Progress Energy

OCT 03 2006

Serial: HNP-06-116
10 CFR 50.90

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

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT NO. 1
DOCKET NO. 50-400/LICENSE NO. NPF-63
RESPONSE TO THE REQUEST FOR ADDITIONAL INFORMATION (RAI) REGARDING THE LICENSE AMENDMENT REQUEST APPLICATION FOR TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM GENERATOR TUBE INTEGRITY

Ladies and Gentlemen:

On August 29, 2006, the NRC requested additional information to facilitate the review of the proposed request (HNP-06-060 dated May 23, 2006) for a license amendment to the Technical Specifications (TS) of the Harris Nuclear Plant (HNP). The proposed amendment would revise the TS requirements related to steam generator tube integrity consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity."

Attachment 1 provides the response to the RAI.

Attachment 2 provides the proposed TS changes discussed in the response to the RAI.

Attachment 3 provides the revised TS pages discussed in the response to the RAI.

Attachment 4 provides the proposed TS Bases changes discussed in the response to the RAI (for information only).

For clarity, Attachment 4 has been resubmitted in its entirety and completely replaces the same attachment from the original letter.

Since the proposed revision provided by this submittal does not change the intent or the justification for the requested amendment, HNP has determined that this revision does not result in any change to the No Significant Hazards Consideration contained in the original letter. Therefore, the 10 CFR 50.92 Evaluation provided in the May 23, 2006 HNP letter remains valid.

HNP requests that the proposed amendment be issued prior to May 31, 2007, with the amendment being implemented within 90 days, as originally requested.

Progress Energy Carolinas, Inc.
Harris Nuclear Plant
P. O. Box 165
New Hill, NC 27562

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In addition, this document contains no new or revised Regulatory Commitments.

Please refer any question regarding this submittal to Mr. Dave Corlett at (919) 362-3137.

I declare, under penalty of perjury, that the attached information is true and correct (Executed on OCT 03 2006).

Sincerely,

C.S. Kamilaris

C. S. Kamilaris
Manager, Support Services
Harris Nuclear Plant

CSK/jpy

Attachments:

1. Response to the Request for Additional Information (RAI) Regarding the License Amendment Request Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity
2. Proposed Technical Specifications (TS) Changes
3. Revised Technical Specifications (TS) Pages
4. Proposed Technical Specification (TS) Bases Changes (For Information Only)

c:
Mr. R. A. Musser, NRC Senior Resident Inspector
Ms. B. O. Hall, N.C. DENR Section Chief
Ms. B. L. Mozafari, NRC Project Manager
Dr. W. D. Travers, NRC Regional Administrator

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TECHNICAL SPECIFICATION IMPROVEMENT REGARDING STEAM
GENERATOR TUBE INTEGRITY

Request 1:

The existing bases for the Harris 1 technical specifications (TS) reference 10 CFR part 100 guideline values for dose consequences in the event of a steam generator tube rupture (SGTR) or steam line break (page B 3/4 4-4). The proposed revision to the bases (insert 3/4.4.5) references 10 CFR 50.67 for an SGTR event. Please explain the reason for this change. In addition, please discuss why there was no reference to GDC 19 or your approved licensing basis. (Alternately, please modify your proposal to be consistent with TSTF 449.)

Response 1:

Page B 3/4 4-4 has been revised to delete reference to 10 CFR part 100, and this revised page has been included in the proposed Technical Specification (TS) Bases Changes (Attachment 4) of this letter. HNP Technical Specification Amendment No. 107, dated October 12, 2001 (Ascension No. ML012830516 and TAC Nos. MB0199 and MB0782), adopted the alternate source term (AST) methodology, using the guidance of NRC Regulatory Guide 1.183. As part of implementing the AST, the total effective dose equivalent (TEDE) acceptance criteria in the Standard Review Plan (SRP) Section 15.0.1 and 10 CFR 50 67 replaced the previous whole body and thyroid dose guidelines provided in 10 CFR 100 and GDC 19. Therefore, GDC 19 and 10 CFR 100 are not applicable in this section because 10 CFR 50.67 is the approved licensing basis for HNP.

Request 2:

Action statement 3.4.5.a in the proposed TS insert 3/4.4.5 states, in part, that the reactor shall be in hot standby within 6 hours and cold shutdown within the next 30 hours. However, the proposed bases Insert 3/4.4.5 BASES, corresponding to this portion of the TS, states that the reactor must be brought to hot standby within 6 hours and cold shutdown within 36 hours. Please discuss your plans to modify your proposed bases to be consistent with the proposed TS. Alternately, discuss your plans to modify the proposed TS to be consistent with the proposed bases.

