FPL Energy
Seabrook Station

FPL Energy Seabrook Station
P.O. Box 300
Seabrook, NH 03874
(603) 773-7000

March 23, 2006

SBK-L-06064
Docket No. 50-443

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Seabrook Station
License Amendment Request 06-02

"Application for Technical Specification Improvement Regarding Steam Generator Integrity Using the Consolidated Line Item Improvement Process"

In accordance with the provisions of Section 50.90 of Title 10 of the *Code of Federal Regulations* (10 CFR), FPL Energy Seabrook, LLC (FPL Energy Seabrook) is submitting License Amendment Request (LAR) 06-02 for an amendment to the technical specifications (TS) for Seabrook Station.

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).

The enclosure contains FPL Energy Seabrook's evaluation of the proposed change, including a description of the proposed change and confirmation of applicability. Attachment 1 provides a mark-up of the technical specification (TS) pages showing the proposed changes, and Attachment 2 contains markups showing proposed changes to the TS bases. The retyped TS pages will be provided at a later date upon request from the NRC.

A copy of this letter and the enclosed LAR have been forwarded to the New Hampshire State Liaison Officer pursuant to 10 CFR 50.91(b). The Station Operation Review Committee and the Company Nuclear Review Board have reviewed this LAR.

FPL Energy Seabrook requests NRC Staff review and approval of LAR 06-02 with issuance of a license amendment by March 30, 2007 and implementation of the amendment within 90 days.

A001

Should you have any questions regarding this letter, please contact Mr. James M. Peschel, Regulatory Programs Manager, at (603) 773-7194.

Very truly yours,

FPL Energy Seabrook, LLC.

Gene F. St. Pierre
Site Vice President

Enclosures:
Notarized Affidavit
FPL Energy Seabrook's Evaluation of the Proposed Change

Attachments:
1. Proposed Technical Specification Changes (markup)
2. Proposed Technical Specification Bases Changes (markup)

cc: S. J. Collins, NRC Region I Administrator
G. E. Miller, NRC Project Manager, Project Directorate I-2
G.T. Dentel, NRC Senior Resident Inspector

Mr. Bruce G. Cheney, ENP, Director, Division of Emergency Services
N.H. Department of Safety
Division of Emergency Services, Communications, and Management
Bureau of Emergency Management
33 Hazen Drive
Concord, NH 03305

FPL Energy

Seabrook Station

SEABROOK STATION UNIT 1

Facility Operating License NPF-86

Docket No. 50-443

License Amendment Request 06-02

"Application for Technical Specification Improvement Regarding Steam Generator Integrity Using the Consolidated Line Item Improvement Process"

The following information is enclosed in support of this License Amendment Request:

- **Enclosure** — **FPL Energy Seabrook's Evaluation of the Proposed Change**
- **Attachment 1** — **Proposed Technical Specification Changes (markup)**
- **Attachment 2** — **Proposed Technical Specification Bases Changes (markup)**

I, Gene St. Pierre, Site Vice President of FPL Energy Seabrook, LLC hereby affirm that the information and statements contained within this License Amendment Request are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

Sworn and Subscribed
before me this
23rd day of March, 2006

Shirley Sweeney
Notary Public

SHIRLEY SWEENEY
MY COMMISSION EXPIRES OCT. 6, 2009
NOTARY PUBLIC
NEW HAMPSHIRE

Gene St. Pierre
Gene St. Pierre
Site Vice President

FPL ENERGY SEABROOK'S EVALUATION

Subject: License Amendment Request 06-02, "Application for Technical Specification Improvement Regarding Steam Generator Integrity Using the Consolidated Line Item Improvement Process"

1.0 INTRODUCTION

2.0 PROPOSED CHANGE

3.0 BACKGROUND

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

5.0 TECHNICAL ANALYSIS

6.0 REGULATORY ANALYSIS

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

8.0 ENVIRONMENTAL EVALUATION

9.0 PRECEDENT

10.0 REFERENCES

1.0 INTRODUCTION

The proposed license amendment (LAR) revises the requirements in the 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 (70 FR 24126)] as part of the consolidated line item improvement process (CLIIP).

2.0 DESCRIPTION OF PROPOSED AMENDMENT

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

- Revised TS definition of IDENTIFIED LEAKAGE
- Revised TS 3.4.6.2, "Reactor Coolant System Operational Leakage"
- Revised TS 3.4.5, "Steam Generator (SG) Tube Integrity"
- New TS 6.7.6.k, "Steam Generator (SG) Program"
- New TS 6.8.1.7, "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.

