FPL

Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

April 24, 2006

L-2006-089
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
Proposed License Amendment
Steam Generator Tube Integrity

Pursuant to 10 CFR 50.90, Florida Power and Light Company (FPL) requests to amend Facility Operating License DPR-67 for St. Lucie Unit 1.

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

Attachment 1 provides a description of the proposed change and confirmation of applicability. Attachment 2 provides the existing TS pages marked-up to show the proposed changes. Attachment 3 provides the word-processed TS pages. Attachment 4 provides an information only markup of the TS Bases.

FPL requests that the proposed amendment be processed by January 2007 to support SL1-21 April 2007 refueling outage inspections and that the amendment be effective on the date of issuance with implementation within 90 days.

The license amendment proposed by FPL has been reviewed by the St. Lucie Plant Facility Review Group and the FPL Company Nuclear Review Board. In accordance with 10 CFR 50.91(b)(1), a copy of this proposed license amendment is being forwarded to the State Designee for the State of Florida.

A001

Please contact Ken Frehafer at 772-467-7748 if there are any questions about this submittal.

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

Executed on April 24, 2006

Very truly yours,

Gordon L. Johnston

Gordon L. Johnston
Acting Vice President
St. Lucie Plant

GLJ/KWF

Attachments

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

Evaluation of Proposed Change

INTRODUCTION

Pursuant to 10 CFR 50.90, Florida Power and Light Company (FPL) requests to amend Facility Operating License DPR-67 for St. Lucie Unit 1. This proposed license amendment request (LAR) revises the requirements in the St. Lucie Unit 1 Technical Specification (TS) related to steam generator tube integrity and Reactor Coolant System Operational Leakage. The change is based on the NRC approved Revision 4 to industry/Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity." The availability of this TS improvement was announced in the Federal Register on May 6, 2005 (70 FR 24126) as part of the consolidated line item improvement process (CLIIP).

DESCRIPTION OF PROPOSED AMENDMENT

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

- Revise the TS definitions of Identified Leakage (1.15c) and Pressure Boundary Leakage (1.22).
- Replace the existing TS 3/4.4.5, "Steam Generators," requirements with the new "Steam Generator Tube Integrity" requirements.
- Revise TS 3/4.4.6, "Reactor Coolant System Leakage."
- Add new TS 6.8.4 l. (lima), "Steam Generator Program."
- Add new TS 6.9.1.12, "Steam Generator Tube Inspection Report."

Proposed revisions to the TS Bases are also included with this LAR. As discussed in the NRC's model safety evaluation, adoption of the revised TS Bases associated with TSTF-449, Rev. 4 is an integral part of implementing this TS improvement. Departure from the wording proposed in the TS Bases associated with TSTF-449, Rev. 4 is taken only when necessary to maintain consistency with the St. Lucie Unit 1 licensing basis. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program.

BACKGROUND

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

The table below provides a summary of the proposed changes. It also identifies the Improved Standard Technical Specifications (ITS) sections based on TSTF-449, Rev. 4 and the corresponding sections in the St. Lucie Unit 1 TS.

