



September 14, 2006

L-2006-212
10 CFR 50.90

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

RE: St. Lucie Unit 1
Docket No. 50-335
Reply to Request for Additional Information
Steam Generator Tube Integrity Amendment Request

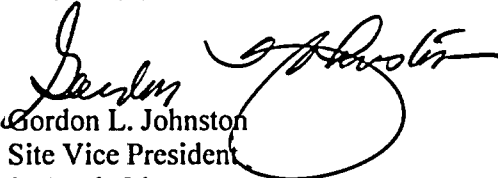
Via letter L-2006-089 dated April 24, 2006, Florida Power and Light Company (FPL) requested to amend Facility Operating License DPR-67 for St. Lucie Unit 1 to change the Technical Specification (TS) requirements related to steam generator tube integrity. The change was based on NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF – 449, "Steam Generator Tube Integrity."

Interactions between the NRC staff and FPL resulted in the NRC transmitting a request for additional information (RAI) in the NRC letter from Mr. Brendan T. Moroney to Mr. J. A. Stall dated August 22, 2006. This letter forwards FPL's reply to the RAI.

Attachment 1 provides the RAI reply. Attachment 2 provides marked-up TS pages to support the RAI reply, and Attachment 3 provides the word-processed TS pages. Attachment 4 provides an information only markup of the TS Bases to support the RAI reply.

The results of the no significance hazards evaluation in the original submittal remain unaffected by the RAI reply. If there are any questions on this submittal, please contact Mr. Ken Frehafer at (772) 467-7748.

Very truly yours,


Gordon L. Johnston
Site Vice President
St. Lucie Plant

GLJ/KWF

Attachments

cc: Mr. William A. Passetti, Florida Department of Health

A122

St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendment
Steam Generator Tube Integrity

L-2006-212
Attachment 1
Page 1 of 4

REQUEST FOR ADDITIONAL INFORMATION
ST LUCIE UNIT 1 STEAM GENERATOR
TUBE INTEGRITY TECHNICAL SPECIFICATION AMENDMENT
DOCKET NO. 50-335

1. *On page 13 of Attachment 2 in your submittal you proposed to include the statement "...per Surveillance Requirement 4.4.6.2.c..." under Action a and b. The purpose of adding this statement in this section is not clear since SR 4.4.6.2.c has no additional details. In addition, this action statement states that the Reactor Coolant System (RCS) water inventory balance should be performed at least once per 24 hours, which appears to conflict with the 72 hours requirement stated in your current TS 4.4.6.2.c. Please explain the purpose of adding this statement or discuss your plans to remove this statement.*

Reply to RAI 1 – FPL agrees with the Staff's comments and the statement has been removed in response to this RAI. The proposed TS wording is presented in Attachments 2 and 3 of this letter.

2. *On page 11 of Attachment 4 in your submittal, your current TS states that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values. Then, on page 6 and 12 of the same Attachment, your proposed TS states that the dose consequences are within the limits of 10 CFR Part 100 as well as 10 CFR Part 50.67. Please clarify whether your current NRC approved accident source term is based on Part 100 (which is referenced in your current TS Bases), Part 50.67 or both?*

Reply to RAI 2 – The current NRC approved accident source term for St. Lucie Unit 1 is based on Part 100, which is referenced in the current TS Bases. Therefore, Part 50.67 is removed in response to this RAI.

3. *On page 12 of Attachment 4 in your submittal there appears to be two words missing from the second sentence of the 3rd paragraph. The sentence currently reads: "Therefore, monitoring reactor coolant leakage..." The sentence should read: "Therefore, detecting and monitoring..." according to the Standard Technical Specifications. Please discuss your plans to add these two words to your proposed TS.*

Reply to RAI 3 – FPL agrees with the Staff's comment and the words "detecting and" have been added to the TS Bases in response to this RAI.

4. *On page 6 of Attachment 4 in your submittal, the last sentence of the 2nd paragraph currently reads: "...or the NRC approved licensing basis (e.g., a small fraction of these limits)." Then, on page 12 of the same Attachment, the same sentence appears on the last paragraph but statement included in the parenthesis is missing. Please discuss your plans to add this statement in this section.*

Reply to RAI 4 – FPL agrees with the Staff's comment and the statement included in the parenthesis has been added in response to this RAI.