3.0 BACKGROUND

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

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

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

5.0 TECHNICAL ANALYSIS

FPL Energy Seabrook has reviewed the safety evaluation (SE) published on March 2, 2005 (70 FR10298) 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. FPL Energy Seabrook has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to Seabrook Station and justify this amendment for the incorporation of the changes to the Seabrook Station 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	Seabrook Station Unit 1
Steam Generator Models	Westinghouse Model F
Effective Full Power Years (EFPY) of service for currently installed SGs	12.40 EFPY (through the end of refueling outage 10 in April 2005)
Tubing Material	Alloy 600TT
Number of tubes per SG	5626
Number and percentage of tubes plugged in each SG	SG-A: 31, 0.55% SG-B: 23, 0.41% SG-C: 28, 0.50% SG-D: 58, 0.89%
Number of tubes repaired in each SG	None
Degradation mechanism(s) identified	1. Anti-vibration Bar (AVB) wear 2. Outside diameter stress corrosion cracking (ODSCC)
Current primary to secondary leakage limits	• Per SG: 500 gallons per day • Total: 1.0 gallon per minute • Leakage is evaluated at a density of 1.0 gm/cc (cold liquid).

Approved Alternate Tube Repair Criteria	NONE
Approved SG Tube Repair Methods	NONE
Performance criteria for accident leakage	• 1.0 gallon per minute total leakage • 500 gallons per day though any one SG • The primary-to-secondary leak rate is based on a density of 1.0 gm/cc (cold liquid).

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

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

8.0 ENVIRONMENTAL EVALUATION

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

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. FPL Energy Seabrook is not proposing significant 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). However, this LAR also revises a Surveillance Requirement (SR) for TS 3.4.6.2., Reactor Coolant System Operational Leakage, to be consistent with NUREG-1431, Improved Standard Technical Specifications (ISTS), revision 3. Current SR 4.4.6.2.1.requires:

> 4.4.6.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:
>
> d. Performance of a Reactor Coolant System water inventory balance within 12 hours after achieving steady-state operation* and at least once per 72 hours thereafter during steady-state operation, except that not more than 96 hours shall elapse between any two successive inventory balances; and

A footnote defines steady-state operation: "*T_{avg} being changed by less than 5°F/hour."

With this proposed change, SR 4.4.6.2.1.d is revised to state:

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady-state operation, except that not more than 96 hours shall elapse between any two successive inventory balances; (1) (2)

(1) Not applicable to primary to secondary leakage.

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

This revision deletes the current footenote (*). T_{avg} changing at a rate less than 5°F/hour does not accurately define the steady-state conditions required for performing a RCS water inventory balance. In lieu of the footnote, the proposed bases for this SR discuss the steady-state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) that must exist to perform this surveillance. In addition, the proposed change clarifies the frequency of this SR by adding footnote 2, which states the SR is not required to be performed until 12 hours after establishing steady-state operation. The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established. The proposed changes to SR 4.4.6.2.1.d are consistent with ISTS SR 3.4.13.1 concerning the RCS water inventory balance.

10.0 REFERENCES

Federal Register Notices:

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

Attachment 1

Proposed Technical Specification Change (mark-up)

Refer to the attached markup of the proposed changes to the Technical Specifications. The attached markup reflects the currently issued version of the Technical Specifications. At the time of submittal, the Technical Specifications were revised through Amendment No. 106. Pending Technical Specifications or Technical Specification changes issued subsequent to this submittal are not reflected in the enclosed markup.

Listed below are the license amendment requests that are awaiting NRC approval and may impact the currently issued version of the Technical Specifications affected by this LAR.

LAR	Title	FPL Energy Seabrook Letter	Date Submitted
05-04	Measurement Uncertainty Recapture Power Uprate	SBK-L-05205	9-22-2005
05-08	Limited Inspection of the Steam Generator Tube Portion Within the Tube Sheet	SBK-L-05185	9-29-2005
06-03	Amendment to the Technical Specifications for Miscellaneous Changes	SBK-L-06059	

The following Technical Specifications are included in the attached markup:

Technical Specification	Title	Page	
N/A	Index	v	
		xiv	
		x	
Definitions	Unidentified Leakage	1-4	
	Pressure Boundary Leakage	1-5	
TS 3/4.4.5	Steam Generators	3/4 4-13	3/4 4-17
		3/4 4-14	3/4 4-18
		3/4 4-15	3/4 4-19
		3/4 4-16	
TS 3.4.6.2	Operational Leakage	3/4 4-21	
		3/4 4-22	
TS 6.7.6	Procedures and Programs	6-11	
TS 6.8	Reporting Requirements	6-19	

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

6.0 ADMINISTRATIVE CONTROLS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

This page contains no changes and is only provided for information

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same TEDE dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed under inhalation in Federal Guidance Report No. 11 (FGR 11), "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1989.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.13 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the sample) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample with half-lives greater than 10 minutes.