TSTF-449 ITS Section	Condition or Requirement	Current Licensing Basis	St. Lucie TS Location & Proposed Change	
3.4.13d	Operational primary-to-secondary leakage	≤1 GPM total through all SGs and ≤500 gallons per day through any one SG (accident conditions).	3.4.6.2c	RCS Operational Leakage TS ≤ 150 gallons per day through any one SG (room temperature).
3.4.13	RCS primary-to-secondary leakage through any one SG not within limits	Reduce 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.	3.4.6.2	RCS Operational Leakage, ACTION a. - be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
3.4.13.1	RCS leakage determined by water inventory balance	RCS leakage is determined by water inventory balance.	4.4.6.2c 3.4.6.1, ACTION a. and b.	Relocate extemporaneous information to footnote and revise to state: "Not required to be performed until 12 hours after establishment of steady state operation. Not applicable to primary to secondary leakage." Add conforming changes to other affected specifications.
3.4.13.2	SG Tube integrity verification	Sample and analysis program requires Gross Radioactivity Determination every 72 hours.	4.4.6.2	Add new RCS Operational Leakage TS 4.4.6.2.g to verify primary-to-secondary leakage within LCO limit at least once per 72 hours. Add Note stating "Not required to be performed until 12 hours after establishment of steady state operation."
3.4.13, 3.4.20	ACTIONS	Performance Criteria not defined. Primary to secondary leakage limit and actions included in the Tech Specs. Plug or repair tubes exceeding repair criteria.	3.4.6.2, 3/4.4.6 3/4.4.5	RCS Operational Leakage TS and SG Tube Integrity TS – Contains primary-to-secondary leakage limit. SG tube integrity requirements and ACTIONS required upon failure to meet performance criteria. Plug or repair tubes satisfying repair criteria.

TSTF-449 ITS Section	Condition or Requirement	Current Licensing Basis	St. Lucie TS Location & Proposed Change	
3.4.13d	Performance criteria	Operational leakage ≤ 1 gpm total or ≤ 500 gallons per day through any one SG (accident conditions).	3.4.6.2c	RCS Operational leakage TS – Operational leakage ≤ 150 gallons per day through any one SG (room temperature).
3.4.20		No criteria specified for structural integrity or accident induced leakage.	3/4.4.5	SG Tube Integrity TS 3/4.4.5 – Requires that tube integrity be maintained.
5.5.9			6.8.4 l	TS 6.8.4 l – Defines structural integrity and accident induced leakage performance criteria, which are dependent on design basis limits. Provides provisions for condition monitoring assessment to verify compliance.
5.5.9	Frequency of verification of tube integrity	6 to 40 months depending on SG category defined by previous inspection results.	6.8.4 l	SG Tube Integrity TS – Requires Surveillance Frequency in accordance with TS 6.8.4 l, Steam Generator Program. Frequency is dependent on tubing material and the previous inspection results and the anticipated defect growth rate. Steam Generator Program – Establishes maximum inspection intervals.
5.5.9	Tube sample selection	Based on SG Category, industry experience, random selection, existing indications, and results of the initial sample set - 3% times the number of SGs at the plant as a minimum.	6.8.4 l	Steam Generator Program and implementing procedures – Dependent on a pre-outage evaluation of actual degradation locations and mechanisms, and operating experience – 20% of all tubes as a minimum.

TSTF-449 ITS Section	Condition or Requirement	Current Licensing Basis	St. Lucie TS Location & Proposed Change	
5.5.9	Inspection techniques	Not specified	6.8.4 l	SG Tube Integrity TS – SR 4.4.5.1 requires that tube integrity be verified in accordance with the Steam Generator Program. TS 6.8.4 l Steam Generator Program and implementing procedures – Establishes requirements for qualifying NDE techniques. Requires use of qualified techniques in SG inspections. Requires a pre-outage evaluation of potential tube degradation morphologies and locations and an identification of NDE techniques capable of finding the degradation.
5.5.9	Inspection scope	From the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg, or from the point of entry (cold leg side) completely round the U-bend and to the bottom of the hot leg.	6.8.4 l	TS 6.8.4 l Steam Generator Program procedures – Inspection scope is defined by the degradation assessment that considers existing and potential degradation morphologies and locations. Explicitly requires consideration of entire length of tube from tube-sheet weld to tubesheet weld.
5.5.9	Repair criteria	Plug tubes with imperfections extending $\geq$40% through wall.	6.8.4 l	TS 6.8.4 l – Criteria unchanged.
5.5.9	Repair methods	Methods (except plugging) require prior approval by the NRC. Approved methods listed in TS.	6.8.4 l	TS 6.8.4 l –Requirements unchanged.