5. *You made several changes to the Bases in the Reactor Coolant System leakage section that go beyond TSTF-449. Please confirm that all of the proposed changes are consistent with your current NRC approved design and licensing basis. If they are not consistent, please provide that technical justification or discuss your plans to remove them.*

Reply to RAI 5 – TSTF-449 details a number of changes to NUREG-1432 Bases Section B3.4.13, “RCS Operational LEAKAGE”. In order to adopt these changes, it was necessary to replace St. Lucie Unit 1 Bases Section B3/4.4.6.2 with the content of NUREG-1432 Bases Section B3.4.13. However, changes were necessary to maintain consistency with the current NRC approved design and licensing basis for St. Lucie Unit 1. A review of our submittal provided in letter L-2006-009 dated April 24, 2006 identified differences from TSTF-449 as explained below. Additional changes in response this RAI are also explained.

- In our April 24, 2006 submittal, the third paragraph in the Background section of NUREG-1432 B3.4.13 was omitted (i.e., beginning with “10CFR50, Appendix, A, GDC 30 (Ref.1) . . .”). Upon further review it was determined that this paragraph is consistent with the current NRC approved design and licensing basis for St. Lucie Unit 1 and was added in response to this RAI. Conforming changes were also made to add the references for this paragraph.
- In our April 24, 2006 submittal, the first, second and third paragraphs in the Applicable Safety Analyses section of NUREG-1432 B3.4.13 were omitted. Upon further review and in response to this RAI, these paragraphs are added to St. Lucie Unit 1 Bases section B3/4.4.6.2. The first paragraph, however, is modified to retain only the last sentence as the remainder of this paragraph does not pertain to the St. Lucie Unit 1 design and licensing basis. The second paragraph is added without modification. The third paragraph is modified because the St. Lucie Unit 1 accident analyses do not credit contaminated secondary fluid being steamed to the condenser.
- In our April 24, 2006 submittal, the fourth paragraph in the Applicable Safety Analyses of NUREG-1432 B3.4.13 was modified to be consistent with the current NRC approved design and licensing basis. Also see the response to RAI #4 for additional change.
- In our April 24, 2006 submittal, the first line in the Limiting Condition for Operation (LCO) of NUREG-1432 B3.4.13, “RCS” was changed to “Reactor Coolant System” to maintain consistency with the format of the current TS.
- In our April 24, 2006 submittal, the second paragraph under c. Identified LEAKAGE in the Limiting Condition for Operation (LCO) in NUREG-1432 B3.4.13 (i.e., beginning “LCO 3.4.14, “RCS Pressure Isolation Valve (PIV) Leakage . . .”), was substantially modified and relocated to St. Lucie Unit 1 Bases

B3/4.4.6.2.e. Bases section B3/4.4.6.2.e is subsequently replaced in this RAI response using the wording in the second paragraph under c. Identified LEAKAGE in NUREG-1432 B3.4.13, and modified slightly to be consistent with the current NRC approved St. Lucie Unit 1 design and licensing basis.

- In our April 24, 2006 submittal, the first paragraph of NUREG-1432 Bases SR 3.4.13.1 is used as the first paragraph for St. Lucie Unit 1 Bases SR 4.4.6.2. The remaining paragraphs in NUREG-1432 Bases SR 3.4.13.1 are used in item c under St. Lucie Unit 1 Bases SR 4.4.6.2. Items a, b, d, e and f under St. Lucie Unit 1 Bases SR 4.4.6.2 were added to address monitoring of containment sump, containment atmosphere radioactivity levels, pressure isolation valves and reactor head flange leak-off to maintain consistency with the current NRC approved St. Lucie Unit 1 design and licensing basis.

Several editorial changes are also made to the Bases 3/4.4.6.2 "Reactor Coolant System Leakage" (Attachment 4 of our submittal dated April 24, 2006) with this submittal to remove the underline from "made" and "and" (in the Background), "of" and "operational" (in the Applicable Safety Analyses), "Coolant", "POWER", "COLD", and "allowed" (in the Limiting Condition for Operation).

6. *On page 9 of Attachment 4, you indicate that the affected tubes(s) must be plugged prior to entering Hot Standby. Please discuss your plans to modify this statement since the corresponding requirement to plug the tube(s) is before entering Hot Shutdown.*

Reply to RAI 6 – FPL agrees with the Staff's comment and Hot Standby has been replaced with Hot Shutdown in response to this RAI.

7. *Under Item "d" on page 13 of Attachment 4, there appears to be a typographical error in the 1st sentence. It appears that "with" should be "within."*

Reply to RAI 7 – FPL agrees with the Staff's comment and "with" has been replaced with "within" in response to this RAI.

8. *On page 14 of 17, the action statements do not address the need to be in Hot Standby within 6 hours and Cold Shutdown within the following 30 hours in the extent unidentified or identified leakage can not be reduced to within limits within 4 hours. Please discuss your plans to modify your proposed Bases to address this issue.*

Reply to RAI 8 – FPL agrees with the Staff's comment and the words "Otherwise, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours." has been added after the second sentence in Action b for clarification.

St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendment
Steam Generator Tube Integrity

L-2006-212
Attachment 2
Page 1 of 2

Technical Specification Markups
TS Page 3/4 4-12

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following RCS leakage detection systems will be OPERABLE:

- a. The reactor cavity sump inlet flow monitoring system; and
- b. One containment atmosphere radioactivity monitor (gaseous or particulate).

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

ACTION:

- a. With the required reactor cavity sump inlet flow monitoring system inoperable, perform a RCS water inventory balance at least once per 24 hours and restore the sump inlet flow monitoring system to OPERABLE status within 30 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With the required radioactivity monitor inoperable, analyze grab samples of the containment atmosphere or perform a RCS water inventory balance at least once per 24 hours, and restore the required radioactivity monitor to OPERABLE status within 30 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With all required monitors inoperable, enter LCO 3.0.3 immediately.
- d. The provisions of Specification 3.0.4 are not applicable if at least one of the required monitors is OPERABLE.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The RCS leakage detection instruments shall be demonstrated OPERABLE by:

- a. Performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor at the frequencies specified in Table 4.3-3.
- b. Performance of the CHANNEL CALIBRATION of the required reactor cavity sump inlet flow monitoring system at least once per 18 months.

* Not required to be performed until 12 hours after establishment of steady state operation.

St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendment
Steam Generator Tube Integrity

L-2006-212
Attachment 3
Page 1 of 2

Word-Processed Technical Specifications
TS Page 3/4 4-12

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following RCS leakage detection systems will be OPERABLE:

- a. The reactor cavity sump inlet flow monitoring system; and
- b. One containment atmosphere radioactivity monitor (gaseous or particulate).

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

ACTION:

- a. With the required reactor cavity sump inlet flow monitoring system inoperable, perform a RCS water inventory balance at least once per 24* hours and restore the sump inlet flow monitoring system to OPERABLE status within 30 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With the required radioactivity monitor inoperable, analyze grab samples of the containment atmosphere or perform a RCS water inventory balance at least once per 24* hours, and restore the required radioactivity monitor to OPERABLE status within 30 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With all required monitors inoperable, enter LCO 3.0.3 immediately.
- d. The provisions of Specification 3.0.4 are not applicable if at least one of the required monitors is OPERABLE.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The RCS leakage detection instruments shall be demonstrated OPERABLE by:

- a. Performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor at the frequencies specified in Table 4.3-3.
- b. Performance of the CHANNEL CALIBRATION of the required reactor cavity sump inlet flow monitoring system at least once per 18 months.

* Not required to be performed until 12 hours after establishment of steady state operation.

