NMC
Committed to Nuclear Excellence

Nuclear Management Company, LLC

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10 CFR 50.90

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U. S. Nuclear Regulatory Commission
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Point Beach Nuclear Plant Units 1 and 2
Dockets 50-266 and 50-301
Renewed License Nos. DPR-24 and DPR-27

Supplement to Application For Technical Specification Improvement Regarding Steam Generator Tube Integrity

Reference 1) License Amendment Request (LAR) titled, "Application For Technical Specification Improvement Regarding Steam Generator Tube Integrity", dated February 16, 2006

By letter dated February 16, 2006, Nuclear Management Company (NMC) submitted the referenced LAR to adopt Technical Specification (TS) improvements regarding steam generator tube integrity provided in Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler TSTF-449, "Steam Generator Tube Integrity", Revision 4. This letter supplements the referenced LAR to address April 27, 2006 telephone discussions with the Nuclear Regulatory Commission (NRC) Staff regarding Enclosures 4B and 5B which apply to the Point Beach Nuclear Plant, Units 1 and 2 (PBNP). NMC is submitting this supplement in accordance with the provisions of 10 CFR 50.90.

NMC proposes in this supplement to revise PBNP TS 5.5.8 paragraph b.2 on pages 5.5-7 and 5.5-8 to include the clause, "total leakage rate for all SGs", which is consistent with the PBNP accident analyses and the guidance of TSTF-449. Enclosure 1, which includes the marked up changes to pages 5.5-7 and 5.5-8, replaces Enclosure 4B of the Reference LAR in its entirety. Enclosure 2, which includes the revised changes to pages 5.5-7 and 5.5-8, replaces Enclosure 5B of the Reference LAR in its entirety.

The additional information provided in this supplement does not impact the conclusions of the Determination of No Significant Hazards Consideration and Environmental Assessment presented in the Reference February 16, 2006 submittal.

In accordance with 10 CFR 50.91, NMC is providing a copy of this letter and enclosures to the designated State Official.

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Summary of Commitments

This letter contains no new commitments and no revisions to existing commitments.

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

Edward J. Weinkam
Director, Nuclear Licensing and Regulatory Services
Nuclear Management Company, LLC

Enclosures (2)

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
Senior Resident Inspector, Point Beach Nuclear Plant, USNRC
State Official, Ms. Ave M. Bie – Public Service Commission of WI

ENCLOSURE 1

Proposed Technical Specification and Bases Pages (markup)

Point Beach Nuclear Plant Units 1 and 2

Technical Specification Pages

1.1-3	5.5-10
3.4.13-1	5.5-11
3.4.13-2	5.5-12
3.4.17-1	5.5-13
3.4.17-2	5.5-14
5.5-7	5.6-6
5.5-8	5.6-7
5.5-9	

Bases pages

B 3.4.4-2	B 3.4.13-7
B 3.4.5-3	B 3.4.17-1
B 3.4.6-2	B 3.4.17-2
B 3.4.7-3	B 3.4.17-3
B 3.4.13-2	B 3.4.17-4
B 3.4.13-3	B 3.4.17-5
B 3.4.13-4	B 3.4.17-6
B 3.4.13-5	B 3.4.17-7
B 3.4.13-6	

32 pages follow

1.1 Definitions

L_a	The maximum allowable primary containment leakage rate, L_a, shall be 0.4% of primary containment air weight per day at the peak design containment pressure (P_a).
LEAKAGE	LEAKAGE shall be: a. Identified LEAKAGE 1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator ~~(SG)~~ to the Secondary System (primary to secondary LEAKAGE); b. Unidentified LEAKAGE All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; c. Pressure Boundary LEAKAGE LEAKAGE (except primary to secondary ~~SG~~ LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150~~500~~ gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	---------------------------NOTES------------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS Operational LEAKAGE~~eakage~~ is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.13.2	---------------------------NOTE--------------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify ~~steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program~~ primary to secondary LEAKAGE is < 150 gallons per day through any one SG.	~~In accordance with the Steam Generator Tube Surveillance Program~~ 72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

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

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

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SG tube.

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for

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all SGs and leakage rate for an individual SG. Leakage is not to exceed 500 gallons per day per SG.

