Proposed Steam Generator Tech Spec

REACTOR COOLANT SYSTEM (RCS)

3.4.20 Steam Generator Tube Integrity

LCO 3.4.20 Steam Generator tube integrity shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3. <u>AND</u> A.2. Be in MODE 5	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.20.1	Verify steam generator tube integrity **<u>satisfies</u>** the structural integrity and accident induced **<u>LEAKAGE</u>** performance criteria **<u>in accordance with</u>** the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.20.2	Verify that **<u>steam generator</u>** tubes that exceed the repair criteria in the Steam Generator Program are plugged or repaired in accordance with repair methods in the Steam Generator Program.	Prior to **<u>entering</u>** MODE 4

Template for an
Administrative Section Technical Specification
for a
Steam Generator Program

Revision to NRC Submittal

Text in bold blue underlined font indicates a change from the version of the Steam Generator Generic License Change Package sent to the NRC in December 2000.

5.5.9 Steam Generator Program

A Steam Generator Program shall be established and implemented to ensure that steam generator tube integrity is maintained. Steam generator tube integrity is maintained by meeting the performance criteria as defined in the **Steam Generator Program.**

a. Condition Monitoring Assessment - Condition Monitoring Assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural and accident leakage integrity. The "as found" condition refers to the condition of the tubing during a steam generator inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the steam generator tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met. Requirements for condition monitoring are defined in the Steam Generator Program.

b. Performance Criteria - The steam generator performance criteria are defined in the Steam Generator Program. Revisions to performance criteria (and their associated definitions as used in the Steam Generator Program) require review and approval by the NRC. The performance criteria (and their associated definitions as used in the Steam Generator Program) may be revised to incorporate changes approved generically by the NRC subject to the limitations and conditions set forth in the staff's approving document.

c. Tube Repair Criteria and Repair Methods - Tube repair criteria and repair methods shall be described in and implemented by the Steam Generator Program. Repair criteria and repair methods may be implemented after review and approval by the NRC. In addition, repair criteria and repair methods approved generically by the NRC may be used subject to the limitations and conditions set forth in the staff's approving document. Note that tube plugging is not a repair and does not need to be reviewed or approved by the NRC.

NOTE: For plants that have not converted to the Improved Standard Technical Specifications or do not have a Technical Specification Bases Control Program in the Administrative Controls Section, the following sentence shall be added to 5.5.9b. and 5.5.9c.:

> **"Changes approved generically by the NRC shall be processed in accordance with 10 CFR 50.59."**

This statement is not required for those plants that have a TS Bases Control Program as the adoption of NRC approved generic changes would be made to the TS Bases and are required to be reviewed pursuant to 10 CFR 50.59. NOTE: This paragraph will not be included in the TSTF.

5.6.10 Steam Generator Tube Inspection Report

If the results of the steam generator inspection indicate greater than 1% of the inspected tubes in any steam generator exceed the repair criteria in accordance with the requirements of the Steam Generator Program, a Special Report shall be submitted within 120 days after the initial entry into MODE 4 following completion of the inspection. The report shall summarize:

a) The scope of inspections performed on each steam generator inspected in the affected unit during the current outage,
b) Active degradation mechanisms found,
c) NDE 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 or repaired during the inspection outage for each active degradation mechanism,
f) Repair method utilized and the number of tubes repaired by each repair method,
g) Total number and percentage of tubes plugged and/or repaired to date,
h) The effective plugging percentage for all plugging and tube repairs in each steam generator, and
i) The results of condition monitoring including the results of tube pulls and in-situ testing.

~~5.6.x Steam Generator Performance Criteria Reports~~

~~5.6.X.1 If steam generator tube integrity is not maintained in accordance with [3.4.X], the NRC report required within 60 days of the discovery shall provide the Performance Criteria exceeded and the basis for the planned operating period immediately following the discovery.~~

~~5.6.X.2 If a steam generator tube(s) is discovered to not meet the Performance Criteria while in MODES 5 or 6, the NRC report required within 60 days shall provide information on the Performance Criteria exceeded. In addition, a special report that provides the basis for the planned operating period immediately following the discovery shall be submitted to the NRC within 120 days of entering MODE 4.~~

Steam Generator Tube Integrity
Technical Specification Bases

<u>Changes from the TRM Bases are bold, blue and underlined.</u>

<u>Changes resulting from the 4/6/01 telecon with the NRC are bold, red, and underlined.</u>

TRM Steam Generator Integrity

BASES

BACKGROUND

The three Steam Generator Performance Criteria defined by the Steam Generator Program: Accident Induced Leakage, Structural Integrity, and Operational Leakage, act together to provide reasonable assurance of tube integrity at normal and faulted conditions. Steam generator tube integrity means that the tubes are capable of performing their intended safety functions consistent with their licensing basis, including applicable regulatory requirements. The Performance Criteria and the processes required to meet them are defined by the Steam Generator Program.

