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October 5, 2004
NOC-AE-04001794
10CFR50.90

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

South Texas Project
Units 1 and 2
Docket No. STN 50-498 and STN 50-499
License Amendment Request -
Revision to Proposed Amendment to Technical Specifications for Steam Generators

Reference: Letter, T. J. Jordan to NRC Document Control Desk, "License Amendment Request - Proposed Amendment to Technical Specifications for Steam Generators," dated August 12, 2004 (NOC-AE-04001765)

STP Nuclear Operating Company (STPNOC) hereby submits a revision to the referenced letter as a result of a conference call with the NRC staff on September 22, 2004. Attachment 1 to this letter addresses a request for additional information; Attachment 2 provides the revised evaluation; Attachment 3 provides the revised Technical Specification pages; and Attachment 4 provides the revised Bases pages (for information only). All revisions are marked with heavy change bars. This submittal supercedes the referenced letter in its entirety.

The Plant Operations Review Committee has recommended approval of this revised license amendment request and STPNOC has notified the State of Texas in accordance with 10 CFR 50.91(b).

STPNOC requests approval of the proposed change by November 1, 2004 to allow timely decisions regarding the scope of refueling outage 1RE12. If SG tube inspections are required for 1RE12, the level 3 schedule will have to be revised significantly and the contract award process will have to be expedited.

There are no new commitments in this letter.

A001

STI: 31793861

If there are any questions regarding this revision to the proposed license amendment, please contact John Conly, at (361) 972-7336 or me at (361) 972-7902.

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

Executed on October 5, 2004

T. J. Jordan
Vice President,
Engineering & Technical Services

jtc

Attachments:

1. Response to Request for Additional Information
2. Revised Licensee's Evaluation
3. Revised Technical Specification Pages (Marked Up)
4. Revised Bases Pages (For Information Only)
5. Steam Generator Design Information

cc:
(paper copy)

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Attachment 1

Response to Request for Additional Information

Request for Additional Information

NRC Staff Conference Call on September 22, 2004

1. Please confirm that there is no contradiction between the following sentences:

Att 3, page 2/12	"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."
Att 3, page 8/12	"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)"
Att 3, page 9/12	"The SLB is more limiting for primary-to-secondary leakage."

Response:

The primary-to-secondary rupture flow from an SGTR accident is the dominant release path for a SGTR accident and is used to determine the offsite dose consequences for the accident. The SGTR constitutes the limiting accident regarding SG tube integrity. However, in this accident, and in other accidents involving a steam release from the secondary side, an additional amount of operational primary-to-secondary leakage is assumed. The primary example of this is the main steam line break (MSLB). While the additional primary-to-secondary leakage is almost negligible in a SGTR accident, it is the major release source for a MSLB.

The rupture flow is dominant in the SGTR accident. The allowable operational primary-to-secondary leakage is determined by the consequences of the MSLB analysis. The two values and the statements made on pages 2/12, 8/12, and 9/12 are not mutually exclusive or contradictory.

2. Regarding the following sentence from Attachment 3, page 8/12:

"The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary-to-secondary leakage from all steam generators is 1 gpm as a result of accident induced conditions."

What temperature is assumed for the 1 gpm? Does the computer program convert the temperature?

Response:

The primary-to-secondary leakage is assumed to be 1 gpm at cold conditions (8.33 lbm/cu ft). This is used in a hand calculation of source terms that are inputs to a computer program.

Attachment 2

Revised Licensee's Evaluation

LICENSEE'S EVALUATION

1.0 DESCRIPTION

This letter is a request to amend Operating Licenses NPF-76 and NPF-80 for South Texas Project (STP) Units 1 and 2. The proposed change is based on draft Technical Specification Task Force (TSTF) Improved Standard Technical Specifications Change Traveler TSTF-449, Rev. 2. The change will implement guidance for the industry initiative on NEI 97-06, "Steam Generator Program Guidelines."

STP Nuclear Operating Company (STPNOC) requests approval of the proposed change by November 1, 2004 to allow timely decisions regarding the scope of refueling outage 1RE12. If SG tube inspections are required for 1RE12, the level 3 schedule will have to be revised significantly and the contract award process will have to be expedited.

2.0 PROPOSED CHANGE

The proposed change will:

- Replace Technical Specification (TS) 3/4.4.5, "Steam Generators" with 3/4.4.5, "Steam Generator Tube Integrity"

The proposed change replaces the detail of 3/4.4.5 with one action requiring that steam generator (SG) tube integrity be maintained and that each SG tube that satisfies the repair criterion be plugged in accordance with the Steam Generator Program. The detail of the replaced TS 3/4.4.5 is provided in the Steam Generator Program document.

- Revise TS 3/4.4.6.2, "Reactor Coolant System Operational Leakage"

The limit of 150 gpd primary-to-secondary leakage through any one SG had already been implemented in the TS after approval by the NRC in September 1997. The existing allowance of 4 hours to reduce primary-to-secondary leakage before having to shut down is eliminated. Primary-to-secondary leakage will be determined by monitoring or analyzing secondary coolant activity levels.

- Add TS 6.8.3.o, "Steam Generator Program"

The proposed change adds a new TS requiring a 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.