ENGINEERED SAFETY FEATURES (ESF) RESPONSE TIME

1.14 The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its 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 methodology for verification have been previously reviewed and approved by the NRC.

FREQUENCY NOTATION

1.15 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.1.

GASEOUS RADWASTE TREATMENT SYSTEM

1.16 A GASEOUS RADWASTE TREATMENT SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System offgases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.17 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 Coolant System. (primary to secondary leakage)

MASTER RELAY TEST

1.18 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.

MEMBER(S) OF THE PUBLIC

1.19 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

1.20 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 6.7.6 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 6.8.1.3 and 6.8.1.4.

OPERABLE - OPERABILITY

1.21 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

1.22 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 1.2.

PHYSICS TESTS

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

(primary to secondary)

PRESSURE BOUNDARY LEAKAGE

1.24 PRESSURE BOUNDARY LEAKAGE shall be leakage (except ~~steam generator tube~~ leakage) through a nonisolable fault in a Reactor Coolant System component body, pipe wall, or vessel wall.

PROCESS CONTROL PROGRAM

1.25 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.

PURGE - PURGING

1.26 PURGE or PURGING shall be any controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

QUADRANT POWER TILT RATIO

1.27 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.28 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3587 Mwt.

REACTOR TRIP SYSTEM (RTS) RESPONSE TIME

1.29 The RTS RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its RTS Trip Setpoint at the channel sensor until loss of stationary gripper coil voltage. 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 methodology for verification have been previously reviewed and approved by the NRC.

INSERT TS 3/4.4.5

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 TS 3/4.4.5

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

ACTION:

........................NOTE..

Separate action entry is allowed for each steam generator tube.

....

a. With one or more steam generator tubes satisfying the tube repair criteria and not plugged 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 SG tube inspection, or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours, and

2. Plug 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 SG tube integrity in accordance with the Steam Generator Program.

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

SURVEILLANCE REQUIREMENTS

4.4.5.2b. (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, and

3) A tube inspection (pursuant to Specification 4.4.5.4a.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, and

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.

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS

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 shall be performed after 6 Effective Full-Power Months but no later than restart after first refueling. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, *not including the preservice inspection, result in all inspection results falling in* Category C-1 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 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and

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 tubes leak (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of Specification 3.4.6.2, or

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

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

4) A main steam line or feedwater line break.

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS

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 the 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) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

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 Specification 4.4.5.3c., 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; and

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 prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

STEAM GENERATORS

SURVEILLANCE REQUIRMENTS

4.4.5.4 (Continued)

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 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 6.8.2;

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.8.2 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, and

3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report to the Commission pursuant to Specification 6.8.2 within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

No. of Steam Generators per Unit	Four
Preservice Inspection	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One (1)

TABLE NOTATION

(1) The third and fourth steam generators that were not inspected during the first inservice inspection shall be inspected during the second and third inspections, respectively. For the fourth and subsequent inspections, the inservice inspection may be limited to one steam generator on a rotating schedule encompassing 12% of the tubes if the results of the previous inspections of the four steam generators 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 GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result for 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 defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to §50.72 (b)(3) of 10CFR Part 50	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. in C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to §50.72 (b)(3) of 10CFR Part 50	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.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:
operational

a. No PRESSURE BOUNDARY LEAKAGE,

150 gallons per day primary to secondary leakage

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. ~~1 gpm total reactor-to-secondary leakage through all steam generators and 500 gallons per day~~ through any one steam generator,
(SG)

d. 10 gpm IDENTIFIED LEAKAGE from the Reactor Coolant System,

e. 40 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 psig ± 20 psig, and

f. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve.*

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

ACTION:
UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or CONTROLLED LEAKAGE
or with primary to secondary leakage not within the limit,

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. 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 above 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.