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

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

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

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. i. Unit 1 (alloy 600 Thermally Treated tubes): Inspect 100% of the tubes at sequential periods of 120, 90, 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 48 effective full power months or two refueling outages (whichever is less) without being inspected.

ii. Unit 2 (alloy 690 Thermally Treated tubes): 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

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outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

~~This program provides controls for inservice inspection and testing of steam generator tubing.~~

~~a. Definitions.~~

~~1. Tube Inspection – Entry from the hot-leg side with examination from the point of entry completely around the U-bend to the top support of the cold-leg is considered a tube inspection.~~

~~2. Imperfection – An exception to the dimension, 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 – A service induced cracking, wastage, wear or general corrosion occurring on either the inside or outside of a tube.~~

~~4. Degraded Tube – A tube that contains imperfections caused by degradation greater than 20% of the nominal tube wall thickness.~~

~~5. Defect – An imperfection of such severity that it exceeds the minimum acceptable tube wall thickness of 50%. A tube containing a defect is defective.~~

~~6. Plugging Limit – The imperfection depth beyond which the tube must be removed from service or repaired, because the tube~~

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~~may become defective prior to the next scheduled inspection.~~
~~The plugging limit is 40% of the nominal wall thickness.~~

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~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~b. Sample Selection and Testing~~

~~Selection and testing of steam generator tubes shall be made on the following basis:~~

~~1. One steam generator of each unit may be selected for inspection during inservice inspection in accordance with the following requirements:~~

~~i. The inservice inspection may be limited to one steam generator on an alternating sequence basis. This examination shall include at least 6% of the tubes if the results of the first or a prior inspection indicate that both generators are performing in a comparable manner.~~

~~ii. When both steam generators are required to be examined by Table 5.5.8-1 and if the condition of tubes in one steam generator is found to be more severe than in the other steam generator of a unit, the steam generator sampling sequence at the subsequent inservice inspection shall be modified to examine the steam generator with the more severe condition.~~

~~2. The minimum sample size, inspection result classification and the associated required action shall be in conformance with the requirements of Table 5.5.8-1. The results of each sampling examination of a steam generator shall be classified in the following three categories:~~

~~i. Category C-1: Less than 5% of the total tubes examined are degraded but none are defective.~~

~~ii. Category C-2: Between 5% and 10% of the total number of tubes examined are degraded, but none are defective or one tube to not more than 1% of the sample is defective.~~

~~iii. Category C-3: More than 10% of the total number of tubes examined are degraded, but none are defective or more than 1% of the sample is defective.~~

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~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~If the first sample of a given steam generator during any inservice inspection, degraded tubes not beyond the plugging limit detected by prior examinations in that steam generator shall be included in the above percentage calculations, only if these tubes are demonstrated to have a further wall penetration of greater than 10% of the tube nominal wall thickness.~~

~~3. Tubes shall be selected for examination primarily from those areas of the tube bundle where service experience has shown the most severe tube degradation.~~

~~4. In addition to the sample size specified in Table 5.5.8-1, the tubes examined in a given steam generator during the first examination of any inservice inspection shall include all non-plugged tubes in that steam generator that from prior examination were degraded.~~

~~5. During the second and third sample examinations of any inservice inspection, the tube inspection may be limited to those sections of the tube lengths where imperfections were detected during the prior examination.~~

~~c. Examination Method and Requirements.~~

~~The examination method shall meet the intent of the requirements in ASME Section XI Appendix IV. This includes equipment, personnel, and procedure requirements, certification and calibration along with records and reports. The actual technique may be the latest industry accepted technique, provided the flaw detection capability is as good or better than the technique endorsed by the code in effect per 10 CFR 50, Section 50.55a(g). This allows the use of improvements in inspection techniques that were not included in the code in effect. However, it means that word-for-word compliance with Appendix IV of ASME Section XI may not be possible.~~

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~~5.5.8 Steam Generator (SG) Tube Surveillance Program (continued)~~

~~d. Inspection Intervals~~

~~1. Inservice inspections shall not be more than 24 calendar months apart.~~

~~2. The inservice inspections may be scheduled to be coincident with refueling outages or any plant shutdown, provided the inspection intervals of 5.5.8.d.1 are not exceeded.~~

~~3. If two consecutive inservice inspections covering a time span of at least 12 months yield results that fall in C-1 category, the inspection frequency may be extended to 40 month intervals.~~

~~4. If the results of the inservice inspection of steam generator tubing conducted in accordance with Table 5.5.8-1 requires that a third sample examination must be performed, and the results of this fall in the category C-3, the inspection frequency shall be reduced to not more than 20 month intervals. The reduction shall apply until a subsequent inspection demonstrates that a third sample examination is not required.~~

