FPL Energy
Seabrook Station

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

August 16, 2006

Docket No. 50-443
SBK-L-06158

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

Seabrook Station
Facility Operating License NPF-86

Response to Request for Additional Information Regarding
License Amendment Request 06-02
"Application for Technical Specification Improvement Regarding Steam Generator Integrity
Using the Consolidated Line Item Improvement Process"

References:

1. FPL Energy Seabrook, LLC letter SBK-L-06064, License Amendment Request 06-02, Application for Technical Specification Improvement Regarding Steam Generator Integrity Using the Consolidated Line Item Improvement Process, March 23, 2006.

2. NRC letter to FPL Energy Seabrook, LLC, Draft Request for Additional Information (TAC NO. MD 0696), May 3, 2006.

By letter dated March 23, 2006, (Reference 1) FPL Energy Seabrook, LLC submitted License Amendment Request 06-02, Application for Technical Specification Improvement Regarding Steam Generator Integrity Using the Consolidated Line Item Improvement Process. In Reference 2, the NRC requested additional information in order to complete its evaluation.

Enclosed in Attachment 1 is the FPL Energy Seabrook, LLC response to the requested additional information. Attachments 2 and 3 contain revised markups of the Technical Specifications pages and proposed Bases, respectively, as modified in response to the request for additional information. The changes do not alter the conclusion discussed in Reference 1 that the proposed change does not involve a significant hazard consideration pursuant to 10 CFR 50.92. A copy of this letter has 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 the proposed change to Technical Specification 3.4.6.2, Reactor Coolant System Operational Leakage.

A001

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

Very truly yours,

FPL Energy Seabrook, LLC

Gene St. Pierre

Gene St. Pierre
Site Vice President

Enclosure

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

OATH AND AFFIRMATION

I, Gene St. Pierre, Site Vice President of FPL Energy Seabrook, LLC, hereby affirm that the information and statements contained within this response to the request for additional information to License Amendment Request 06-02 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

16th day of August, 2006

Shirley Sweeney
Notary Public

Gene St. Pierre

Gene St. Pierre
Site Vice President

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

Attachment 1

FPL Energy Seabrook Response to Request for Additional Information

Attachment 1

Response to NRC Request for Addition Information
License Amendment Request 06-02
"Application for Technical Specification Improvement Regarding Steam Generator Integrity Using the Consolidated Line Item Improvement Process"

NRC RAI 1:

Proposed Action statement "b" under TS 3.4.6.2 indicates that identified leakage greater than the limits should be reduced to within the limits, within 4 hours or the plant should be shut down. Since identified leakage includes primary-to-secondary leakage (per TS definition 1.17), this proposed revision appears to contradict Action statement "a" under TS 3.4.6.2 and Technical Specification Task Force (TSTF) Traveler 449. Please clarify this apparent contradiction or correct Action statement "b" to indicate that it does not apply to primary-to-secondary leakage. In addition, discuss your plans to modify your Bases to reflect this change (page 4 of Insert Bases 3.4.6.2).

FPL Energy Seabrook Response 1:

Proposed TS 3.4.6.2, action b is revised as follows to exclude primary to secondary leakage from Action b:

> b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, primary to secondary 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.

The originally proposed Bases for TS 3.4.6.2 in the section ACTIONS (page 4 of Insert Bases 3.4.6.2), are similar to the bases for TS 3.4.13 in TSTF-449, which states "Unidentified LEAKAGE or identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours." Nonetheless, the proposed bases are modified similar to the Action to exclude applying the four-hour time limit to primary to secondary leakage:

> Unidentified leakage, identified leakage *(excluding primary to secondary leakage)*, or controlled leakage in excess of the LCO limits must be reduced to within limits within 4 hours.

NRC RAI 2:

Please discuss why the proposed TS 6.7.6.k does not include the acronym for steam generator (i.e., SG) in the title (since "SG" is used throughout this section and to make your proposal consistent with TSTF-449, proposed insert 6.8.1.7, and your proposed Bases).

FPL Energy Seabrook Response 2:

The title for proposed TS 6.7.6.k is modified to include the acronym for steam generator consistent with TSTF-449, revision 4:

6.7.6.k Steam Generator *(SG)* Program

NRC RAI 3:

In the Background section of Insert B3/4.4.5 (top of second page of the insert), the leakage assumptions for your steam generator tube rupture (SGTR) accident analysis are not clear. For example, in one sentence, you indicate that the leakage is apportioned between the SG (1.0 gallon per minute (1 gpm) total, 500 gallons per day (gpd) to any one SG). However, in the next sentence, the assumption appears to be that one of the non-faulted SGs is assigned a leak rate of 500 gpd (as part of the 1126.67 gpd total). This sentence appears to indicate that a tube rupture is only assigned a leak rate of 313.33 gpd since (unlike TSTF-449) it does not indicate that the leak rate associated with a double-ended rupture of a single tube is added to the 313.33 gpd. The next sentence then appears to indicate that the SGTR analysis only considers the leakage rate associated with the instantaneous rupture of a SG tube (and it is not clear if this is the 313.33 gpd value mentioned above, or the leak rate associated with a double-ended rupture of a single tube). Please clarify the NRC staff's understanding of this section.

