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May 1, 2006

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: R.E. Ginna Nuclear Power Plant
Docket No. 50-244

License Amendment Request: Technical Specification Improvement Regarding
Steam Generator Tube Integrity Using the Consolidated Line Item Improvement
Process

In accordance with the provisions of 10 CFR 50.90 of Title 10 of the Code of Federal Regulations, R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC) is submitting a request for a license amendment to modify the Technical Specifications (TS) for the R.E. Ginna Nuclear Power Plant.

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

It has been determined that this amendment application does not involve a significant hazard consideration as determined by 10 CFR 50.92. Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of this amendment.

This proposed change to the Technical Specifications and our determination of significant hazards have been reviewed by our Plant Operations Review Committee (PORC) and Nuclear Safety Review Board (NSRB), and they have concluded that implementation of these changes will not result in an undue risk to the health and safety of the public.

Attachment (1) provides a description of the proposed change and confirmation of applicability.
Attachment (2) provides the existing TS and Bases pages marked up to show the proposed change. Final

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ADD1

TS Bases changes will be implemented pursuant to TS 5.5.13, Technical Specification Bases Control Program and are provided for information only. Ginna LLC will adopt these TS Bases changes upon implementation of the license amendment. This is the only commitment associated with this amendment request.

In accordance with 10 CFR 50.91, a copy of this application, with enclosures, is being provided to the designated New York State Official.

Ginna LLC requests approval of the proposed license amendment by May 1, 2007, with the amendment being implemented within 90 days.

Should you have questions regarding this matter, please contact Mr. Robert Randall at (585) 771-3734, or robert.randall@constellation.com.

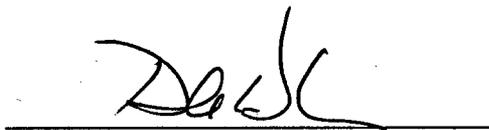
Very truly yours,



Dave A. Holm

STATE OF NEW YORK :
: TO WIT:
COUNTY OF WAYNE :

I, Dave A. Holm, begin duly sworn, state that I am Plant General Manager, R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC), and that I am duly authorized to execute and file this request on behalf of Ginna LLC. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other Ginna LLC employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.



Subscribed and sworn before me, a Notary Public in and for the State of New York and County of Wayne, this 1st day of May, 2006.

WITNESS my Hand and Notarial Seal:


Notary Public

My Commission Expires: 12-21-06

SHARON L. MILLER
Notary Public, State of New York
Registration No. 01MI6017755
Monroe County
Commission Expires December 21, 2006

- Attachments:** (1) **Description and Assessment**
(2) **Proposed Technical Specification and Bases Changes**

cc: S. J. Collins, NRC
P.D. Milano, NRC
Resident Inspector, NRC (Ginna)

J. P. Spath, NYSERDA
P.D. Eddy, NYSDPS

ATTACHMENT (1)

DESCRIPTION AND ASSESSMENT

ATTACHMENT (1)
DESCRIPTION AND ASSESSMENT

1.0 INTRODUCTION

The proposed license amendment revises the requirements in the Technical Specification (TS) related to steam generator tube integrity. The changes are consistent with Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4. The availability of this Technical Specification 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 PROPOSED CHANGE

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

- Revised TS 1.1, definition of LEAKAGE
- Revised TS 3.4.13, "RCS Operational Leakage"
- New TS 3.4.17, "Steam Generator (SG) Tube Integrity"
- Revised TS 5.5.8, "Steam Generator Program"
- New TS 5.6.7, "Steam Generator Tube Inspection Report"

Proposed revisions to the TS Bases are also included in this application. As discussed in the NRC's model safety evaluation, adoption of the revised TS Bases associated with TSTF-449, Revision 4 is an integral part of implementing this TS improvement. The changes to the affected TS Bases pages will be incorporated in accordance with TS 5.5.14, Technical Specification Bases Control Program.

3.0 BACKGROUND

The background for this application 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.

4.0 REGULATORY REQUIREMENTS AND GUIDANCE

The applicable regulatory requirements and guidance associated with this application are 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.

