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SOUTHERN COMPANY
Energy to Serve Your World℠

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Joseph M. Farley Nuclear Plant
Request for Technical Specifications Change
Steam Generator Program - Revision 1

Ladies and Gentlemen:

By letter dated June 28, 2004, Southern Nuclear Operating Company (SNC) submitted a request to revise the subject technical specifications. Subsequent discussions with the NRC staff have prompted SNC to revise this submittal. The changes are minor and the significant hazards evaluation previously performed per 10 CFR 50.92 remains valid, but for convenience both the affected and unaffected pages are provided with this letter and SNC's June 28, 2004 submittal is superceded in its entirety.

In accordance with 10 CFR 50.90, SNC proposes to revise the Technical Specifications (TS) to establish a new Steam Generator Program for the Farley Nuclear Plant (FNP), Units 1 and 2, Facility Operating License Nos. NPF-2 and NPF-8. The proposed change is based on draft Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-449, Rev. 2. Revision of the FNP TS per TSTF-449 is necessary to implement guidance for the industry initiative on NEI 97-06, "Steam Generator Program Guidelines." In conjunction with this proposed change SNC plans to complete an analysis per Regulatory Guide 1.121 by September 1, 2004. This analysis will reflect forthcoming industry guidance relating to implementation of the SG tube structural integrity performance criterion.

The steam generators (SGs) were replaced at FNP Unit 1 during the spring 2000 refueling outage and at FNP Unit 2 during the spring 2001 refueling outage. The Westinghouse Model 54F replacement SGs used in both units incorporate significant improvements, including thermally treated Alloy 690 (690TT) tubing. One-time TS revisions permitting SG inspection interval extension were previously approved for FNP Units 1 and 2 by NRC letters dated September 20, 2002 and July 14, 2003, respectively.

Stress corrosion cracking, including circumferential cracking, is not expected to occur in the 690TT tubing used in the FNP SGs for a number of years. Service-induced stress corrosion cracking has not been detected in any 690TT tubing. The only cracking confirmed to have occurred in 600TT tubing was a result of high residual stresses in the tubing. Testing has indicated that 690TT tubing will have improved corrosion resistance compared to 600TT tubing.

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The Model 54F Replacement Steam Generator Stress Report Tube Analysis, Alabama Power Company, J. M. Farley Nuclear Plant, Units 1 and 2, WNEP-9380, Volume 7, dated January, 2000, documents compliance with ASME Code requirements for stress intensity limits, fatigue usage factor limit, external pressure collapse limits, and thermal stress ratchet limits. The analyses for the large steam line break faulted conditions and the LOCA faulted conditions included primary bending stresses. Additionally, these analyses include allowances for corrosion and wear.

A new section, TS 3.4.17, "SG Tube Integrity" is added to the FNP TS by the proposed change, while existing sections TS 3.4.13, "RCS Operational LEAKAGE," TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program" and TS 5.6.10, "Steam Generator Tube Inspector Report" are revised.

Enclosure 1 provides the basis for the proposed change, including an evaluation determining that the proposed change involves no significant hazards consideration as defined in 10 CFR 50.92. Enclosures 2 and 3 include the marked-up and clean-typed TS and Bases pages incorporating the proposed change.

SNC requests approval of the proposed change by September 17, 2004 to permit the SG inspections now scheduled to be performed during the FNP Unit 1 fall 2004 refueling outage to be rescheduled for spring 2006. Likewise, the Unit 2 SG inspections scheduled for fall 2005 would be rescheduled for spring 2007. The amendment will be implemented within 30 days of approval.

SNC has reviewed the proposed change pursuant to 10 CFR 50.92 and determined that it does not involve a significant hazards consideration. In addition, there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite and there is no significant increase in individual or cumulative occupational radiation exposure. Consequently, the proposed TS change has no significant effect on the human environment and satisfies the criteria of 10 CFR 51.22 for categorical exclusion from the requirements for an environmental assessment.

A copy of the proposed change has been sent to Dr. D. E. Williamson, the Alabama State Designee, in accordance with 10 CFR 50.91(b)(1).

This letter contains no NRC commitments. If you have any questions, please advise.

Mr. L. M. Stinson states he is a Vice President of Southern Nuclear Operating Company, is authorized to execute this oath on behalf of Southern Nuclear Operating Company and to the best of his knowledge and belief, the facts set forth in this letter are true.

Respectfully submitted,

SOUTHERN NUCLEAR OPERATING COMPANY

L. M. Stinson

L. M. Stinson

Sworn to and subscribed before me this 5th *day of* August *, 2004.*

Gloria H. Bui
Notary Public

My commission expires: 6-7-05

LMS/DWD/sdl

Enclosures: 1. Basis for Proposed Change
2. Marked-Up Technical Specifications and Bases Pages
3. Clean Typed Technical Specifications and Bases Pages

cc: Southern Nuclear Operating Company
Mr. J. B. Beasley, Jr., Executive Vice President
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U. S. Nuclear Regulatory Commission
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Mr. S. E. Peters, NRR Project Manager – Farley
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Alabama Department of Public Health
Dr. D. E. Williamson, State Health Officer

Enclosure 1 - Basis for the Proposed Change

1.0 DESCRIPTION

The proposed change revises the FNP Unit 1 and 2 Technical Specification (TS) sections as follows:

- Table of Contents
- TS 3.4.13, "RCS Operational LEAKAGE,"
- TS 5.5.9, "Steam Generator Tube Surveillance Program,
- TS 5.6.10, "Steam Generator Tube Inspector Report," and adds
- TS 3.4.17, "Steam Generator Tube Integrity." (New Section)

The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97-06, "Steam Generator Program Guidelines," (Reference 1).

2.0 PROPOSED CHANGE

The proposed change will:

- Revise Technical Specification Table of Contents

  The proposed change revises the Table of Contents to add a new Technical Specification 3.4.17 entitled "Steam Generator Tube Integrity." The TS Bases Table of Contents is likewise revised.

- Revise Technical Specification 3.4.13, "RCS Operational LEAKAGE"

  The proposed change revises TS 3.4.13, "RCS Operational LEAKAGE," to delete the existing LCO 3.4.13.d since it is enveloped by the existing LCO 3.4.13.e and revises the Conditions and Surveillances to clarify the requirements related to primary to secondary LEAKAGE.

  SR 3.4.13.2 is changed from verifying SG tube integrity to requiring verification that primary to secondary LEAKAGE is within the limit. SG tube integrity is verified under a new LCO. A new Note is added to SR 3.4.13.1 to indicate that this surveillance in not applicable to primary to secondary LEAKAGE. A Note is added to SR 3.4.13.2 stating that the SR is not required to be performed until 12 hours after establishment of steady state operation. This is consistent with the existing Note in SR 3.4.13.1.

  TS Bases changes are made to reflect the changes proposed to the Technical Specifications.

- Revise Technical Specification 5.5.9, "Steam Generator (SG) Tube Surveillance Program"

  The proposed change revises TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to delete the existing SG Tube Surveillance Program and to require a new Steam Generator Program to be established and implemented to ensure that SG tube integrity is maintained, and to describe SG condition monitoring, performance criteria and inspection intervals. The title of TS 5.5.9 is revised from "Steam Generator Tube Surveillance Program" to "Steam Generator (SG) Program."

- Revise Technical Specification 5.6.10, "Steam Generator Tube Inspector Report"

  The proposed change to TS 5.6.10, "Steam Generator Tube Inspector Report," revises the requirements for and contents of the SG tube inspection report. The reporting requirements are revised to require a report within 180 days of initial entry into MODE 4 following a steam generator inspection.

- Add Technical Specification 3.4.17, "Steam Generator Tube Integrity"

  The proposed change adds a new Technical Specification entitled "Steam Generator Tube Integrity," and associated Bases. The proposed Specification requires that SG tube integrity be maintained and requires that all SG tubes that satisfy the repair criterion be plugged in accordance with the Steam Generator Program.

- Revise the TS Bases for Specifications 3.4.4, 3.4.5, 3.4.6, and 3.4.7

  The TS Bases for TS 3.4.4, "RCS Loops – MODES 1 and 2," TS 3.4.5, "RCS Loops – MODE 3," TS 3.4.6, "RCS Loops – MODE 4," and TS 3.4.7, "RCS Loops – MODE 5, Loops Filled," are revised to eliminate the reference to the Steam Generator Tube Surveillance Program as the method of establishing Steam Generator OPERABILITY.

3.0 BACKGROUND

The SG tubes in pressurized water reactors 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 upon to maintain the primary system's pressure and inventory. As part of the RCPB, the SG tubes are unique in that they act as a heat transfer surface between the primary and secondary systems to remove heat from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system.

Steam generator tube integrity is necessary in order to satisfy the tubing's safety functions. Maintaining tube integrity ensures that the tubes are capable of performing their intended safety functions consistent with the plant licensing basis, including applicable regulatory requirements.

Concerns relating to the integrity of the tubing stem from the fact that the SG tubing is subject to a variety of degradation mechanisms. Steam generator tubes have experienced 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. When the degradation of the tube wall reaches a prescribed criterion, the tube is considered defective and corrective action is taken.

The criteria governing structural integrity of SG tubes were developed in the 1970s and assumed uniform tube wall thinning. This led to the establishment of a through wall SG tube repair criteria (e.g. ≥40 %) that has historically been incorporated into most pressurized water reactor (PWR) Technical Specifications and has been applied, in the absence of other repair criteria, to all forms of SG tube degradation where sizing techniques are available. Since the basis of the through wall depth criterion was 360° wastage, it is generally considered to be conservative for other mechanisms of SG tube degradation. The repair criterion does not allow licensees the flexibility to manage different types of SG tube degradation. Licensees must either use the through wall criterion for all forms of degradation or obtain approval for use of more appropriate repair criteria that consider the structural integrity implications of the given mechanism.