The purpose of the steam generator integrity LCO is to require compliance **with the two Performance Criteria that are necessary for primary to secondary pressure boundary integrity: Accident Induced Leakage and Structural Integrity**. **These two** Performance Criteria apply to steam generator tubes and associated appurtenances (e.g. plugs, sleeves, and other repairs).

The third Performance Criterion, Operational Leakage, is addressed by the Operational Leakage Technical Specification [3.4.13].

The steam generator (SG) tubes in pressurized water reactors 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 upon 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 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. **This steam generator tube integrity technical specification addresses the RCPB integrity function of the steam generator. The SG heat removal function is addressed by the RCS Loop Operability technical specifications.**

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.

The steam generator Performance Criteria identify the standards against which performance is to be measured. Meeting the Performance Criteria provides reasonable assurance that the steam generator tubing remains capable of fulfilling its specific safety function of maintaining RCPB integrity. structural under the full range

APPLICABLE SAFETY ANALYSIS

Incomplete. Also ensures against unacceptable accident induced leakage, also burst.

Satisfying the **steam generator** Performance Criteria provides reasonable assurance against tube Burst and ~~the~~ resulting primary to secondary leakage that might occur at normal and faulted conditions. The consequences of design basis accidents that include primary to secondary leakage are, in part, functions of the accident induced primary-to-secondary leakage rates and the dose equivalent I^{131} in the primary coolant.

The typical analysis for an event resulting in steam discharge to the atmosphere, except a steam generator tube rupture (SGTR), assumes that **the total** primary-to-secondary leakage **from** all steam generators is [1 gallon per minute] or increases to [1 gallon per minute] as a result of accident induced stresses. For accidents that do not involve fuel damage, the reactor coolant activity levels of dose equivalent I^{131} are **based on** the technical specification values. For accidents that do involve fuel damage, the primary coolant activity values are a function of the accident conditions.

For most PWRs, the SGTR accident is the limiting design basis event that establishes limits for these parameters. In the analysis of a SGTR event, a bounding primary-to-secondary leakage rate equal to the operational leakage rate limits in the technical specifications plus the leakage rate associated with a double-ended rupture of a single tube is assumed. The accident analysis for a SGTR assumes the contaminated secondary fluid is only briefly released to the atmosphere via safety valves and the majority is steamed to the main condenser.

BASES (continued)

For other design basis accidents such as main steam line break (MSLB), rod ejection, and reactor coolant pump locked rotor the tubes are assumed to retain their structural integrity (i.e., they are assumed not to rupture). The leakage is assumed to be at the design basis value, which is consistent with the Accident Induced Leakage Performance Criterion.

referenced

The steam generator **Accident Induced Leakage and Structural Integrity** Performance Criteria ~~addressed~~ **by this Technical Specification** and the limits included in the plant technical specifications for operational leakage and for dose equivalent I^{131} in primary coolant ensure the plant is operated within its analyzed condition. The dose consequences resulting from the most Limiting Design Basis Accident are within the limits defined in GDC 19 [1], 10 CFR 100 [2] or the NRC approved licensing basis (e.g., a small fraction of these limits).

No

"Specification 3.4.20" **Steam generator tube integrity" satisfies criterion 2 of 10 CFR 50.36 (c.)(2)(ii).**

LCO

The LCO requires that steam generator tube integrity be maintained. This means that the Accident Induced Leakage and Structural Integrity Performance Criteria are met. These Performance Criteria include design basis parameters that define acceptable steam generator performance. during operation in modes 1-4

met & maintained

Since conformance with the **Accident Induced Leakage and Structural Integrity** Performance Criteria can only be determined during SG inspections, compliance with the LCO during MODES 1 through 4 is ~~determined by verifying~~: verified by:

- ~~**satisfactory completion of an integrity**~~ **assessment during** each ~~**the last**~~ **steam generator inspection and** condition monitoring
- **that plant operation is within the operating cycle defined by the ~~Integrity~~ assessment.** operation assessment

Inconsistent with NEI-97-01

Performance Criteria

Accident Induced Leakage and Structural Integrity are two of the three Performance Criteria defined by the

