3/4 4-5	3/4.4.7	1	Repaginate.	Repagination caused by revision to preceding pages.

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. 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

c. Reactor coolant system leakage through a steam generator to the secondary system (primary-to-secondary leakage).

MASTER RELAY TEST

1.16 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except ~~steam generator tube~~ primary-to-secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes wil be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

INSERT 3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.5.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators. The tubes selected for these inspections shall be selected on a random basis except:

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

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

INSERT 3/4.4.5

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Steam generator tube integrity shall be maintained.

AND

All steam generator tubes satisfying the tube repair criteria shall be plugged or repaired in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

The ACTIONS may be entered separately for each steam generator tube.

a. With one or more steam generator tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program,

   1. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or steam generator tube inspection, or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and

   2. Plug or repair the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or steam generator tube inspection.

b. With steam generator tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify steam generator tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected steam generator tube that satisfies the tube repair criteria is plugged or repaired in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a steam generator tube inspection.

SURVEILLANCE REQUIREMENTS (Continued)

1. All nonplugged tubes that previously had detectable wall penetrations greater than 20%.

2. Tubes in those areas where experience has indicated potential problems.

3. A tube inspection (pursuant to Specification 4.4.5.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:

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

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

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

Category	Inspection Results
C-1	Less than 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

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

a. The first inservice inspection after the steam generator replacement shall be performed after at least 6 Effective Full Power Months from the time of the replacement but within 24 calendar months of initial criticality after the steam generator replacement. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.*

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:

1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2.

2. A seismic occurrence greater than the Operating Basis Earthquake.

3. A loss-of-coolant accident requiring actuation of the engineered safeguards.

4. A main steam line or feedwater line break.

* A one-time inspection interval of once per 58 months is allowed for the inspection performed immediately following refueling outage RF-12.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

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

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

3. Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

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

6. Tube Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging and is equal to 40% of the nominal tube wall thickness.

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.

8. Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

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

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit) required by Table 4.4-2.

4.4.5.5 Reports

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to Specification 6.9.2.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

2. Location and percent of wall-thickness penetration for each indication of an imperfection.

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to 10 CFR 50.72(b)2(i) prior to resumption of plant operation. A report pursuant to 10 CFR 50.73(a)2(ii) shall be submitted to provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Number of Steam Generators per Unit	Three
First Inservice Inspection	Two
Second and Subsequent Inservice Inspections	One*

*The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections may be limited to one steam generator on a rotating schedule encompassing 9% of the tubes if the results of the previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.4-2

STEAM GEN TOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S. G.	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S. G.	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S. G., plug or repair defective tubes and inspect 2S tubes in each other S. G. Prompt notification to NRC pursuant to 10 CFR 50.72 (b)2(ii) and 10 CFR 50.73 (a) 2(ii)	All other S. G.s are C-1	None	N/A	N/A
			Some S. G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug or repair defective tubes. Prompt notification to NRC pursuant to 10 CFR 50.72 (b)2(ii) and 10 CFR 50.73(a)2(ii)	N/A	N/A

S = $3\frac{N}{n}$% Where N is the number of steam generators in the unit, and n is the number of steam generators inspected during an inspection

The sample size, S, will also include 3% of the total number of sleeved tubes in all 3 steam generators or all of the sleeved tubes in the generator chosen for the inspection, whichever is less.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM ~~UNIDENTIFIED LEAKAGE~~,

c. 150 gallons per day ~~through any one steam generator not isolated from the Reactor Coolant System,~~ primary-to-secondary leakage through any one steam generator

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 33 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. The leakage rate specified for each Reactor Coolant System Pressure Isolation Valve in Table 3.4-1 at a Reactor Coolant System pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, or with primary-to-secondary leakage, not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or CONTROLLED LEAKAGE greater than the above limits, ~~With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and Leakage from Reactor Coolant System Pressure Isolation Valves~~, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve Leakage greater than the limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 The Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the reactor building atmosphere (gaseous or particulate) radioactivity monitor at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the reactor building sump inventory at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours.(1) This requirement is not applicable to primary-to-secondary leakage.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit.

a. During startup following each refueling outage.

b. Prior to returning the valve to service following maintenance repair or replacement work on the valve.

c. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve for valves denoted on Table 3.4-1 by an asterisk*.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

4.4.6.2.3 Primary-to-secondary leakage shall be verified ≤ 150 gallons per day through any one steam generator at least once per 72 hours (1).