For the last several years, the industry, through the Electric Power Research Institute (EPRI) Steam Generator Management Program (SGMP), has developed a generic approach to improving SG performance referred to as "Steam Generator Degradation Specific Management" (SGDSM). Under this approach, different methods of inspection and different repair criteria may be developed for different types of degradation. A degradation specific approach to managing SG tube integrity has several important benefits. These include:

- improved scope and methods for SG inspection,
- industry incentive to continue to improve inspection methods, and
- development of repair criteria based on appropriate NDE parameters.

As a result, the assurance of SG tube integrity is improved and unnecessary conservatism is eliminated.

Over the course of this effort, the SGMP has developed a series of EPRI guidelines that define the elements of a successful SG Program. These guidelines include:

- "Steam Generator Examination Guideline" (Reference 2),
- "Steam Generator Integrity Assessment Guideline" (Reference 3),
- "Steam Generator In-situ Pressure Test Guideline" (Reference 4),
- "PWR Primary-to-Secondary Leak Guideline" (Reference 5),
- "Primary Water Chemistry Guideline" (Reference 6), and
- "Secondary Water Chemistry Guideline" (Reference 7).

These EPRI Guidelines, along with NEI 97-06 (Ref. 1), tie the entire Steam Generator Program together, while defining a comprehensive, performance based approach to managing SG performance.

In parallel with the industry efforts, the NRC pursued resolution of SG performance issues. In December of 1998, the NRC Staff acknowledged that the Steam Generator Program described by NEI 97-06 (Ref. 1) and its referenced EPRI Guidelines provides an acceptable starting point to use in the resolution of differences between it and the staff's proposed Generic Letter and draft Regulatory Guide (DG-1074). Since then, the industry and the NRC have participated in a series of meetings to resolve the differences and develop the regulatory framework necessary to implement a comprehensive Steam Generator Program.

Revising the existing regulatory framework to accommodate degradation specific management is the most appropriate way to address the issues of regulatory stability, resource expenditure, use of state-of-the-art inservice inspection techniques, repair criteria, and enforceability. The NRC Staff has stated that an integrated approach for addressing SG tube integrity is essential and that materials, systems, and radiological issues that pertain to tube integrity need to be considered in the development of the new regulatory framework.

4.0 TECHNICAL ANALYSIS

The proposed changes do not affect the design of the SGs or their method of operation. In addition, the proposed changes do not require any change to FNP's primary coolant chemistry controls because FNP currently follows the Primary Water Chemistry Guideline referenced in Section 3.0. The primary coolant activity limit and its assumptions are not affected by the proposed changes to the Technical Specifications. The proposed changes are an improvement to the existing SG inspection requirements and provide additional assurance that the plant licensing basis will be maintained between SG inspections.

A steam generator tube rupture (SGTR) event is one of the design basis accidents that is analyzed as part of a plant's licensing basis. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the leakage rate associated with a double-ended rupture of a single tube.

For design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor, the SG tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). For FNP Units 1 and 2, these analyses assume that primary to secondary LEAKAGE for all SGs is 1 gallon per minute (gpm), except in the case of the rod ejection accident, where the primary to secondary LEAKAGE is assumed to be 150 gallons per day (gpd) per SG. The rod ejection accident does not result in increased accident induced leakage. For accidents that do not involve fuel damage, the reactor coolant activity levels are at the technical specification values. For accidents that do involve fuel damage, the primary coolant activity values are a function of the amount of activity released from the damaged fuel.

The consequences of these design basis accidents are, in part, functions of the radioactivity levels in the primary coolant and the accident primary to secondary LEAKAGE rates. As a result, limits are included in the plant technical specifications for operational LEAKAGE and for DOSE EQUIVALENT I-131 in primary coolant to ensure the plant is operated within its analyzed condition.

The FNP Units 1 and 2 current technical specification RCS operational LEAKAGE limit, 150 gpd of primary to secondary LEAKAGE through any one SG, is based on operating experience as an indication of one or more tube leaks. This reduced LEAKAGE limit provides additional assurance that leaking flaws will not propagate to burst prior to plant shutdown.

The other technical specification changes proposed are in general a significant improvement over current requirements. They replace an outdated prescriptive technical specification with one that references Steam Generator Program requirements that incorporate the latest knowledge of SG tube degradation morphologies and the techniques developed to manage them.

The requirements being proposed are more effective in detecting SG degradation and prescribing corrective actions than required by current technical specifications. As a result, these proposed changes will result in added assurance of the function and integrity of SG tubes.

The following table and associated sections describe in detail and provide the technical justification for the proposed changes. Note that many of the requirements discussed in the following sections are part of the Steam Generator Program and are not specifically included in the technical specifications.

Enclosure 1 - Basis for the Proposed Change

Condition or Requirement	Current Licensing Basis	Location in TS (or other document) - Proposed Change	Section
Operational primary to secondary LEAKAGE	450 gpd total through all SGs; and 150 gpd through any one SG	RCS Operational LEAKAGE TS 3.4.13: ≤ 150 gpd through any one SG; delete 450 gpd total through all SGs.	1
RCS primary to secondary LEAKAGE through any one SG not within limits	Reduce LEAKAGE to within limits in 4 hours or be in MODE 3 in 6 hours and in MODE 5 in 36 hours	RCS Operational LEAKAGE TS 3.4.13: Be in MODE 3 in 6 hours and in MODE 5 in 36 hours.	2
RCS LEAKAGE determined by water inventory balance (SR 3.4.13.1)	Note states: Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation	RCS Operational LEAKAGE TS 3.4.13: New Note indicating SR not applicable to primary to secondary LEAKAGE.	3
SG Tube integrity verification (SR 3.4.13.2)	Verify in accordance with the SG Tube Surveillance Program	RCS Operational LEAKAGE TS 3.4.13: Revised the SR to verify primary to secondary LEAKAGE every 72 hours. Added Note stating "Not required to be performed until 12 hours after establishment of steady state operation."	4
Frequency of verification of tube integrity	6 to 40 months depending on SG category defined by previous inspection results.	SG Tube Integrity TS 3.4.17: Requires Surveillance Frequency in accordance with TS 5.5.9, "Steam Generator (SG) Program ". Frequency adopted from TSTF-449 Rev. 2 for FNP Units 1 and 2 tube material type (Alloy 690 thermally treated). Steam Generator Program – Establishes maximum inspection intervals.	5

Enclosure 1 - Basis for the Proposed Change

Condition or Requirement	Current Licensing Basis	Location in TS (or other document) - Proposed Change	Section
Tube sample selection	Based on SG Category, industry experience, random selection, existing indications, and results of the initial sample set - 3% times the number of SGs at the plant as a minimum	Steam Generator Program TS 5.5.9 and implementing procedures - Dependent on a pre-outage evaluation of actual degradation locations and mechanisms, and operating experience – 20% of all tubes as a minimum.	6
Inspection techniques	Not specified	SG Tube Integrity TS – SR 3.4.17.1 requires that tube integrity be verified in accordance with the Steam Generator Program. Steam Generator Program TS 5.5.9 and implementing procedures – Establishes requirements for qualifying NDE techniques. Requires use of qualified techniques in SG inspections. Requires a pre-outage evaluation of potential tube degradation morphologies and locations and an identification of NDE techniques capable of finding the degradation.	7
Inspection Scope	Hot leg point of entry to (typically) the first support plate on the cold leg side of the U-bend	Steam Generator Program TS 5.5.9 implementing procedures – Inspection scope is defined by the degradation assessment that considers existing and potential degradation morphologies and locations. Explicitly requires consideration of entire length of tube from tube-sheet weld to tube-sheet weld.	8

Enclosure 1 - Basis for the Proposed Change

Condition or Requirement	Current Licensing Basis	Location in TS (or other document) - Proposed Change	Section
Performance criteria	Operational LEAKAGE Primary to secondary through SGs: 450 gpd total through all SGs 150 gpd through any one SG No criteria specified for structural integrity or accident induced leakage.	RCS Operational LEAKAGE TS 3.4.13 – Operational leakage ≤150 gpd through any one SG. SG Tube Integrity TS 3.4.17 – Requires that tube integrity be maintained. TS 5.5.9 – Defines structural integrity and accident induced leakage performance criteria which are dependent on design basis limits. Provides provisions for condition monitoring assessment to verify compliance.	9
Repair criteria	Plug tubes with imperfections extending ≥40% through wall.	TS 5.5.9 – Criteria unchanged	10
ACTIONS	Performance Criteria not defined. Primary to secondary LEAKAGE limit and actions included in the Tech Specs. Plug tubes exceeding repair criteria.	RCS Operational LEAKAGE TS 3.4.13 and SG Tube Integrity TS 3.4.17 – Contains primary to secondary LEAKAGE limit, SG tube integrity requirements and ACTIONS required upon failure to meet performance criteria. Plug tubes exceeding repair criteria.	11

Enclosure 1 - Basis for the Proposed Change

Condition or Requirement	Current Licensing Basis	Location in TS (or other document) - Proposed Change	Section
Repair methods	Methods (except plugging) require previous approval by the NRC. No approved methods are listed in Technical Specification.	TS 5.5.9 –Requirements unchanged	12
Reporting requirements	Plugging report required 15 days after each inservice inspection, 12 month report documenting inspection results, and reports in accordance with 10 CFR 50.72 when the inspection results fall into category C-3.	10 CFR 50 - Serious SG tube degradation (i.e., tubing fails to meet the structural integrity and accident induced leakage criteria) requires reporting in accordance with 50.72 and 50.73. TS 5.6.10 - 180 days after the initial entry into MODE 4 after performing a SG inspection	13
Definitions SG Terminology	Normal TS definitions (i.e., Definitions Section) did not address SG Program issues.	TS 5.5.9, TS Bases, Steam Generator Program procedures – Includes Steam Generator Program terminology applicable only to SGs.	14

Enclosure 1 - Basis for the Proposed Change

Section 1: Operational LEAKAGE (ref. TS 3.4.13)

The existing primary to secondary LEAKAGE limit for FNP Units 1 and 2 is 150 gpd through any one SG. This leakage rate limit provides assurance against tube rupture at normal operating and faulted conditions. This together with the allowable accident induced leakage limit helps to ensure that the dose contribution from tube leakage will be limited to less than the 10 CFR 100 and GDC 19 dose limits or other NRC approved licensing basis for postulated faulted events.