- Add TS 6.9.1.7, "Steam Generator Tube Inspection Report"

The proposed change adds a new TS stating the requirements for and contents of the SG tube inspection report. The TS requires a report within 180 days of initial entry into Mode 4 following a steam generator inspection.

3.0 BACKGROUND

The SG tubes in pressurized water reactors (PWRs) 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 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 throughwall SG tube repair criteria (e.g., ≥ 40 %) that has historically been incorporated into most 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 throughwall 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 throughwall criterion for all forms of degradation or obtain NRC 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." 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 Steam Generator 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.

The NRC pursued resolution of SG performance issues in parallel with the industry efforts. In December 1998, the NRC staff acknowledged that the Steam Generator Program described by NEI 97-06 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 the primary coolant chemistry controls because STP currently follows the Primary Water Chemistry Guideline (Ref. 6). The primary coolant activity limit and its assumptions are not affected by the proposed changes to the

TS. 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 an 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). At STP, these analyses assume that the total primary-to-secondary leakage is 1 gpm. For accidents that do not involve fuel damage, the reactor coolant activity levels are at the TS values. For accidents that involve fuel damage, the primary coolant activity values are a function of the amount of activity released from the damaged fuel.

The current TS RCS operational leakage limit of 150 gpd primary-to-secondary leakage through any one SG is based on operating experience as an indication of one or more tube leaks. This leakage limit provides an effective measure for minimizing the frequency of steam generator tube rupture prior to plant shutdown.

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 TS for operational leakage and for dose equivalent I-131 in primary coolant to ensure the plant is operated within its analyzed condition.

The other proposed TS changes are in general a significant improvement over current requirements. They replace an outdated prescriptive TS with one that references Steam Generator Program requirements, which 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 TS. As a result, these proposed changes will result in added assurance of the function and integrity of SG tubes.

The following table and 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 TS.

Condition or Requirement	Current Licensing Basis	Location – Proposed Change	Section
Operational primary-to-secondary leakage	≤ 150 gpd through any one SG	No change; this is already a requirement in TS 3.4.6.2	1
RCS primary-to-secondary leakage through any one SG not within limits	Reduce leakage to within limits in 4 hours or be in Hot Standby within 6 hours and in Cold Shutdown within the next 30 hours	TS 3.4.6.2a – delete the 4-hour period to reduce leakage. Revise Action 3.4.6.2a to require shutdown if primary-to-secondary leakage is not within the limit.	2
RCS leakage determined by water inventory balance (TS 4.4.6.2.1c)	Primary-to-secondary leakage is determined by water inventory balance.	Add Notes to TS 4.4.6.2.1c. regarding performance of water inventory balance every 72 hours: 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary-to-secondary leakage.	3
SG tube integrity verification		Add TS 4.4.6.2.3 and add a Note: 1. Not required to be performed until 12 hours after establishment of steady state operation.	4

Condition or Requirement	Current Licensing Basis	Location – Proposed Change	Section
Frequency of tube integrity verification	6 to 40 months depending on SG category defined by previous inspection results	TS 3.4.5 – revise to require surveillance frequency in accordance with TS 6.8.3.o. Frequency adopted from TSTF-449, Rev. 2 for Alloy 690TT. Steam Generator Program - establishes maximum inspection intervals	5
Tube sample selection	Based on SG category, industry experience, random selection, existing indications, and results of the initial sample set. The minimum is 3% times the number of SGs at the plant.	Steam Generator Program and implementing procedures - dependent on pre-outage evaluation of actual degradation locations and mechanisms, and operating experience; 20% of all tubes as a minimum.	6
Inspection techniques	Not specified	TS 3.4.5 requires that tube integrity be verified in accordance with the Steam Generator Program. Steam Generator Program and implementing procedures - establish requirements for qualifying NDE techniques. Require use of qualified techniques in SG inspections. Require a pre-outage evaluation of potential tube degradation morphologies and locations, and identification of NDE techniques capable of finding the degradation.	7

Condition or Requirement	Current Licensing Basis	Location – Proposed Change	Section
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 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 tubesheet weld to tubesheet weld.	8
Performance criteria	RCS operational leakage ≤ 150 gpd through any one SG No criteria specified for structural integrity or accident induced leakage	TS 3.4.6.2: RCS operational leakage ≤ 150 gpd through any one SG TS 3.4.5 requires that tube integrity be maintained. TS 6.8.3.o - defines structural integrity and accident induced leakage performance criteria which are dependent on design basis limits. Provides for condition monitoring assessment to verify compliance.	9
Repair criteria	Plug tubes with imperfections extending ≥40% throughwall and alternate criteria approved by the NRC and throughwall depth-based criteria for repair techniques approved by the NRC	TS 6.8.3.o: criteria unchanged.	10

Condition or Requirement	Current Licensing Basis	Location – Proposed Change	Section
Actions	Performance Criteria not defined. Primary-to-secondary leakage limit and actions included in the TS. Plug tubes satisfying repair criteria.	TS 3.4.5 and TS 3.4.6.2 contain primary-to-secondary leakage limit, SG tube integrity requirements, and actions required upon failure to meet performance criteria. Plug tubes satisfying repair criteria	11
Repair methods	Methods except plugging require previous approval by the NRC. Approved methods are listed in the TS (none at STP).	TS 6.8.3.o: requirements unchanged	12
Reporting requirements	Plugging and repair 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.	CFR - serious SG tube degradation (i.e., tubing fails to meet the structural integrity or accident induced leakage criteria) requires reporting in accordance with 10 CFR 50.72 and 50.73. TS 6.9.1.7 - 180 days after initial entry into Mode 4 after performing SG inspection	13
Definitions of SG terminology	TS 1.0, "Definitions," did not address SG program issues.	TS 6.8.3.o, TS Bases, Steam Generator Program procedures - include SG program terminology applicable only to SGs	14