St. Lucie Unit 1
Docket No. 50-335
Proposed License Amendment
Steam Generator Tube Integrity

L-2006-212
Attachment 4
Page 1 of 16

TS Bases Markups

| | | |
|-----------------------|---|------------------|
| SECTION NO.: 3/4.4 | TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 6 OF ADM-25.04 REACTOR COOLANT SYSTEM ST. LUCIE UNIT 1 | PAGE: 2 of 16 |
| REVISION NO.: 1 | | |

| TABLE OF CONTENTS | |
|---|------|
| SECTION | PAGE |
| BASES FOR SECTION 3/4.4 | 3 |
| 3/4.4 REACTOR COOLANT SYSTEM..... | 3 |
| BASES | 3 |
| 3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION | 3 |
| 3/4.4.2 DELETED | 4 |
| 3/4.4.3 SAFETY VALVES..... | 4 |
| 3/4.4.4 PRESSURIZER..... | 5 |
| 3/4.4.5 STEAM GENERATORS..... | 5 |
| 3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE | 7 |
| 3/4.4.6.1 LEAKAGE DETECTION SYSTEMS | 7 |
| 3/4.4.6.2 REACTOR COOLANT SYSTEM LEAKAGE | 7 |
| 3/4.4.7 CHEMISTRY | 8 |
| 3/4.4.8 SPECIFIC ACTIVITY..... | 8 |
| 3/4.4.9 PRESSURE/TEMPERATURE LIMITS | 9 |
| 3/4.4.10 STRUCTURAL INTEGRITY | 11 |
| TABLE B 3/4.4-1 REACTOR VESSEL TOUGHNESS | 12 |
| 3/4.4.11 DELETED | 15 |
| 3/4.4.12 PORV BLOCK VALVES | 15 |
| 3/4.4.13 POWER OPERATED RELIEF VALVES and | 15 |
| 3/4.4.14 REACTOR COOLANT PUMP - STARTING | 15 |
| 3/4.4.15 REACTOR COOLANT SYSTEM VENTS | 16 |

| | | |
|-----------------------|--|------------------|
| SECTION NO.: 3/4.4 | TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 6 OF ADM-25.04 REACTOR COOLANT SYSTEM ST. LUCIE UNIT 1 | PAGE: 5 of 16 |
| REVISION NO.: 1 | | |

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

3/4.4.3 SAFETY VALVES (continued)

The pressurizer code safety valve as-found setpoint is 2500 psia $\pm 3/-2.5\%$ for OPERABILITY; however, the valves are reset to 2500 psia $\pm 1\%$ during the Surveillance to allow for drift. The LCO is expressed in units of psig for consistency with implementing procedures.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer-Pressure-High signal minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The required pressurizer heater capacity is capable of maintaining natural circulation sub-cooling. Operability of the heaters, which are powered by a diesel generator bus, ensures ability to maintain pressure control even with loss of offsite power.

3/4.4.5 STEAM GENERATORS (SG) TUBE INTEGRITY

One ~~OPERABLE~~ steam generator provides sufficient heat removal capability to ~~remove~~ decay heat after a reactor shutdown. The requirement for two steam generators capable of removing decay heat, combined with the requirements of Specifications 3.7.1.1, 3.7.1.2 and 3.7.1.3 ensures adequate decay heat removal capabilities for RCS temperatures greater than 325°F if one steam generator becomes inoperable due to single failure considerations. Below 325°F, decay heat is removed by the shutdown cooling system.

INSERT
83/4.4.5

| | | |
|-----------------------|---|------------------|
| SECTION NO.: 3/4.4 | TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 6 OF ADM-25.04 REACTOR COOLANT SYSTEM ST. LUCIE UNIT 1 | PAGE: 6 of 16 |
| REVISION NO.: 1 | | |

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued) (SG) TUBE INTEGRITY

3/4.4.5 STEAM GENERATORS (continued)

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 in-service conditions that lead to corrosion. Inservice inspection of Steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 1 gallon per minute, total). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 1 gallon per minute 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 the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.4.5.4.a, is 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.

| | | |
|-----------------------|--|------------------|
| SECTION NO.: 3/4.4 | TITLE: TECHNICAL SPECIFICATIONS BASES ATTACHMENT 6 OF ADM-25.04 REACTOR COOLANT SYSTEM ST. LUCIE UNIT 1 | PAGE: 7 of 16 |
| REVISION NO.: 1 | | |