BASES (continued)

<u>Steam Generator Program. These two, along with the third Performance Criteria, Operational Leakage, act together to provide reasonable assurance of tube integrity at normal and faulted conditions.</u>

The NRC must approve all Performance Criteria prior to use. The **<u>required process</u>** for approval of changes to the Performance Criteria **<u>is described in Administrative</u>** Technical Specification **<u>[5.5.9]. The three Performance Criteria approved for use at [Plant] are described below.</u>**

(i) <u>Structural Integrity Criterion</u>

The Structural Integrity Criterion is:

"Steam Generator Tubing shall retain Structural Integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against Burst under Normal Steady State Full Power Operation and a safety factor of 1.4 against Burst under the Limiting Design Basis Accident. Any additional loading combinations shall be included as required by existing design and licensing basis.."

Steam Generator Tubing refers to the entire length of the tube, including the tube wall and any repairs made to it, 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.

Changed No

In the context of the Structural Integrity Criterion, the Limiting Design Basis Accident is defined as the accident that results in the largest differential pressure across the steam generator tubes.

The Structural Integrity Criterion can be broken into two separate considerations:

BASES (continued)

- Providing a margin of safety against tube Burst under normal and accident conditions, and
- Ensuring Structural Integrity (preventing yield or Burst) of the SG tubes under all anticipated transients included in the design specification.

Tube Burst

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.

The Structural Integrity Criterion provides reasonable assurance that a steam generator tube will not Burst during normal or postulated accident conditions. The Structural Integrity Criterion requires that the tubes not Burst when subjected to differential pressures equal to three (3) times those experienced during normal steady state **full power** operation and 1.4 times accident differential pressures. In addition, other loading combinations are included as required by the design and licensing basis. The safety factors of 3 and 1.4 and the requirement to include applicable design basis loads are based on ASME Code Section III subsection NB [7] requirements and Draft Regulatory Guide 1.121 [8] guidance.

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For most plants the Normal Steady State Full Power Operation condition defines the most limiting parameters ~~under which the tubes are tested~~. In the context of the Structural Integrity Criterion, 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 their effects on differential pressure should be **included** if significant. Guidance on

BASES (continued)

accounting for changes in these parameters is provided in the EPRI Integrity Assessment Guidelines [6].

Tube Structural Integrity

In accordance with the ~~The~~ Structural Integrity Criterion, [redacted]t the primary pressure stresses must ~~do~~ not exceed the yield strength for the full range of normal operating conditions including startup, operation in the power range, hot standby, cooldown, and all anticipated transients included in the design specification. All appropriate loads contributing to combined primary plus secondary stress are evaluated so as to ensure that these loads do not significantly reduce the Burst pressure for the full range of normal operating conditions including postulated accidents. For example, axial loads due to tube-to-shell temperature differences in once-through steam generator designs during postulated MSLB, or axial loading associated with locked tube supports in recirculating steam generator designs are addressed to ensure that the test conditions are at least as severe as those expected during operating and accident events.

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(ii) Accident Induced Leakage Criterion

The Accident Induced Leakage Criterion is:

"The primary to secondary Accident Induced Leakage Rate for the Limiting Design Basis Accident, 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 steam generators and leakage rate for an individual steam generator. Leakage is not to exceed [1 gpm per steam generator, except for specific types of degradation at specific locations where the NRC has approved greater accident-induced leakage as part of a plant's licensing basis. Exceptions to the 1 gpm limit can be applied if approved by the NRC in conjunction with approved Alternate Repair Criteria]."

BASES (continued)

In the context of the Accident Induced Leakage Criterion:

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

- For steam generator primary-to-secondary pressure boundary integrity considerations, Limiting Design Basis Accident is defined as the accident that results in the minimum margin to the applicable dose limits.

The Accident Induced Leakage Criterion can be broken down into two separate considerations:

- Meeting design basis conditions, and
- Limiting Accident Induced Leakage to less than 1 gpm per steam generator under all circumstances.

Design Basis

Primary to secondary leakage is a factor in the dose releases outside containment resulting from a Limiting Design Basis Accident. The radiological dose consequences resulting from a potential primary-to-secondary leak during postulated design basis accidents must not exceed the offsite dose limits required by 10 CFR Part 100 [2] or the control room personnel dose limits required by GDC-19 [1] or the NRC approved licensing basis.