---

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.6.2.1 Reactor Coolant System operational leakage(s) shall be demonstrated to be within each of the above limits by:

a. Monitoring the containment atmosphere particulate radioactivity monitor at least once per 12 hours;

b. Monitoring the containment drainage sump inventory and discharge 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 ~~within 12 hours after achieving steady-state operation* and~~ at least once per 72 hours ~~thereafter~~ during steady-state operation, except that not more than 96 hours shall elapse between any two successive inventory balances; ~~and~~ (1) (2)

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours(2), and

f. Verifying primary to secondary leakage is ≤ 150 gallons per day through any one SG at least once per 72 hours(2)

(1) Not applicable to primary to secondary leakage
(2) Not required to be performed until 12 hours after establishment of steady state operation

~~*T_{avg} being changed by less than 5°F/hour.~~

PAGE INTENTIONALLY BLANK

Deleted

TABLE 3.4-2

(THIS TABLE NUMBER IS NOT USED)

Deleted

PROCEDURES AND PROGRAMS

6.7.6 (Continued)

j. Technical Specification (TS) Bases Control Program

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

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

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

1. A change in the TS incorporated in the license or

2. A change to the updated FSAR (UFSAR) or Bases that requires NRC approval pursuant to 10 CFR 50.59.

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

d. Proposed changes that meet the criteria of Specification 6.7.6j.b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

INSERT 6.7.6.K

K.

6.8 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.8.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.8.1.1 A summary report of station startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the station.

INSERT 6.7.6.k

k. Steam Generator Program

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

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

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

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm total or 500 gpd through any one SG.

   3. The operational LEAKAGE performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Leakage."

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

(INSERT 6.7.6.k (con't)

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

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

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

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

e. Provisions for monitoring operational primary to secondary leakage.

SPECIAL REPORTS INSERT 6.8.1.7

6.8.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attn: Document Control Desk, with a copy to the NRC Regional Administrator within the time period specified for each report.

6.9 (THIS SPECIFICATION NUMBER IS NOT USED)

6.10 RADIATION PROTECTION PROGRAM

6.10.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.11 HIGH RADIATION AREA

6.11.1 Pursuant to paragraph 20.1601(c) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) and (b), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less-than 1000 mR/h at 30 cm (12 in.) from the radiation source or from any surface that the radiation penetrates shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

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

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

c. An individual qualified in radiation protection procedures with a radiation dose rate monitoring device, who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified in the Radiation Work Permit.

INSERT 6.8.1.7

STEAM GENERATOR TUBE INSPECTION REPORT

6.8.1.7 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.7.6.k, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

d. Location, orientation (if linear), and measured sizes (if available) of service induced indications,

e. Number of tubes plugged during the inspection outage for each active degradation mechanism,

f. Total number and percentage of tubes plugged to date,

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

h. The effective plugging percentage for all plugging in each SG.

Attachment 2

Proposed Technical Specifications Bases Changes (markups)

The following TS bases are included in this attachment:

Technical Specification	Title
Bases 3/4.4.1	Reactor Coolant Loops and Coolant Circulation
Bases 3/4.4.4	Relief Valves
Bases 3/4.4.5	Steam Generators
Bases 3/4.4.6.2	Operational Leakage

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

~~A reactor coolant loop is comprised of its associated steam generator and reactor coolant pump.~~ An OPERABLE reactor coolant system loop consists of an OPERABLE reactor coolant pump and an OPERABLE steam generator ~~in accordance with the Steam Generator Tube Surveillance Program~~.

The plant is designed to operate with all reactor coolant loops in operation and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 6 hours.

In MODE 3, two reactor coolant loops provide sufficient heat removal capability for removing core decay heat even in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by placing the Control Rod Drive System in a condition incapable of rod withdrawal. Single failure considerations require that two loops be OPERABLE at all times.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

A Reactor Coolant "loops filled" condition is defined as follows: (1) Having pressurizer level greater than or equal to 55% if the pressurizer does not have a bubble, and greater than or equal to 17% when there is a bubble in the pressurizer. (2) Having the air and non-condensables evacuated from the Reactor Coolant System by either operating each reactor coolant pump for a short duration to sweep air from the Steam Generator U-tubes into the upper head area of the reactor vessel, or removing the air from the Reactor Coolant System via an RCS evacuation skid, and (3) Having vented the upper head area of the reactor vessel if the pressurizer does not have a bubble. (4) Having the Reactor Coolant System not vented, or if vented capable of isolating the vent paths within the time to boil. 4

The operation of one reactor coolant pump (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP in MODES 4 and 5 are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant System, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold-leg temperatures.