5.0 TECHNICAL ANALYSIS

R.E. Ginna Nuclear Power Plant, LLC (Ginna LLC). has reviewed the safety evaluation (SE) published on March 2, 2005 (70 FR 10298) as part of the CLIP 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. Ginna LLC has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are

ATTACHMENT (1)
DESCRIPTION AND ASSESSMENT

applicable to the R.E. Ginna Nuclear Power Plant (Ginna), and justify this amendment for the incorporation of the changes to the Ginna 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 Verification and Commitments

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

Plant Name, Unit No.	R.E. Ginna
Steam Generator Model(s)	Babcock & Wilcox Replacement Steam Generators
Effective Full Power Years (EFPY) of service for currently installed SGs	9.0
Tubing Material	690TT
Number of tubes per SG	4,765
Number and percentage of tubes plugged in each SG	SG A – 1 (0.02%), SG B – 5 (0.10%)
Number of tubes repaired in each SG	None
Degradation mechanism(s) identified	None
Current primary-to-secondary leakage limits	TS Criteria: <ul style="list-style-type: none"> • No pressure boundary leakage • 1 gpm unidentified leakage • 10 gpm identified leakage • 0.1 gpm total primary to secondary LEAKAGE through each steam generator (SG) when averaged over 24 hours.
Approved Alternate Tube Repair Criteria	None
Approved SG Tube Repair Methods	None
Performance criteria for accident leakage	Accident induced primary to secondary leak rate values (cold conditions) assumed in licensing basis accident analysis are 500 gpd per steam generator for the Locked Rotor and Rod Ejection accident, and 1 gpm per steam generator for the Steam Line Break accident.

7.0 NO SIGNIFICANT HAZARDS EVALUATION

Ginna LLC has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (70 FR 10298) as part of the CLIIP. Ginna LLC has concluded that

ATTACHMENT (1)
DESCRIPTION AND ASSESSMENT

the proposed determination presented in the notice is applicable to Ginna and the determination is hereby incorporated by Reference (1) to satisfy the requirements of 10 CFR 50.91(a).

8.0 ENVIRONMENTAL EVALUATION

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

9.0 PRECEDENT

This application is being made in accordance with the CLIIP. Ginna LLC 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).

10.0 REFERENCES

1. Federal Register Notices:
 - Notice for Comment published on March 2, 2005 (70 CFR 10298)
 - Notice of Availability published on May 6, 2005 (70 FR 24126)

ATTACHMENT (2)

PROPOSED TECHNICAL SPECIFICATION AND BASES CHANGES

INSERT 3.4.13 A

----- NOTE -----

1. Not required to be performed until 12 hours after establishment of steady state operation.
 2. Not applicable to primary to secondary LEAKAGE.
-

Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.

INSERT 3.4.13 B

----- NOTE -----

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

Verify primary to secondary LEAKAGE is \leq 150 gallons per day through any one SG.

INSERT B 3.4.13 A

The safety analysis for an event resulting in steam discharge to the atmosphere assumes that the secondary specific activity is equal to the limit in LCO 3.7.14, and this value is consistent with the LCO 3.4.13 primary to secondary LEAKAGE and the LCO 3.4.16 RCS Specific Activity.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident and other accidents or transients which involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR), reactor coolant pump locked rotor (LR), and a rod ejection (RE) accident. The leakage contaminates the secondary fluid.

The UFSAR (Ref. 3) analysis for SGTR assumes that the intact SG primary to secondary LEAKAGE is 150 gallons per day, which is relatively inconsequential. The safety analysis for the SLB accident assumes 1 gpm primary to secondary LEAKAGE in each SG as a result of the accident. The LR and RE accidents are assumed to result in a 500 gallon per day primary to secondary LEAKAGE in each SG as a result of the accident. The dose consequences resulting from the accidents outside of containment are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

INSERT B 3.4.13 B

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). The Steam Generator Program operational LEAKAGE performance criterion in NEI 97-06 states, "The RCS operational primary to secondary leakage through any one SG shall be

limited to 150 gallons per day.” The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.

INSERT B 3.4.13 C

The RCS water inventory balance must be met with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. Note 1 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.

INSERT B 3.4.13 D

Note 2 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.

INSERT B 3.4.13 E

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, “Steam Generator Tube Integrity,” should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

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 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).