In most cases when calculating offsite doses, the safety analysis for the Limiting Design Basis Accident, other than a steam generator tube rupture, assumes a total of [1 gpm] primary to secondary leakage as an initial condition. Plant specific assumptions for Accident Induced Leakage are defined in each licensee's licensing basis. The leakage value used in the

BASES (continued)

Accident Induced Leakage Criterion must be consistent with the licensing basis.

Limiting Accident Induced Leakage to 1 gpm per SG

Probabilistic safety analysis sensitivity studies have shown that accident risk is sensitive to certain design basis parameters such as 1 gpm Accident Induced Leakage per SG. As a result, leakage greater than the design basis or 1 gpm per steam generator **(whichever is less)** is not allowed unless the NRC has approved greater leakage rates as part of an Alternate Repair Criterion.

(iii) Operational Leakage Criterion

The Operational Leakage Criterion and its associated action and surveillance requirements are contained in the RCS Operational Leakage Technical Specification. The Operational Leakage Criterion is not included in **the Steam Generator Tube Integrity Technical Specification because it is one of the forms of RCS leakage that are addressed by the RCS Operational LEAKAGE technical specification and because, unlike Structural Integrity and Accident Induced leakage, it is measurable and observable by the operator during MODES 1 through 4. The Operational Leakage Criterion is presented below to facilitate an understanding of all of the Performance Criteria since they act together to ensure tube integrity**.

The Operational Leakage Criterion is:

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

An explanation of the Operational Leakage Criterion is

BASES (continued)

provided in the Basses for the Operational LEAKAGE technical specification [3.4.13].

APPLICABILITY

Steam generator tubes are designed to withstand the stresses due to differential pressures as large as 3 times those experienced under normal full power operations or 1.4 times those experienced during a Limiting Design Basis Accident. This requirement is delineated in the Structural Integrity Criterion. This magnitude of differential pressure is only possible during MODES 1, 2, 3, and 4.

RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1 through 4. When the plant is shutdown, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for leakage. In addition, primary coolant activity is also low. Therefore this LCO is applicable in MODES 1 through 4 only.

ACTIONS

A. **The Accident Induced Leakage and Structural Integrity Performance Criteria must be met in order to ensure tube integrity.**

No, operational assessments must always be performed (see NEI-97-06

Restate consistent with NEI 97-06

If there is reason to believe that the Accident Induced Leakage or Structural Integrity Performance Criteria are not being met during MODES 1 through 4, an evaluation must be performed to determine compliance. The evaluation process and corresponding acceptance criteria are defined in the Steam Generator Program.

If an operating plant determines through operational assessment that either **Performance Criteria (i) or (ii) is not met, the reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the factors that tend to challenge tube integrity.**

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly

BASES (continued)

manner and without challenging plant systems. In MODE 5 the pressure stresses acting on the RCPB are much lower and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

A. During shutdown periods the steam generators will be inspected as required by the Steam Generator Program. The existence of the Steam Generator Program is required by **Administrative** Technical Specification **[5.6.10]**. NEI 97-06, Steam Generator Program Guidelines [5], and its referenced EPRI Guidelines establish the content of the Steam Generator Program.

Not the same

During steam generator inspections the licensee will perform an integrity assessment of the steam generator tubes. The purpose of the integrity assessment is to ensure that the Performance Criteria have been met for the previous operating period (i.e., condition monitoring), and will continue to be met for the next period (i.e., operational assessment).

The condition monitoring assessment determines the "as found" condition of the steam generator tubes with respect to the Structural Integrity and Accident Induced Leakage Performance Criteria. **The Steam Generator Program defines the methods used to determine compliance with the Performance Criteria. Use of the Steam Generator Program ensures that the methods used to determine tube condition with respect to the Performance Criteria are appropriate and consistent with accepted industry practices.**

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The condition of the steam generator tubes with respect to the Performance Criteria is then used to assess tube integrity and the effectiveness of the Steam Generator program. ~~This assessment may be performed analytically or by test.~~ ??

The Steam Generator Program defines the frequency of SR 3.4.X.1. The frequency is determined as part of the integrity assessment. The integrity assessment determines the length of the surveillance period by using information on existing degradations and growth rates to define a cycle length that provides reasonable

operational assessment

assurance that the tubing will meet the Performance Criteria at the next scheduled inspection.