(1) Not required to be performed/completed until 12 hours after establishment of steady state operation.

i. Technical Specifications (TS) Bases Control Program

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

1. Changes to the Bases shall be made under appropriate administrative control and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   a) A change in the TS incorporated in the license or

   b) A change to the updated FSAR or bases that requires NRC approval pursuant to 10 CFR 50.59.

3. The Bases Control Program shall contain provisions to insure that the Bases are maintained consistent with the FSAR.

4. Proposed changes that meet the criteria of Specification 6.8.4.i.2.b above shall be reviewed and approved 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).

j. Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Positions C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

INSERT k. Steam Generator Program

INSERT 6.8.4.K

k. Steam Generator Program

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

1. 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 a 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 or repaired to confirm that the performance criteria are being met.

2. Performance criteria for SG tube integrity. Steam generator tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

   a) Structural integrity performance criterion. All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, HOT STANDBY, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 (3deltaP) 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.

   b) 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. Accident induced leakage is not to exceed 1 gpm total for all three SGs.

   c) The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

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

4. 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 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 4a, 4b, and 4c 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.

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

   b) Inspect 100% of the tubes at sequential periods of 144, 108, 72 and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

   c) 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.

5. Provisions for monitoring operational primary-to-secondary leakage.

INSERT 6.9.1.12

Steam Generator Tube Inspection Report

6.9.1.12 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of a Steam Generator tube inspection performed in accordance with Specification 6.8.4.k. 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 or repaired 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.

ATTACHMENT III

PROPOSED TECHNICAL SPECIFICATION CHANGES (RETYPED)

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.11 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant.

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

FREQUENCY NOTATION

1.13 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

GASEOUS RADWASTE TREATMENT SYSTEM

1.14 A GASEOUS RADWASTE TREATMENT SYSTEM is any system designed and installed to reduce radioactive gaseous effluents by collecting primary coolant system offgases from the primary system and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. 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

c. Reactor coolant system leakage through a steam generator to the secondary system (primary-to-secondary leakage).

MASTER RELAY TEST

1.16 A MASTER RELAY TEST shall be the energization of each master relay and verification of OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.19 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.1.

PHYSICS TESTS

1.20 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

1.21 PRESSURE BOUNDARY LEAKAGE shall be leakage (except primary-to-secondary leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, tests, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Steam generator tube integrity shall be maintained.

AND

All steam generator tubes satisfying the tube repair criteria shall be plugged or repaired in accordance with the Steam Generator Program.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

The ACTIONS may be entered separately for each steam generator tube.

a. With one or more steam generator tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program,

   1. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or steam generator tube inspection, or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and

   2. Plug or repair the affected tube(s) in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following the next refueling outage or steam generator tube inspection.

b. With steam generator tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify steam generator tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected steam generator tube that satisfies the tube repair criteria is plugged or repaired in accordance with the Steam Generator Program prior to entering HOT SHUTDOWN following a steam generator tube inspection.

The following pages were deleted:

3/4 4-12
3/4 4-13
3/4 4-14
3/4 4-15
3/4 4-16
3/4 4-17

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day primary-to-secondary leakage through any one steam generator,

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and

e. 33 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.

f. The leakage rate specified for each Reactor Coolant System Pressure Isolation Valve in Table 3.4-1 at a Reactor Coolant System pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE or with primary-to-secondary leakage, not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or CONTROLLED LEAKAGE greater than the above limits, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve Leakage greater than the limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 The Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:

a. Monitoring the reactor building atmosphere (gaseous or particulate) radioactivity monitor at least once per 12 hours.

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the reactor building sump inventory at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours[1]. This requirement is not applicable to primary-to-secondary leakage and controlled leakage.

e. Monitoring the reactor head flange leakoff system at least once per 24 hours.

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit.

a. During startup following each refueling outage.

b. Prior to returning the valve to service following maintenance repair or replacement work on the valve.

c. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve for valves denoted on Table 3.4-1 by an asterisk*.

d. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

4.4.6.2.3 Primary-to-Secondary leakage shall be verified ≤ 150 gallons per day through any one steam generator at least once per 72 hours[1].