This limit also contributes to meeting the GDC 14 requirement that the reactor coolant pressure boundary "have an extremely low probability of abnormal leakage, of rapidly propagating to failure, and of gross rupture." The Steam Generator Program uses the EPRI Primary-to-Secondary Leak Guideline (Ref. 5) to establish sampling requirements for determining primary to secondary LEAKAGE and plant shutdown requirements if leakage limits are exceeded. The guidelines ensure leakage is effectively monitored and timely action is taken before a leaking tube exceeds the performance criteria. The Frequency for determining primary to secondary LEAKAGE is unchanged (i.e., 72 hours and within 12 hours after establishing stable operating conditions).

The technical specification requirement to limit primary to secondary LEAKAGE through all SGs to 450 gpd is a redundant restatement of the requirement to limit primary to secondary LEAKAGE through any one SG to 150 gpd (each FNP unit has three SGs, so 3 x 150 = 450 gpd total LEAKAGE through all SGs) and is therefore proposed to be deleted.

Section 2: Operational LEAKAGE Actions (ref. TS 3.4.13)

If primary to secondary LEAKAGE exceeds 150 gpd through any one SG, a plant shutdown must be commenced. MODE 3 must be achieved in 6 hours and MODE 5 in 36 hours. The existing technical specifications allow 4 hours to reduce primary to secondary LEAKAGE to less than the limit. The proposed technical specification removes this allowance.

The removal of the 4 hour period during which primary to secondary LEAKAGE can be reduced to avoid a plant shutdown results in a technical specification that is significantly more conservative than the earlier RCS Operational LEAKAGE specification. This change is consistent with the Steam Generator Program that also does not allow 4 hours before commencing a plant shutdown.

Section 3: RCS Operational LEAKAGE Determined by Water Inventory Balance (ref. TS 3.4.13)

The proposed change adds a second Note to SR 3.4.13.1 that makes the water inventory balance method not applicable to determining primary to secondary LEAKAGE. This change is proposed because primary to secondary LEAKAGE as low as 150 gpd through any one SG cannot be measured accurately by an RCS water inventory balance. This change is necessary to make the surveillance requirement appropriate for the proposed LCO.

Section 4: SG Tube Integrity Verification (ref. TS 3.4.13)

The current SR 3.4.13.2 requires verification of tube integrity in accordance with the SG Tube Surveillance Program. This surveillance is no longer appropriate since tube integrity is addressed through the addition of a new SG Tube Integrity Specification. Specification 3.4.13 now applies specifically to primary to secondary LEAKAGE. Surveillance Requirement 3.4.13.2 has been changed to verify the LCO requirement on primary to secondary LEAKAGE only. Steam generator tube integrity is verified in accordance with a SR in the SG Tube Integrity Specification.

The Steam Generator Program and the EPRI "Pressurized Water Reactor Primary-to-Secondary Leak Guideline" (Ref. 5) provide guidance on leak rate monitoring. During normal operation the program depends upon continuous process radiation monitors and/or radiochemical grab sampling. The monitoring and sampling frequency increases as the amount of detected LEAKAGE increases or if there are no continuous radiation monitors available.

Primary to secondary LEAKAGE is determined through the analysis of secondary coolant activity levels. At low power, primary and secondary coolant activity is sufficiently low that an accurate determination of primary to secondary LEAKAGE may be difficult. Immediately after shutdown, some of the short lived isotopes are usually at sufficient levels to monitor for LEAKAGE by normal power operational means as long as other plant conditions allow the measurement. During startup, especially after a long outage, there are no short lived isotopes in either the primary or secondary system. This limits measurement of the LEAKAGE to chemical or long lived radiochemical means. Because of these effects, an accurate primary to secondary leakage measurement is highly dependent upon plant conditions and may not be obtainable prior to reactor criticality (e.g., MODES 1 and 2). If SG water samples have less than the minimum detectable activity (5.0E-7 microcuries/ml with current technology) for each principal gamma emitter, primary to secondary LEAKAGE may be assumed to be less than 150 gpd through any one SG.

The Surveillance Frequency is unchanged. Determination of the primary to secondary LEAKAGE is required every 72 hours. The SR is modified by a Note stating the SR is not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. As stated above, additional monitoring of primary to secondary LEAKAGE is also required by the Steam Generator Program based upon guidance provided in Reference 5.

Section 5: Frequency of Verification of SG Tube Integrity (ref. TS 3.4.17)

The current technical specifications contain prescriptive inspection intervals which depend on the condition of the tubes as determined by the last SG inspection. The tube condition is classified into one of three categories based on the number of tubes found degraded and defective. The minimum inspection interval is no less than 12 and no more than 24 months unless the results of two consecutive inspections are in the best category (no additional degradation), and then the interval can be extended to 40 months.

The surveillance Frequency in the proposed Steam Generator Tube Integrity specification is governed by the requirements in the Steam Generator Program and specifically by References 2 and 3. The proposed Frequency is also prescriptive, but has a stronger engineering basis than the existing technical specification requirements. The interval is dependent on tubing material and whether any active degradation is found. The interval is limited by existing and potential degradation mechanisms and their anticipated growth rate. In addition, a maximum inspection interval is established in Specification 5.5.9.

The maximum inspection interval for Alloy 690 thermally treated tubing, the type used in the FNP Unit 1 and 2 SGs, is "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." Even though the maximum interval for Alloy 690 thermally treated tubing is slightly longer than allowed by current technical specifications, it is only applicable to SGs with advanced materials and is only achievable early in SG life, and only if the SGs are free from active degradation. In addition, the interval must be supported by an evaluation that shows that the performance criteria will continue to be met at the next SG inspection. Taken in total, the proposed inspection intervals provide a larger margin of safety than the current requirements because they are based on an engineering evaluation of the tubing condition and potential degradation mechanisms and growth rates, not only on the previous inspection results. As an added safety measure, the Steam Generator Program requires a minimum sample size at each inspection that is significantly larger than that required by current technical specifications (20 percent versus 3 percent times the number of SGs in the plant); thus providing added assurance that any degradation within the SGs will be detected and accounted for in establishing the inspection interval.

The proposed maximum inspection intervals are based on the historical performance of advanced SG tubing materials. Reference 8 shows that the performance of Alloy 600TT and 690TT is significantly better than the performance of 600MA tubing, the material used in SG tubing at the time that the current technical specifications were written. There are no known instances of cracking in 690TT tubes in either the U.S. or international SGs.

In summary, the proposed change is an improvement over the current technical specification. The current technical specification bases inspection intervals on the results of previous inspections; it does not require an evaluation of expected performance. The proposed technical specification uses information from previous plant inspections as well as industry experience to evaluate the length of time that the SGs can be operated and still provide reasonable assurance that the performance criteria will be met at the next inspection. The actual interval is the shorter of the evaluation results and the requirements in Reference 3. Allowing plants to use the proposed inspection intervals maximizes the potential that plants will use improved techniques and knowledge since better knowledge of SG conditions supports longer intervals.

Section 6: SG Tube Sample Selection (ref. TS 5.5.9)

The current technical specifications base tube selection on SG conditions and industry and plant experience. The minimum sample size is 3% of the tubes times the number of SGs in the plant. The proposed change refers to the Steam Generator Program degradation assessment guidance for sampling requirements. The minimum sample size is 20% of the total number of active tubes in all SGs for each unit.

The Steam Generator Program requires the preparation of a degradation assessment before every SG inspection. The degradation assessment is the key document used for planning a SG inspection, where inspection plans and related actions are determined, documented, and communicated prior to the outage. The degradation assessment addresses the various reactor coolant pressure boundary components within the SG (e.g., plugs, sleeves, tubes, and components that support the pressure boundary). In a degradation assessment, tube sample selection is performance based and is dependent upon actual SG conditions, plant operational experience and industry experience in general. Existing and potential degradation mechanisms and their locations are evaluated to determine which tubes will be inspected. Tube sample selection is adjusted to minimize the possibility that tube integrity might degrade during an operating cycle beyond the limits defined by the performance criteria. The EPRI Steam Generator Examination Guideline (Ref. 2) and EPRI Steam Generator Integrity Assessment Guideline (Ref. 3) provide guidance on degradation assessment.

Enclosure 1 - Basis for the Proposed Change

In general, the sample selection considerations required by the current technical specifications and the requirements in the Steam Generator Program as proposed by this change are consistent, but the Steam Generator Program provides more guidance on selection methodologies and incorporation of industry experience and requires more extensive documentation of the results. Therefore the sample selection method proposed by this change is more conservative than the current technical specification requirements. In addition, the minimum sample size in the proposed requirements is larger.

Section 7: SG Inspection Techniques (ref. TS 3.4.17)

The Surveillance Requirements proposed in the Steam Generator Tube Integrity specification require that tube integrity be verified in accordance with the requirements of the Steam Generator Program. The Steam Generator Program uses the EPRI Steam Generator Examination Guidelines (Ref. 2) to establish requirements for qualifying NDE techniques and maintains a list of qualified techniques and their capabilities.

The Steam Generator Program requires the performance of a degradation assessment before every SG inspection and refers utilities to EPRI Steam Generator Examination Guidelines (Ref. 2) and EPRI Steam Generator Integrity Assessment Guidelines (Ref. 3) for guidance on its performance. The degradation assessment will identify current and potential new degradation locations and mechanisms and NDE techniques that are effective in detecting their existence. Tube inspection techniques are chosen to reliably detect flaws that might progress during an operating cycle beyond the limits defined by the performance criteria.

The current technical specifications contain no requirements on NDE inspection techniques. The proposed change is an improvement over the current technical specifications that contain no similar requirement.