~~5. Unscheduled inspections shall be conducted in accordance with Specification 5.5.8.b on any steam generator with primary-to-secondary tube leakage exceeding Specification 3.4.13.d. All steam generators shall be inspected in the event of a seismic occurrence greater than an operating basis earthquake, a LOCA requiring actuation of engineered safeguards, or a main steam line or feedwater line break.~~

~~e. Corrective Measures~~

~~All tubes that leak or have degradation exceeding the plugging limit shall be plugged or repaired by a process such as sleeving prior to return to power from a refueling or inservice inspection condition. Sleeved tubes having sleeve degradation exceeding 40% of the nominal sleeve wall thickness shall be plugged. (Brazed joints shall not be employed.~~

~~Tubes previously subject to explosive plugging shall not be sleeved)~~

~~The provisions of SR 3.0.2 are applicable to the SG Tube Surveillance Program test frequencies.~~

~~TABLE 5.5.8-1~~
~~STEAM GENERATOR TUBE INSPECTION PER UNIT~~
~~POINT BEACH UNITS 1 & 2~~

	~~1ST SAMPLE EXAMINATION~~		~~2ND SAMPLE EXAMINATION~~		~~3RD SAMPLE EXAMINATION~~	
~~Sample Size~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~
~~A minimum of S tubes per Steam Generator (S.G.)~~	~~C-1~~	~~Acceptable for continued service~~	~~N/A~~	~~N/A~~	~~N/A~~	~~N/A~~
~~S=3(N/n) %~~	~~C-2~~	~~Plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in same steam generator~~	~~C-1~~	~~Acceptable for continued service~~	~~N/A~~	~~N/A~~
~~Where:~~			~~C-2~~	~~Plug or repair tubes exceeding the plugging limit and proceed with 3rd sample examination of 4S tubes in same steam generator~~	~~C-1~~	~~Acceptable for continued service~~
~~N is the number of steam generators in the plant = 2~~					~~C-2~~	~~Plug or repair tubes exceeding plug limit. Acceptable for continued service~~
					~~C-3~~	~~Perform action required under C-3 of 1st sample examination~~
			~~C-3~~	~~Perform action required under C-3 of 1st sample examination~~	~~N/A~~	~~N/A~~
~~n is the number of steam generators inspected during an examination~~	~~C-3~~	~~Inspect essentially all tubes in this S.G., plug or repair tubes exceeding the plugging limit and proceed with 2nd sample examination of 2S tubes in the other steam generator. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).~~	~~C-1 in other S.G.~~	~~Acceptable for continued service~~	~~N/A~~	~~N/A~~
			~~C-2 in other S.G.~~	~~Perform action required under C-2 of 2nd sample examination above~~	~~N/A~~	~~N/A~~
			~~C-3 in other S.G.~~	~~Inspect essentially all tubes in S.G. & plug or repair tubes exceeding the plugging limit. Reportable in accordance with 10 CFR 50.73(a)(2)(ii).~~	~~N/A~~	~~N/A~~

5.6 Reporting Requirements

5.6.7 Tendon Surveillance Report (continued)

Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

h. The effective plugging percentage for all plugging in each SG.

~~(a) After each inservice examination, the number of tubes plugged or repaired in each steam generator shall be reported to the commission as soon as practicable.~~

5.6 Reporting Requirements

~~(b) The complete results of the steam generator tube inservice inspection shall be included in a report for the period in which the inspection was completed.~~

~~Reports shall include:~~

~~1. Number and extent of tubes inspected.~~
~~2. Location and percent of all thickness penetration for each indication.~~
~~3. Identification of tubes plugged or repaired.~~

~~(c) Reports required by Table 5.5.8-1, "Steam Generator Tube Inspection," shall provide the information required by Specification 5.6.8.(b) and a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence. The report shall be submitted to the Commission prior to resumption of plant operation.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the two pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

Steady state DNB analysis has been performed for the two RCS loop operation. For two RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 120% RTP. This is the design overpower condition for two RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 118% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops — MODES 1 and 2 satisfy Criterion 2 of the NRC Policy Statement.

LCO

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

In MODES 1 and 2, an OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

BASES

LCO (continued)

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction; and

c. The Rod Control System is not capable of rod withdrawal, to preclude the possibility of an inadvertent control rod withdrawal and associated power excursion.