FPL Energy Seabrook Response 3:

The 313.33 gpd leak rate referred to in the Background section of Insert B3/4.4.5 is the pre-accident leak for the faulted steam generator and is not the leak rate assigned to the double-ended rupture of the single tube in the steam generator tube rupture dose consequences analysis. A thermal-hydraulic analysis was performed to determine a conservative maximum break flow, break flashing flow and steam release through the atmospheric steam dump valve (ASDV) associated with the faulted steam generator. These flow rates are time dependent based on changing primary-to-secondary differential pressure. Although not required by NUREG-800, Section 15.6.3, Rev. 2, the pre-accident leak is conservatively added to the maximum tube break flow determined by the above thermal-hydraulic analysis for the faulted steam generator.

For the intact steam generators, one steam generator is assigned a leak rate of 500 gpd and the remaining two intact steam generators are assigned a leak rate of 313.33 gpd for a total of 1126.67 gpd.

NRC RAI 4:

In the Background section of Insert B3/4.4.5 (middle of second page of the insert), please clarify the statement that your SGTR analysis "considers any leakage changes as a result of the accident induced changes in primary-to-secondary pressure differential." Is this statement implying that in your current accident analysis, you are constantly adjusting your leakage rate based on the actual primary-to-secondary pressure differential throughout the SGTR accident.

FPL Energy Seabrook Response 4:

The statement quoted in the RAI is in reference to analyses of design basis accidents *other than a SGTR*. For analyses of design basis events other than a SGTR, a continuous leakage rate consistent with a limit of 500 gpd in any one steam generator and total leakage of 1 gpm from all steam generators is assumed. The intent of the statement in this section of B3/4.4.5 is that this leak rate assumed by the dose consequence analyses conservatively bounds the expected actual leakage that is inclusive of operational leakage existing prior to the accident plus any increased leakage resulting from accident induced changes in primary-to-secondary pressure differential.

NRC RAI 5:

Near the end of the Background Section for the SG Tube Integrity Program in TSTF-449, there is a phrase that reads "or the NRC approved licensing basis (e.g., a small fraction of these limits)." This phrase is not in the corresponding section of your proposed Bases. In addition, this phrase is also missing from the proposed "Applicable Safety Analysis" Section under Bases Insert 3.4.6.2. Please discuss why this statement was not included in your proposal, or alternatively, propose to include it. In addition, discuss your plans to incorporate reference to General Design Criteria 19, consistent with TSTF-449, in your proposed Bases Section for the SG Tube Integrity Program.

FPL Energy Seabrook Response 5:

The proposed Bases for TS 3/4.4.5, Steam Generator (SG) Tube Integrity, and TS 3/4.4.6.2, Reactor Coolant System Operational Leakage, are modified to include the phrase "10 CFR 100 (Ref. 7), or the NRC approved licensing basis (e.g., a small fraction of these limits)," and "10 CFR 100, or the staff approved licensing basis (i.e., a small fraction of these limits)," respectively, consistent with TSTF-449.

The proposed Bases for TS 3/4.4.5, Steam Generator (SG) Tube Integrity, refer to General Design Criterion (GDC) 19 at the end of the discussion under Applicable Safety Analyses and list GDC in item 2 of the Reference section. The proposed bases change contains the following, which matches the bases for TS 3.4.10 in TSTF-449 exactly: "The dose consequences of these events are within the limits of GDC 19 (Ref. 2)..."

NRC RAI 6:

In the Limiting Condition of Operation (LCO) Section of the proposed B3/4.4.5 (top of fourth page), the wording is modified from that of TSTF-449 In several places. For each of the following areas, please justify the exception taken to TSTF-449 (by addressing the questions below) or modify the proposed TSs to be consistent with TSTF-449:

a. TSTF-449 indicates that "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." In your proposal you indicate that "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. Please discuss why you only focused on changes in "primary-to-secondary pressure differential" in your proposal given that other factors can affect leakage under certain circumstances (e.g., axial thermal loads, bending loads, etc.) and that accident-induced leakage includes not only leakage induced by the accident but also any pre-existing leakage. In addition, discuss why you specified "dose consequence analysis" rather than the broader term "accident analysis".

b. TSTF-449 indicates that "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." In your proposal, you indicate that "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 statement in TSTF-449 is intended to indicate that the term "accident induced leakage" includes not just any additional leakage that may be induced as a result of the accident loadings, but also the leakage that was present prior to the accident Your proposed wording, however, does not contain this "definition". In addition, your proposed wordings seems to imply that you adjust your leakage (up and down) based on changes in primary-to-secondary pressure. Please clarity whether your existing accident analyses varies the leakage rate as a result of changes in primary-to-secondary differential pressure. In addition, please discuss why leakage can not be affected by loading conditions other than differential pressure.