Response 2:

Page B 3/4 4-5 has been revised to be consistent with the proposed TS (i.e., cold shutdown within the next 30 hours), and this revised page has been included in the proposed TS Bases changes (Attachment 4) of this letter.

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GENERATOR TUBE INTEGRITY

Request 3:

In the proposed TS Insert 3/4.4.5, the second part of TSTF-449 Condition B is not included in the SG Tube Integrity LCO. Condition B in the TSTF is as follows: "Required action and associated completion time of Condition A not met OR SG tube integrity not maintained." In the proposed TS for Harris 1, the second part of this condition is excluded (i.e., "SG tube integrity not maintained"). Please provide justification for removing the key requirement to shut down the reactor if SG tube integrity is not being maintained, or alternatively, discuss your plans to modify your proposed TS LCO to include this key requirement and be consistent with TSTF-449.

Response 3:

TS 3/4 4.5 has been revised to be consistent with TSTF-449, and this revised page has been included in the proposed TS changes (Attachment 2) and the revised TS pages (Attachment 3) of this letter. The associated TS Bases (Attachment 4) have also been updated to be consistent with these proposed changes.

Request 4:

The information contained in several paragraphs in section 3/4.4.6.2, page B3/4 4-4 of the existing bases is repeated in the proposed Insert 3/4.4.6.2 BASES. Specifically, the fourth and fifth paragraphs are repeated on page 4 of 7 in the insert, and the sixth paragraph is repeated on page 6 of 7 in the insert. Please discuss your plans to eliminate any redundancy in the bases.

Response 4:

The three redundant paragraphs have been deleted from the existing page B 3/4 4-4 mark-up, and this revised page has been included in the proposed TS Bases changes (Attachment 4) of this letter.

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Request 5:

The action statements on page 5 of 7 in Insert 3/4.4.5 BASES are not consistent with the action statements in TS Insert 3/4.4.5. The action statements in the proposed bases are taken directly from TSTF 449, however the action statements in the proposed TS were altered from their tabular form in order to maintain the format of the Harris 1 TS. As a result, some actions that are part of action (a) in the proposed TS are listed under action (b) in the proposed bases. Similarly, some actions associated with action (b) in the proposed TS are listed under action (a) in the proposed bases. Please discuss your plans to ensure consistency between the proposed TS and the proposed bases.

Response 5:

The action statements on page 5 of 7 of Insert 3/4.4.5 Bases have been revised to be consistent with the proposed TS, and this revised page has been included in the proposed TS Bases changes (Attachment 4) of this letter.

Request 6:

The staff identified several inconsistencies and/or typographical errors between the proposed TS and TSTF 449. Please discuss your plans to make the proposed TS and bases consistent with TSTF 449. Alternately, provide a justification for the inconsistencies.

a.) Insert 6.9.1.7,first paragraph, "completion of a steam generator tube inspection" should read "completion of an inspection"

b.) Insert 3/4.4.5 BASES, page 1 of 7, third paragraph, "mechanically phenomena" should read "mechanically induced phenomena"

c.) Insert 3/4.4.5 BASES, page 3 of 7, last paragraph, "(upset conditions)" should read "(upset or abnormal conditions)"

d.) Insert 3/4.4.6.2 BASES, page 2 of 7,"...primary to secondary leakage from all steam generators is 1 gpm" should read, "primary to secondary leakage from all steam generators is 1 gpm, or is assumed to increase to 1 gpm as a result of accident induced conditions"

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e.) Insert 3/4.4.6.2 BASES, page 3 (not 2) of 7, "rate criterion is conjunction" should read, "rate criterion in conjunction"

f.) Insert 3/4.4.6.2 BASES, page 5 of 7, last sentence, "by performance or a RCS" should read, "by performance of a RCS"

g.) Insert 3/4.4.6.2 BASES, page 6 of 7, first paragraph, "ank levels, makeup letdown," should read, "tank levels, makeup and letdown,"

h.) Proposed TS 4.4.6.2.3, page 3/4 4-24, "verified ≤150 gallons per day" should read "verified to be ≤150 gallons per day"

Response 6:

The proposed TS (i.e., a and h) and the proposed TS Bases (i.e., b, d, e, f, and g) referenced in the request above have been revised to be consistent with the intent of TSTF-449, and these revised pages have been included in the proposed TS, revised TS, and proposed TS Bases changes (Attachments 2, 3, and 4), respectively, of this letter. However, the proposed TS Bases (i.e., c) on page 3 of 7 of Insert 3/4.4.5 Bases, last paragraph, referenced in the request above, was not revised because the HNP FSAR does not include "or abnormal" in the loading conditions for American Society of Mechanical Engineers (ASME) Class 1 components.