3/4.4 REACTOR COOLANT SYSTEM (continued)

BASES (continued)

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. The LCO is consistent with NUREG-1432, Revision 1, and is satisfied when leakage detection monitors of diverse measurement means are OPERABLE in MODES 1, 2, 3, and 4. Monitoring the reactor cavity sump inlet flow rate, in combination with monitoring the containment particulate or gaseous radioactivity, provides an acceptable minimum to assure that unidentified leakage is detected in time to allow actions to place the plant in a safe condition when such leakage indicates possible pressure boundary degradation.

OPERATIONAL

3/4.4.6.2 REACTOR COOLANT SYSTEM LEAKAGE

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

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

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents.

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.

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.

INSERT
B3/4.4.6.2
(follows
Insert for
B3/4.4.5)

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT 3/4.4.5

Background

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. SG tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.1.1, "Reactor Coolant Loops and Coolant Circulation, Startup and Power Operation," LCO 3.4.1.2, "Hot Standby," LCO 3.4.1.3, "Hot Shutdown," LCO 3.4.1.4.1, "Cold Shutdown – Loops Filled," and LCO 3.4.1.4.2, "Cold Shutdown – Loops Not Filled."

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

SG tubing is subject to a variety of degradation mechanisms. SG tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other 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 6.8.4.1, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.4.1, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. The SG performance criteria are described in Specification 6.8.4.1. 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. The analysis of a SGTR event assumes a bounding primary-to-secondary leakage rate equal to the operational leakage rate limits in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released via the main steam safety valves and/or atmospheric dump valves. The majority of the activity released to the atmosphere results from the tube rupture.

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT 3/4.4.5

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses the steam discharge to the atmosphere is based on the total primary-to-secondary leakage from all SGs of 1 gpm and 0.5 gpm through any one SG 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 limits in LCO 3.4.8, "Reactor Coolant System Specific Activity." For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

Limiting Condition for Operation (LCO)

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

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

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

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.1, "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 verses displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT 3/4.4.5

secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

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

The accident induced leakage performance criterion ensures that the primary-to-secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm total from all SGs and 0.5 gpm through any one SG. 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 gpd at room temperature. 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

SG tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in POWER OPERATION, START UP, HOT STANDBY and HOT SHUTDOWN.

RCS conditions are far less challenging in COLD SHUTDOWN and REFUELING than during POWER OPERATION, START UP, HOT STANDBY and HOT SHUTDOWN. In COLD SHUTDOWN and REFUELING, primary-to-secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.

ACTIONS

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

a.1 and a.2

ACTIONS a.1 and a.2 apply if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by Surveillance Requirement (SR) 4.4.5.2.

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT 3/4.4.5

An evaluation of SG tube integrity of the affected tube(s) must be made. SG 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.

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

If the evaluation determines that the affected tube(s) have tube integrity, ACTION a.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering HOT SHUTDOWN following the next refueling outage or SG inspection. This allowable completion time is acceptable since operation until the next inspection is supported by the operational assessment.

b.

If the requirements and associated allowable completion time of ACTION a are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours. The allowable completion times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

Surveillance Requirements

SR 4.4.5.1 During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, "Steam Generator Program Guidelines" (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

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT 3/4.4.5

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, non-destructive 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 6.8.4.1 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 6.8.4.1 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 HOT SHUTDOWN following a SG tube 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 100
 4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB
 5. Draft Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," August 1976
 6. EPRI "Pressurized Water Reactor Steam Generator Examination Guidelines"
-

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT 3/4.4.6.2

Background

Components that contain or transport the coolant to or from the reactor core make up the reactor coolant system (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

The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 1 gpm primary to secondary leakage as the initial condition.

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) analysis for SGTR assumes the contaminated secondary fluid is released via the safety valves or atmospheric dump valves. The 1 gpm primary to secondary leakage is relatively inconsequential.