INSERT B 3.4.13 F

4. NEI 97-06, Steam Generator Program Guidelines
5. EPRI, Pressurized Water Reactor Primary-to-Secondary Leak Guidelines

INSERT 5.5.8

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, plugged, to confirm that the performance criteria are being met.
- b. Performance criteria for SG tube integrity. Steam generator 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 steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down 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 a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for each SG. Leakage is not to exceed 1 gpm per SG.
 3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS 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 tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.
2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.
3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary to secondary LEAKAGE.

INSERT 5.6.7

5.6.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, 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,

LEAKAGE

LEAKAGE from the RCS shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or return), that is captured and conducted to collection systems or a sump or collecting tank;
2. 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
3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. Unidentified LEAKAGE

(primary to secondary LEAKAGE)

All LEAKAGE (except RCP seal water injection or return) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

primary to secondary

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

**MODE
- MODES**

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

**OPERABLE
- OPERABILITY**

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

- a. No pressure boundary LEAKAGE;
- b. 1 gpm unidentified LEAKAGE;
- c. 10 gpm identified LEAKAGE; and
- d. 0.1 gpm total primary to secondary LEAKAGE through each steam generator (SG) when averaged over 24 hours.

150 gallons per day

any one

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. <u>operational</u> RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits. <u>or primary to secondary LEAKAGE</u>	4 hours
B. Required Action and associated Completion Time not met. <u>OR</u> RCS pressure boundary LEAKAGE exists.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

OR
Primary to secondary LEAKAGE not within limit.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<p>SR 3.4.13.1</p> <p style="text-align: center;">NOTE</p> <p>Only required to be performed during steady state operation.</p> <p>Perform RCS water inventory balance.</p>	<p>Once during initial 12 hours of steady state operation</p> <p>AND</p> <p>72 hours thereafter</p>
<p>SR 3.4.13.2</p> <p>Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.</p>	<p>In accordance with the Steam Generator Tube Surveillance Program</p>

Insert
3.4.13A

Insert
3.4.13B

72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

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

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

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection. <u>AND</u> A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	7 days Prior to entering MODE 4 following the next refueling outage or SG tube inspection
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

5.5.7

Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components including applicable supports. The program shall include the following:

- a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

<u>ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities</u>	<u>Required Frequencies for performing inservice testing activities</u>
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

- b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;
- c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and
- d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

5.5.8

Steam Generator (SG) Tube Surveillance Program

Insert
5.5.8

~~Each SG shall be demonstrated OPERABLE by performance of an inservice inspection program in accordance with the Nuclear Policy Manual. This inspection program shall define the specific requirements of the edition and Addenda of the ASME Boiler and Pressure Code, Section XI, as required by 10 CFR 50.55a(g). The program shall include the following:~~

- ~~a. The inspection intervals for SG tubes shall be specified in the Inservice Inspection Program:~~

- ~~b. SG tubes that have imperfections > 40% through wall, as indicated by eddy current, shall be repaired by plugging or sleeving.~~
- ~~c. SG sleeves that have imperfections > 30% through wall, as indicated by eddy current, shall be repaired by plugging.~~

5.5.9

Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. This program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.10

Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature filter ventilation systems and the Spent Fuel Pool (SFP) Charcoal Adsorber System. The test frequencies will be in accordance with Regulatory Guide 1.52, Revision 2, except that in lieu of 18 month test intervals, a 24 month interval will be implemented. The test methods will be in accordance with Regulatory Guide 1.52, Revision 2, except as modified below.

- a. Containment Recirculation Fan Cooler System
 - 1. Demonstrate the pressure drop across the high efficiency particulate air (HEPA) filter bank is < 3 inches of water at a design flow rate ($\pm 10\%$).
 - 2. Demonstrate that an in-place dioctylphthalate (DOP) test of the HEPA filter bank shows a penetration and system bypass < 1.0%.

Insert
5.6.7



- d. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for revisions or supplement thereto.
-

Steady state DNB analysis has been performed for the two RCS loop operation. For two RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 109% RTP. This is the design overpower condition for two RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 118% and is based on an analysis assumption that bounds all possible instrumentation errors (Ref. 2). The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with both RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant. Adequate heat transfer between the reactor coolant and the secondary side is ensured by maintaining $\geq 16\%$ SG level in accordance with LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation," which provides sufficient water inventory to cover the SG tubes.

RCS Loops - MODE 1 > 8.5% RTP satisfies Criterion 2 of the NRC Policy Statement.