Section 1 - Operational Leakage

The existing primary-to-secondary leakage limit is ≤150 gallons per day through any one SG. This leakage rate limit provides an effective measure for minimizing the frequency of steam generator 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), but a Note has been added to state that the surveillance need not be performed until 12 hours after establishment of steady state operation.

Section 2 - Operational Leakage Actions

If primary-to-secondary leakage exceeds 150 gallons per day through any one SG, a plant shutdown must be commenced. Mode 3 must be achieved within 6 hours and Mode 5 within the next 30 hours. The existing TS allow 4 hours to reduce primary-to-secondary leakage to less than the limit. The proposed change 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 TS that is significantly more conservative than the existing RCS operational leakage specification. This change is consistent with the Steam Generator Program.

Section 3 - RCS Operational leakage Determined by Water Inventory Balance

The proposed change adds a Note to TS 4.4.6.2.1c 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 gallons per day 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

The Steam Generator Program and the EPRI "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines" (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). Leakage verification is provided by chemistry procedures that provide alternate means of calculating and confirming primary-to-secondary leakage is less than or equal to 150 gpd through any one SG.

The surveillance frequency is unchanged. Determination of the primary-to-secondary leakage is required every 72 hours. The Surveillance Requirement is modified by a Note stating the requirement is not required to be performed until 12 hours after establishment of steady state operation. The 12-hour allowance provides sufficient time to collect and process necessary data after stable plant conditions are established. 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

The current TS 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 TS 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 TS 6.8.3.o.

The maximum inspection interval for Alloy 690 thermally treated tubing, the type used at STP, 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 TS, 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 TS (20% versus 3% 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 600TTand Alloy 690TT is significantly better than the performance of Alloy 600 mill annealed tubing, the material used in SG tubing at the time that the current TS were written. There are no known instances of cracking in Alloy 690TT tubes in either the U.S. or international SGs.

In summary, the proposed change is an improvement over the current TS. The current TS bases inspection intervals on the results of previous inspections; it does not require an evaluation of expected performance. The proposed TS 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.

<u>Section 6 - Steam Generator Tube Sample Selection</u>

The current TS 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 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 on 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.

In general, the sample selection considerations required by the current TS 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 TS requirements. In addition, the minimum sample size in the proposed requirements is larger.

Section 7 - Steam Generator Inspection Techniques

The Surveillance Requirements proposed in the Steam Generator Tube Integrity specification (TS 3/4.4.5) 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 TS 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 - Steam Generator Inspection Scope

The current TS 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.

The proposed change is an improvement over the current TS 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

The proposed change adds a performance-based Steam Generator Program to the TS. 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 an effective measure for minimizing the frequency of 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 TS 6.8.3.o. The requirements and methodologies established to meet the performance criteria are documented in the Steam Generator Program. The current TS 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 ($3\Delta P$) 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 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.

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 ($3\Delta P$) 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 $3\Delta P$ 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 $3\Delta P$ 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 steam generator 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 steam generator. Accident induced leakage is not to exceed 1 gpm total for all four SGs in a unit.

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 100 guidelines, or the radiological limits to control room personnel imposed by GDC-19, or other NRC approved licensing basis. As measured at

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 STP, the total 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 are approved for STP) would be limited to the specific criteria and type of degradation for which it was granted.

The operational leakage performance criterion is:

> "The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day."

Plant shutdown will commence if primary-to-secondary leakage exceeds 150 gallons per day (as measured at room temperature conditions) from any one SG. The operational leakage performance criterion is documented in the Steam Generator Program and implemented in Specification 3.4.6.2, "Reactor Coolant System Operational Leakage."

<u>Section 10 - Steam Generator Repair Criteria</u>

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

Tube repair criteria are established for each active degradation mechanism. Tube repair criteria are either the standard throughwall depth-based criterion (e.g., $\geq$ 40% throughwall) or throughwall 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 in accordance with approved methods. Steam generator tubes experiencing a damage form or mechanism for which no depth sizing capability exists are "plugged on detection" and their integrity should be assessed.

No plant-specific ARC are currently approved for STP.

Section 11 - Actions

The RCS Operational Leakage and Steam Generator Tube Integrity specifications require STP 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 TS 3.4.6.2. 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.

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 the Steam Generator Tube Integrity specification. If a performance criterion has been exceeded, a principal safety barrier has been challenged, and 10 CFR 50.72(b)(3)(ii)(A) and 10 CFR 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 additionally requires that the report contain information on the performance criteria exceeded and the basis for the planned operating cycle. The current TS 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 TS change to the actions required upon exceeding the operational leakage criterion is conservative with respect to the current TS as explained in Section 2 above.

The current TS 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 TS 6.9.1.7. The changes in the required reports are discussed below.