Section 8: SG Inspection Scope (ref. TS 5.5.9)

The current technical specifications include a definition of inspection that specifies the end points of the eddy current examination of each tube. Inspection is required from the point of entry of the tube on the hot leg side completely around the U-bend to the top support of the cold leg. This definition is overly prescriptive and simplistic and has led to interpretation questions in the past.

The Steam Generator Program states, "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." The Steam Generator Program provides extensive guidance and a defined process, the degradation assessment, for determining the extent of a tube inspection. This guidance takes into account industry and plant specific history to determine potential degradation mechanisms and the location that they might occur within the SG. This information is used to define a performance based inspection scope targeted on plant specific conditions and SG design.

Enclosure 1 - Basis for the Proposed Change

The proposed change is an improvement over the current technical specifications because it focuses the inspection effort on the areas of concern, thereby minimizing the unnecessary data that the NDE analyst must review to identify indication of tube degradation.

Section 9: SG Performance Criteria (ref. TS 3.4.13 and 3.4.17)

The proposed change adds a performance-based Steam Generator Program to the Technical Specifications. A performance-based approach has the following attributes:

- measurable parameters,
- objective criteria to assess performance based on risk-insights,
- deterministic analysis and/or performance history, and
- licensee flexibility to determine how to meet established performance criteria.

The performance criteria used for SGs are based on tube structural integrity, accident induced leakage, and operational LEAKAGE. The structural integrity and accident induced leakage criteria were developed deterministically and are consistent with the plant's licensing basis. The operational LEAKAGE criterion was based on providing added assurance against tube rupture at normal operating and faulted conditions. The proposed structural integrity and accident induced leakage performance criteria are new requirements. The performance criteria are specified in Specification 5.5.9. The requirements and methodologies established to meet the performance criteria are documented in the Steam Generator Program. The current technical specifications contain only the operational LEAKAGE criterion; therefore the proposed change is more conservative than the current requirements.

The SG performance criteria identify the standards against which performance is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity throughout each operating cycle.

The structural integrity performance criterion is:

> "All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 (3ΔP) 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."

The structural integrity performance criterion is based on providing reasonable assurance that a SG tube will not burst or collapse during normal operation or postulated accident conditions.

Enclosure 1 - Basis for the Proposed Change

Normal steady state full power operation is defined as the conditions existing during MODE 1 operation at the maximum steady state reactor power as defined in the design or equipment specification. Changes in design parameters such as plugging or sleeving levels, primary or secondary modifications, or T_{hot} should be assessed and included if significant.

The definition of normal steady state full power operation is important as it relates to application of the safety factor of three in the structural integrity performance criterion. The criterion requires "...retaining a safety factor of 3.0 (3ΔP) against burst under normal steady state full power operation primary to secondary pressure differential ..." The application of the safety factor of three to normal steady state full power operation is founded on past NRC positions, accepted industry practice, and the intent of the ASME Code for original design and evaluation of inservice components. The assumption of normal steady state full power operating pressure differential has been consistently used in the analysis, testing and verification of tubes with stress corrosion cracking for verifying a safety factor of three against burst. Additionally, the 3ΔP criterion is measurable through the condition monitoring process.

The actual operational parameters may differ between cycles. As a result of changes to these parameters, reaching the differential pressure in the equipment specification may not be possible during plant operations. Evaluating to the pressure in the design or equipment specification in these cases would be an unnecessary conservatism. Therefore, the definition allows adjustment of the 3ΔP limit for changes in these parameters when necessary. Further guidance on this adjustment is provided in Appendix M of the EPRI Steam Generator Integrity Assessment Guidelines (Ref. 3).

The accident induced leakage performance criterion is:

> "The primary to secondary accident induced leakage rate for all design basis accidents, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Accident induced leakage is not to exceed 1 gpm total for all three SGs."

Primary to secondary LEAKAGE is a factor in the activity releases outside containment resulting from a limiting design basis accident. The potential dose consequences from primary to secondary LEAKAGE during postulated design basis accidents must not exceed the radiological limits imposed by 10 CFR Part 100 guidelines, or the radiological limits to control room personnel imposed by GDC-19, or other NRC approved licensing basis.

When calculating offsite doses the safety analysis for the limiting Design Basis Accident assumes a total of 1 gpm primary to secondary LEAKAGE as an initial condition. Revision 2 of the Standard Technical Specifications limited the amount of RCS Operational LEAKAGE to 1 gpm from all SGs, with 500 gpd from the worst generator, since the initial safety analyses assumed that leakage under accident conditions would not exceed the limit on Operational LEAKAGE. More recent experience with degradation mechanisms involving tube cracking has revealed that leakage under accident conditions can exceed the level of operating leakage by orders of magnitude. The NRC has concluded (Item Number 3.4 in Attachment 1 to Reference 14) that additional research is needed to develop an adequate methodology for fully predicting the effects of leakage on the outcome of some accident sequences. Therefore, a separate performance criterion was established for accident induced leakage. For FNP Units 1 and 2 the limit for accident induced leakage is 1 gpm. Use of an increased accident induced leakage limit approved in conjunction with alternate repair criteria (currently none

Enclosure 1 - Basis for the Proposed Change

are approved for FNP Units 1 and 2) would be limited to the specific criteria and type of degradation for which it was granted.

The operational LEAKAGE performance criterion is:

> "RCS operational LEAKAGE shall be limited to: 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG)."

Plant shutdown will commence if primary to secondary LEAKAGE exceeds 150 gallons per day at room temperature conditions from any one SG.

The operational LEAKAGE performance criterion is documented in the Steam Generator Program and implemented in Specification 3.4.13, "RCS Operational LEAKAGE."

Proposed Administrative Specification 5.5.9 contains the performance criteria and is more conservative than the current technical specifications. The current technical specifications do not address the structural integrity and accident induced leakage criteria.

Section 10: SG Repair Criteria (ref. TS 5.5.9)

Repair criteria are those NDE measured parameters at or beyond which the tube must be repaired or removed from service by plugging.

Tube repair criteria are established for each active degradation mechanism. Tube repair criteria are either the standard through-wall depth-based criterion (e.g., ≥40% through-wall) or through-wall depth based criteria for repair techniques approved by the NRC, or other Alternate Repair Criteria (ARC) approved by the NRC such as a voltage-based repair limit per Generic Letter 95-05 (Ref. 12). A SG degradation-specific management strategy is followed to develop and implement an ARC.

The surveillance requirements of the proposed Steam Generator Integrity specification require that tubes that satisfy the tube repair criteria be plugged or repaired in accordance with approved methods. SG tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "repaired (or plugged)-on-detection" and their integrity should be assessed

No plant-specific alternate repair criteria are currently approved for FNP Units 1 and 2.

Section 11: ACTIONS (ref. TS 3.4.13 and 3.4.17)

The RCS Operational LEAKAGE and Steam Generator Tube Integrity specifications require the licensee to monitor SG performance against performance criteria in accordance with the Steam Generator Program.

During plant operation, monitoring is performed using the operational LEAKAGE criterion. Exceeding that criterion will lead to a plant shutdown in accordance with Technical Specification 3.4.13. Once shutdown, the Steam Generator Program will ensure that the cause of the operational LEAKAGE is determined and corrective actions are taken to prevent recurrence. Operation may resume when the requirements of the Steam Generator Program have been met. This requirement is unchanged from the current technical specifications.

Enclosure 1 - Basis for the Proposed Change

Also during plant operation the licensee may discover an error or omission that indicates a failure to implement a required plugging during a previous SG inspection. Under these circumstances, the licensee is expected to take the actions required by Condition A in the Steam Generator Tube Integrity specification, TS 3.4.17. If a performance criterion has been exceeded, a principal safety barrier has been challenged and 10 CFR 50.72 (b) (3) (ii) (A) and 50.73 (a) (2) (ii) (A) require NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence. The Steam Generator Program (TS 5.5.9) additionally requires that the report contain information on the performance criteria exceeded and the basis for the planned operating cycle. The current technical specifications only address operational LEAKAGE during operations and therefore do not include the proposed requirement.

During MODES 5 and 6, the operational LEAKAGE criterion is not applicable, and the SGs will be inspected as required by the surveillance in the Steam Generator Tube Integrity specification. A condition monitoring assessment of the "as found" condition of the SG tubes will be performed to determine the condition of the SGs with respect to the structural integrity and accident leakage performance criteria. If the performance criteria are not met, the Steam Generator Program requires ascertaining the cause and determining corrective actions to prevent recurrence. Operation may resume when the requirements of the Steam Generator Program have been met.

The proposed technical specification's change to the ACTIONS required upon exceeding the operational leakage criterion is conservative with respect to the current technical specifications as explained in Section 2 above.

The current technical specifications do not address ACTIONS required while operating if it is discovered that the structural integrity or accident induced leakage performance criteria or repair criteria are exceeded, so the proposed change is conservative with respect to the current technical specifications.

If performance or repair criteria are exceeded while shutdown, the affected tubes must be plugged. A report will be submitted to the NRC in accordance with Technical Specification 5.6.10. The changes in the required reports are discussed in Section 13 below.

Section 12: SG Repair Methods (ref. TS 5.5.9)

Repair methods are those means used to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a SG tube is not a repair.

The purpose of a repair is typically to reestablish or replace the RCPB. The proposed Steam Generator Tube Integrity surveillance requirements require that tubes that satisfy the tube repair criteria be plugged in accordance with the Steam Generator Program, TS 5.5.9. No repair methods are currently listed in TS 5.5.9 or are proposed by this change.

Steam generator tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "plugged-on-detection" and their integrity is assessed. This requirement is unchanged by the proposed technical specifications. Note that SG plug designs do not require NRC review and therefore plugging is not considered a repair in the context of this requirement.

The proposed approach is not a change to the technical specifications.

Enclosure 1 - Basis for the Proposed Change

Section 13: Reporting Requirements (ref. TS 5.6.10)

The current technical specifications require the following reports:

- A report listing the number of tubes plugged in each SG submitted within 15 days of the end of the inspection.
- A SG inspection results report submitted within 12 months after the inspection.
- Reports required pursuant to 10 CFR 50.73.