FPL Energy Seabrook Response 6:

The intent of the proposed wording in the Limiting Condition of Operation (LCO) Section of proposed B3/4.4.5 is to maintain consistency with TSTF-449, and add additional information providing an engineering justification that discussed one aspect (pressure differential) for the conclusion that the limit on operational leakage contained in LCO 3.4.6.2 is significantly less then the conditions assumed in the accident analyses.

Therefore, FPLE proposes to modify the proposed B3/4.4.5 to be consistent with TSTF-449 to simplify this submittal.

NRC RAI 7:

Proposed TS 6.7.6.k.b.3 refers to LCO 3.4.6.2 as "Reactor Coolant System Leakage;" however, in the LCO Section of Insert B3/4.4.5 (middle of fourth page of the insert), you refer to LCO 3.4.6.2 as "RCS [reactor coolant system] Operational Leakage." Please clarify this apparent discrepancy.

FPL Energy Seabrook Response 7:

The reference to TS 3.4.6.2 in proposed TS 6.7.6.k.b.3 and in the LCO Section of Insert B3/4.4.5 is changed to the following for consistency:

TS 3.4.6.2, "Reactor Coolant System Operational Leakage"

NRC RAI 8:

In the LCO Section of Insert B3/4.4.5 (bottom of fourth page of the Insert), you have added a paragraph regarding the details of your tube integrity procedures. Either provide full technical justification for each of these conclusions covering all possible degradation mechanisms or modify the proposed TSs consistent with TSTF-449.

FPL Energy Seabrook Response 8:

The intent of the proposed wording in the Limiting Condition of Operation (LCO) Section of proposed B3/4.4.5 is to maintain consistency with TSTF-449, and add additional information providing an engineering justification that discussed one aspect (pressure differential) for the conclusion that the limit on operational leakage contained in LCO 3.4.6.2 is significantly less then the conditions assumed in the accident analyses. Therefore, FPLE proposes to modify the proposed B3/4.4.5 to be consistent with TSTF-449 to simplify this submittal.

NRC RAI 9:

Reference 7 is cited on the last page of Insert B3/4.4.5 in the text under Surveillance Requirement 4.4.5.2; however, Reference 7 is not listed in the References Section. Please clarify whether there should be a Reference 7 since TSTF-449 has no Reference 7. Also, TSTF-449, Revision 3 is referred to three times in the References Section (with no numbering). Please confirm that these entries should be there. If so, confirm that Revision 3 is the correct reference since Revision 4 was approved by the NRC staff.

FPL Energy Seabrook Response 9:

The references to TSTF-449, revision 3 and Reference 7 in the originally proposed bases for TS 3.4.5 are incorrect. The reference to TSTF-449, revision 3 is deleted. The response to RAI #5 added Reference 7 (for 10 CFR 100) to the proposed bases for TS 3.4.5.

NRC RAI 10:

Several sentences appear to have been eliminated from TS Bases 3/4.4.6.2 and not incorporated into your new Bases section. Specifically, the first three sentences under "Unidentified Leakage" and the last two sentences under "Identified Leakage" do not appear to be incorporated into your new Bases section on operational leakage. Discuss the reason for deleting these sentences (or discuss where they may elsewhere be incorporated into your new proposed Bases).

FPL Energy Seabrook Response 10:

The information that was eliminated from the proposed bases is discussed in Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, and the Seabrook Station UFSAR. Because the information is available in other documents, and because the information is not discussed in the bases in TSTF-449, the sentences were eliminated from the proposed bases for consistency with TSTF-449.

Attachment 2

Revised Markups of Technical Specifications Pages

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,

b. 1 gpm UNIDENTIFIED LEAKAGE,

150 gallons per day primary to secondary 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:

or with primary to secondary leakage not within 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.

operational | primary to secondary leakage,

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.

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 (SG) Program

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

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

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

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

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

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

(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.

Attachment 3

Revised Technical Specifications Bases

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), 10 CFR 50.67 (Ref. 3), 10 CFR 100 (Ref. 7), or the NRC approved licensing basis (e.g., a small fraction of these limits). 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 have a significant affect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

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

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

The accident induced leakage performance criterion ensures that the primary-to secondary leakage caused by ~~any changes in primary-to-secondary pressure differential during~~ a design basis accident, other than SGTR, is ~~considered in the accident dose consequences~~ within the accident analysis assumptions. The accident ~~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~~ 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.* ~~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, "Reactor Coolant System 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~~ 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.
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~~
7. 10 CFR 100

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.

INSERT BASES 3.4.6.2

3/4.4.6.2 OPERATIONAL LEAKAGE

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. 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 (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, 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).

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 (excluding primary to secondary 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).

4.4.6.2.2

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."