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Request 7:

The proposed bases have several changes that go beyond the bases in TSTF 449. Please confirm that the proposed bases are consistent with your current design and licensing basis.

Response 7:

The proposed bases were reviewed, and HNP confirms that the proposed bases are consistent with the current design and licensing basis. For example, in the bases of TSTF-449, the accident radiological analysis for a Steam Generator Tube Rupture (SGTR) assumes that the ruptured Steam Generator (SG) secondary fluid is only briefly released to the atmosphere via safety valves and that the majority is discharged to the main condenser. However, in the bases for HNP (page 5 of 7 of Insert 3/4.4.5 Bases), the accident radiological analysis for a SGTR assumes that the ruptured SG secondary fluid is released directly to the atmosphere due to a failure of the SG Power-Operated Relief Valve (PORV) in the open position. This difference is consistent with the HNP design. In addition, some of the TS Bases information is not from the TSTF, but it is from the existing TS Bases and from other non-Improved Technical Specifications (non-ITS) submittals.

Request 8:

On page 3/4 4-21 of the proposed TS you proposed to include the statement "...per Surveillance Requirement 4.4.6.2.1.d..." under Action Statement #3. The purpose of adding this statement in this section is not clear since SR 4.4.6.2.1.d has no additional details. Please explain the purpose of adding this statement or discuss your plans to remove this statement from page 3/4 4-21.

Response 8:

For clarity, the statement on page 3/4 4-21 of the proposed TS (i.e., TS 3.4.6.1 ACTION c.3) referenced in the request above was removed, and instead, the additional information from the footnote of SR 4.4.6.2.1.d was added to the footnote of TS 3.4.6.1 ACTION c.3.

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PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGES

PROPOSED TECHNICAL SPECIFICATIONS (TS) CHANGES

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Steam generator tube integrity shall be maintained.

AND

All steam generator tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator Program.

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

INSERT A

ACTION*:

a. With one or more steam generator tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program, within 7 days verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection, or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

AND

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

SURVEILLANCE REQUIREMENTS

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

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

* Separate ACTION entry is allowed for each SG tube.

INSERT A

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.

AND

b. With the requirements and associated allowed 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

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 Airborne Gaseous Radioactivity Monitoring System,
b. The Reactor Cavity Sump Level and Flow Monitoring System, and
c. The Containment Airborne Particulate Radioactivity Monitoring System.

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

ACTION:

a. With a. or c. of the above required Leakage Detection Systems INOPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for airborne gaseous and particulate radioactivity at least once per 24 hours when the required Airborne Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With b. of the above required Leakage Detection Systems inoperable be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a. and c. of the above required Leakage Detection Systems inoperable:

   1. Restore either Monitoring System (a. or c.) to OPERABLE status within 72 hours and

   2. Obtain and analyze a grab sample of the containment atmosphere for gaseous and particulate radioactivity at least once per 24 hours, and

   3. Perform a Reactor Coolant ~~System/~~water inventory balance ~~per Surveillance Requirement 4.4.6.2.1.d~~ at least once per 8 hours*.

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

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

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 Airborne Gaseous or Particulate Radioactivity Monitor at least once per 12 hours;

b. Monitoring the containment sump inventory and Flow Monitoring System at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours*; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 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,

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

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

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

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

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

6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

n. Mechanical Design Methodologies

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

ANF-81-58(P)(A), "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," approved version as specified in the COLR.

XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup," approved version as specified in the COLR.

ANF-88-133(P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnups of 62 GWd/MTU," approved version as specified in the COLR.

XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results," approved version as specified in the COLR.

EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs," approved version as specified in the COLR.

(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

an

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of a steam generator tube inspection performed in accordance with Specification 6.8.4.l. 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,

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REVISED TECHNICAL SPECIFICATIONS (TS) PAGES

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3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 Steam generator tube integrity shall be maintained.

AND

All steam generator tubes satisfying the tube repair criteria shall be plugged in accordance with the Steam Generator 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.

AND

b. With the requirements and associated allowed 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 steam generator tube integrity in accordance with the Steam Generator Program.

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

* Separate ACTION entry is allowed for each SG tube.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

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a. The Containment Airborne Gaseous Radioactivity Monitoring System,
b. The Reactor Cavity Sump Level and Flow Monitoring System, and
c. The Containment Airborne Particulate Radioactivity Monitoring System.