(1) Not required to be performed/completed until 12 hours after establishment of steady state operation.

i. Technical Specifications (TS) Bases Control Program

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

1. Changes to the Bases shall be made under appropriate administrative control and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   a) A change in the TS incorporated in the license or

   b) A change to the updated FSAR or bases that requires NRC approval pursuant to 10 CFR 50.59.

3. The Bases Control Program shall contain provisions to insure that the Bases are maintained consistent with the FSAR.

4. Proposed changes that meet the criteria of Specification 6.8.4.i.2.b above shall be reviewed and approved 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).

j. Reactor Coolant Pump Flywheel Inspection Program

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

In lieu of Positions C.4.b(1) and C.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and/or PT) of exposed surfaces of the removed flywheels may be conducted at 20 year intervals.

k. Steam Generator Program

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

1. 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 a 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 or repaired to confirm that the performance criteria are being met.

2. Performance criteria for SG tube integrity. Steam generator tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

   a) Structural integrity performance criterion. All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, HOT STANDBY, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 (3deltaP) 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.

   b) 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. Accident induced leakage is not to exceed 1 gpm total for all three SGs.

   c) The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

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

4. 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 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 4.a, 4.b, and 4.c 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.

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

b) Inspect 100% of the tubes at sequential periods of 144, 108, 72 and thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

c) 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.

5. Provisions for monitoring operational primary-to-secondary leakage.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.12 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of a Steam Generator tube inspection performed in accordance with Specification 6.8.4.k. 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 or repaired 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.

ATTACHMENT IV

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES (MARK-UP)

Insert 3/4.4.5 Steam Generator Tube Integrity Bases

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage-type degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to 10CFR50.72(b)2(i) prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

TECHNICAL SPECIFICATION BASES
Insert 3/4.4.5

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

Background

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary-to-secondary heat exchangers. The SG tubes have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.1.1, "Reactor Coolant System, Reactor Coolant Loops and Coolant Circulation, Startup and Power Operation," LCO 3.4.1.2, "Reactor Coolant System, Hot Standby," LCO 3.4.1.3, "Reactor Coolant System, Hot Shutdown," and LCO 3.4.1.4.1, "Reactor Coolant System, Cold Shutdown-Loops Filled."

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

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

Specification 6.8.4.k, "Steam Generator Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.4.k, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. The SG performance criteria are described in Specification 6.8.4.k. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

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

Applicable Safety Analyses

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

primary-to-secondary leakage is assumed to occur in the intact steam generators where it can be released during the subsequent cooldown of the plant.

The analyses for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 1 gpm, or is assumed to increase to 1 gpm as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be greater than or equal to the limits in LCO 3.4.8, "Reactor Coolant System, Specific Activity." For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Reference 2), 10 CFR 100 (Reference 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

Limiting Condition for Operation (LCO)

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

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

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

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.k and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given

structure, collapse occurs at the top of the load verses displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Reference 4) and Draft Regulatory Guide 1.121 (Reference 5).

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

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2 and limits primary-to-secondary leakage through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

Applicability

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

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

Actions

The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the required ACTIONS provide appropriate compensatory actions for each affected SG tube. Complying with the required

ACTIONS may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated required ACTIONS.

a. The Condition applies if it is discovered that one or more SG tubes examined in an Inservice Inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by Surveillance Requirement 4.4.5.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, LCO 3.4.5 Action b. applies.

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

   If the evaluation determines that the affected tube(s) have tube integrity, the ACTION statement allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This completion time is acceptable since operation until the next inspection is supported by the operational assessment.

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

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

Surveillance Requirements (SR)

4.4.5.1 During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Reference 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

A condition monitoring assessment of the SG tubes is performed during SG inspections. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

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

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

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

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

References

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

insert New 3/4.4.6.2 Operational Leakage

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

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

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

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

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

OPERATIONAL LEAKAGE (Continued)

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

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

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

The maximum allowed steam generator tube leakage of 450 GPD (3 steam generators with 150 GPD each) for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 150 GPD per steam generator limit preserves the assumptions used in the analysis of these accidents and ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.7 CHEMISTRY

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

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

TECHNICAL SPECIFICATION BASES
Insert 3/4.4.6.2

3/4.4.6.2 OPERATIONAL LEAKAGE

Background

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

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

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

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

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leak tight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

Applicable Safety Analyses

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary-to-secondary leakage from all steam generators is 1 gpm or increases to 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary-to-secondary leakage through any one steam generator to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

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

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

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

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

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

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

b. UNIDENTIFED LEAKAGE

One gallon per minute (gpm) of UNIDENTIFED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the leakage is from the pressure boundary.

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

The limit of 150 gallons per day (gpd) per steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Reference 1). The Steam Generator Program operational leakage performance criterion

in NEI 97-06 states, "The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gpd." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFED LEAKAGE and is well within the capability of the Reactor Coolant System Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or CONTROLLED LEAKAGE. Violation of this LCO could result in continued degradation of a component or system.

e. CONTROLLED LEAKAGE

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

f. Reactor Coolant System Pressure Isolation Valve Leakage

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

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

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low-pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting system

are degraded or degrading. Excessive PIV leakage could lead to overpressure of the low-pressure piping or components, potentially resulting in a loss of coolant accident (LOCA) outside of containment.

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

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

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

Applicability

In MODES 1, 2, 3, and 4, the potential for RCPB leakage is greatest when the Reactor Coolant System is pressurized.

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

Actions

a. If any PRESSURE BOUNDARY LEAKAGE exists, or primary-to-secondary leakage is not within limit, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within the next 30 hours. This ACTION reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

   The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

b. UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or CONTROLLED LEAKAGE in excess of the LCO limits must be reduced to within the limits within 4 hours. This allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. This ACTION is necessary to prevent further deterioration of the RCPB.

c. With PIV leakage in excess of the limit, the high pressure portion of the affected system must be isolated within 4 hours, or be in at least hot standby within the next 6 hours, and cold shutdown within the following 30 hours. This ACTION is necessary to prevent over pressurization of low pressure systems, and the potential for intersystem LOCA.

Surveillance Requirements

4.4.6.2.1 Verifying Reactor Coolant System leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained.

PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of a Reactor Coolant System water inventory balance.

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

For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows.

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity and containment sump level. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Reactor Coolant System, Leakage Detection Systems."

Part (d) notes that this SR is not applicable to primary-to-secondary leakage because leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72-hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

4.4.6.2.2 This Surveillance Requirement verifies RCS Pressure Isolation Valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

4.4.6.2.3 This Surveillance Requirement verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one steam generator. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5 should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 2. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator.

The Surveillance Requirement is modified by a note, which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For Reactor Coolant System primary-to-secondary leakage determination, steady state is defined as stable Reactor Coolant System pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows.

The frequency of 72 hours is a reasonable interval to trend primary-to-secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Reference 2).

References

1. NEI 97-06, "Steam Generator Program Guidelines"

2. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines"

ATTACHMENT V

PROPOSED TECHNICAL SPECIFICATION BASES CHANGES (RETYPED)

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

Background

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary-to-secondary heat exchangers. The SG tubes have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.1.1, "Reactor Coolant System, Reactor Coolant Loops and Coolant Circulation, Startup and Power Operation," LCO 3.4.1.2, "Reactor Coolant System, Hot Standby," LCO 3.4.1.3, "Reactor Coolant System, Hot Shutdown," and LCO 3.4.1.4.1, "Reactor Coolant System, Cold Shutdown-Loops Filled."

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

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

Specification 6.8.4.k, "Steam Generator Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.4.k, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. The SG performance criteria are described in Specification 6.8.4.k. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

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

STEAM GENERATOR TUBE INTEGRITY (Continued)

Applicable Safety Analyses

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

The analyses for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 1 gpm, or is assumed to increase to 1 gpm as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be greater than or equal to the limits in LCO 3.4.8, "Reactor Coolant System, Specific Activity." For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Reference 2), 10 CFR 100 (Reference 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

Limiting Condition for Operation (LCO)

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

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

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

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.k and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

STEAM GENERATOR TUBE INTEGRITY (Continued)

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

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

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Reference 4) and Draft Regulatory Guide 1.121 (Reference 5).