The proposed change to Technical Specification 5.6.10 replaces the 15 day and the SG inspection reports with one report required within 180 days after initial entry into MODE 4 following inspection. The proposed report also contains more information than the current SG inspection report. This provision expands the report to provide more substantive information and will be sent earlier (180 days versus 12 months). This allows the licensee and NRC to focus their attention on the more significant conditions.

The guidance in NUREG-1022, Rev. 2, "Event Reporting Guidelines 10 CFR 50.72 and 50.73," identifies serious SG tube degradation as an example of an event or condition that results in the condition of the nuclear power plant, including its principal safety barriers, being seriously degraded. Steam generator tube degradation is considered serious if the tubing fails to meet structural integrity or accident induced leakage performance criteria. Serious SG tube degradation would be reportable in accordance with 10 CFR 50.72 (b) (3) (ii) (A) and 50.73 (a) (2) (ii) (A) requiring NRC notification and the submittal of a report containing the cause and corrective actions to prevent recurrence.

The proposed reporting requirements are an improvement as compared to those required by the current technical specifications. The proposed reporting requirements are more useful in identifying the degradation mechanisms and determining their effects. In the unlikely event that a performance criterion is not met, NEI 97-06 (Ref. 1) directs the licensee to submit additional information on the root cause of the condition and the basis for the next operating cycle.

The changes to the reporting requirements are performance based. The new requirements remove the burden of unnecessary reports from both the NRC and the licensee, while ensuring that critical information related to problems and significant tube degradation is reported more completely and, when required, more expeditiously than under the current technical specifications.

Section 14: SG Terminology (ref. TS 5.5.9, 3.4.17 and Bases)

The proposed Steam Generator Tube Integrity specification Bases explain a number of terms that are important to the function of the Steam Generator Program. The Technical Specification Bases are controlled by the Technical Specification Bases Control Program, which appears in the Administrative Technical Specifications.

The terms are described below.

1. Accident induced leakage rate means the primary to secondary LEAKAGE rate occurring during postulated accidents other than a steam generator tube rupture. This includes the primary to secondary LEAKAGE rate existing immediately prior to the accident plus additional primary to secondary LEAKAGE induced during the accident.

Enclosure 1 - Basis for the Proposed Change

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a limiting design basis accident. The potential primary to secondary leak rate during postulated design basis accidents must not cause radiological dose consequences in excess of the 10 CFR Part 100 guidelines for offsite doses, or the GDC-19 requirements for control room personnel, or other NRC approved licensing basis.

2. The LCO section of Steam Generator Tube Integrity Bases (B 3.4.17) defines the term "burst" 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."

Since a burst definition is required for condition monitoring, a definition that can be analytically defined and is capable of being assessed via in situ and laboratory testing is necessary. Furthermore, the definition must be consistent with ASME Code requirements, and apply to most forms of tube degradation.

The definition developed for tube burst is consistent with the testimony of James Knight (Ref. 9), and the historical guidance of draft Regulatory Guide 1.121 (Ref. 10). The definition of burst per these documents is in relation to gross failure of the pressure boundary; e.g., "the degree of loading required to burst or collapse a tube wall is consistent with the design margins in Section III of the ASME B&PV Code (Ref. 11)." Burst, or gross failure, according to the Code would be interpreted as a catastrophic failure of the pressure boundary.

The above definition of burst was chosen for a number of reasons:

- The burst definition supports field application of the condition monitoring process. For example, verification of structural integrity during condition monitoring may be accomplished via in situ testing. Since these tests do not have the capability to provide an unlimited water supply or the capability to maintain pressure under certain leakage scenarios, opening area may be more a function of fluid reservoir rather than tube strength. Additionally, in situ designs with bladders may not be reinforced. In certain cases, the bladder may rupture when tearing or extension of the defect has not occurred. This condition may simply mean the opening of the flanks of the defect was sufficient to permit extrusion of the bladder, and that the actual, or true, burst pressure was not achieved during the test. The burst definition addresses this issue.

- The definition does not characterize local instability or "ligament pop-through", as a burst. The onset of ligament tearing need not coincide with the onset of a full burst. For example, an axial crack about 0.5" long with a uniform depth at 98% of the tube wall would be expected to fail the remaining ligament, (i.e., extend the crack tip in the radial direction) due to deformation during pressurization at a pressure below that required to cause extension at the tips in the axial direction. Thus, this would represent a leakage situation as opposed to a burst situation and a factor of safety of three against crack extension in the axial direction may still be demonstrated. Similar conditions have been observed for localized deep wear indications.

Enclosure 1 - Basis for the Proposed Change

3. The LCO section of Steam Generator Tube Integrity Bases (B 3.4.17) defines a SG tube 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."

   This definition ensures that all portions of SG tubes that are part of the RCPB, with the exception of the tube-to-tubesheet weld, are subject to Steam Generator Program requirements. The definition is also intended to exclude tube ends that can not be NDE inspected by eddy current. If there are concerns in the area of the tube end, they will be addressed by NDE techniques if possible or by using other methods if necessary.

   For the purposes of SG tube integrity inspection, any weld metal in the area of the tube end is not considered part of the tube. This is necessary since the acceptance requirements for tubing and weld metals are different.

4. The LCO section of Steam Generator Tube Integrity Bases (B 3.4.17) defines the term "collapse" 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."

   In dealing with pure pressure loadings, burst is the only failure mechanism of interest. If bending loads are introduced in combination with pressure loading, the definition of failure must be broadened to encompass both burst and bending collapse. Which failure mode applies depends on the relative magnitude of the pressure and bending loads and also on the nature of any flaws that may be present in the tube.

Conclusion:

The proposed changes will provide greater assurance of SG tube integrity than that offered by the current technical specifications. The proposed requirements are performance based and provide the flexibility to adopt new technology as it matures. These changes are consistent with the guidance in NEI 97-06, "Steam Generator Program Guidelines," (Ref. 1).

Adopting the proposed changes will provide added assurance that SG tubing will remain capable of fulfilling its specific safety function of maintaining RCPB integrity.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

The proposed change revises the FNP Units 1 and 2 Technical Specifications (TS) sections TS 3.4.13, "RCS Operational LEAKAGE," TS 5.5.9, "Steam Generator Tube Surveillance Program," and TS 5.6.10, "Steam Generator Tube Inspector Report," and adds a new section TS 3.4.17, "Steam Generator Tube Integrity." The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97-06, "Steam Generator Program Guidelines," (Reference 1).

SNC has evaluated whether or not a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, Issuance of amendment," as discussed below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

The proposed change requires a Steam Generator Program that includes performance criteria that will provide reasonable assurance that the steam generator (SG) tubing will retain integrity over the full range of operating conditions (including startup, operation in the power range, hot standby, cooldown and all anticipated transients included in the design specification). The SG performance criteria are based on tube structural integrity, accident induced leakage, and operational LEAKAGE.

The structural integrity performance criterion is:

> "All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 ($3\Delta P$) 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."

The accident induced leakage performance criterion is:

> "The primary to secondary accident induced leakage rate for all design basis accidents, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Accident induced leakage is not to exceed 1 gpm total for all three SGs."

Enclosure 1 - Basis for the Proposed Change

The operational LEAKAGE performance criterion is:

> The RCS operational primary to secondary LEAKAGE through any one SG shall be limited to 150 gpd.

A steam generator tube rupture (SGTR) event is one of the design basis accidents analyzed as part of the plant licensing basis. In the analysis of a SGTR event, a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in the licensing basis plus the LEAKAGE rate associated with a double-ended rupture of a single tube is assumed.

For other design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). For FNP Units 1 and 2 these analyses assume that primary to secondary LEAKAGE for all SGs is 1 gpm. The accident induced leakage criterion introduced by the proposed changes accounts for tubes that may leak during design basis accidents. The accident induced leakage criterion limits this leakage to no more than the value assumed in the accident analysis.

The SG performance criteria proposed in this change to the TS identify the standards against which tube integrity is to be measured. Meeting the performance criteria provides reasonable assurance that the SG tubing will remain capable of fulfilling its specific safety function of maintaining reactor coolant pressure boundary integrity throughout each operating cycle and in the unlikely event of a design basis accident. The performance criteria are only a part of the Steam Generator Program required by the proposed change to the TS. The program, defined by NEI 97-06, Steam Generator Program Guidelines, includes a framework that incorporates a balance of prevention, inspection, evaluation, plugging, and leakage monitoring.

The consequences of design basis accidents are, in part, functions of the DOSE EQUIVALENT I-131 in the primary coolant and the primary to secondary LEAKAGE rates resulting from an accident. Therefore, limits are included in the TS for operational leakage and for DOSE EQUIVALENT I-131 in primary coolant to ensure the plant is operated within its analyzed condition. The analysis of the limiting design basis accident assumes that primary to secondary leak rate after the accident is 1 gpm with no more than 500 gpd in any one SG, and that the reactor coolant activity levels of DOSE EQUIVALENT I-131 are at the technical specification values before the accident.

The proposed change does not affect the design of the SGs, their method of operation, or primary coolant chemistry controls. The proposed approach updates the current TS and enhances the requirements for SG inspections. The proposed change does not adversely impact any other previously evaluated design basis accident and is an improvement over the current TS.

Therefore, the proposed change does not affect the consequences of a SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of a MSLB, rod ejection, or a reactor coolant pump locked rotor event.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

Enclosure 1 - Basis for the Proposed Change

The proposed performance based requirements are an improvement over the requirements imposed by the current TS.

Implementation of the proposed Steam Generator Program will not introduce any adverse changes to the plant design basis or postulated accidents resulting from potential tube degradation. The result of the implementation of the Steam Generator Program will be an enhancement of SG tube performance. Primary to secondary LEAKAGE that may be experienced during all plant conditions will be monitored to ensure it remains within current accident analysis assumptions.