TSTF-449 ITS Section	Condition or Requirement	Current Licensing Basis	St. Lucie TS Location & Proposed Change	
5.6.9	Reporting requirements	Plugging and repair report required 15 days after each inservice inspection, 12 month report documenting inspection results, and reports in accordance with §50.72 when the inspection results fall into category C-3.	6.9.1.12	CFR - Serious SG tube degradation (i.e., tubing fails to meet the structural integrity or accident induced leakage criteria) requires reporting in accordance with 50.72 or 50.73. TS 6.9.1.12 - 180 days after the initial entry into MODE 4 after performing a SG inspection
Definitions	Definitions SG Terminology	Normal TS definitions (i.e., Definitions Section) did not address SG Program issues. The Definitions Section uses the term "SG leakage."	Definitions	TS 6.8.4 l, TS Bases, Steam Generator Program procedures – Includes Steam Generator Program terminology applicable only to SGs. TS Definitions 1.15c and 1.22 are revised to use the term "primary-to-secondary leakage."

REGULATORY REQUIREMENTS AND GUIDANCE

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

TECHNICAL ANALYSIS

FPL has reviewed the safety evaluation (SE) published on March 2, 2005 (70 FR 10298) as part of the CLIIP Notice for Comment. This included the NRC staff's SE, the supporting information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. FPL has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to St. Lucie Unit 1 and justify this amendment for the incorporation of the changes to the St. Lucie TS considering the differences described in this application. These differences from the TS changes described in TSTF-449, Revision 4 are necessary due to the non-standard format of the St. Lucie Unit 1 TS.

REGULATORY ANALYSIS

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published on May 6, 2005 (70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (70 FR 10298), and TSTF-449, Revision 4.

Supporting Information

The following information is provided to support the NRC staff's review of this LAR:

Plant Name, Unit No.	St. Lucie Unit 1
Steam Generator Model(s):	B&W Canada Replacement for CE Series-67
Approximate Effective Full Power Years (EFPY) of service for currently installed SGs	7.3 EFPY as of the October 2005 refueling outage.
Tubing Material	Alloy 690 Thermally Treated
Number of tubes per SG	8523
Number and percentage of tubes plugged in each SG as of 7/2005	1A 14 (0.17%) 1B 0 (0.0%)
Number of tubes repaired in each SG	None
Degradation mechanism(s) identified	Mechanical Wear
Current primary-to-secondary leakage limits:	Per SG: Not specified. Total: 1 gallon per minute (operating temp.)

Approved Alternate Tube Repair Criteria:	None
Approved SG Tube Repair Methods	None
Performance criteria for accident leakage	1 gpm total (operating temp.)

NO SIGNIFICANT HAZARDS CONSIDERATION

FPL has reviewed the no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIIP. FPL has concluded that the proposed determination presented in the notice is applicable to St. Lucie Unit 1 considering the differences described in this application. Therefore, the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

ENVIRONMENTAL EVALUATION

FPL has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (70 FR 10298) as part of the CLIIP. FPL has concluded that the NRC staff's findings presented in that evaluation are applicable to St. Lucie Unit 1 and the evaluation is hereby incorporated by reference for this application.

PRECEDENT

This application is being made in accordance with the CLIIP. FPL is not proposing variations or deviations from the TS changes described in TSTF-449, Revision 4, or the NRC staff's model SE published on March 2, 2005 (70 FR 10298). The following differences from the TS changes described in TSTF-449, Revision 4 are necessary due to the non-standard format of the St. Lucie Unit 1 TS:

1. The current format and terminology used in the St. Lucie Unit 1 TS is retained to maintain consistency with the current specifications. For example:
   - The general format and numbering convention associated with the current TS for Limiting Conditions for Operation (LCOs), Actions, Surveillance Requirements (SRs) and Notes are retained.
   - Terminology used in the current TS Actions is maintained. For example, HOT STANDBY, HOT SHUTDOWN and COLD SHUTDOWN are used in lieu of MODE 3, MODE 4 and MODE 5, respectively.
2. Necessary conforming changes regarding the proper timing and conditions for performing the RCS water inventory balance were made to Specifications 3.4.6.1 ACTIONS a. and b

