



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

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Application for Technical Specification Improvement
Regarding Steam Generator Tube Integrity

Pursuant to 10 CFR 50.90, Florida Power and Light Company (FPL) requests to amend Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4.

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

Enclosure 1 provides a description of the proposed change and confirmation of applicability. Enclosure 2 provides the existing TS pages marked-up to show the proposed changes. Enclosure 3 provides the proposed revised TS pages. Enclosure 4 provides the existing TS Bases pages marked-up to show the proposed changes. The marked-up TS Bases pages are provided for information only.

FPL requests approval of the proposed amendment with an amendment implementation period of 90 days from the date of issuance.

The license amendments proposed by FPL have been reviewed by the Turkey Point Plant Nuclear Safety Committee and the FPL Company Nuclear Review Board. In accordance with 10 CFR 50.91(b)(1), a copy of these proposed license amendments is being forwarded to the State Designee for the State of Florida.

Please contact Mr. Walter J. Parker, Licensing Manager at 305-246-6632 if there are any questions about this submittal.

Very truly yours,

A handwritten signature in black ink that reads "Terry O. Jones".

Terry O. Jones
Vice President
Turkey Point Nuclear Plant

A001

ENCLOSURES

- 1 Description and Assessment
- 2 Proposed Technical Specification Changes (Mark up)
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- 4 Proposed Technical Specification Bases Pages (Mark up for information only)

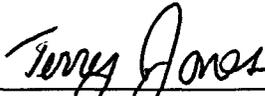
cc: Regional Administrator, Region II, USNRC
USNRC Project Manager, Turkey Point
Senior Resident Inspector, USNRC, Turkey Point
W. A. Passetti, Florida Department of Health

STATE OF FLORIDA)
)
COUNTY OF MIAMI-DADE)

Terry O. Jones, being first duly sworn, deposes and says:

That he is Vice President, Turkey Point Plant, of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

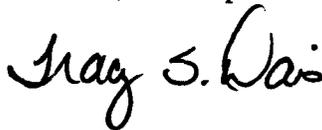


TERRY O. JONES

Sworn to and subscribed before me

This 21 day of April, 2006

By Terry O. Jones, who is personally known to me.



 Tracy S Davis
My Commission DD364248
Expires September 12, 2008

ENCLOSURE 1
DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

Pursuant to 10 CFR 50.90, Florida Power and Light Company (FPL) requests to amend Facility Operating Licenses DPR-31 and DPR-41 for Turkey Point Units 3 and 4. This proposed license amendment request (LAR) revises the requirements in the Turkey Point Units 3 and 4 Technical Specification (TS) related to steam generator tube integrity and Reactor Coolant System Operational Leakage. The change is consistent with 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 (CLIP).

2.0 DESCRIPTION OF PROPOSED AMENDMENT

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

- Revise the TS definitions of Identified Leakage (1.16) and Pressure Boundary Leakage (1.20).
- Revise TS 3/4.3.3, "Monitoring Instrumentation".
- 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.j, "Steam Generator Program."
- Add new TS 6.9.1.8, "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 Turkey Point Units 3 and 4 licensing basis. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program.

3.0 BACKGROUND

The background for this 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 Turkey Point Units 3 and 4 TS.

TSTF-449 ITS Section	Condition or Requirement	Current Licensing Basis	Turkey Point 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.	4.4.6.2.1c 3.3.3.1 3.4.6.1	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.	3/4.4.5	Add new RCS Operational Leakage TS 4.4.6.2.1e 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, 3/4.4.5	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	Turkey Point TS Location & Proposed Change	
3.4.13d 3.4.20 5.5.9	Performance criteria	Operational leakage \leq 1 gpm total or \leq 500 gallons per day through any one SG (accident conditions). No criteria specified for structural integrity or accident induced leakage.	3.4.6.2c 3/4.4.5 6.8.4.j	RCS Operational leakage TS – Operational leakage \leq 150 gallons per day through any one SG (room temperature). SG Tube Integrity TS 3/4.4.5 – Requires that tube integrity be maintained. TS 6.8.4.j – 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.j	SG Tube Integrity TS – Requires Surveillance Frequency in accordance with TS 6.8.4.j, 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.j	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	Turkey Point TS Location & Proposed Change	
5.5.9	Inspection techniques	Not specified	6.8.4.j	<p>SG Tube Integrity TS – SR 4.4.5.1 requires that tube integrity be verified in accordance with the Steam Generator Program.</p> <p>TS 6.8.4.j 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.</p>
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.j	TS 6.8.4.j 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.j	TS 6.8.4.j – Criteria unchanged.
5.5.9	Repair methods	Methods (except plugging) require previous approval by the NRC. Approved methods listed in TS.	6.8.4.j	TS 6.8.4.j –Requirements unchanged.
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.	6.9.1.8	<p>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.</p> <p>TS 6.9.1.8 - 180 days after the initial entry into MODE 4 after performing a SG inspection</p>