B. During a steam generator inspection, any tube that exceeds Steam Generator Program Repair Criteria is repaired or removed from service by plugging. Repair Criteria are those NDE measured parameters at or beyond which a tube must be repaired using an approved Repair Method or removed from service by plugging. The tube Repair Criteria establish limits for tube degradation that provide reasonable assurance that an affected tube will meet the Performance Criteria at the next scheduled inspection by allowing for anticipated growth during the intervening time interval. Because of this allowance for growth, exceeding a tube repair criterion does not necessarily **mean that** the **steam generator tube integrity ~~is~~** are **not met.** performance criteria

Tube Repair Criteria are either the **standard** through-wall (TW) depth-based criterion (**e.g.**, 40% TW for most plants) or other Alternate Repair Criteria (ARC) approved by the NRC such as a voltage-based repair limit per Generic Letter 95-05.

Delete. Alternatives discuss "plug on detection" practice, consistent with response to GL-97-xx

The depth based criterion, approved for use at all plants by the NRC, was established when the most frequent form of degradation was general wastage corrosion. This type of degradation structurally bounds other forms of degradation and is characterized by a volumetric loss of the tube wall. This criterion was established to allow for NDE uncertainties and growth and still provide a reasonable assurance that the affected tube would not fail in the event of an accident. "Repair / plug in detection" is considered a subset of the depth based criterion. Additional basis information is provided in Draft Regulatory Guide 1.121 [8]. ?

In recent years, improved inspection techniques, knowledge of corrosion mechanisms, and experience have revealed additional types of tube degradation in the form of cracks in the tube wall. In some instances, a reliable method of characterizing specific types of cracks at defined locations within certain steam generator designs has been developed. In these cases, the industry has developed, and the NRC has approved Alternate Repair Criteria (ARC) to permit leaving a tube in service (as opposed to plugging) when the tube has indications that fall within the limits established by the ARC. ~~Plug or repair on detection is not an ARC.~~

The NRC must approve all Repair Criteria prior to use. The **required process** for approval of changes to the Repair Criteria **is described in Administrative** Technical Specification **[5.5.9].** New plugging designs or methods are not ARCs and do not require prior approval by the NRC. Repair Criteria currently approved for use at [Plant] are:

- [40%] nominal tube wall thickness
- [Other Repair Criteria that are currently approved for use – list.] describe – wording should be consistent with wording in existing TS

Due to technique and analyst uncertainties, sampling plans, and probability of detection there is a possibility that tube(s) exceeding the Repair Criteria will not be detected during a particular steam generator inspection. If the flaw(s) is detected during a subsequent inspection, the condition is not considered a reportable event unless it is determined that the Performance Criteria are not met.

Steam generator tube repairs are only performed using approved Repair Methods. Repair Methods are those means used to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. Plugging a steam generator tube is not a repair.

The NRC must approve all Repair Methods prior to use. The **required process** for approval of changes to the Repair Methods **is described in Administrative** Technical Specification **[5.5.9].** The Repair Methods currently approved for use at [Plant] are:

- [...]

During steam generator inspections inspected steam generator tubes that exceed the Repair Criteria **are** repaired or removed from service by plugging prior to entry into MODE 4. This is necessary in order to provide reasonable assurance that tube integrity will be maintained until the next scheduled inspection.

BASES (continued)

Operational Assessment

<u>During shutdowns without steam generator inspections</u>, the surveillance requirement is **<u>satisfied</u>** by **<u>verifying</u>** that an acceptable **<u>~~Integrity~~</u>** Assessment exists and that the plant is within the operating cycle determined by the **<u>Assessment</u>**.

REFERENCES

1. 10 CFR 50 Appendix A, GDC 19, *Control Room*
2. 10 CFR 100, *Reactor Site Criteria*
3. 10 CFR 50.36, *Technical Specifications*
4. NEI 97-06, *Steam Generator Program Guidelines*
5. EPRI Report TR-107621, *Steam Generator Integrity Assessment Guidelines*
6. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, *Rules for Construction of Nuclear Facility Components, Class 1 Components*
7. Draft Regulatory Guide 1.121, *Basis for Plugging Degraded Steam Generator Tubes*, August 1976
8. List applicable UFSAR sections.