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

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One RCS loop provides sufficient circulation for these purposes. However, one additional RCS loop is required to be OPERABLE to ensure redundant capability for decay heat removal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

BASES

LCO (continued)

that are designed to validate various accident analyses values. An example of one of the tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed during the initial startup testing program, and as such should only be performed once. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

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

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ the Low Temperature Overpressure Protection (LTOP) enabling temperature specified in the PTLR. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. SG secondary side water temperature can be approximated by using the SG metal temperature indicator.

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG ~~in accordance with the Steam Generator Tube Surveillance Program~~, which has the minimum water level specified in SR 3.4.6.2. The OPERABLE RCP and SG must be in the same loop for the RCS loop to be considered OPERABLE.

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

BASES

LCO (continued)

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

Note 3 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops. Note 4 also allows both RHR loops to be removed from operation when at least one RCS loop is in operation to allow for the performance of leakage or flow testing, as required by Technical Specifications or by regulation. This allowance is necessary based on the design of the Point Beach RHR System configuration, which requires the system to be removed from service to perform the required PIV testing.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An ~~OPERABLE~~ SG can perform as a heat sink via natural circulation (Ref. 1) when it has an adequate water level and is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program~~.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes.

However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least one SGs is required to be ≥ 30% narrow range.

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 analysis for an event resulting in steam discharge to the atmosphere assumes that primary to secondary LEAKAGE from each steam generator (SG) is 500 gpd or increases to 500 gpd as a result of accident induced conditions. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis. ~~500 gpd primary to secondary LEAKAGE as the initial condition.~~

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. 2) analysis for SGTR assumes the contaminated secondary fluid is only briefly released via safety valves. The 500 gpd primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential.

The SLB is more limiting for site radiation releases. The safety analysis for the SLB accident assumes 500 gpd primary to secondary LEAKAGE is through the affected~~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 SG 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 SG shall be limited to 150 gallons per day." The limit is based on operating experience with SG tube degradation mechanisms that result in tube leakage. The operational leakage rate criterion in conjunction with the implementation of the Steam Generator Program is an effective measure for minimizing the frequency of steam generator tube ruptures.
~~The 500 gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.~~

APPLICABILITY

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

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

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

ACTIONS

A.1

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

BASES

ACTIONS (continued)

B.1 and B.2

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

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

The RCS water inventory balance must be met with the reactor at steady state operating conditions (i.e., stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows). The Surveillance is modified by two Notes. ~~Therefore, a~~ N~~n~~ote 1 states~~is added allowing~~ that this SR is not required to be performed until 12 hours after establishing steady state operation.

The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

SURVEILLANCE REQUIREMENTS (continued)

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

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

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

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

SR 3.4.13.2

This SR verifies that primary to secondary LEAKAGE is less or equal to 150 gallons per day through any one SG. Satisfying the primary to secondary LEAKAGE limit ensures that the operational LEAKAGE performance criterion in the Steam Generator Program is met. If this SR is not met, compliance with LCO 3.4.17, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 4. The operational LEAKAGE rate limit applies to LEAKAGE through any one SG. If it is not practical to assign the LEAKAGE to an individual SG, all the primary to secondary LEAKAGE should be conservatively assumed to be from one SG.

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

The Surveillance Frequency of 72 hours is a reasonable interval to trend primary to secondary LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. The primary to secondary LEAKAGE is determined using continuous process radiation monitors or radiochemical grab sampling in accordance with the EPRI guidelines (Ref. 4).

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

REFERENCES

1. FSAR Section 1.3.3.

2. FSAR, Section 14.

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

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.17 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

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

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

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

Specification 5.5.8, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.8, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.8. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

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

BASES

APPLICABLE SAFETY ANALYSES	The steam generator tube rupture (SGTR) accident is the limiting design basis event for SG tubes and avoiding an SGTR is the basis for this Specification. The analysis of a SGTR event assumes a bounding primary to secondary LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.13, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via safety valves. The analysis for design basis accidents and transients other than a SGTR assume the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere is based on primary to secondary LEAKAGE from each SG of 500 gallons per day or is assumed to increase to 500 gallons per day as a result of accident induced conditions. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits). Steam generator tube integrity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	The LCO requires that SG tube integrity be maintained. The LCO also requires that all SG tubes that satisfy the repair criteria be plugged in accordance with the Steam Generator Program. During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged, the tube may still have tube integrity. In the context of this Specification, an SG tube is defined as the entire length of the tube, including the tube wall, between the tube-to-tubesheet weld at the tube inlet and the tube-to-tubesheet weld at the tube outlet. The tube-to-tubesheet weld is not considered part of the tube. A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.8, "Steam Generator Program," and describe acceptable SG tube performance.