TSTF-449 ITS Section	Condition or Requirement	Current Licensing Basis	Turkey Point TS Location & Proposed Change	
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.j, TS Bases, Steam Generator Program procedures – Includes Steam Generator Program terminology applicable only to SGs. TS Definitions 1.16 and 1.20 are revised to use the term "primary-to-secondary leakage."

4.0 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.

5.0 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 Turkey Point Units 3 and 4 and justify this amendment for the incorporation of the changes to the Turkey Point TS considering the differences described in Section 9.0 below. These differences from the TS changes described in TSTF-449, Revision 4 are necessary due to the non-standard format of the Turkey Point Units 3 & 4 TS.

6.0 REGULATORY ANALYSIS

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

6.1 Supporting Information

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

Plant Name, Unit No.	Turkey Point Units 3 and 4
Steam Generator Model(s):	Westinghouse Replacement Model 44F
Approximate Effective Full Power Years (EFPY) of service for currently installed SGs	Unit 3 – 17.7 EFPY as of the 2006 refueling outage. Unit 4 – 16.1 EFPY as of the 2005 refueling outage.
Tubing Material	Alloy 600 Thermally Treated
Number of tubes per SG	3214
Number and percentage of tubes plugged in each SG as of 7/2005	3A 47 (1.5%), 3B 69 (2.1%), 3C 53 (1.6%) 4A 23 (0.7%), 4B 13 (0.4%), 4C 11 (0.3%)
Number of tubes repaired in each SG	None
Degradation mechanism(s) identified	Mechanical Wear
Current primary-to-secondary leakage limits:	500 gallons per day through any one SG (operating temp.) 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 through all steam generators and 500 gallons per day through any one steam generator (operating temp.)

7.0 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 CLIP. FPL has concluded that the proposed determination presented in the notice is applicable to Turkey Point Units 3 and 4 considering the differences described in Section 9.0 below. Therefore, the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91(a).

8.0 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 CLIP. FPL has concluded that the NRC staff's findings presented in that evaluation are applicable to Turkey Point Units 3 and 4 and the evaluation is hereby incorporated by reference for this application.

9.0 PRECEDENT

This application is being made in accordance with the CLIP. 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 Turkey Point Units 3 and 4 TS:

1. The current format and terminology used in the Turkey Point Units 3 and 4 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 is 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. RCS Operational Leakage Water Inventory Balance requirement contained in TSTF-449, Rev. 4, SR 3.4.13.1, is performed at a frequency of 72 hours whereas the current frequency specified in Turkey Point Units 3 and 4 TS 4.4.6.2.1c requires this to be performed at least once per 24 hours, and not to exceed 48 hours between any two successive inventory balances. This requirement is retained except that the frequency is reduced to once every 72 hours and the 48 hour limitation between any two successive inventory balances is eliminated. This change maintains consistency with TSTF-449, Rev. 4, ITS SR 3.4.13.1 and is acceptable because a 72-hour frequency is a reasonable interval to trend leakage and provide early indications of gradual RCS deterioration. The 72-hour frequency also provides conformance with the primary-to-secondary monitoring and trending frequency of the changes herein for Turkey Point TS 4.4.6.2.1e and TSTF-449, Rev. 4, ITS SR 3.4.13.2. Furthermore, the inventory balance verification is not used for the prompt identification of rapid changes in RCS leakage rates. Other methods are available to the operators to provide prompt

indication of any significant increases in RCS leakage, including the RCS leakage detection instrumentation required by TS LCO 3.3.3.1 (Radiation Monitoring) and LCO 3.4.6.1 (Leakage Detection Systems). Therefore, reducing the frequency for evaluating RCS leakage using an inventory balance will not affect the timely identification and response to RCS leakage that is indicative of significant RCS pressure boundary deterioration. Accordingly, this change will have no significant adverse impact on safety.

3. Necessary conforming changes regarding the proper timing and conditions for performing the RCS water inventory balance were made to Specifications 3.3.3.1 and 3.4.6.1.

ENCLOSURE 2

Technical Specification Mark Up and Inserts

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DEFINITIONS

FREQUENCY NOTATION

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.