Section 12 - Steam Generator Repair Methods

Repair methods are those means used to reestablish the RCPB 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 tubes that satisfy the tube repair criteria to be plugged in accordance with the Steam Generator Program. No repair methods are currently listed in the STP Steam Generator Program 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 TS. 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 TS.

Section 13 - Reporting Requirements

The current TS 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 addition of TS 6.9.1.7 replaces the 15-day and the SG 12-month inspection reports with one report required within 180 days after initial entry into Mode 4 following inspection. The proposed report 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 10 CFR 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.

<u>Section 14 - SG Terminology</u>

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 Bases are controlled by the STP Technical Specification Bases Control Program, which appears in the TS Administrative Controls as TS 6.8.3.m.

The terms are described below.

1. Accident induced leakage rate means the primary-to-secondary leakage rate occurring during postulated accidents other than a SGTR. This includes the primary-to-secondary leakage rate existing immediately prior to the accident plus additional primary-to-secondary leakage induced during the accident.

   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 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 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.

3. The LCO section of Steam Generator Tube Integrity Bases 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 cannot 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 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.

<u>Conclusion</u>:

The proposed changes will provide greater assurance of SG tube integrity than that offered by the current TS. 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 (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 TS Sections 3/4.4.5 and 3/4.4.6.2, and adds Sections 6.8.3.o and 6.9.1.7. The proposed changes are necessary in order to implement the guidance for the industry initiative on NEI 97-06, "Steam Generator Program Guidelines," (Ref. 1).

STPNOC has evaluated whether a significant hazards consideration is involved with the proposed generic change by focusing on the three standards set forth in 10 CFR 50.92, 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 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 any 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 four SGs in a unit.

The operational leakage performance criterion is:

> "The RCS operational primary-to-secondary leakage through any one SG shall be limited to 150 gallons per day."

An SGTR event is one of the design basis accidents analyzed as part of the plant licensing basis. In the analysis of an 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 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). At STP these analyses assume that the total primary-to-secondary leakage is 1 gpm. The accident induced leakage criterion introduced by the proposed change 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 RCPB 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, includes a framework that incorporates a balance of prevention, inspection, evaluation, repair, 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 TS 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 an SGTR accident and the probability of such an accident is reduced. In addition, the proposed changes do not affect the consequences of an 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

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 are an integral part of the RCPB and, as such, are relied upon to maintain the primary system pressure and inventory. As part of the RCPB, 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 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 I 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.

The proposed change and the Steam Generator Program requirements which underlie it are in full compliance with the ASME Code. The proposed TS 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.

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 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," (Ref. 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.

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 50, Appendix A, define requirements for the RCPB with respect to structural and leakage integrity. Steam generator tubing and tube repairs constitute a major fraction of the RCPB surface area. Steam generator 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 STP), 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 area, 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 "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.

Attachment 3

Revised Technical Specifications Changes
(Mark-up)

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

BASES

SECTION PAGE

ADMINISTRATIVE CONTROLS

3/4.4.5 STEAM GENERATOR TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 ~~Each steam generator shall be OPERABLE.~~ Steam generator tube integrity shall be maintained.

AND

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

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

ACTION:

NOTE: Separate entry is allowed for each steam generator tube

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

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

AND

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

b. With steam generator tube integrity not maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVELLANCE REQUIREMENTS

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

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

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

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

~~3) A tube inspection (pursuant to Specification 4.4.5.4a.9) 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 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:~~

~~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, and~~

~~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.~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

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

~~a. The first inservice inspection following steam generator replacement shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality after the steam generator replacement. 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, 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;~~

~~Note: Inservice Inspection is not required during the steam generator replacement outage.~~

~~Note: For Unit 1, a one-time inspection interval of a maximum of once per 44 months is allowed for the inspection performed immediately following 1RE10. This is an exception to 4.4.5.3a in that the interval extension is based on all of the results of one inspection falling into the C-1 category.~~

~~b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40-month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3a.; the interval may then be extended to a maximum of once per 40 months; and~~

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

~~1) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.6.2, or~~

~~2) A seismic occurrence greater than the Operating Basis Earthquake, or~~

~~3) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or~~

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

~~4.4.5.4 Acceptance Criteria~~

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

~~1) Tube means that portion of the tube which forms the primary system to secondary system pressure boundary;~~

~~2) Imperfection means an exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;~~

~~3) Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;~~

~~4) Degraded Tube means a tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;~~

~~5) % Degradation means the percentage of the tube wall thickness affected or removed by degradation;~~

~~6) Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;~~

~~7) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness:~~

~~8) 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 Specification 4.4.5.3c., above;~~

~~9) 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;~~

~~10) 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 prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.~~

~~b) The steam generator shall be determined OPERABLE after plugging all tubes exceeding the plugging limit and all tubes containing trough-wall cracks required by Table 4.4-2.~~

~~4.4.5.5 Reports~~

~~a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be reported to the Commission in a Special Report;~~

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

~~1) Number and extent of tubes inspected,~~

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

~~3) Identification of tubes plugged or repaired.~~

~~c. Results of steam generator tube inspections which fall into Category C-3 shall be reported in a Special Report within 30 days and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