BASES

LCO (continued) 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." The structural integrity performance criterion provides guidance on assessing loads that have a significant effect on burst or collapse. In that context, the term "significant" is defined as "An accident loading condition other than differential pressure is considered significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

Structural integrity requires that the primary membrane stress intensity in a tube not exceed the yield strength for all ASME Code, Section III, Service Level A (normal operating conditions) and Service Level B (upset or abnormal conditions) transients included in the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB (Ref. 4) and Draft Regulatory Guide 1.121 (Ref. 5).

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident analysis assumes that accident induced leakage does not exceed 500 gallons per day per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to

BASES

LCO (continued)

primary to secondary LEAKAGE induced during the accident. The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

APPLICABILITY

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

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

ACTIONS

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

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged in accordance with the Steam Generator Program as required by SR 3.4.17.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation

BASES

ACTIONS (continued) prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Condition B applies.

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

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection provided the inspection interval continues to be supported by an operational assessment that reflects the affected tubes. However, the affected tube(s) must be plugged prior to entering MODE 4 following the next refueling outage or SG inspection. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

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

The allowed Completion Times are reasonable, based on operating experience, to reach the desired plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.17.1

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

During SG inspections a condition monitoring assessment of the SG tubes is performed. The condition monitoring assessment determines the "as found" condition of the SG tubes. The purpose of the condition monitoring assessment is to ensure that the SG performance criteria have been met for the previous operating period.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Steam Generator Program determines the scope of the inspection and the methods used to determine whether the tubes contain flaws satisfying the tube repair criteria. Inspection scope (i.e., which tubes or areas of tubing within the SG are to be inspected) is a function of existing and potential degradation locations. The Steam Generator Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.17.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.8 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.17.2

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

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

BASES (continued)

REFERENCES

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

2. 10 CFR 50 Appendix A, GDC 19.

3. 10 CFR 100.

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

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

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

ENCLOSURE 2

Proposed Technical Specification Pages (revised)

Point Beach Nuclear Plant Units 1 and 2

Technical Specification Pages

1.1-3	5.5-8
3.4.13-1	5.5-9
3.4.13-2	5.5-10
3.4.17-1	5.5-11
3.4.17-2	5.6-6
5.5-7	

11 pages follow

1.1 Definitions

L_a

The maximum allowable primary containment leakage rate, L_a, shall be 0.4% of primary containment air weight per day at the peak design containment pressure (P_a).

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

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

c. Pressure Boundary LEAKAGE

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

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing all master relays in the channel required for OPERABILITY and verifying the OPERABILITY of each required master relay. The MASTER RELAY TEST shall include a continuity check of each associated required slave relay. The MASTER RELAY TEST may be performed by means of any series of sequential, overlapping, or total channel steps.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	---------------------------NOTES------------------------- 1. Not required to be performed until 12 hours after establishment of steady state operation. 2. Not applicable to primary to secondary LEAKAGE. --- Verify RCS Operational LEAKAGE is within limits by performance of RCS water inventory balance.	72 hours
SR 3.4.13.2	---------------------------NOTE--------------------------- Not required to be performed until 12 hours after establishment of steady state operation. --- Verify primary to secondary LEAKAGE is $\leq$ 150 gallons per day through any one SG.	72 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

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

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

ACTIONS

---NOTE---
Separate Condition entry is allowed for each SG tube.

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

SURVEILLANCE REQUIREMENTS

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

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program

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

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

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

   1. Structural integrity performance criterion: All in-service steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cool down and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.

   2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate

5.5.8 Steam Generator (SG) Program (continued)

for all SGs and leakage rate for an individual SG. Leakage is not to exceed 500 gallons per day per SG.

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

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

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

1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.

2. i. Unit 1 (alloy 600 Thermally Treated tubes): Inspect 100% of the tubes at sequential periods of 120, 90, 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 48 effective full power months or two refueling outages (whichever is less) without being inspected.

   ii. Unit 2 (alloy 690 Thermally Treated tubes): 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

5.5 Programs and Manuals

5.5.8 Steam Generator (SG) Program (continued)

considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

5.5 Programs and Manuals

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5.5 Programs and Manuals

This page retained for page numbering

5.6 Reporting Requirements

5.6.7 Tendon Surveillance Report (continued)

Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

5.6.8 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.8, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

c. Nondestructive examination techniques utilized for each degradation mechanism,

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

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

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

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

h. The effective plugging percentage for all plugging in each SG.