The proposed change does not affect the design of the SGs, their method of operation, or primary or secondary coolant chemistry controls. In addition, the proposed change does not impact any other plant system or component. The change enhances SG inspection requirements.

Therefore, the proposed change does not create the possibility of a new or different type of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The SG tubes in pressurized water reactors are an integral part of the reactor coolant pressure boundary and, as such, are relied upon to maintain the primary system's pressure and inventory. As part of the reactor coolant pressure boundary, the SG tubes are unique in that they are also relied upon as a heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system. In addition, the SG tubes also isolate the radioactive fission products in the primary coolant from the secondary system. In summary, the safety function of a SG is maintained by ensuring the integrity of its tubes.

Steam generator tube integrity is a function of the design, environment, and the physical condition of the tube. The proposed change does not affect tube design or operating environment. The proposed change is expected to result in an improvement in the tube integrity by implementing the Steam Generator Program to manage SG tube inspection, assessment and plugging. The requirements established by the Steam Generator Program are consistent with those in the applicable design codes and standards and are an improvement over the requirements in the current TS.

For the above reasons, the margin of safety is not changed and overall plant safety will be enhanced by the proposed change to the TS.

5.2 Applicable Regulatory Requirements/Criteria

The regulatory requirements applicable to SG tube integrity are the following:

10 CFR 50.55a, Codes and Standards - Section (b), ASME Code - c) *Reactor coolant pressure boundary.* (1) Components which are part of the reactor coolant pressure boundary must meet the requirements for Class 1 components in Section III of the ASME Boiler and Pressure Vessel Code, except as provided in paragraphs (c)(2), (c)(3), and (c)(4) of this section.

Enclosure 1 - Basis for the Proposed Change

The proposed change and the Steam Generator Program requirements which underlie it are in full compliance with the ASME Code. The proposed technical specifications are more effective at ensuring tube integrity and, therefore, compliance with the ASME Code, than the current technical specifications as described in Section 4.0 (Technical Analysis).

10 CFR 50.65 Maintenance Rule – Each holder of a license to operate a nuclear power plant under 10 CFR 50.21(b) or 50.22 shall monitor the performance or condition of structures, systems, or components, against licensee-established goals, in a manner sufficient to provide reasonable assurance that such structures, systems, and components, as defined in paragraph (b), are capable of fulfilling their intended functions. Such goals shall be established commensurate with safety and, where practical, take into account industry-wide operating experience. When the performance or condition of a structure, system, or component does not meet established goals, appropriate corrective action shall be taken. For a nuclear power plant for which the licensee has submitted the certifications specified in 10 CFR 50.82(a)(1), this section only shall apply to the extent that the licensee shall monitor the performance or condition of all structures, systems, or components associated with the storage, control, and maintenance of spent fuel in a safe condition, in a manner sufficient to provide reasonable assurance that such structures, systems, and components are capable of fulfilling their intended functions.

Under the Maintenance Rule, licensees classify SGs as risk significant components because they are relied on to remain functional during and after design basis events. The performance criteria included in the proposed technical specifications are used to demonstrate that the condition of the SG "is being effectively controlled through the performance of appropriate preventive maintenance" (Maintenance Rule section (a)(2)). If the performance criteria are not met, a root cause determination of appropriate depth is done and the results evaluated to determine if goals should be established per section (a)(1) of the Maintenance Rule.

NEI 97-06, Steam Generator Program Guidelines, and its referenced EPRI guidelines define a SG program that provides the appropriate preventive maintenance that meets the intent of the Maintenance Rule. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," (Reference 13) offers guidance for implementing the Maintenance Rule should a licensee elect to incorporate additional monitoring goals beyond the scope of those documented in NEI 97-06.

10 CFR 50, Appendix A, General Design Criterion (GDC) 14 – *Reactor Coolant Pressure Boundary*. The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, or rapidly propagating failure, and of gross rupture.

There are no changes to the SG design that impact this general design criterion. The evaluation performed in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

10 CFR 50, Appendix A, GDC 30 - *Quality of reactor coolant pressure boundary*. Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

There are no changes to the SG design that impact this general design criterion. The evaluation performed in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

Enclosure 1 - Basis for the Proposed Change

10 CFR 50, Appendix A, GDC 32 – *Inspection of reactor coolant pressure boundary.* Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

There are no changes to the SG design that impact this general design criterion. The evaluation performed in Section 4.0 concludes that the proposed change will continue to comply with this regulatory requirement.

GDC 14, 30, and 32 of 10 CFR Part 50, Appendix A, define requirements for the reactor coolant pressure boundary with respect to structural and leakage integrity. SG tubing and tube repairs constitute a major fraction of the reactor coolant pressure boundary surface area. SG tubing and associated repair techniques and components, such as plugs and sleeves, must be capable of maintaining reactor coolant inventory and pressure. The Steam Generator Program required by the proposed TS establishes performance criteria, repair criteria (no repair methods are currently approved or proposed for FNP Units 1 and 2), inspection intervals and the methods necessary to meet them. These requirements provide reasonable assurance that tube integrity will be met in the interval between SG inspections.

The proposed change provides requirements that are more effective in detecting SG degradation and prescribing corrective actions. The proposed change results in added assurance of the function and integrity of SG tubes. Therefore, based on the considerations discussed above:

1) There is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner;

2) Such activities will be conducted in compliance with the Commission's regulations; and

3) Issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed change would change a requirement with respect to installation or use of a facility component located within the restricted areas, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed change does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that would be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed change.

7.0 REFERENCES

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

2. EPRI, "Steam Generator Examination Guideline."

3. EPRI, "Steam Generator Integrity Assessment Guideline."

4. EPRI, "Steam Generator In-situ Pressure Test Guideline."

5. EPRI, "PWR Primary-to-Secondary Leak Guideline."

6. EPRI, "Primary Water Chemistry Guideline."

7. EPRI, "Secondary Water Chemistry Guideline."

8. EPRI Report R-5515-00-2, "Experience of US and Foreign PWR Steam Generators with Alloy 600TT and Alloy 690TT Tubes and Sleeves," June 5, 2002.

9. Testimony of James Knight Before the Atomic Safety and Licensing Board, Docket Nos. 50-282 and 50-306, January 1975.

10. Draft Regulatory Guide 1.121, "Bases for Plugging Degraded Steam Generator Tubes," August 1976.

11. ASME B&PV Code, Section III, Rules for Construction of Nuclear Facility Components.

12. Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," August 3, 1995.

13. NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 3.

14. S. C. Collins memo to W. D. Travers, "Steam Generator Action Plan Revision to Address Differing Professional Opinion on Steam Generator Tube Integrity," May 11, 2001.

Joseph M. Farley Nuclear Plant
Request for Technical Specifications Change
Steam Generator Program

Enclosure 2

Marked-Up Technical Specification and Bases Pages

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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. 450 gallons per day total primary to secondary LEAKAGE through all steam generators (SGs); and~~

d. ~~e.~~ 150 gallons per day primary to secondary LEAKAGE through any one ~~SG~~ steam generator (SG).

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

ACTIONS

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

S

SURVEILLANCE REQUIREMENTS

1.

	SURVEILLANCE	FREQUENCY
SR 3.4.13.1	-----------------------------NOTE----------------------------- Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. --- 2. Not applicable to primary to secondary LEAKAGE. o Verify RCS O perational LEAKAGE is within limits by performance of RCS water inventory balance.	-------NOTE-------- Only required to be performed during steady state operation --------------------- 72 hours
SR 3.4.13.2	~~Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.~~	~~In accordance with the Steam Generator Tube Surveillance Program~~

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

72 hours

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

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 inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	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. OR SG tube integrity not maintained.	B.1 Be in MODE 3. AND 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 Programs and Manuals

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. 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:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
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 TS.

5.5.9 Steam Generator (SG) ~~Tube Surveillance~~ Program

Insert 5.5.9

~~The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program Test Frequencies. [Specification 5.5.9 is not required to be performed on the replacement steam generators during the shutdown when the steam generators are replaced.]~~

~~5.5.9.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program.~~

(continued)

5.5 Programs and Manuals

~~5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~5.5.9.1 Steam Generator Sample Selection and Inspection~~

~~Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 5.5.9-1.~~

~~5.5.9.2 Steam Generator Tube Sample Selection and Inspection~~

~~The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 5.5.9.3, and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators. The tubes selected for these inspections shall be selected on a random basis except:~~

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

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

~~1. All nonplugged tubes that previously had detectable wall penetrations greater than 20%.~~

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

~~3. A tube inspection (pursuant to Specification 5.5.9.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.~~

~~c. The tubes selected as the second and third samples (if required by Table 5.5.9-2) during each inservice inspection may be subjected to a partial tube inspection provided:~~

~~(continued)~~

5.5 Programs and Manuals

~~5.5.9.2 Steam Generator Tube Sample Selection and Inspection (continued)~~

~~1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found.~~

~~2. The inspections include those portions of the tubes where imperfections were previously found.~~

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

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

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

~~5.5.9.3 Inspection Frequencies~~

~~The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:~~

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

~~(continued)~~

5.5 Programs and Manuals

~~5.5.9.3 Inspection Frequencies (continued)~~

~~for each unit, a one-time inspection interval extension of a maximum of once per 40 months is allowed for the inspection performed immediately after the Unit 1 1R17 inspection and for the inspection performed immediately after the Unit 2 2R15 inspection. These are exceptions to the Extension Criteria in that the inspection interval for each unit is based on the results of only one inspection falling into the C-1 category for that unit.]~~

~~b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 5.5.9-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.3.a; the interval may then be extended to a maximum of once per 40 months.~~

~~c. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:~~

~~1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of Specification 3.4.13.~~

~~2. A seismic occurrence greater than the Operating Basis Earthquake.~~

~~3. A loss-of-coolant accident requiring actuation of the engineered safeguards.~~

~~4. A main steam line or feedwater line break.~~

~~5.5.9.4 Acceptance Criteria~~

~~a. As used in this Specification:~~

~~1. Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal wall thickness, if detectable, may be considered as imperfections.~~