E-AVERAGE DISINTEGRATION ENERGY

1.13 \bar{E} shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95 percent of the total non-iodine activity in the coolant.



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



GAS DECAY TANK SYSTEM

1.15 A GAS DECAY TANK SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.



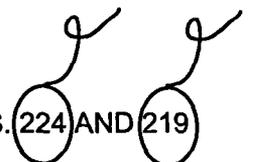
IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

- a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or
- b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or
- c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.



(primary-to-secondary leakage)



DEFINITIONS

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.18 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

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 13.5 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

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

PURGE - PURGING

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

TABLE 3.3-4 (Continued)
TABLE NOTATIONS

- * During CORE ALTERATIONS or movement of irradiated fuel within the containment comply with Specification 3/4.9.13.
- ** With irradiated fuel in the spent fuel pits.
- # Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

Note 1 Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

Note 2 Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{F}$ CPM,

Where $F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in the Offsite Dose Calculation Manual.

①

ACTION STATEMENTS

ACTION 26 - In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

- 1) A Containment sump level monitoring system is OPERABLE, per Surveillance Requirement 4.4.6.2.1.c
- 2) Appropriate grab samples are obtained and analyzed at least once per 24 hours,
- 3) 8*** A Reactor Coolant System water inventory balance is performed at least once per 8 hours during steady state operation except when operating in shutdown cooling mode, and
- 4) Containment Purge, Exhaust and Instrument Air Bleed Valves are maintained closed.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours (ACTION 27 applies in MODES 5 and 6).

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

REACTOR COOLANT SYSTEM

(SG) TUBE INTEGRITY

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

INSERT A

INSERT B

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

*

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

SURVEILLANCE REQUIREMENTS

4.4.5.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

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

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

- a. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from these critical areas;
- b. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each steam generator shall include:

INSERT C

INSERT D

* Separate Action entry is allowed for each SG tube.

REACTOR COOLANT SYSTEM

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

- 1) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),
 - 2) Tubes in those areas where experience has indicated potential problems, and
 - 3) A tube inspection (pursuant to Specification 4.4.5.4a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.
- c. The tubes selected as the second and third samples in the inservice inspection may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be 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:

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

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

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REACTOR COOLANT SYSTEM

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

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 following replacement of steam generators. 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.
- b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a; the interval may then be extended to a maximum of once per 40 months; and
- c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.4-2 during the shutdown subsequent to any of the following conditions:
 - 1) Primary-to-secondary tubes leak (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or
 - 2) A seismic occurrence greater than the Operating Basis Earthquake, or
 - 3) A loss-of-coolant accident resulting in rapid depressurization of the primary system, or
 - 4) A main steam line or feedwater line break resulting in rapid depressurization of the affected steam generator.

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REACTOR COOLANT SYSTEM

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

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 greater than or equal to 20% of the nominal wall thickness caused by degradation;
- 4) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;
- 5) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.
- 6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service 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 4.4.5.3c, above;
- 8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg, or from the point of entry (cold leg side) completely around the U-bend and to the bottom of the hot leg and

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REACTOR COOLANT SYSTEM

STEAM GENERATORS

SURVEILLANCE REQUIREMENTS (Continued)

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing.

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 the completion of the inspection. This Special Report shall include:

- 1) Number and extent of tubes inspected,
- 2) Location and percent of wall-thickness penetration for each indication of an imperfection, and
- 3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to 10 CFR Part 50.72 and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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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	Three	Three
First Inservice Inspection	All	Two
Second & Subsequent Inservice Inspections	One ¹	One ²

Table Notation

1. The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 9% of the tubes 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 instruction described in 1 above.

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TABLE 4-2

STEAM GENERATOR TUBE INSPECTION

SAMPLE SIZE	1st SAMPLE INSPECTION		2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
	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 inspect additional 4S tubes in this S.G.	C-1	None
			C-3	Perform action for C-3 result of first sample	C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
C-3	Inspect all tubes in this S.G. plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to Section 4.4.5.5c.	All other S.G.s are C-1	None	N/A	N/A	
		Some S.G.s C-2 but no additional S.G.s 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. Notification to NRC pursuant to Section 4.4.5.5c.	N/A	N/A	

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$S = \frac{9}{n} \%$ Where n is the number of steam generators inspected during an inspection.