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT 3/4.4.6.2

Primary-to-secondary leakage contaminates the secondary fluid. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary-to-secondary leakage from all steam generators is 1 gpm and 0.5 gpm through any one SG as a result of accident induced conditions. The dose consequences of these events are within the limits of GDC 19, 10 CFR 100 or the NRC approved licensing basis (e.g., a small fraction of these limits). The LCO requirement to limit primary-to-secondary leakage through any one steam generator to less than or equal to 150 gpd is significantly less than the conditions assumed in the safety analysis.

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

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

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

b. 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.

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

The limit of 150 gpd per steam generator 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 9706 states, "The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion is conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT 3/4.4.6.2

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the Reactor Coolant System Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump seal leak-off (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

e. Reactor Coolant System Pressure Isolation Valve Leakage

Leakage is measured 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 leaktight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable IDENTIFIED LEAKAGE.

Applicability

In POWER OPERATION, START UP, HOT STANDBY and HOT SHUTDOWN, the potential for PRESSURE BOUNDARY LEAKAGE is greatest when the RCS is pressurized.

In COLD SHUTDOWN and REFUELING, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage.

ACTIONS

- a. If any PRESSURE BOUNDARY LEAKAGE exists, or primary-to-secondary leakage is not within limit, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours. This ACTION reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.
 - b. UNIDENTIFIED LEAKAGE or IDENTIFIED LEAKAGE in excess of the LCO limits must be reduced to within the limits within 4 hours. This allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. Otherwise, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours. This ACTION is necessary to prevent further deterioration of the Reactor Coolant Pressure Boundary.
 - c. The leakage from any RCS Pressure Isolation Valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is
-

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT 3/4.4.6.2

provided by two in-series 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. With one or more RCS Pressure Isolation Valves with leakage greater than that allowed by Specification 3.4.6.2.e, within 4 hours, at least two valves, including check valves, in each high pressure line having a non-functional valve must be closed and remain closed to isolate the affected line(s). In addition, the ACTION statement for the affected system must be followed and the leakage from the remaining Pressure Isolation Valves in each high pressure line having a valve not meeting the criteria of Table 3.4.6-1 shall be recorded daily. If these requirements are not met, the reactor must be brought to at least HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

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 COLD SHUTDOWN, the pressure stresses acting on the Reactor Coolant Pressure Boundary are much lower, and further deterioration is much less likely.

Surveillance Requirements

4.4.6.2

Verifying Reactor Coolant System leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained. PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance or a Reactor Coolant System water inventory balance.

a and b.

These SRs demonstrate that the RCS operational leakage is within the LCO limits by monitoring the containment atmosphere gaseous or particulate radioactivity monitor and the containment sump level at least once per 12 hours.

c.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and reactor coolant pump seal injection and return flows). The Surveillance is modified by a note that states that this Surveillance Requirement is not required to be performed until 12 hours after establishment of steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT 3/4.4.6.2

Steady state operations 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 Reactor Coolant Pump seal injection and return flows.

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity, containment sump level, and reactor head flange leak-off. The reactor cavity (containment) sump and containment atmosphere radioactivity leakage detection systems are specified in LCO 3.4.6.1, "Reactor Coolant System Leakage Detection Systems."

The note also 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.

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

d.

This SR demonstrates that the RCS operational leakage is within the LCO limits by monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.

e. and f.

This Surveillance Requirement verifies RCS Pressure Isolation Valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

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 overpressurization 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.

Whenever integrity of a pressure isolation valve listed in Table 3.4.6-1 cannot be demonstrated the integrity of the remaining check valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in each high pressure line having a leaking valve shall be recorded daily.

Technical Specification Bases Attachment 6 of ADM-25.04 - INSERT 3/4.4.6.2

g.

This Surveillance Requirement verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one steam generator. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5, "Steam Generator Tube Integrity" should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 5. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator.

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

The 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).

References

1. 10 CFR 50, Appendix A, GDC 30.
 2. Regulatory Guide 1.45, May 1973.
 3. UFSAR, Section 15.4.4.
 4. NEI 97-06, "Steam Generator Program Guidelines."
 5. EPRI "PWR Primary-to-Secondary Leak Guidelines."
-