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

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2 and limits primary-to-secondary leakage through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

STEAM GENERATOR TUBE INTEGRITY (Continued)

Applicability

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

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

Actions

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

a. The Condition applies if it is discovered that one or more SG tubes examined in an Inservice Inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by Surveillance Requirement 4.4.5.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, LCO 3.4.5 Action b. applies.

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

   If the evaluation determines that the affected tube(s) have tube integrity, the ACTION statement allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This completion time is acceptable since operation until the next inspection is supported by the operational assessment.

STEAM GENERATOR TUBE INTEGRITY (Continued)

ACTIONS (Continued)

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

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

Surveillance Requirements (SR)

4.4.5.1 During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Reference 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

A condition monitoring assessment of the SG tubes is performed during SG inspections. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

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

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

STEAM GENERATOR TUBE INTEGRITY (Continued)

Surveillance Requirements (Continued)

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

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

References

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Background

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

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

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

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

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leak tight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

OPERATIONAL LEAKAGE (Continued)

Applicable Safety Analyses

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary-to-secondary leakage from all steam generators is 1 gpm or increases to 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary-to-secondary leakage through any one steam generator to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

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

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

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

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

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

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

OPERATIONAL LEAKAGE (Continued)

b. UNIDENTIFED LEAKAGE

One gallon per minute (gpm) of UNIDENTIFED LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the Reactor Coolant Pressure Boundary, if the leakage is from the pressure boundary.

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

The limit of 150 gallons per day (gpd) per steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Reference 1). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gpd." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFED LEAKAGE and is well with the capability of the Reactor Coolant System Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or CONTROLLED LEAKAGE. Violation of this LCO could result in continued degradation of a component or system.

e. CONTROLLED LEAKAGE

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

f. Reactor Coolant System Pressure Isolation Valve Leakage

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

OPERATIONAL LEAKAGE (Continued)

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

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

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

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

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

Applicability

In MODES 1, 2, 3, and 4, the potential for Reactor Coolant Pressure Boundary leakage is greatest when the Reactor Coolant System is pressurized.

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

OPERATIONAL LEAKAGE (Continued)

Actions

a. If any PRESSURE BOUNDARY LEAKAGE exists, or primary-to-secondary leakage is not within limit, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within the next 30 hours. This ACTION reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

   The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the Reactor Coolant Pressure Boundary are much lower, and further deterioration is much less likely.

b. UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or CONTROLLED LEAKAGE in excess of the LCO limits must be reduced to within the limits within 4 hours. This allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. This ACTION is necessary to prevent further deterioration of the Reactor Coolant Pressure Boundary.

c. With PIV leakage in excess of the limit, the high pressure portion of the affected system must be isolated within 4 hours, or be in at least hot standby within the next 6 hours, and cold shutdown within the following 30 hours. This ACTION is necessary to prevent over pressurization of low pressure systems, and the potential for intersystem LOCA.

Surveillance Requirements

4.4.6.2.1 Verifying Reactor Coolant System leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained.

PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of a Reactor Coolant System water inventory balance.

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

For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and Reactor Coolant Pump seal injection and return flows.

OPERATIONAL LEAKAGE (Continued)

Surveillance Requirements (Continued)

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity and containment sump level. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Reactor Coolant System, Leakage Detection Systems."

Part (d) notes that this SR is not applicable to primary-to-secondary leakage because leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72-hour frequency is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.

4.4.6.2.2 This Surveillance Requirement verifies RCS Pressure Isolation Valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA.

4.4.6.2.3 This Surveillance Requirement verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one steam generator. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5 should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 2. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator.

The Surveillance Requirement is modified by a note, which states that the Surveillance is not required to be performed until 12 hours after establishment of steady state operation. For Reactor Coolant System primary-to-secondary leakage determination, steady state is defined as stable Reactor Coolant System pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows.

The frequency of 72 hours is a reasonable interval to trend primary-to-secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary-to-secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Reference 2).

References

1. NEI 97-06, "Steam Generator Program Guidelines"

2. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines"

3/4.4.7 CHEMISTRY

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

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

ATTACHMENT VI

LIST OF REGULATORY COMMITMENTS

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

SCE&G has identified the development of a Steam Generator Program to adopt the provisions of TSTF-449 in TS 6.8.4. There are no additional regulatory commitments created due to this License Amendment Request.