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

- a. The Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System, and
- b. A Containment Sump Level Monitoring System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

- a. With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:
 - 1) A Containment Sump Level Monitoring System is OPERABLE; per Surveillance Requirement 4.4.6.2.1.c
 - 2) Appropriate grab samples are obtained and analyzed at least once per 24 hours;
 - 3) A Reactor Coolant System water inventory balance is performed at least once per 8 hours 8* during steady state operation except when operating in shutdown cooling mode; and
 - 4) Containment Purge, Exhaust and Instrument Air Bleed valves are maintained closed.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With no Containment Sump Level Monitoring System operable, restore at least one Containment Sump Level Monitoring System to OPERABLE status within 7 days, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection System shall be demonstrated OPERABLE by:

- a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3, and
- b. Containment Sump Level Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months.

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

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATING

operational

3.4.6.2 Reactor Coolant System leakage shall be limited to:

- a. No PRESSURE BOUNDARY LEAKAGE,
- b. 1 GPM UNIDENTIFIED LEAKAGE,
- c. 1 GPM total primary-to-secondary leakage through all steam generators and 500 gallons per day through any one steam generator, (SG)
- d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, and
- e. Leakage as specified in Table 3.4-1 up to a maximum of 5 GPM at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

150 gallons per day

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

or with primary-to-secondary leakage not within limit.

- a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. operational primary-to-secondary leakage.
- b. With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE, and leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than allowed by 3.4.6.2.e above operation may continue provided:
 - 1. Within 4 hours verify that at least two valves in each high pressure line having a non-functional valve are in, and remain in that mode corresponding to the isolated condition, i.e., manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supplies deenergized. Follow applicable ACTION statement for the affected system, and

* Test pressure less than 2235 psig are allowed. Minimum differential test pressure shall not be less than 150 psid. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

REACTOR COOLANT SYSTEM
OPERATIONAL LEAKAGE
LIMITING CONDITION FOR OPERATION (Continued)

2. The leakage* from the remaining isolating valves in each high pressure line having a valve not meeting the criteria of Table 3.4-1, as listed in Table 3.4-1, shall be determined and recorded daily. The positions of the other valves located in the high pressure line having the leaking valve shall be recorded daily unless they are manual valves located inside containment.

Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

- d. With any Reactor Coolant System Pressure Isolation Valve leakage greater than 5 gpm, reduce leakage to below 5 gpm within 1 hour, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

operational

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

- a. Monitoring the containment atmosphere gaseous or particulate radioactivity monitor at least once per 12 hours.

72*** ; and

- b. Monitoring the containment sump level at least once per 12 hours.

**

- c. Performance of a Reactor Coolant System water inventory balance within 12 hours after achieving steady-state operation** and at least once per 24 hours thereafter during steady-state operation, except that not more than 48 hours shall elapse between any two successive inventory balances; and

- d. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours; and

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage* to be within its limit:

- a. At least once per 18 months.
- b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months, and
- c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

e. Verifying primary-to-secondary leakage is < 150 gallons per day through any one SG 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.

* To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

** RCS average coolant temperature being changed by less than 5°F/hour.

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

- The combined As-left leakage rates determined on a maximum pathway leakage rate basis for all penetrations shall be verified to be less than $0.60 L_a$, prior to increasing primary coolant temperature above 200°F following an outage or shutdown that included Type B and Type C testing only.
 - The As-found leakage rates, determined on a minimum pathway leakage rate basis, for all newly tested penetrations when summed with the As-left minimum pathway leakage rate leakage rates for all other penetrations shall be less than $0.6 L_a$, at all times when containment integrity is required.
- 3) Overall air lock leakage acceptance criteria is $\leq 0.05 L_a$, when pressurized to P_a .

The provisions of Specification 4.0.2 do not apply to the test frequencies contained within the Containment Leakage Rate Testing Program.

i. Technical Specifications (TS) Bases Control Program

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

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
 - 1. Change in the TS incorporated in the license or
 - 2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- d. Proposed changes that meet the criteria of Specification 6.8.4 i.b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

INSERT
6.8.4.j

6.8.5 Administrative procedures shall be developed and implemented to limit the working hours of plant staff who perform safety-related functions, e.g. licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel. The procedures shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime for individuals.

Any deviation from the working hour guidelines shall be authorized by the applicable department manager or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant General Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the working hour guidelines shall not be authorized.

ADMINISTRATIVE CONTROLS - INSERT 6.8.4.j

j. 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 500 gallons per day through any one SG.

3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."
 - c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.
 - 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 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.
 3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.
 - e. Provisions for monitoring operational primary-to-secondary leakage.