Technical Specification Markups

TS Page V
TS Page XV
TS Page 1-4
TS Page 1-5
TS Page 3/4 4-5
TS Page 3/4 4-6
TS Page 3/4 4-7
TS Page 3/4 4-8
TS Page 3/4 4-10
TS Page 3/4 4-11
TS Page 3/4 4-12
TS Page 3/4 4-14
TS Page 3/4 4-14a
TS Page 6-15d
TS Page 6-19c

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

ADMINISTRATIVE CONTROLS

(1)

DEFINITIONS

IDENTIFED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

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

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system~~.~~ (Primary-to-secondary leakage)

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

1.16 The LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE is that operating condition when (1) the cold leg temperature is $\leq$ 304°F during heatup or $\leq$ 281°F during cooldown and (2) the Reactor Coolant System has pressure boundary integrity. The Reactor Coolant System does not have pressure boundary integrity when the Reactor Coolant System is open to containment and the minimum area of the Reactor Coolant System opening is greater than 1.75 square inches.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 THE OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8.

DEFINITIONS

OPERABLE – OPERABILITY

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

OPERATIONAL MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

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

primary-to-secondary

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM (PCP)

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

PURGE – PURGING

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

REACTOR COOLANT SYSTEM

STEAM GENERATORS (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE. INSERT A

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: *

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

SURVEILLANCE REQUIREMENTS

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

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

1

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

b. ~~The first inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:~~

1. ~~All nonplugged tubes that previously had detectable wall penetrations (> 20%), and~~

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

* Separate Action entry is allowed for each SG tube.

3/4.4.5 INSERT A

SG tube integrity shall be maintained

AND

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

3/4.4.5 INSERT B

a. With one or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program;

   1. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, and

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

b. With the requirements and associated allowable outage time of Action a above not met or SG tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

3/4.4.5 INSERT C

Verify SG tube integrity in accordance with the Steam Generator Program.

3/4.4.5 INSERT D

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

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

c. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to the inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

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

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

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

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

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

THIS PAGE DELETED

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be reduced to at least once per 20 months. The reduction in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

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

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2,
2. A seismic occurrence greater than the Operating Basis Earthquake,
3. A loss-of-coolant accident requiring actuation of the engineered safeguards, or
4. A main steam line or feedwater line break.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.
2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.
3. Degraded Tube means a tube containing imperfections ≥ 20% of the nominal wall thickness caused by degradation.
4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

THIS PAGE DELETED

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

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

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

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

4.4.5.5 Reports

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

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

1. Number and extent of tubes inspected.

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

3. Identification of tubes plugged.

THIS PAGE DELETED

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

Preservice Inspection	No			Yes		
No. of Steam Generators per Unit	Two	Three	Four	Two	Three	Four
First Inservice Inspection	All			One	Two	Two
Second & Subsequent Inservice Inspections	One[1]			One[1]	One[2]	One[3]

Table Notation:

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.
2. The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.
3. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1st SAMPLE INSPECTION			2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample.	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G.*	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes.	N/A	N/A

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

* The requirement to inspect all tubes may be relaxed for Cycle 5 Refueling since an engineering evaluation has shown that the condition(s) has been adequately bounded by inspection.