~~TABLE 4.4-1~~

~~MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION~~

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

~~TABLE NOTATIONS~~

1. ~~The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.~~

2. ~~The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections should follow the instructions described in 1 above.~~

3. ~~Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described in 1 above.~~

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N.A.	N.A.	N.A.	N.A.
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N.A.	N.A.
			C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S. G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action C-3 result of first sample
			C-3	Perform action for C-3 result of first sample.	N.A.	N.A.
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Notify NRC pursuant to 10CFR50.72 (b)(3)(ii)	All other S.G.s are C-1	None	N.A.	N.A.
			Some S.G.s C-2 but no additional S. G. are C-3	Perform action for C-2 result of second sample	N.A.	N.A.
			Additional S. G. is C-3	Inspect all tubes in each S. G. and plug defective tubes. Notify NRC pursuant to 10CFR50.72 (b)(3)(ii)	N.A.	N.A.

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

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System **operational** leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

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

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

e. 0.5 gpm leakage per nominal inch of valve size up to a maximum of 5 gpm at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.*

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

ACTION:

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

b. ~~With any Reactor Coolant System leakage greater than any one of the above limits, excluding PRESSURE BOUNDARY LEAKAGE and leakage from Reactor Coolant System Pressure Isolation Valves,~~ **With Reactor Coolant System operational UNIDENTIFIED or IDENTIFIED LEAKAGE greater than the above limits**, reduce ~~the~~ leakage ~~rate~~ to within limits within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With any Reactor Coolant System Pressure Isolation Valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed manual or deactivated automatic valves, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*Test pressures less than 2235 psig but greater than 150 psig are allowed. Observed leakage shall be adjusted for the actual test pressure up to 2235 psig assuming the leakage to be directly proportional to pressure differential to the one-half power.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.2.1 Note: this requirement is not applicable to primary-to-secondary leakage (refer to 4.4.6.2.3)

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

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

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

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours; and [(1)]

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

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

a. At least once per 18 months,

b. Prior to entering MODE 2 whenever the plant has been in COLD SHUTDOWN for 72 hours or more and if leakage testing has not been performed in the previous 9 months,

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

d. Prior to entering MODE 2 following valve actuation due to automatic or manual action or flow through the valve except for valves XRH0060 A, B, C, and XRH0061 A, B, C.

4.4.6.2.3 Primary-to-secondary leakage shall be verified $\leq$ 150 gallons per day through any one steam generator at least once per 72 hours [(1)]

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

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

6.8.3.n (continued)

2) The ODCM shall also contain descriptions of the radioactive effluent controls and radiological environmental monitoring activities, and descriptions of the information that should be included in the Annual Radiological Environmental Operating Report and the Radiological Effluent Release Report required by Specifications 6.9.1.3 and 6.9.1.4.

3) Licensee-initiated changes to the ODCM:

a) Shall be documented and records of reviews performed shall be retained.

This documentation shall contain:

1. Sufficient information to support the changes together with the appropriate analyses or evaluations justifying the changes and

2. A determination that the changes maintain the levels of radioactive effluent control required by 10 CFR 20.1 302, 40 CFR 190, 10 CFR 50.36a, and 10 CFR 50, Appendix I, and do not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

b) Shall become effective after approval of the plant manager.

c) Shall be submitted to the NRC in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (month and year) the change was implemented.

o. Steam Generator Program

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

6.8.3.o (continued)

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. Steam generator tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational leakage.

   1. Structural integrity performance criterion. All 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.

   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 four SGs in one unit.

6.8.3.o (continued)

3. The operational leakage performance criterion is specified in LCO 3.4.6.2, "Reactor Coolant System Operational Leakage."

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged

d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the 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.

6.8.3.o (continued)

3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.

e. Provisions for monitoring operational primary-to-secondary leakage.

6.9.1.6 (continued)

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

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

6.9.1.7 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.3.o. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

6.9.2 Not Used

Attachment 4

Bases Pages (For Information Only)

3/4.4.5 STEAM GENERATOR~~S~~ TUBE INTEGRITY

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

~~The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to minimize corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the 3.4.6.2.c limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System. Cracks having a primary-to-secondary leakage less than this limit during operation have a reasonably high likelihood of achieving "leak-before-break" conditions. Operating plants have demonstrated that primary-to-secondary leakage as low as 150 gallons per day per steam generator can readily be detected. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.~~

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

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

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.

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

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

Specification 6.8.3.o, "Steam Generator Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.8.3.o, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational leakage. The SG performance criteria are described in Specification 6.8.3.o. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

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

Applicable Safety Analyses

The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding 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 3.4.6.2, "Reactor Coolant System Operational Leakage," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released via the main steam safety valves. 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 1-131 is assumed to be equal to the limits in LCO 3.4.8, "Reactor Coolant System Specific Activity." For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

Limiting Condition for Operation (LCO)

The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program.

During 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. Refer to Action a. below.

In the context of this Specification, a SG tube is defined as the entire length of the tube, including the tube wall between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.8.3.o 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 gpm total from all SGs. The accident induced leakage rate includes any primary-to-secondary leakage existing prior to the accident in addition to primary-to-secondary leakage induced during the accident.

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

<u>Applicability</u>

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.

<u>ACTIONS</u>

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. The condition 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 Surveillance Requirement 4.4.5.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next 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, the plant must be shut down in accordance with the ACTION.