ADMINISTRATIVE CONTROLS

3. WCAP-10054-P, Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection in the Broken Loop and Improved Condensation Model," October 1995.*
4. WCAP-12945-P, "Westinghouse Code Qualification Document For Best Estimate LOCA Analysis," Volumes I-V, June 1996.**
5. USNRC Safety Evaluation Report, Letter from R. C. Jones (USNRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis,' " June 28, 1996.**
6. Letter dated June 13, 1996, from N. J. Liparulo (W) to Frank R. Orr (USNRC), "Re-Analysis Work Plans Using Final Best Estimate Methodology."***
7. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and T. L. Ryan, April 1995.



The analytical methods used to determine Rod Bank Insertion Limits and the All Rods Out position shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

The ability to calculate the COLR nuclear design parameters are demonstrated in:

1. Florida Power & Light Company Topical Report NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants."

Topical Report NF-TR-95-01 was approved by the NRC for use by Florida Power & Light Company in:

1. Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 174 to Facility Operating License DPR-31 and Amendment No. 168 to Facility Operating License DPR-41, Florida Power & Light Company Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251.

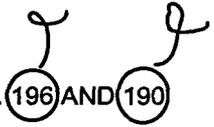
The AFD, F_Q(Z), F_AH, K(Z), and Rod Bank Insertion Limits shall be determined such that all applicable limits of the safety analyses are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector, unless otherwise approved by the Commission.

INSERT
6.9.1.8

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report as stated in the Specifications within Sections 3.0, 4.0, or 5.0.

*This reference is only to be used subsequent to NRC approval.
**As evaluated in NRC Safety Evaluation dated December 20, 1997.



ADMINISTRATIVE CONTROLS - INSERT 6.9.1.8

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.8 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.j, 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.

ENCLOSURE 3

Proposed Revised Technical Specification Pages

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DEFINITIONS

FREQUENCY NOTATION

DOSE EQUIVALENT I-131

1.12 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

E-AVERAGE DISINTEGRATION ENERGY

1.13 \bar{E} shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample isotopes, other than iodines, with half lives greater than 30 minutes, making up at least 95 percent of the total non-iodine activity in the coolant.

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

GAS DECAY TANK SYSTEM

1.15 A GAS DECAY TANK SYSTEM shall be any system designed and installed to reduce radioactive gaseous effluents by collecting Reactor Coolant System off gases from the Reactor Coolant System and providing for delay or holdup for the purpose of reducing the total radioactivity prior to release to the environment.

IDENTIFIED LEAKAGE

1.16 IDENTIFIED LEAKAGE shall be:

- a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or
- b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of Leakage Detection Systems or not to be PRESSURE BOUNDARY LEAKAGE, or
- c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System (primary-to-secondary leakage)

DEFINITIONS

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

1.18 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

1.19 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation: (1) described in Chapter 13.5 of the FSAR, (2) authorized under the provisions of 10 CFR 50.59, or (3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

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

PURGE - PURGING

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

TABLE 3.3-4 (Continued)
TABLE NOTATIONS

* During CORE ALTERATIONS or movement of irradiated fuel within the containment comply with Specification 3/4.9.13.

** With irradiated fuel in the spent fuel pits.

Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

Note 1 Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

Note 2 Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{F}$ CPM,

$$\text{Where } F = \frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in the Offsite Dose Calculation Manual.

ACTION STATEMENTS

ACTION 26 - In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

- 1) A Containment sump level monitoring system is OPERABLE,
- 2) Appropriate grab samples are obtained and analyzed at least once per 24 hours,
- 3) A Reactor Coolant System water inventory balance is performed per Surveillance Requirement 4.4.6.2.1.c at least once per 8^{***} hours except when operating in shutdown cooling mode, and
- 4) Containment Purge, Exhaust and Instrument Air Bleed Valves are maintained closed.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours (ACTION 27 applies in MODES 5 and 6).

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

REACTOR COOLANT SYSTEM

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

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

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REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

- a. The Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System, and
- b. A Containment Sump Level Monitoring System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

- a. With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:
 - 1) A Containment Sump Level Monitoring System is OPERABLE;
 - 2) Appropriate grab samples are obtained and analyzed at least once per 24 hours;
 - 3) A Reactor Coolant System water inventory balance is performed per Surveillance Requirement 4.4.6.2.1.c at least once per 8* hours except when operating in shutdown cooling mode; and
 - 4) Containment Purge, Exhaust and Instrument Air Bleed valves are maintained closed.Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With no Containment Sump Level Monitoring System operable, restore at least one Containment Sump Level Monitoring System to OPERABLE status within 7 days, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection System shall be demonstrated OPERABLE by:

- a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3, and
- b. Containment Sump Level Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months.