Proposed RCS Operational Leakage Tech Spec

REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS Operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;
b. 1 gpm unidentified LEAKAGE;
c. 10 gpm identified LEAKAGE;
d. 150 gallons per day primary to secondary LEAKAGE through any one Steam Generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS Operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required action and associated completion time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limits.	B.1 Be in MODE 3 <u>AND</u> B.2. Be in MODE 5	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.13.1 ------------------------NOTE------------------------ 1. Not required to be performed in MODE 3 or 4 until 12 hours of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. -- Perform RCS water inventory balance.	--------------NOTE--------------- Only required to be performed during steady state operation -------------------------------------- 72 hours
SR 3.4.13.2 Verify primary to secondary LEAKAGE <u>is less than 150 gallons per day through any one SG in accordance with the Steam Generator Program.</u>	In accordance with the Steam Generator Program

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS. During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE. 10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

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

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

BASES

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analyses for events resulting in steam discharge to the atmosphere assumes **that primary to secondary LEAKAGE from all steam generators** (SGs) **is [one gallon per minute] or increases to [1 gallon per minute] as a result of accident induced conditions. The LCO requirement to limit primary to secondary** LEAKAGE **through any one** SG **to less than 150 gallons per day 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 FSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves and the majority is steamed to the condenser. The [1 gpm] primary to secondary LEAKAGE **assumption in the safety analysis** is relatively inconsequential.

The [SLB] is more limiting for site radiation releases. The safety analysis for the [SLB] accident assumes [1 gpm] primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the [SLB] accident are well within the limits defined in 10 CFR 100 or the staff approved licensing basis (i.e., a small fraction of these limits).

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

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

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

BASES

LCO (continued)

b. Unidentified LEAKAGE

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

c. Identified LEAKAGE

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

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per steam generator (SG) is based on the Operational LEAKAGE Performance Criterion in the Steam Generator Program. The Steam Generator Program criterion states:
"The RCS operational primary-to-secondary leakage through any one steam generator shall be limited to 150 gallons per day."

The RCS Operational primary to secondary **LEAKAGE** is measured at standard temperature and pressure.

The operational **LEAKAGE** rate limit applies to **LEAKAGE** in any one steam generator. If it is not practical to assign the **LEAKAGE** to an individual steam generator, all the **LEAKAGE** should be conservatively assumed to be from one steam generator.

The limit in this criterion is based on operating experience gained from SG tube degradation mechanisms that result in tube **LEAKAGE**. **The LEAKAGE rate criterion along with the other two Steam Generator Program Performance Criteria**

LCO (continued)

d. Primary to Secondary LEAKAGE through Any One SG

(Structural Integrity and Accident Induced LEAKAGE) provide reasonable assurance that a single flaw leaking this amount will not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture prior to detection by **LEAKAGE** monitoring methods and commencement of plant shutdown. If leaked through many flaws, the flaws are very small and the above assumption is conservative.

The other two Steam Generator Performance Criteria are addressed by the Steam Generator Tube Integrity technical specification ([3.4.X]).

APPLICABILITY

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

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

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

ACTIONS

A.1

Unidentified LEAKAGE**, or** identified LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion

Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

B.1 and B.2

If any pressure boundary LEAKAGE exists, **or if primary to secondary LEAKAGE is not within limits,** or if unidentified LEAKAGE **or** identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE to be within the LCO limits ensures the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and can only be positively identified by inspection. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. Unidentified LEAKAGE and identified LEAKAGE are determined by performance of an RCS water inventory balance

The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. **Calculations during maneuvering are not useful.** Therefore, **Note 1 states that** this SR is not required to be performed in MODES 3 and 4 until 12 hours of steady state operation near operating pressure have been established.

Note 2 states that this SR is not applicable to primary-to-secondary LEAKAGE because LEAKAGE limits as low as 150 gallons per day cannot be measured accurately by an RCS water inventory balance.

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. Steady state operation is required to perform a proper inventory balance and **therefore** a Note requires the Surveillance to be met when steady state is established. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

This SR requires the verification of the primary to secondary LEAKAGE limit specified in the LCO. Satisfying the primary to secondary LEAKAGE ~~limits~~ ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. The 150 gallons per day limit is measured at standard temperature and pressure.

The operational LEAKAGE Performance Criterion along with the other two Performance Criteria in the Steam Generator Program provide reasonable assurance that a single flaw leaking this amount will not propagate to an SGTR under the stress conditions of a LOCA or a main steam line rupture prior to detection by LEAKAGE monitoring methods and commencement of plant shutdown.

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, 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. The Steam Generator Program provides guidance on leak rate monitoring during MODES 3 and 4.

The surveillance frequency is determined by the Steam Generator Program requirements. The Steam Generator Program's primary – to - secondary LEAKAGE test frequencies are described in the EPRI PWR Primary-To-Secondary Leak Guidelines. The leak testing frequency changes as the amount of detected LEAKAGE increases. The greater the LEAKAGE, the more monitoring is required.

REFERENCES

1. 10 CFR 50, Appendix A, 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section [15].