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, two pumps are required to be in operation at rated power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG ~~(in accordance with the Steam Generator Tube Surveillance Program).~~

APPLICABILITY

In MODE 1 > 8.5% RTP, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, both RCS loops are required to be OPERABLE and in operation in this MODE to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower MODES as indicated by the LCOs for MODES 1 $\leq 8.5\%$ RTP, 2, 3, 4, and 5.

The Note permits all RCPs to be de-energized for \leq 1 hour per 8 hour period in MODE 3. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test was satisfactorily performed during the initial startup testing program (Ref. 5). If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again.

The no flow test may be performed in MODE 3, 4, or 5. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed. The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

An OPERABLE RCS loop consists of an OPERABLE RCP and an OPERABLE SG in accordance with the ~~Steam Generator Tube Surveillance Program~~, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and able to provide forced flow if required.

Note 1 permits all RCPs and RHR pumps to be de-energized for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. One of the tests performed during the startup testing program was the validation of rod drop times during cold conditions, both with and without flow (Ref. 1). If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by test procedures:

- a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and
- b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that the pressurizer water volume be < 324 cubic feet (38% level), or that the secondary side water temperature of each SG be $\leq 50^\circ\text{F}$ above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature less than or equal to the LTOP enable temperature specified in the PTLR. The water volume limit ensures that the pressurizer will accommodate the swell resulting from an RCP start. Restraints on the pressurizer water volume and SG secondary side water temperature prevent a low temperature overpressure event due to a thermal transient when an RCP is started and the colder RCS water enters the warmer SG and expands. Violation of this Note places the plant in an unanalyzed condition.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~, which has the minimum water level specified in SR 3.4.6.2. RCPs are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

or equal to the LTOP enable temperature specified in the PTLR. The water volume limit ensures that the pressurizer will accommodate the swell resulting from an RCP start. Restraints on the pressurizer water volume and SG secondary side water temperature are to prevent a low temperature overpressure event due to a thermal transient when an RCP is started and the colder RCS water enters the warmer SG and expands. Violation of this Note places the plant in an unanalyzed Condition.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops. A planned heatup is a scheduled transition to MODE 4 within a defined time period.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. Also included are all necessary support systems not addressed by applicable LCOs (e.g., component cooling water and service water). A SG can perform as a heat sink when it is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program~~, with the minimum water level specified in SR 3.4.7.2.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The RCS loops are considered filled until the isolation valves are opened to facilitate draining of the RCS. The loops are also considered filled following the completion of filling and venting the RCS. However, in both cases, loops filled is based on the ability to use a SG as a backup. To be able to take credit for the use of one SG the ability to pressurize to 50 psig and control pressure in the RCS must be available. This is to prevent flashing and void formation at the top of the SG tubes which may degrade or interrupt the natural circulation flow path (Ref. 2). One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least one SG is required to be $\geq 16\%$.

APPLICABLE
SAFETY
ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event (Ref. 2).

Insert
B 3.4.13A

~~Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 0.5 gpm primary to secondary LEAKAGE as the initial condition. The leakage contaminates the secondary fluid.~~

~~The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The assumed 0.5 gpm primary to secondary LEAKAGE is relatively inconsequential.~~

~~The SLB outside of containment is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 0.5 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident outside of containment are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits). However, a lower LEAKAGE limit is assumed for all SLBs to prevent a coincident SGTR due to the large stresses placed on the SG tubes as a result of the rapid cooldown and depressurization. These stress calculations conservatively assume a tube with a 0.4 inch long through wall crack in a location with 40% local wall thinning. The analyses demonstrate that the integrity of the selected tube is maintained with sufficient margin after the SLB. The assumed through wall crack of 0.4 inches corresponds to 0.1 gpm leakage under normal operating conditions (Ref. 4). Therefore, the primary to secondary LEAKAGE is limited to 0.1 gpm per SG.~~

The RCS operational LEAKAGE satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS operational LEAKAGE shall be limited to:

1. 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.

2. 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.