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

ACTION:

a. With a. or c. of the above required Leakage Detection Systems INOPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed for airborne gaseous and particulate radioactivity at least once per 24 hours when the required Airborne Gaseous or Particulate Radioactivity Monitoring System is inoperable; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With b. of the above required Leakage Detection Systems inoperable be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a. and c. of the above required Leakage Detection Systems inoperable:

   1. Restore either Monitoring System (a. or c.) to OPERABLE status within 72 hours and

   2. Obtain and analyze a grab sample of the containment atmosphere for gaseous and particulate radioactivity at least once per 24 hours, and

   3. Perform a Reactor Coolant System water inventory balance at least once per 8 hours*.

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

---

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

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 Airborne Gaseous or Particulate Radioactivity Monitor at least once per 12 hours;

b. Monitoring the containment sump inventory and Flow Monitoring System at least once per 12 hours;

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4;

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours*; and

e. Monitoring the Reactor Head Flange Leakoff System at least once per 24 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,

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

d. Within 24 hours following valve actuation due to automatic or manual action or flow through the valve.

The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

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

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

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6.9.1.6 CORE OPERATING LIMITS REPORT (Continued)

n. Mechanical Design Methodologies

XN-NF-81-58(P)(A), "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," approved version as specified in the COLR.

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(Methodologies for Specification 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor).

6.9.1.6.3 The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.4 The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements, shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

6.9.1.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into HOT SHUTDOWN following completion of an inspection performed in accordance with Specification 6.8.4.1. 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,

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PROPOSED TECHNICAL SPECIFICATIONS (TS) BASES CHANGES
(FOR INFORMATION ONLY)

PROPOSED TECHNICAL SPECIFICATIONS (TS) BASES CHANGES (FOR INFORMATION ONLY)

RELIEF VALVES (Continued)

Surveillance Requirement 4.4.4.3 provides assurance of operability of the accumulators and that the accumulators are capable of supplying sufficient air to operate the PORV(s) if they are needed for RCS pressure control and normal air and nitrogen systems are not available.

Surveillance Requirement 4.4.4.2 addresses the block valves. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with ACTION statements "b" or "c". This precludes the need to cycle the valves with a full system differential pressure or when maintenance is being performed to restore an inoperable PORV to OPERABLE status.

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.

INSERT 3/4.4.5 BASES

INSERT 3/4.4.5 BASES

REACTOR COOLANT SYSTEM

BASES

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

Background

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. 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, "Reactor Coolant System, Hot Standby," LCO 3.4.1.3, "Reactor Coolant System, Hot Shutdown," and LCO 3.4.1.4.1, "Reactor Coolant System, Cold Shutdown-Loops Filled."

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

SG tubing is subject to a variety of degradation mechanisms. SG tubes may experience tube degradation related to corrosion phenomena, such as wastage, pitting, intergranular attack, and stress corrosion cracking, along with other mechanically induced phenomena such as denting and wear. These degradation mechanisms can impair tube integrity if they are not managed effectively. The SG performance criteria are used to manage SG tube degradation.

Specification 6.8.4.l, "Steam Generator Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.4.l, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. The SG performance criteria are described in Specification 6.8.4.l. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

The processes used to meet the SG performance criteria are defined by the Steam Generator Program Guidelines (Reference 1).

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

Applicable Safety Analyses

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary-to-secondary leakage rate equal to 1 gpm, plus the leakage rate associated with a double-ended rupture of a single tube. The accident radiological analysis for a SGTR assumes the ruptured SG secondary fluid is released directly to the atmosphere due to a failure of the PORV in the open position.

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 some analyses developed by the industry, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions. The HNP accident analyses assume the amount of primary to secondary steam generator tube leakage is 1 gpm. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the limits in LCO 3.4.8, "Reactor Coolant System Specific Activity." For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of 10 CFR 50.67 (Reference 2).

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 an 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.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.4.l and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

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

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load verses displacement curve where the slope of the curve becomes zero." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A, (normal operating conditions) and Service Level B, (upset conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Reference 3) and Draft Regulatory Guide 1.121 (Reference 4).

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

The accident induced leakage performance criterion ensures that the primary-to-secondary leakage caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 1 gpm total from all SGs. The accident induced leakage rate includes any primary-to-secondary leakage existing prior to the accident in addition to primary-to-secondary leakage induced during the accident.

The operational leakage performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational leakage is contained in LCO 3.4.6.2 and limits primary-to-secondary leakage through any one SG to 150 gpd. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of leakage is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODES 1, 2, 3, or 4.