3/4.4.4 RELIEF VALVES (Continued)

The PORVs are equipped with automatic actuation circuitry and manual control capability. The PORVs are considered OPERABLE in either the automatic or manual mode for the following reasons:

(1) No credit is taken in any FSAR accident analysis for automatic PORV actuation to mitigate the consequences of an accident.

(2) No Surveillance Requirement (ACOT or TADOT) exists for verifying automatic operation.

(3) The required ACTION for an inoperable PORV(s) (closing the block valve) conflicts with any presumed requirement for automatic actuation.

INSERT B3/4.4.5

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.~~

STEAM GENERATOR (SG) 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. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG.

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

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

Specification 6.7.6.k, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.7.6.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 6.7.6.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 (Ref. 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. In the analysis of a SGTR, the primary-to-secondary leak rate is apportioned between the SGs (1.0 gpm total, 500 gpd to any one SG). The tube leakage is conservatively apportioned as 313.33 gpd to the faulted SG and 1126.67 gpd total to the other three SGs in order to maximize dose consequences. The analysis assumes the leakage rate associated with the instantaneous rupture of a SG tube that relieves to the lower pressure secondary system. The analysis assumes the contaminated fluid is released to the atmosphere through the main steam safety valves or the atmospheric steam dump valves. The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In the analyses of the dose consequences for these events, the activity level in the steam discharged to the atmosphere is based on a conservative value for the total primary-to-secondary leakage which bounds the operational leakage rate as an initial condition and considers any leakage changes as a result of the accident induced changes in primary-to-secondary pressure differential. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.8, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2) and 10 CFR 50.67 (Ref. 3). The LCO limit of 150 gpd primary to secondary leakage through any one SG is significantly less than the initial conditions assumed in the dose consequence analysis. Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program. During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. 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 6.7.6.k, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria. There are three SG performance criteria: structural integrity, accident-induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

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

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

Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary-to secondary leakage caused by any changes in primary-to-secondary pressure differential during a design basis accident other than SGTR, is considered in the accident dose consequences analysis. The dose consequence analyses assumes that the accident-induced leakage does not exceed 500 gpd in any SG and that the total accident leakage does not exceed 1 gpm. This accident induced leakage rate conservatively bounds the expected total accident primary-to-secondary leakage and considers any leakage changes as a result of the accident induced changes in primary-to-secondary pressure differential.

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, "RCS Operational leakage," and limits primary to secondary leakage through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to 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.

SG tube structural and leakage integrity is assessed at each SG inspection in accordance with FPL Steam Generator Integrity Program procedures. These assessments support the statement that accident induced leakage is bounded by the leakage rate assumed in the accident analysis. The analysis for the limiting depressurization event, steam line break, indicates that there is little if any increase in primary to secondary differential pressure, and that any such increase would be limited in duration and magnitude such that any increase in leakage would be inconsequential to the dose consequences calculated for this event. Furthermore, use of the expected pressure differential profile over the 24-hour term of the accident would result in a reduction of the integrated leakage and resultant dose consequences. Additionally, since the steam line break event results in rapid RCS cool down and depressurization, a combination of high primary-side pressure and a depressurized secondary system ("high-dry condition) leading to increased heating of the leaking tube will not occur. Therefore, the Seabrook Station accident analysis assumption of a constant primary-to-secondary leak at the assumed primary-to-secondary leak rate throughout the term of the accident 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 MODES 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 actions may be entered independently for each SG tube. This is acceptable because the actions provide appropriate compensatory actions for each affected SG tube. Complying with the actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent entry and application of associated actions.

<u>a and b</u>

Action a applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 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, Action b applies.

A completion time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Action a 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.

If SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. The shutdown 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

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 (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

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

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

The Steam Generator Program defines the Frequency of SR 4.4.5.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.7.6.k contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 4.4.5.2

During a SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.7.6.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 and Reference 7 provide 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.
3. 10 CFR 50.67
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI, "Pressurized Water Reactor Steam Generator Examination
Guidelines."
TSTF-449, Rev. 3
TSTF-449, Rev. 3
TSTF-449, Rev. 3

3/4.4.5 STEAM GENERATORS (Continued)

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 Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-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 reactor-to-secondary leakage of 500 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 the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect 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 in a Special Report pursuant to Specification 6.8.2 within 30 days and 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.

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 BASES 3.4.6.2

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.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

The safety significance of RCS leakage varies depending on the source, rate, and duration of the leak; therefore, detection and monitoring of RCS leakage are necessary. In addition, a means of separating the identified from the unidentified leakage is necessary to permit the operators to take prompt corrective action in the event of a leak that is detrimental to the safety of the facility.

Unidentified Leakage

Uncollected leakage to the containment atmosphere, which is ultimately collected in the containment drainage sumps where the leak rate can be established and monitored, is unidentified leakage. Unidentified leakage to the containment atmosphere is kept to a minimum (normal leakage is estimated to range from 20 to 40 gallons per day) to permit the leakage detection system to detect positively and rapidly a small increase in leakage. Identified leakage and unidentified leakage are separated so that a small unidentified leak will not be masked by larger, acceptable identified leakage. The one-gallon per-minute limit on unidentified leakage is a reasonable, minimum detectable amount that the leakage detection system can detect in a reasonable time period.

Identified Leakage

A limited amount of leakage is expected from auxiliary systems inside containment that cannot practically be made 100% leak tight. Identified leakage, which consists of collectable, detectable, leakage from specifically known and located sources, does not interfere with the ability of the leakage detection system to detect unidentified leakage. Identified leakage is monitored separately from unidentified leakage. Up to 10 gpm of identified leakage is acceptable because the leakage is from known sources that will not mask a small, unidentified leak and is well within the capability of the RCS make up system.

Primary to Secondary Leakage

The total steam generator tube leakage limit of 1 gpm 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 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

~~Controlled Leakage~~

~~The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 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 safety analyses.~~

~~Pressure Isolation Valve Leakage~~

~~The specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing over-pressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.~~

~~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. RCS Pressure Isolation Valve (PIV) Leakage measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS leakage when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE.~~

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 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage. Regulatory Guide 1.45 (Ref. 2) 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% leaktight. 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 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 (SGs) is one

gallon per minute. The LCO requirement to limit primary to secondary leakage through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident or a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analyses for SLB and SGTR assume one gallon per minute primary to secondary leakage. For the SLB, the tube leakage is conservatively apportioned as 500 gpd to the faulted SG and 940 gpd total to the other three SGs in order to maximize dose consequences. Similarly, the SGTR analysis assumes the tube leakage is 313. gpd to the faulted SG and 1127 gpd total to the other three SGs in order to maximize dose consequences. The dose consequences resulting from these accidents are within the limits defined in 10 CFR 50.67.

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

LCO

RCS operational leakage shall be limited to:

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.

Unidentified Leakage

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

LCO
(continued)

Identified Leakage

Up to 10 gpm of identified leakage is considered allowable because leakage is from known sources that do not interfere with detection of unidentified leakage and is well within the capability of the RCS Makeup System. Identified leakage includes leakage to the containment from specifically known and located sources, but does not include pressure boundary leakage or controlled reactor coolant pump (RCP) seal leakoff. Violation of this LCO could result in continued degradation of a component or system.

Primary to Secondary Leakage through Any One SG

The limit of 150 gallons per day per SG is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

Controlled Leakage

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 40 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 safety analyses.

Pressure Isolation Valve Leakage

The specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing over-pressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

APPLICABILITY

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

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

ACTIONS

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

If any pressure boundary leakage exists or primary to secondary leakage is not within limit; or if unidentified leakage, identified leakage, or controlled leakage cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary. The allowed completion times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In MODE 5, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

4.4.6.2.1

Verifying RCS leakage to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary leakage would at first appear as unidentified leakage and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. Unidentified leakage and identified leakage are determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The surveillance is modified by two footnotes. Footnote 1 states that this SR is not applicable to primary to secondary leakage because leakage of 150 gallons per day cannot be measured accurately by an RCS water inventory balance. Footnote 2 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12-hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Steady state operation is required to perform a proper inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary leakage or unidentified leakage is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that leakage past seals and gaskets is not pressure boundary leakage. These leakage detection systems are specified in LCO 3.4.6.1, "RCS Leakage Detection Instrumentation."

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

SR 4.4.6.2.1.f verifies that primary to secondary leakage is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.5, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational leakage rate limit applies to leakage through any one SG. If it is not practical to assign the leakage to an individual SG, the entire primary to secondary leakage should be conservatively assumed to be from one SG.

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

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary leakage and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary leakage is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 5).

<u>4.4.6.2.2</u>

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. RCS Pressure Isolation Valve (PIV) Leakage measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS leakage when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 15.
4. NEI 97-06, "Steam Generator Program Guidelines."

5. EPRI, "Pressurized Water Reactor Primary-to Secondary Leak Guidelines."