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.

per Surveillance Requirement 4.4.6.2.c

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.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total 150 gallons per day primary-to-secondary leakage through any one steam generators, (SG)

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

e. Leakage as specified in Table 3.4.6-1 for each Reactor Coolant System Pressure Isolation Valve identified in Table 3.4.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE, and Reactor Coolant System Pressure Isolation Valve leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit in 3.4.6.2.e above reactor operation may continue provided that at least two valves, including check valves, in each high pressure line having a non-functional valve are in and remain in the mode corresponding to the isolated condition. Motor operated valves shall be placed in the closed position, and power supplies deenergized. (Note, however, that this may lead to ACTION requirements for systems involved.) Otherwise, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the containment atmosphere gaseous and particulate radioactivity at least once per 12 hours.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the containment sump inventory and discharge at least once per 12 hours,

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode, *

d. Monitoring the reactor head flange leakoff system at least once per 24 hours, and

e. Verifying each Reactor Coolant System Pressure Isolation Valve leakage (Table 3.4.6-1) to be within limits:

1. Prior to entering MODE 2 after refueling,

2. Prior to entering MODE 2, whenever the plant has been in COLD SHUTDOWN for 7 days or more if leakage testing has not been performed in the previous 9 months,

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

4. The provision of Specification 4.0.4 is not applicable for entry into MODE 3 or 4.

f. 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. ; and

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

* Not required to be performed until 12 hours after establishment of steady state operation. Not applicable to primary-to-secondary leakage.

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

ADMINISTRATIVE CONTROLS (continued)

k. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ≤ 0.05% when tested in accordance with ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation	2000 ± 200 cfm

2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 0.05% when tested in accordance with ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation	2000 ± 200 cfm

3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Emergency Ventilation	≤ 2.5%	70%

4. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
Control Room Emergency Ventilation	< 4.15" W.G.	2000 ± 200 cfm

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

INSERT 6.8.4 l.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment of 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 plant.

ADMINISTRATIVE CONTROLS - INSERT 6.8.4.l (lima).

l. 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 SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 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 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 through all SGs and 0.5 gpm through any one SG.

3. The operational leakage performance criterion is specified in LCO 3.4.6.2.c, "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.

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

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.

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

INSERT 6.9.1.12

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

INSERT 6.9.1.12 (new)

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.12 A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.l, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

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

Word-Processed Technical Specifications

TS Page V
TS Page XV
TS Page 1-4
TS Page 1-5
TS Page 3/4 4-5
TS Page 3/4 4-6
TS Page 3/4 4-7
TS Page 3/4 4-8
TS Page 3/4 4-10
TS Page 3/4 4-11
TS Page 3/4 4-12
TS Page 3/4 4-14
TS Page 3/4 4-14a
TS Page 6-15d
TS Page 6-15e
TS Page 6-15f
TS Page 6-19c

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

ADMINISTRATIVE CONTROLS

SECTION PAGE

IDENTIFED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

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

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor Coolant System leakage through a steam generator to the secondary system (Primary-to-secondary leakage). |

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

1.16 The LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE is that operating condition when (1) the cold leg temperature is ≤ 304°F during heatup or ≤ 281°F during cooldown and (2) the Reactor Coolant System has pressure boundary integrity. The Reactor Coolant System does not have pressure boundary integrity when the Reactor Coolant System is open to containment and the minimum area of the Reactor Coolant System opening is greater than 1.75 square inches.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 THE OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8.

DEFINITIONS

OPERABLE – OPERABILITY

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

OPERATIONAL MODE

1.20 An OPERATIONAL MODE (i.e., MODE) shall correspond to any one inclusive combination of core reactivity condition, power level and average reactor coolant temperature specified in Table 1.2.

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM (PCP)

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

PURGE – PURGING

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

REACTOR COOLANT SYSTEM

STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: *

a. With one or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program;

   1. Within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, and

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

b. With the requirements and associated allowable outage time of Action a above not met or SG tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

* Separate Action entry is allowed for each SG tube.