~~(continued)~~

5.5 Programs and Manuals

~~5.5.9.4 Acceptance Criteria (continued)~~

~~2. Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.~~

~~3. Degraded Tube means a tube that contains imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation.~~

~~4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.~~

~~5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective.~~

~~6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging and is greater than or equal to 40% of the nominal tube wall thickness.~~

~~7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 5.5.9.3.c, above.~~

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

~~9. Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed using the equipment and techniques expected to be used during subsequent inservice inspections.~~

~~(continued)~~

5.5 Programs and Manuals

~~5.5.9.4 Acceptance Criteria (continued)~~

~~b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plugging of all tubes exceeding the plugging limit) required by Table 5.5.9-2.~~

~~Table 5.5.9-1~~

~~No. of Steam Generators per Unit~~	~~Three~~
~~First Inservice Inspection~~	~~Two~~
~~Second and Subsequent Inservice Inspections~~	~~One*~~

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

Table 5.5.9-2
Steam Generator Tube Inspection

	1st Sample Inspection		2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per SG	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this SG	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this SG	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this SG, plug defective tubes and inspect 2S tubes in each other SG Notification to NRC pursuant to 10 CFR 50.73	All other SGs are C-1	None	N/A	N/A
			Some SGs C-2 but no additional SGs are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional SG is C-3	Inspect all tubes in each SG and plug defective tubes. Notification to NRC pursuant to 10 CFR 50.73	N/A	N/A

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

INSERT 5.5.9

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

    1. Structural integrity performance criterion: All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 ($3\Delta P$) 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 all SGs and leakage rate for an individual SG. Accident induced leakage is not to exceed 1 gpm total for all three SGs.

    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.

INSERT 5.5.9 (continued)

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.

5.6 Reporting Requirements

5.6.9 Tendon Surveillance Report (continued)

Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

(SG)

Inspection

5.6.10 Steam Generator Tube ~~Inspector~~ Report

Insert 5.6.10

~~a. Following each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission within 15 days of the completion of the plugging effort.~~

~~b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission within 12 months following the completion of the inspection. This Report shall include:~~

~~1. Number and extent of tubes inspected.~~

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

~~3. Identification of tubes plugged.~~

~~c. Results of steam generator tube inspections which fall into Category C-3 shall be considered a Reportable Event and shall be reported pursuant to 10 CFR 50.73 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.~~

5.6.11 Alternate AC (AAC) Source Out of Service Report

The NRC shall be notified if the AAC source is out of service for greater than 10 days.

INSERT 5.6.10

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 5.5.9, 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, and

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

B 3.4.17	**Steam Generator (SG) Tube Integrity**	**B 3.4.17-1**

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming three RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the complete loss of forced reactor coolant flow, single RCP locked rotor, partial loss of reactor coolant flow (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

Steady state DNB analysis has been performed for the three RCS loop operation. For three 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 120% RTP. This is the design overpower condition for three 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 possible instrumentation errors. 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 all 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.

RCS Loops — MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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, three pumps are required 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 Surveillance Program~~.

BASES

LCO (continued)

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 stopping 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 initial startup 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 one OPERABLE RCP and one OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~, which has the minimum water level specified in SR 3.4.5.2. This assumes steam removal capability and the availability of a makeup water source (if necessary for extended use of the SG) as required to remove decay heat. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the rod control system capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the rod control system not capable of rod withdrawal.

(continued)

BASES

LCO (continued)

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. This assumes steam removal capability and the availability of a makeup water source (if necessary for extended use of the SG) as required to remove decay heat.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops — MODES 1 and 2";
LCO 3.4.5, "RCS Loops — MODE 3";
LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops — MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level" (MODE 6).

ACTIONS

A.1

If one required RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(continued)

BASES

LCO (continued)

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 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be < 50°F above each of the RCS cold leg temperatures or that the pressurizer water volume is less than 770 cubic feet (24% of wide range, cold, pressurizer level indication) before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ 325°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

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.

Note 5 restricts the number of operating reactor coolant pumps at RCS temperatures less than 110°F. Only one reactor coolant pump is allowed to be in operation below 110°F (except during pump swap operations) consistent with the assumptions of the P/T Limits Curve.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A~~n OPERABLE~~ SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE~~ in accordance with the Steam Generator Tube Surveillance Program~~.

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is typically seen as a precursor to a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes ~~a 150 gpd per SG primary to secondary LEAKAGE as the initial condition.~~

Insert B 3.4.13 A

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

relatively

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via the main steam safety valves. The majority of the activity released to the atmosphere results from the tube rupture. Therefore, the ~~150 gpd per SG~~ primary to secondary LEAKAGE is inconsequential.

safety analysis assumption

1 gpm

The SLB is more limiting for primary to secondary LEAKAGE. The safety analysis for the SLB assumes 500 gpd and 470 gpd primary to secondary LEAKAGE in the ~~ruptured~~ and intact steam generators respectively as an initial condition. The dose consequences resulting from the SLB accident are bounded by a small fraction (i.e., 10%) of the limits defined in 10 CFR 100. The RCS specific activity assumed was 0.5 μCi/gm DOSE EQUIVALENT I-131 at a conservatively high letdown flow of 145 gpm, with either a pre-existing or an accident initiated iodine spike. These values bound the Technical Specifications values.

faulted

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

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher

(continued)

INSERT B 3.4.13 A

that primary to secondary LEAKAGE from all steam generators (SGs) is 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gpd (i.e. total leakage less than or equal to 450 gpd) is significantly less than the conditions assumed in the safety analysis (with leakage assumed to occur at room temperature in both cases).

INSERT B 3.4.13 B

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gpd per each 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.

BASES

LCO

a. Pressure Boundary LEAKAGE (continued)

LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

b. Unidentified LEAKAGE

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

c. Identified LEAKAGE

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

Insert B 3.4.13 B

~~d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)~~

~~The limits for total primary to secondary LEAKAGE through all SGs produce acceptable offsite doses in the SLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.~~

~~e. Primary to Secondary LEAKAGE through Any One SG~~

~~The limit on one SG 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 rupture. If~~

(continued)

BASES

LCO

~~e. Primary to Secondary LEAKAGE through Any One SG (continued)~~

~~leaked through many cracks, the cracks are very small, and the above assumption is conservative.~~

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.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is 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

or

A.1

Unidentified LEAKAGE~~,~~ 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

or primary to secondary LEAKAGE is not within limit,

or

If any pressure boundary LEAKAGE exists, or if unidentified ~~LEAKAGE,~~ identified LEAKAGE~~, or primary to secondary LEAKAGE~~ cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

LEAKAGE. 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.

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 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. ~~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.~~

The Surveillance is modified by two Notes. Note 1 states that

The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. ~~Therefore,~~ this SR is not required to be performed in MODES 3 and 4 until 12 hours of steady state operation near operating pressure have been established.

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 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 the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1 (continued)

containment atmosphere radioactivity and the containment air cooler condensate flow rate. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Insert B 3.4.13 C

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. A Note under the Frequency column states that this SR is required to be performed during steady state operation.

SR 3.4.13.2

Insert B 3.4.13 D

~~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. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 3.1.2.6, 5.2.7, 10.4, 11.0, 12.0 and 15.0.

Insert B 3.4.13 E

INSERT B 3.4.13 C

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE. This is because LEAKAGE of 150 gpd cannot be measured accurately by an RCS water inventory balance.

INSERT B 3.4.13 D

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gpd through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gpd limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one 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. During normal operation the primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with EPRI guidelines.

INSERT B 3.4.13 E

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

5. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

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. 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 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 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.

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 5.5.9, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, 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 5.5.9. 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 via the main steam safety valves. The majority of the activity released to the atmosphere results from the tube rupture.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gpm 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.

(continued)

BASES

LCO (continued)

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

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

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." 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 gallon per minute (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 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gpd. This limit is based on the assumption that a

(continued)

BASES

LCO (continued)

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.

<u>A.1 and A.2</u>

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 SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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.

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1 (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 5.5.9 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 5.5.9 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 and Reference 6 provide 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.

BASES

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 TR-107569, "Pressurized Water Reactor Steam Generator Examination Guidelines."

Joseph M. Farley Nuclear Plant
Request for Technical Specifications Change
Steam Generator Program

Enclosure 3

Clean-Typed Technical Specification and Bases Pages

Technical Specifications Page	Action	Technical Specification Bases Page	Action
TOC ii	Replace	B TOC ii	Replace
3.4.13-1	Replace	B TOC iii*	Replace
3.4.13-2	Replace	B 3.4.4-2	Replace
3.4.17-1	Insert	B 3.4.5-3	Replace
3.4.17-2	Insert	B 3.4.6-3	Replace
5.5-5	Replace	B 3.4.7-3	Replace
5.5-6	Replace	B 3.4.13-2	Replace
5.5-7	Replace	B 3.4.13-3	Replace
5.5-8*	Replace	B 3.4.13-4	Replace
5.5-9*	Replace	B 3.4.13-5	Replace
5.5-10*	Replace	B 3.4.13-6	Replace
5.5-11*	Replace	B 3.4.17-1	Insert
5.5-12*	Replace	B 3.4.17-2	Insert
5.5-13*	Replace	B 3.4.17-3	Insert
5.5-14*	Replace	B 3.4.17-4	Insert
5.5-15*	Remove	B 3.4.17-5	Insert
5.5-16*	Remove	B 3.4.17-6	Insert
5.5-17*	Remove	B 3.4.17-7	Insert
5.5-18*	Remove		
5.6-6	Replace		

* Pages affected by renumbering; content unchanged.