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

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

a. and b. Six hours to reach HOT STANDBY and an additional 30 hours to reach COLD SHUTDOWN 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.

<u>Surveillance Requirements</u>

4.4.5.1 During shutdown periods the SGs are inspected as required by this SR and the Steam Generator Program. NEI 97-06, Steam Generator Program Guidelines (Ref. 1), and its referenced EPRI Guidelines, establish the content of the Steam Generator Program. Use of the Steam Generator Program ensures that the inspection is appropriate and consistent with accepted industry practices.

A condition monitoring assessment of the SG tubes is performed during SG inspections. 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, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the frequency of SR 4.4.5.1. The frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.8.3.o contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

4.4.5.2 During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service (by plugging). The tube repair criteria delineated in Specification 6.8.3.o 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"

3/4.4.6.2 . OPERATIONAL LEAKAGE

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

~~Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.~~

~~The primary-to-secondary accident-induced leakage rate for the limiting design basis accident, other than the steam generator tube rupture, shall not exceed the leakage rate assumed in the safety analysis in terms of the total leakage rate for all steam generators, and the leakage rate for an individual steam generator. The total leakage shall not exceed 1 gpm.~~

~~The steam generator tube leakage limit of 150 gpd for each steam generator not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 150 gpd limit per steam generator is conservative compared to the assumptions used in the analysis of these accidents.~~

~~The 10 gpm IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the Leakage Detection Systems.~~

~~The specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.~~

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

Background

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

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

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

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

Applicable Safety Analyses

Except for primary-to-secondary leakage, the safety analyses do not address operational leakage. However, other operational leakage is related to the safety analyses for a LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes that primary-to-secondary leakage from all steam generators is 1 gpm as a result of accident induced conditions. The LCO requirement to limit primary-to-secondary leakage through any one steam generator to less than or equal to 150 gpd is significantly less than the conditions assumed in the safety analysis.

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 UFSAR analysis for SGTR assumes the contaminated secondary fluid is only briefly released via the main steam safety valves and the majority is steamed to the condenser. The 1 gpm primary-to-secondary leakage safety analysis assumption for the intact loops is relatively inconsequential.

The SLB is more limiting for primary-to-secondary leakage. The safety analysis for the SLB assumes 500 gpd and 936 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 1.0 µCi/gm DOSE EQUIVALENT 1-131 at a conservatively high letdown flow of 250 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).

Limiting Condition for Operation (LCO)

Reactor Coolant System operational leakage shall be limited to:

a. PRESSURE BOUNDARY LEAKAGE

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

b. 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 Reactor Coolant Pressure Boundary, if the leakage is from the pressure boundary.

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

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

d. IDENTIFIED LEAKAGE

Up to 10 gpm of IDENTIFIED LEAKAGE is considered allowable because leakage is from known sources that do not interfere with detection of UNIDENTIFIED LEAKAGE and is well within the capability of the Reactor Coolant System Makeup

System. IDENTIFIED LEAKAGE includes leakage to the containment from specifically known and located sources, but does not include PRESSURE BOUNDARY LEAKAGE or controlled reactor coolant pump seal leakoff (a normal function not considered leakage). Violation of this LCO could result in continued degradation of a component or system.

e. Reactor Coolant System Pressure Isolation Valve Leakage

The specified allowed leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series check valve failure. It is apparent that when pressure isolation is provided by two in-series check valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA that bypasses containment, these valves should be tested periodically to ensure low probability of gross failure.

Applicability

In MODES 1, 2, 3, and 4, the potential for Reactor Coolant Pressure Boundary leakage is greatest when the Reactor Coolant System is pressurized.

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

ACTIONS

a. If any PRESSURE BOUNDARY LEAKAGE exists, or primary-to-secondary leakage is not within limit, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within the next 30 hours. This ACTION reduces the leakage and also reduces the factors that tend to degrade the pressure boundary.

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

b. UNIDENTIFIED LEAKAGE or IDENTIFIED LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This allows time to verify leakage rates and either identify UNIDENTIFIED LEAKAGE or reduce leakage to within limits before the reactor must be shut down. This ACTION is necessary to prevent further deterioration of the Reactor Coolant Pressure Boundary.

Surveillance Requirements

4.4.6.2.1 Verifying Reactor Coolant System leakage to be within the LCO limits ensures the integrity of the Reactor Coolant Pressure Boundary is maintained.

PRESSURE BOUNDARY LEAKAGE would at first appear as UNIDENTIFIED LEAKAGE and can only be positively identified by inspection. It should be noted that leakage past seals and gaskets is not PRESSURE BOUNDARY LEAKAGE. UNIDENTIFIED LEAKAGE and IDENTIFIED LEAKAGE are determined by performance of a Reactor Coolant System 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 Surveillance Requirement is not required to be performed in until 12 hours after establishment of steady state operation.

Steady state operation is required to perform a proper water 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 Reactor Coolant Pump seal injection and return flows.

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

Note 2 states that this Surveillance Requirement is not applicable to primary-to-secondary leakage. This is because leakage of 150 gpd cannot be measured accurately by a RCS water inventory balance.