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

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATING

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 3.4-1 up to a maximum of 5 GPM at a Reactor Coolant System pressure of 2235 \pm 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

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 leakage from Reactor Coolant System Pressure Isolation Valves, reduce the leakage rate to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than allowed by 3.4.6.2.e above operation may continue provided:
 1. Within 4 hours verify that at least two valves in each high pressure line having a non-functional valve are in, and remain in that mode corresponding to the isolated condition, i.e., manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position and power supplies deenergized. Follow applicable ACTION statement for the affected system, and

* Test pressure less than 2235 psig are allowed. Minimum differential test pressure shall not be less than 150 psid. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

REACTOR COOLANT SYSTEM
OPERATIONAL LEAKAGE
LIMITING CONDITION FOR OPERATION (Continued)

2. The leakage* from the remaining isolating valves in each high pressure line having a valve not meeting the criteria of Table 3.4-1, as listed in Table 3.4-1, shall be determined and recorded daily. The positions of the other valves located in the high pressure line having the leaking valve shall be recorded daily unless they are manual valves located inside containment.

Otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

- d. With any Reactor Coolant System Pressure Isolation Valve leakage greater than 5 gpm, reduce leakage to below 5 gpm within 1 hour, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

- 4.4.6.2.1 Reactor Coolant System operational leakages shall be demonstrated to be within each of the above limits by:
- a. Monitoring the containment atmosphere gaseous or particulate radioactivity monitor at least once per 12 hours.
 - b. Monitoring the containment sump level at least once per 12 hours.
 - c.** Performance of a Reactor Coolant System water inventory balance at least once per 72*** hours; and
 - d. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours; and
 - e. Verifying primary-to-secondary leakage is < 150 gallons per day through any one SG at least once per 72*** hours.
- 4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage* to be within its limit:
- a. At least once per 18 months.
 - b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 7 days or more and if leakage testing has not been performed in the previous 9 months, and
 - c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve.

* To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

** Not applicable to primary-to-secondary leakage.

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

ADMINISTRATIVE CONTROLS

PROCEDURES AND PROGRAMS (Continued)

- The combined As-left leakage rates determined on a maximum pathway leakage rate basis for all penetrations shall be verified to be less than $0.60 L_a$, prior to increasing primary coolant temperature above 200°F following an outage or shutdown that included Type B and Type C testing only.
 - The As-found leakage rates, determined on a minimum pathway leakage rate basis, for all newly tested penetrations when summed with the As-left minimum pathway leakage rate leakage rates for all other penetrations shall be less than $0.6 L_a$, at all times when containment integrity is required.
- 3) Overall air lock leakage acceptance criteria is $\leq 0.05 L_a$, when pressurized to P_a .

The provisions of Specification 4.0.2 do not apply to the test frequencies contained within the Containment Leakage Rate Testing Program.

i. Technical Specifications (TS) Bases Control Program

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

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
 - 1. Change in the TS incorporated in the license or
 - 2. A change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- d. Proposed changes that meet the criteria of Specification 6.8.4 i.b. above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

j. 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

PROCEDURES AND PROGRAMS (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 500 gallons per day through any one SG.
 - 3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."
- c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.
- 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

PROCEDURES AND PROGRAMS (Continued)

2. Inspect 100% of the tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outages nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.
 3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.
- e. Provisions for monitoring operational primary-to-secondary leakage.

6.8.5 Administrative procedures shall be developed and implemented to limit the working hours of plant staff who perform safety-related functions, e.g. licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel. The procedures shall include guidelines on working hours that ensure that adequate shift coverage is maintained without routine heavy use of overtime for individuals.

Any deviation from the working hour guidelines shall be authorized by the applicable department manager or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant General Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the working hour guidelines shall not be authorized.

ADMINISTRATIVE CONTROLS

3. WCAP-10054-P, Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection in the Broken Loop and Improved Condensation Model," October 1995.*
4. WCAP-12945-P, "Westinghouse Code Qualification Document For Best Estimate LOCA Analysis," Volumes I-V, June 1996.**
5. USNRC Safety Evaluation Report, Letter from R. C. Jones (USNRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945(P) Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis," June 28, 1996.**
6. Letter dated June 13, 1996, from N. J. Liparulo (W) to Frank R. Orr (USNRC), "Re-Analysis Work Plans Using Final Best Estimate Methodology."***
7. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and T. L. Ryan, April 1995.

