FPL Energy
Seabrook Station

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Docket No. 50-443
SBK-L-06214

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

Seabrook Station
Facility Operating License NPF-86

Clarification of 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.

3. FPL Energy Seabrook, LLC letter SBK-L-06158, 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, August 16, 2006.

By letter dated March 23, 2006, (Reference 1) FPL Energy Seabrook, LLC (FPL Energy Seabrook) 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, and FPL Energy provided the requested information in Reference 3.

On August 22, 2006, the NRC staff requested further clarification of FPL Energy Seabrook's response in Reference 3 regarding item four in the request for additional information. Enclosed in Attachment 1 is the FPL Energy Seabrook clarification to the response. Attachment 2 contains

A001

revised proposed Technical Specification Bases as modified in response to this clarification of 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 has reviewed the proposed changes to the Bases.

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
Site Vice President

Enclosure

cc: S. J. Collins, NRC Region I Administrator
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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 clarification to the 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

28th day of November, 2006

Gene St Pierre
Gene St. Pierre
Site Vice President

Shirley Sweeney
Notary Public

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

Attachment 1

FPL Energy Seabrook Clarification of Response to Request for Additional Information

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

NRC Clarifying Question Regarding Request for Additional Information (RAI) 4:

In the response to the fourth request for additional information in their August 16, 2006 letter, FPLE indicated that the primary-to-secondary leak rate assumed in 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 accident induced changes in primary-to-secondary pressure differential. This wording (as proposed) goes beyond TSTF-449 and is not consistent with operating experience (in which the leakage from a tube can go from a relatively small amount (a few gallons per day) to several hundreds of gallons per minute during normal operation). As a result, please discuss your plans to remove this statement and make your submittal consistent with TSTF-449. The staff notes that the text on page 2 of Insert B3/4.4.5 and page 2 of Insert Bases 3.4.6.2 will need to be changed.

FPL Energy Seabrook Clarification of Response to RAI 4:

FPL Energy Seabrook has revised the proposed Technical Specification (TS) Bases 3/4.4.5, Steam Generator (SG) Tube Integrity, and 3/4.4.6.2, Operational Leakage, to be consistent with TSTF-449. The changes are shown below, and the revised Bases are included in Attachment 2 to this letter.

Change to page 2 of Insert B3/4.4.5:

> The analyses for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In the***se*** analyses, ~~of the dose consequences for these events, the activity level in~~ the steam discharge~~d~~ 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.~~***the total primary to secondary leakage from all SGs of 1 gallon per minute and 500 gallons per day from any one SG or is assumed to increase to these values as a result of accident induced conditions.*** For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.8, "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.~~

Change to page 2 of Insert Bases 3.4.6.2

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 ***and 500 gallons per day from any one SG or increases to these values as a result of accident-induced conditions***. The LCO requirement to limit primary to secondary leakage through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Below is NRC RAI 4 and FPL Energy's August 16, 2006 response:

NRC Request for Additional Information 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.

Attachment 2

Revised Technical Specifications Bases

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 analyses for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 1 gallon per minute and 500 gallons per day from any one SG or is assumed to increase to these values as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.8, "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).

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 a design basis accident, other than SGTR, is within the accident analysis assumptions. The accident 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. The accident induced leakage rate includes any primary to secondary leakage existing prior to the accident in addition to primary to secondary leakage induced during the accident.

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2, "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.

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.

a and b

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.

<u>SR 4.4.5.2</u>

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 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."
7. 10 CFR 100

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 and 500 gallons per day from any one SG or increases to these values as a result of accident-induced conditions. The LCO requirement to limit primary to secondary leakage through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

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:

<u>Pressure Boundary Leakage</u>

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.

<u>Unidentified Leakage</u>

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."