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

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

4.4.6.2.3 This Surveillance Requirement verifies that primary-to-secondary leakage is less than or equal to 150 gpd through any one steam generator. Satisfying the primary-to-secondary leakage limit ensures that the operational leakage performance criterion in the Steam Generator Program is met. If this Surveillance Requirement is not met, compliance with LCO 3.4.5 should be evaluated. The 150-gpd limit is measured at room temperature as described in Reference 1. The operational leakage rate limit applies to leakage through any one steam generator. If it is not practical to assign the leakage to an individual steam generator, all the primary-to-secondary leakage should be conservatively assumed to be from one steam generator.

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

The 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 MODES 3 and 4, the primary system radioactivity level may be very low, making it difficult to measure primary-to-secondary leakage. Leakage verification is provided by chemistry procedures that provide alternate means of calculating and confirming primary-to-secondary leakage is less than or equal to 150 gpd through any one SG (Ref. 2).

References

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

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

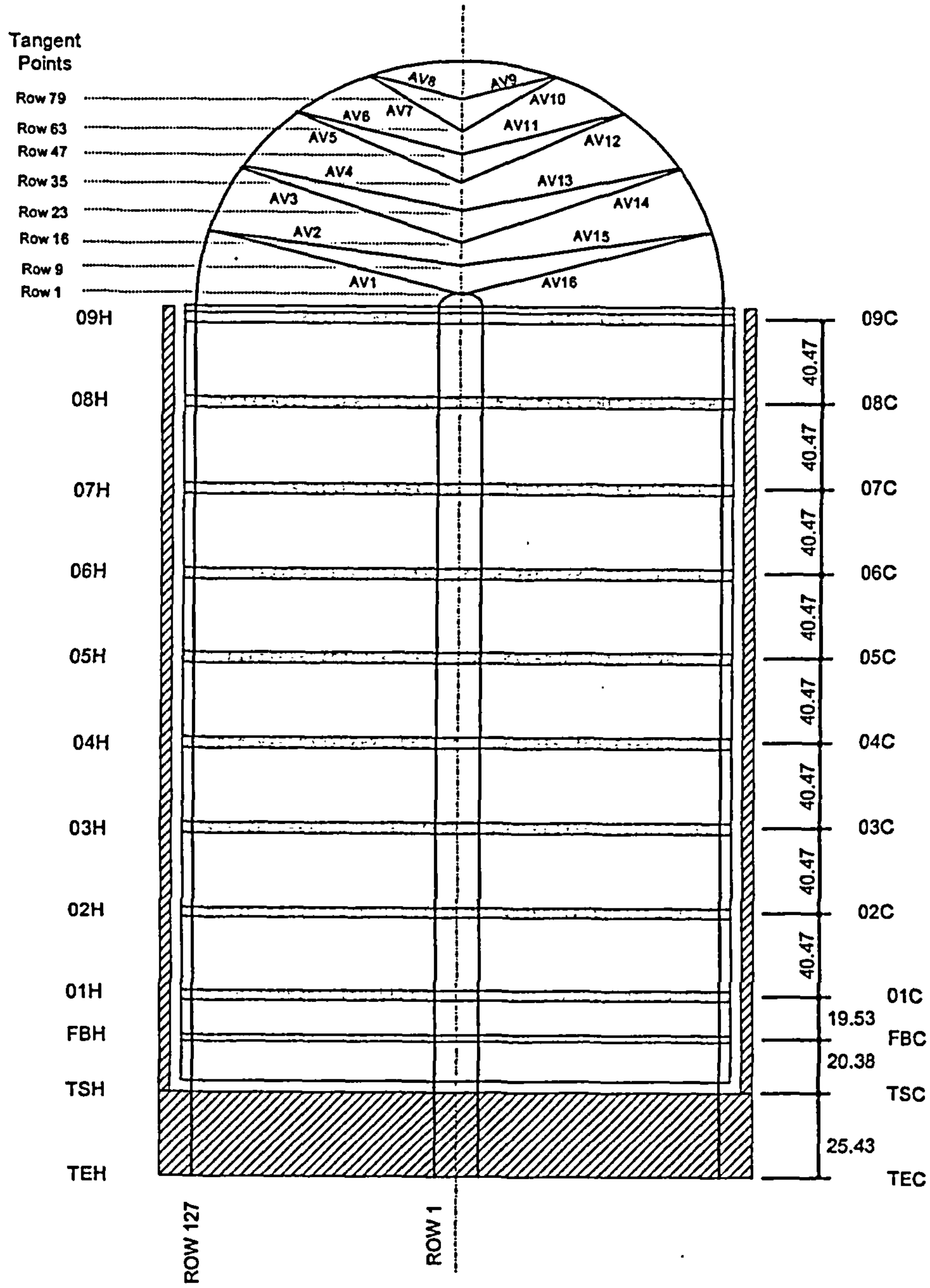
Attachment 5

Steam Generator Design Information

Steam Generator Design Information

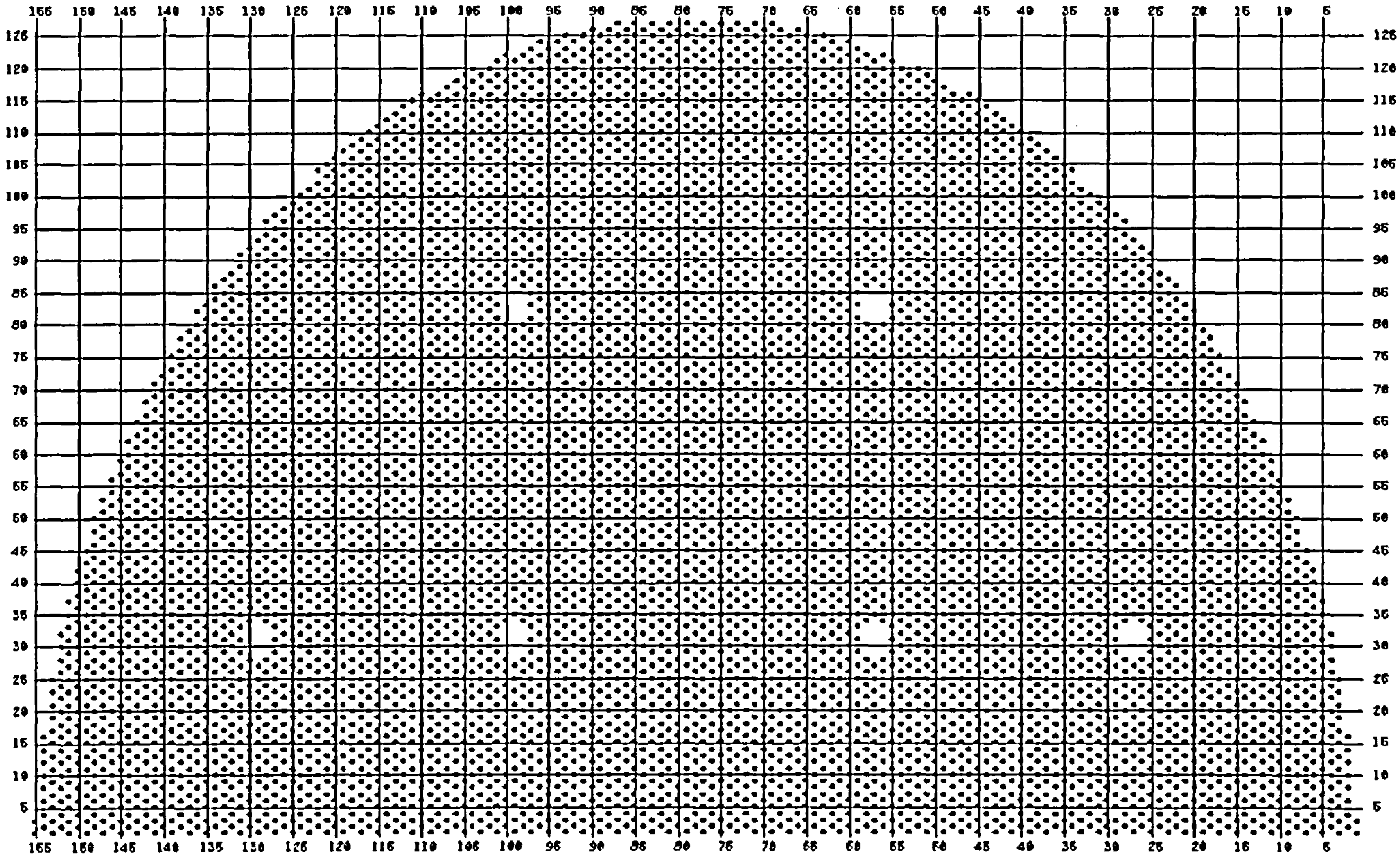
	Unit 1	Unit 2
Tube manufacturer	Sandvik	Sandvik
Steam generator fabricator	Westinghouse	ENSA
Tube spacing (triangular pitch)	0.980"	0.980"
AVB design and material	SA-479 type 405 SS flat bar 0.160" thick by 0.480" wide	SA-479 type 405 SS flat bar 0.160" thick by 0.480" wide
Tubesheet thickness	25.43"	25.43"
Radius of curvature and row number of thermally stress-relieved U-bends (U-bend centerline radius)	Row 1 - 3.250" Row 2 - 3.740" Row 3 - 4.230" Row 4 - 4.720" Row 5 - 5.210" Row 6 - 5.700" Row 7 - 6.190" Row 8 - 6.680" Row 9 - 7.170" Row 10 - 7.660" Row 11 - 8.150" Row 12 - 8.640" Row 13 - 9.130" Row 14 - 9.620" Row 15 - 10.110" Row 16 - 10.600" Row 17 - 11.090"	Row 1 - 3.250" Row 2 - 3.740" Row 3 - 4.230" Row 4 - 4.720" Row 5 - 5.210" Row 6 - 5.700" Row 7 - 6.190" Row 8 - 6.680" Row 9 - 7.170" Row 10 - 7.660" Row 11 - 8.150" Row 12 - 8.640" Row 13 - 9.130" Row 14 - 9.620" Row 15 - 10.110" Row 16 - 10.600" Row 17 - 11.090"
Number of tubes plugged prior to operation	A - 33 B - 40 C - 26 D - 9	A - 1 B - 2 C - 3 D - 0

Support Structure Nomenclature and Measurements



Westinghouse Delta 94

Supports 1 thru 9 = 1.125"
Flow Baffle = .750"



Tubesheet Map