TABLE OF CONTENTS

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. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.13.1	-----NOTES----- 1. Not required to be performed in MODE 3 or 4 until 12 hours 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.	-----NOTE----- Only required to be performed during steady state operation ---------- 72 hours
SR 3.4.13.2	-----NOTE----- Not required to be performed until 12 hours after establishment of steady state operation. ---------- Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.	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 inspection.	7 days
	AND	
	A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	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.	B.1 Be in MODE 3.	6 hours
OR	AND	
SG tube integrity not maintained.	B.2 Be in MODE 5.	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 Programs and Manuals

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. 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:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
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 TS.

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the tubing during an SG inspection outage, as determined from the inservice

(continued)

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.

1. Structural integrity performance criterion: All inservice SG tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 ($3\Delta P$) 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 all SGs and leakage rate for an individual SG. Accident induced leakage is not to exceed 1 gpm total for all three SGs.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."

(continued)

5.5.9 Steam Generator SG Program (continued)

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.

5.5 Programs and Manuals

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The 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, which shall include monitoring the condenser hotwells for evidence of condenser in leakage;

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.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in, and in accordance with, ASME N510-1989. The FNP Final Safety Analysis Report identifies the relevant surveillance testing requirements.

a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 0.5% when tested in accordance with ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Flowrate (CFM)
CREFS Recirculation	2,000 ± 10%
CREFS Filtration	1,000 ± 10%
CREFS Pressurization	300 + 25% to - 10%
PRF Post LOCA Mode	5,000 ± 10%

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 0.5% when tested in accordance ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Flowrate (CFM)
CREFS Recirculation	2,000 ± 10%
CREFS Filtration	1,000 ± 10%
CREFS Pressurization	300 + 25% to - 10%
PRF Post LOCA Mode	5,000 ± 10%

c. Demonstrate for each of the ESF systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in ASME N510-1989, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of ≤ 30°C and greater than or equal to the relative humidity specified below.

ESF Ventilation System	Penetration	RH
CREFS Recirculation	2.5%	70%
CREFS Filtration	2.5%	70%
CREFS Pressurization	0.5%	70%
PRF Post LOCA Mode	5%	95%

NOTE: CREFS Pressurization methyl iodide penetration limit is based on a 6-inch bed depth.

d. Demonstrate for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested in accordance with ASME N510-1989 at the system flowrate specified below.

ESF Ventilation System	Delta P (in. water gauge)	Flowrate (CFM)
CREFS Recirculation	2.3	2,000 ± 10%
CREFS Filtration	2.9	1,000 ± 10%
CREFS Pressurization	2.2	300 + 25% to - 10%
PRF Post LOCA Mode	2.6	5,000 ± 10%

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

e. Demonstrate that the heaters for the CREFS Pressurization System dissipate the value specified below when tested in accordance with ASME N510-1989.

ESF Ventilation System	Wattage (kW)
CREFS Pressurization	2.5 $\pm$ 0.5

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Waste Gas System, the quantity of radioactivity contained in gas storage tanks, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the Waste Gas System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design;

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of $\geq$ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste Treatment System is less than 10 curies.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to the emergency diesel generator storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. a clear and bright appearance.

b. Fuel oil stored in the emergency diesel generator storage tanks is within limits by verifying that a sample of diesel fuel oil from the storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water, and sediment every 92 days.

c. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance test frequencies.

5.5.14 Technical Specifications (TS) Bases Control Program

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

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   1. a change in the TS incorporated in the license; or

   2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

(continued)

5.5 Programs and Manuals

5.5.14 Technical Specifications (TS) Bases Control Program (continued)

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

(continued)

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Main Steamline Inspection Program

The three main steamlines from the rigid anchor points of the containment penetrations downstream to and including the main steam header shall be inspected. The extent of the inservice examinations completed during each inspection interval (IWA 2400, ASME Code, 1974 Edition, Section XI) shall provide 100 percent volumetric examination of circumferential and longitudinal pipe welds to the extent practical. The areas subject to examination are those defined in accordance with examination category C-G for Class 2 piping welds in Table IWC-2520.

5.5.17 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of containment as required by 10 CFR 50.54 (o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following exception to NEI 94-01, Rev. 0, "Industry Guidelines for Implementing Performance-Based Option of 10 CFR 50, Appendix J":

Section 9.2.3: The next Type A test, after the March 1994 test for Unit 1 and the March 1995 test for Unit 2, shall be performed within 15 years. This is a one time exception.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 43.8 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.15% of containment air weight per day.

(continued)

5.5 Programs and Manuals

5.5.17 Containment Leakage Rate Testing Program (continued)

Leakage rate acceptance criteria are:

a. Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During plant startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and C tests, and $\leq 0.75\ L_a$ for Type A tests;

b. Air lock testing acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. For each door, leakage rate is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

c. During plant startup following testing in accordance with this program, the leakage rate acceptance criterion for each containment purge penetration flowpath is $\leq 0.05\ L_a$.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.6 Reporting Requirements

5.6.9 Tendon Surveillance Report (continued)

Program shall be reported to the NRC within 30 days. The report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective action taken.

5.6.10 Steam Generator (SG) 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 5.5.9, 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, and

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

5.6.11 Alternate AC (AAC) Source Out of Service Report

The NRC shall be notified if the AAC source is out of service for greater than 10 days.

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APPLICABLE SAFETY ANALYSES (continued)

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming three RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the complete loss of forced reactor coolant flow, single RCP locked rotor, partial loss of reactor coolant flow (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

Steady state DNB analysis has been performed for the three RCS loop operation. For three 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 120% RTP. This is the design overpower condition for three 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 possible instrumentation errors. 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 all 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.

RCS Loops — MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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, three pumps are required at rated power.

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG.

BASES

LCO
(continued)

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 stopping 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 initial startup 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 one OPERABLE RCP and one OPERABLE SG, which has the minimum water level specified in SR 3.4.5.2. This assumes steam removal capability and the availability of a makeup water source (if necessary for extended use of the SG) as required to remove decay heat. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the rod control system capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the rod control system not capable of rod withdrawal.

(continued)

BASES

LCO (continued)

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG, which has the minimum water level specified in SR 3.4.6.2. This assumes steam removal capability and the availability of a makeup water source (if necessary for extended use of the SG) as required to remove decay heat.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops — MODES 1 and 2";
LCO 3.4.5, "RCS Loops — MODE 3";
LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops — MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level" (MODE 6).

ACTIONS

A.1

If one required RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

(continued)

BASES

LCO (continued)

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 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 requires that the secondary side water temperature of each SG be < 50°F above each of the RCS cold leg temperatures or that the pressurizer water volume is less than 770 cubic feet (24% of wide range, cold, pressurizer level indication) before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ 325°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

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.

Note 5 restricts the number of operating reactor coolant pumps at RCS temperatures less than 110°F. Only one reactor coolant pump is allowed to be in operation below 110°F (except during pump swap operations) consistent with the assumptions of the P/T Limits Curve.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. A SG can perform as a heat sink via natural circulation when it has an adequate water level and is OPERABLE.

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is typically seen as a precursor to a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from all steam generators (SGs) is 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gpd (i.e. total leakage less than or equal to 450 gpd) is significantly less than the conditions assumed in the safety analysis (with leakage assumed to occur at room temperature in both cases).

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released via the main steam safety valves. The majority of the activity released to the atmosphere results from the tube rupture. Therefore, the 1 gpm primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

The SLB is more limiting for primary to secondary LEAKAGE. The safety analysis for the SLB assumes 500 gpd and 470 gpd primary to secondary LEAKAGE in the faulted and intact steam generators respectively as an initial condition. The dose consequences resulting from the SLB accident are bounded by a small fraction (i.e., 10%) of the limits defined in 10 CFR 100. The RCS specific activity assumed was 0.5 μCi/gm DOSE EQUIVALENT I-131 at a conservatively high letdown flow of 145 gpm, with either a pre-existing or an accident initiated iodine spike. These values bound the Technical Specifications values.

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

BASES

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

b. Unidentified LEAKAGE

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

c. Identified LEAKAGE

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

d. Primary to Secondary LEAKAGE Through Any One SG

The limit of 150 gpd per each 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.

BASES

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.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual PIV and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is 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 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 pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to 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.

BASES

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 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.

The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. The Surveillance is modified by two Notes. Note 1 states that this SR is not required to be performed in MODES 3 and 4 until 12 hours of steady state operation near operating pressure have been established.

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 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 the containment atmosphere radioactivity and the containment air cooler condensate flow rate. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

Note 2 states that this SR is not applicable to primary to secondary LEAKAGE. This is because LEAKAGE of 150 gpd 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. A Note under the Frequency column states that this SR is required to be performed during steady state operation.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less than or equal to 150 gpd through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gpd limit is measured at room temperature as described in Reference 5. The operational LEAKAGE rate limit applies to LEAKAGE through any one 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. During normal operation the primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with EPRI guidelines.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section 3.1.2.6, 5.2.7, 10.4, 11.0, 12.0 and 15.0.
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI TR-104788, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

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. 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 3.4.4, "RCS Loops - MODES 1 and 2," LCO 3.4.5, "RCS Loops - MODE 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.

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 5.5.9, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, 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 5.5.9. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

(continued)

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding 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 via the main steam safety valves. The majority of the activity released to the atmosphere results from the tube rupture.

The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture). In these analyses, the steam discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gpm 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.

(continued)

BASES

LCO (continued)

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

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

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification. Tube burst is defined as, "The gross structural failure of the tube wall. The condition typically corresponds to an unstable opening displacement (e.g., opening area increased in response to constant pressure) accompanied by ductile (plastic) tearing of the tube material at the ends of the degradation." Tube collapse is defined as, "For the load displacement curve for a given structure, collapse occurs at the top of the load versus displacement curve where the slope of the curve becomes zero." 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 gallon per minute (gpm) total from all SGs. The accident

(continued)

LCO (continued)

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 3.4.13, "RCS Operational LEAKAGE," 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 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

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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 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.

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.

BASES

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.

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 5.5.9 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 5.5.9

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.17.2 (continued)

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 and Reference 6 provide 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 TR-107569, "Pressurized Water Reactor Steam Generator Examination Guidelines."