The analytical methods used to determine Rod Bank Insertion Limits and the All Rods Out position shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

The ability to calculate the COLR nuclear design parameters are demonstrated in:

1. Florida Power & Light Company Topical Report NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants."

Topical Report NF-TR-95-01 was approved by the NRC for use by Florida Power & Light Company in:

1. Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 174 to Facility Operating License DPR-31 and Amendment No. 168 to Facility Operating License DPR-41, Florida Power & Light Company Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251.

The AFD, $F_Q(Z)$, $F_{\Delta H}$, $K(Z)$, and Rod Bank Insertion Limits shall be determined such that all applicable limits of the safety analyses are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector, unless otherwise approved by the Commission.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.8 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.j, Steam Generator (SG) Program. The report shall include:

- a. The scope of inspections performed on each SG,
- b. Active degradation mechanisms found,

*This reference is only to be used subsequent to NRC approval.

**As evaluated in NRC Safety Evaluation dated December 20, 1997.

ADMINISTRATIVE CONTROLS

STEAM GENERATOR TUBE INSPECTION REPORT (Cont'd)

- 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 Regional Administrator of the Regional Office of the NRC within the time period specified for each report as stated in the Specifications within Sections 3.0, 4.0, or 5.0.

ENCLOSURE 4

Marked Up Pages and Inserts for Technical Specification

Bases Control Program, 0-ADM-536

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(SG) TUBE INTEGRITY

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TECHNICAL SPECIFICATION BASES

3/4.4 REACTOR COOLANT SYSTEM (Continued)

INSERT B3/4.4.5

3/4.4.5 STEAM GENERATORS

(SG) TUBE INTEGRITY

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day per steam generator). Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with the all volatile treatment (AVT) of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

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(SG) TUBE INTEGRITY

TECHNICAL SPECIFICATION BASES

3/4.4 REACTOR COOLANT SYSTEM (Continued)

INSERT B3/4.4.5

3/4.4.5 STEAM GENERATORS (Continued)

~~Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission in a Special Report pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.~~

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary to the containment. The containment sump level system is the normal sump level instrumentation. The Post Accident Containment Water Level Monitor - Narrow range instrumentation also functions as a sump level monitoring system. In addition, gross leakage will be detected by changes in makeup water requirements, visual inspection, and audible detection. Leakage to other systems will be detected by activity changes (e.g., within the component cooling system) or water inventory changes (e.g., tank levels).

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

3/4.4.6.2 OPERATIONAL LEAKAGE

~~PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.~~

~~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 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 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.~~

~~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.~~

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TECHNICAL SPECIFICATION BASES

3/4.4 REACTOR COOLANT SYSTEM (Continued)

Relocated to B3/4.4.6.2, ACTION c.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

~~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. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested periodically to ensure low probability of gross failure.~~

~~The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.~~

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady-State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady-State Limits.

The Surveillance Requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

INSERT B3/4.4.5 (0-ADM-536 - Technical Specification Bases Control Program)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.j, "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.j, 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.j. 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 500 gpd for the two intact SGs plus the leakage rate associated with a double-ended rupture of a single tube in the third (ruptured) SG. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via safety valves or atmospheric dump valves. No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip and concurrent loss of offsite power.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In the dose consequence analysis for these events the activity level in the steam discharged to the atmosphere is based on a primary-to-secondary leakage rate of 1 gpm through all SGs and 500 gpd through any one SG at accident conditions, which is conservative and bounds the allowable operational leakage rate as an initial condition and considers any leakage changes as a result of the accident induced changes in primary-to-secondary pressure differential. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.8, "Reactor Coolant System Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3), 10 CFR 50.67 (Ref. 7) or the NRC approved licensing basis.

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.j and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. Failure to meet any one of these criteria is considered failure to meet the LCO.

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant 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 any changes in primary-to-secondary pressure differential during a design basis accident, other than a SGTR, is considered in the accident dose consequences analysis. The limiting accident analysis (steam line break) assumes that accident induced leakage does not exceed 500 gpd in any one of the three SGs and the total leakage from all SGs does not exceed 1 gpm at accident conditions. This accident induced leakage rate assumption conservatively bounds the expected total accident primary-to-secondary leakage based on the allowable operational leakage rate as an initial condition and considers any leakage changes as a result of the accident induced changes in primary-to-secondary pressure differential.

The 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 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 MODE 1, 2, 3, or 4.

Reactor Coolant System conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary-to-secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage.

ACTIONS

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

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 outage 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 MODE 4 following the next refueling outage or SG inspection. This allowable outage time is acceptable since operation until the next inspection is supported by the operational assessment.

b.