THIS PAGE DELETED

THIS PAGE DELETED

THIS PAGE DELETED

THIS PAGE DELETED

THIS PAGE DELETED

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 per Surveillance Requirement 4.4.6.2.c 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 per Surveillance Requirement 4.4.6.2.c 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.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

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

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

e. Leakage as specified in Table 3.4.6-1 for each Reactor Coolant System Pressure Isolation Valve identified in Table 3.4.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

b. With any Reactor Coolant System operational leakage greater than any one of the above limits, excluding primary-to-secondary leakage, PRESSURE BOUNDARY LEAKAGE, and Reactor Coolant System Pressure Isolation Valve leakage, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the limit in 3.4.6.2.e above reactor operation may continue provided that at least two valves, including check valves, in each high pressure line having a non-functional valve are in and remain in the mode corresponding to the isolated condition. Motor operated valves shall be placed in the closed position, and power supplies deenergized. (Note, however, that this may lead to ACTION requirements for systems involved.) Otherwise, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the containment atmosphere gaseous and particulate radioactivity at least once per 12 hours.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the containment sump inventory and discharge at least once per 12 hours,

c. *Performance of a Reactor Coolant System water inventory balance at least once per 72 hours except when operating in the shutdown cooling mode,

d. Monitoring the reactor head flange leakoff system at least once per 24 hours, and

e. Verifying each Reactor Coolant System Pressure Isolation Valve leakage (Table 3.4.6-1) to be within limits:

   1. Prior to entering MODE 2 after refueling,

   2. Prior to entering MODE 2, whenever the plant has been in COLD SHUTDOWN for 7 days or more if leakage testing has not been performed in the previous 9 months,

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

   4. The provision of Specification 4.0.4 is not applicable for entry into MODE 3 or 4.

f. 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; and

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

* Not required to be performed until 12 hours after establishment of steady state operation. Not applicable to primary-to-secondary leakage.

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

ADMINISTRATIVE CONTROLS (continued)

k. Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Revision 2.

1. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass ≤ 0.05% when tested in accordance with ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation	2000 ± 200 cfm

2. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass ≤ 0.05% when tested in accordance with ANSI N510-1975 at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Emergency Ventilation	2000 ± 200 cfm

3. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Emergency Ventilation	≤ 2.5%	70%

4. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
Control Room Emergency Ventilation	< 4.15" W.G.	2000 ± 200 cfm

The provisions of SR 4.0.2 and SR 4.0.3 are applicable to the VFTP test frequencies.

l. 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.

ADMINISTRATIVE CONTROLS (continued)

I. Steam Generator (SG) Program (continued)

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 SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 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 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 through all SGs and 0.5 gpm through any one SG.

3. The operational leakage performance criterion is specified in LCO 3.4.6.2.c, "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.

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 tube 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.

ADMINISTRATIVE CONTROLS (continued)

l. Steam Generator (SG) Program (continued)

d. (continued)

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

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.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment of 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 plant.

ADMINISTRATIVE CONTROLS

CORE OPERATING LIMITS REPORT (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.12 A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.l, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

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

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

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 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 +3/-2.5% for OPERABILITY; however, the valves are reset to 2500 psia +/- 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 B3/4.4.5

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 GENERATOR~~S~~ (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.~~

INSERT B 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.l, "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.l, 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.l. 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.

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), 10 CFR 50.67 (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).

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.l, "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 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. 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 STANDBY 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 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.l 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.l 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"
7. 10 CFR 50.67

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

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

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

~~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

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.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, 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

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, 10 CFR 50.67 or the NRC approved licensing basis. 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. UNIDENTIFED LEAKAGE

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

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

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

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFED LEAKAGE and is well with the capability of the Reactor Coolant System Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump seal leakoff (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

RCS pressure isolation valve leakage is IDENTIFIED LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

The specified leakage limits for the RCS pressure isolation valves are sufficiently low to ensure early detection of possible in-series check valve failure.

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 with 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. 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 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.

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 normal sump inventory and discharge, and reactor head flange leakoff. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Reactor Coolant System Leakage Detection Systems."

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.

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 should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 1. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator.

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

The 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.2).

References

1. NEI 97-06, "Steam Generator Program Guidelines"
2. EPRI "PWR Primary-to-Secondary Leak Guidelines"
3. UFSAR, Section 15.4.4