3. 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 a charging pump operating at its low speed setting. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, LEAKAGE through two in-series PIVs, and primary to secondary LEAKAGE, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal return (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

4. ~~Primary to Secondary LEAKAGE through Each Steam Generator SG~~

~~Total primary to secondary LEAKAGE amounting to 0.1 gpm through each SG produces acceptable offsite doses and tube stresses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident or result in a coincident SGTR. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE. The SGs shall also be OPERABLE in accordance with the Steam Generator Tube Surveillance Program.~~

Insert
B 3.4.13 B

A PORV which is leaking ≥ 10 gpm must also be declared inoperable per LCO 3.4.11, "Pressurizer PORVs."

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 or 6, the temperature is $\leq 200^\circ\text{F}$ and pressure is maintained low or at atmospheric pressure. Since the temperatures and pressures are far lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is much smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the in-series PIVs in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leak tight. If both valves leak and result in a loss of mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE, ^{or} identified LEAKAGE, ~~or primary to secondary LEAKAGE~~ in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any RCS pressure boundary LEAKAGE exists, ^{or primary to secondary LEAKAGE is not within limit,} or if the Required Action of Condition A cannot be completed within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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

It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

SURVEILLANCE
REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE which is not allowed by this LCO, would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance. Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems.

Insert
B 3.4.13 C

The RCS water inventory balance must be performed with the RCS at steady state operating conditions. Therefore, this SR is required to be performed once during the initial 12 hours of steady state operation and every 72 hours thereafter.

Steady state operation is required to perform a proper inventory balance; calculations during maneuvering are not useful and a Note requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and volume control tank levels, makeup and letdown, and RCP seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the automatic systems that monitor the containment atmosphere radioactivity and the containment sump level. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Insert
B 3.4.13 D

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

SR 3.4.13.2

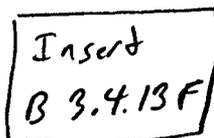
Insert
B 3.4.13 E

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. Atomic Industry Forum (AIF) GDC 16, Issued for comment July 10, 1967.
2. Generic Letter 84-04, "Safety Evaluation of Westinghouse Topical Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops."
3. UFSAR, ~~Section 15.6.3~~ Chapter 15.
4. ~~Letter from R. A. Purple, NRC, to L. D. White, RG&E, Subject: "Issuance of Amendment No. 7 to Provisional Operating License No. DPR-18," dated May 14, 1975.~~

Insert
B 3.4.13 F



B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

Steam generator (SG) tubes are small diameter, thin walled tubes that carry primary coolant through the primary to secondary heat exchangers. The SG tubes have a number of important safety functions. Steam generator tubes are an integral part of the reactor coolant pressure boundary (RCPB) and, as such, are relied on to maintain the primary system's pressure and inventory. The SG tubes isolate the radioactive fission products in the primary coolant from the secondary system. In addition, as part of the RCPB, the SG tubes are unique in that they act as the heat transfer surface between the primary and secondary systems to remove heat from the primary system. This Specification addresses only the RCPB integrity function of the SG. The SG heat removal function is addressed by LCO 3.4.4, "RCS Loops - MODES 1 > 8.5% RTP," LCO 3.4.5, "MODES 1 ≤ 8.5% RTP, 2, and 3," LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, 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.

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

Specification 5.5.8, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.8, 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 5.5.8. 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).

BASES

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 the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via safety valves and the atmospheric relief valves.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on primary to secondary LEAKAGE from each SG which is assumed to increase to 1 gpm (500 gallons per day for a locked rotor or rod ejection accident) as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

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.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator (SG) Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

BASES

LCO (continued)

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 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 (500 gallons per day for a locked rotor or rod ejection accident) per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

BASES

LCO (continued) 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 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 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.

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. 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, Condition B applies.

BASES

Actions (continued)

A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

If the evaluation determines that the affected tube(s) have tube integrity, Required 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 Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

If the Required Actions and associated Completion Times of Condition A are not met or if SG tube integrity is not being maintained, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours.

The 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.

**SURVEILLANCE
REQUIREMENTS**

SR 3.4.17.1

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.

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 3.4.17.1. 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 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

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 5.5.8 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

The Frequency of prior to entering MODE 4 following a SG inspection ensures that the Surveillance has been completed and all tubes meeting the repair criteria are plugged prior to subjecting the SG tubes to significant primary to secondary pressure differential.

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
 2. 10 CFR 50 Appendix A, GDC 19.
 3. 10 CFR 100.
 4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
 5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
 6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."
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