If the requirements and associated allowable outage 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 outage 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.j 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.j 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, "Accident source term"

INSERT B3/4.4.6.2 (0-ADM-536 - Technical Specification Bases Control Program)**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 UFSAR (Ref. 1) describes the design criteria requirement to provide means for detecting significant uncontrolled leakage from the reactor coolant pressure boundary (RCPB). Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS leakage varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant leakage into the containment area is necessary. Quickly separating the IDENTIFIED LEAKAGE from the UNIDENTIFIED LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the 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 (SGs) is 1 gpm or 500 gallons per day from a single SG at accident conditions. 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. 6) or the NRC approved licensing basis. The LCO requirement to limit primary-to-secondary leakage through any one SG to less than or equal to 150 gpd at room temperature is significantly less than the conditions assumed in the safety analysis.

INSERT B3/4.4.6.2 (0-ADM-536 - Technical Specification Bases Control Program)

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released to the atmosphere via the atmospheric dump valves and/or main steam safety valves for a limited period of time. Operator action is taken to isolate the affected SG within the time period. The 500 gpd primary-to-secondary leakage in each SG at accident conditions assumed in the safety analysis assumption is relatively inconsequential.

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

Limiting Condition for Operation (LCO)

RCS operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

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

b. UNIDENTIFIED LEAKAGE

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

INSERT B3/4.4.6.2 (0-ADM-536 - Technical Specification Bases Control Program)**c. IDENTIFIED LEAKAGE**

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

d. Primary-to-Secondary Leakage Through Any One SG

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

e. RCS 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 MODES 1, 2, 3, and 4, the potential for reactor coolant PRESSURE BOUNDARY LEAKAGE is greatest when the RCS is pressurized.

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

INSERT B3/4.4.6.2 (0-ADM-536 - Technical Specification Bases Control Program)**ACTIONS****a.**

If any **PRESSURE BOUNDARY LEAKAGE** exists, or primary-to-secondary leakage is not within limit, the reactor must be brought to lower pressure conditions to reduce the severity of the leakage and its potential consequences. It should be noted that Leakage past seals and gaskets is not **PRESSURE BOUNDARY LEAKAGE**. The reactor must be brought to **HOT STANDBY** within 6 hours and **COLD SHUTDOWN** within the next 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 allowable outage time allows time to verify leakage rates and either identify **UNIDENTIFIED LEAKAGE** or reduce leakage to within limits before the reactor must be shut down. This **ACTION** is necessary to prevent further deterioration of the RCPB.

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 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 3.4-1 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 next 30 hours.

d.

With one or more RCS Pressure Isolation Valves with leakage greater than 5 gpm, the leakage must be reduced to below 5 gpm within 1 hour or the reactor must be brought to at least **HOT STANDBY** within 6 hours and **COLD SHUTDOWN** within the next 30 hours.

The allowable outage times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. In **MODE 5**, the pressure stresses acting on the RCPB are much lower, and further deterioration is much less likely.

INSERT B3/4.4.6.2 (0-ADM-536 - Technical Specification Bases Control Program)**Surveillance Requirements****SR 4.4.6.2.1**

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 and near operating pressure. The Surveillance is modified by two notes. Note *** states that this SR is not required to be performed until 12 hours after establishment of steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

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."

Note ** 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.

INSERT B3/4.4.6.2 (0-ADM-536 - Technical Specification Bases Control Program)

e.

This SR verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one SG. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.5, "Steam Generator (SG) Tube Integrity," 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 SG. If it is not practical to assign the leakage to an individual SG, all the primary-to-secondary leakage should be conservatively assumed to be from one SG.

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

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

SR 4.4.6.2.2

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.

This SR 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.

References

1. UFSAR, Sections 4.1.3 and 6.5.1
2. Regulatory Guide 1.45, May 1973
3. UFSAR, Section 14.2.4.1
4. NEI 97-06, "Steam Generator Program Guidelines"
5. EPRI "PWR Primary-to-Secondary Leak Guidelines"
6. 10 CFR 50.76, "Accident source term"

ATTACHMENT 1
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Enclosure 4, Bases Mark up and Inserts
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TECHNICAL SPECIFICATION BASES

3/4.4 REACTOR COOLANT SYSTEM (Continued)

1 gpm total through all steam generators and 500 gpd through any one steam generator

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of ~~1 gpm~~. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Turkey Point site, Units 3 and 4 site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

primary

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

The sample analysis for determining the gross specific activity and \bar{E} can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 30 minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 30 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.