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

ACTIONS

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

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

ACTIONS a.1 and a.2

ACTIONS a.1 and a.2 apply if it is discovered that one or more SG tubes examined in an Inservice Inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by Surveillance Requirement 4.4.5.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, ACTION b. applies.

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

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

ACTION b.

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

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

Surveillance Requirements

4.4.5.1 During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Reference 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 method used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.4.5.1. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Reference 5). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.4.l contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY (continued)

4.4.5.2 During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.8.4.l are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The frequency of "Prior to entering HOT SHUTDOWN following a SG 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 50.67

3. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB

4. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976

5. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines"

STEAM GENERATORS (Continued)

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 150 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 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

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

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission in a Special Report pursuant to Specification 4.4.5.5.c within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These Detection Systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

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.

INSERT 3/4.4.6.2 BASES

OPERATIONAL LEAKAGE (Continued)

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 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 150 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.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 31 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the safety analyses.

The maximum allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing 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.

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

INSERT 3/4.4.6.2 BASES

REACTOR COOLANT SYSTEM

BASES

3/4.4.6.2 OPERATIONAL LEAKAGE

Background

Components that contain or transport the coolant to or from the reactor core make up the reactor coolant system (RCS). Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant leakage, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational leakage LCO is to limit system operation in the presence of leakage from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of leakage.

10 CFR 50, Appendix A, GDC 30 (Reference 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant leakage. Regulatory Guide 1.45 (Reference 2) describes acceptable methods for selecting leakage detection systems.

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

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

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA).

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Applicable Safety Analyses

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for a LOCA; the amount of leakage can affect the probability of such an event. In some analyses developed by the industry, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute or is assumed to increase to 1 gallon per minute as a result of accident induced conditions. The HNP accident analyses assume the amount of primary to secondary steam generator tube leakage is 1 gpm. The LCO requirement to limit primary-to-secondary leakage through any one steam generator is limited to less than or equal to 150 gpd, which is significantly less than the conditions assumed in the safety analysis.

Primary-to-secondary leakage is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident or a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR analysis for a SGTR assumes the contaminated secondary fluid is released directly to the atmosphere due to a failure of the PORV in the open position and will continue atmospheric release until the time that the PORV can be isolated. The FSAR analysis for the SLB assumes that the SG with the failed steam line boils dry releasing all of the iodine directly to the environment and that iodine carried over to the faulted SG by tube leaks are also released directly to the environment until the RCS has cooled to below 212°F. The dose consequences resulting from the SGTR and the SLB accidents are within the limits defined in 10 CFR 50.67.

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

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

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

b. UNIDENTIFED LEAKAGE

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

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

The limit of 150 gpd per steam generator is based on the operational leakage performance criterion in NEI 97-06, Steam Generator Program Guidelines (Reference 3). The Steam Generator Program operational leakage performance criterion in NEI 97-06 states, "The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day." The limit is based on operating experience with steam generator tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFED LEAKAGE and is well within the capability of the Reactor Coolant System Makeup System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or CONTROLLED LEAKAGE. Violation of this LCO could result in continued degradation of a component or system.

e. CONTROLLED LEAKAGE

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 31 gpm with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analysis.

f. Reactor Coolant System Pressure Isolation Valve Leakage

The maximum allowable leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing 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.

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

Applicability

In MODES 1, 2, 3, and 4, the potential for RCPB leakage is greatest when the RCS is pressurized.

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

ACTIONS

a. If any PRESSURE BOUNDARY LEAKAGE exists, or primary-to-secondary leakage is not within limit, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the next 30 hours. This action reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

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

b. UNIDENTIFIED LEAKAGE, IDENTIFIED LEAKAGE, or CONTROLLED LEAKAGE in excess of the LCO limits must be reduced to within the limits within 4 hours. This completion time allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

Surveillance Requirements

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

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

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

Steady-state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational leakage determination by water inventory balance, steady-state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of PRESSURE BOUNDARY LEAKAGE or UNIDENTIFIED LEAKAGE is provided by the automatic systems that monitor containment atmosphere radioactivity and reactor cavity sump level. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Reactor Coolant System Leakage Detection Systems."

Part (d) notes that this SR is not applicable to primary-to-secondary leakage. This is 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.

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

3/4.4.6.2 OPERATIONAL LEAKAGE (continued)

4.4.6.2.3 This SR verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one steam generator. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5 should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 4. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator.

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

The 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 (Reference 4).

References

1. 10 CFR 50, Appendix A, GDC 30

2. Regulatory Guide 1.45, May 1973

3. NEI 97-06, "Steam Generator Program Guidelines"

4. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines"