FENOC

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August 3, 2006
L-06-119

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: **Beaver Valley Power Station, Unit Nos. 1 and 2**
BV-1 Docket No. 50-334, License No. DPR-66
BV-2 Docket No. 50-412, License No. NPF-73
Supplement to License Amendment Request Nos. 324 and 196
Steam Generator Tube Integrity (TAC Nos. MC8861 and MC8862)

By letter dated June 1, 2006 (L-06-088), the FirstEnergy Nuclear Operating Company (FENOC) submitted a supplement to License Amendment Request (LAR) Nos. 324 and 196 that would revise steam generator tube integrity technical specifications for Beaver Valley Power Station (BVPS) Unit Nos. 1 and 2. Subsequently, by E-mail message dated June 19, 2006, the NRC provided several draft questions involving clarity and precision of proposed technical specification wording in the June 1, 2006 FENOC submittal. These were subsequently discussed in a teleconference on July 13, 2006. A follow-up E-mail message was received on July 18, 2006. Attachment A provides responses to the NRC staff's draft questions in the June 19, 2006 E-mail message.

Attachments B-1 and B-2 are proposed BVPS-1 and BVPS-2 Technical Specification (TS) changes. Attachments C-1 and C-2 are proposed BVPS-1 and BVPS-2 TS Bases changes. Changes in these attachments include refinements that incorporate responses to NRC draft questions in the June 19, 2006 E-mail message, resolution of a comment received in the July 18, 2006 E-mail message, and minor editorial enhancements and corrections that do not affect the intended meaning. The proposed TS Bases changes are provided for information only. These attachments supersede the corresponding attachments contained in the initial LAR submittal and the June 1, 2006 supplement.

FENOC has determined that the revisions proposed by this supplement do not affect the original evaluation of proposed changes or No Significant Hazards Consideration Determination provided in the November 7, 2005 submittal.

No new regulatory commitments are contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Gregory A. Dunn, Manager, FENOC Fleet Licensing, at (330) 315-7243.

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I declare under penalty of perjury that the foregoing is true and correct. Executed on August 3, 2005.

Sincerely,

Richard L Mende

for James H. Lash

Attachments:

A	Responses to Draft Questions Regarding June 1, 2006 LAR Supplement
B-1	Proposed BVPS-1 Technical Specification Changes
B-2	Proposed BVPS-2 Technical Specification Changes
C-1	Proposed BVPS-1 Technical Specification Bases Changes
C-2	Proposed BVPS-2 Technical Specification Bases Changes

c: Mr. T. G. Colburn, NRR Senior Project Manager
Mr. P. C. Cataldo, NRC Senior Resident Inspector
Mr. S. J. Collins, NRC Region I Administrator
Mr. D. A. Allard, Director BRP/DEP
Mr. L. E. Ryan (BRP/DEP)

Attachment A to L-06-119

FENOC Response to Draft Questions Regarding June 1, 2006 Supplement to License Amendment Request Nos. 324 and 196

Beaver Valley Power Station, Unit Nos. 1 and 2 (BVPS-1 and 2) Steam Generator (SG) Tube Integrity Technical Specification (TS)

1. **Please discuss your plans to correct the typographical error in LCO 3.4.6.2 Action a for Unit 2. This LCO should read as follows:**

   "With any Reactor Coolant System operational Leakage not within limits for reasons other than..."

   Response

   Typographical error "then" has been corrected to "than" in the proposed TS.

2. **Given that Technical Specification Section (TS) 6.19.c.4 does not provide safety factor requirements, please discuss your plans to modify 6.19.b.1 to further clarify that all flaws except the flaws addressed in 6.19.c.4 will include a safety factor of 3.0 against burst under normal power operation. For example, "This includes retaining a safety factor of 3.0 against burst under normal steady state full power operation primary-to-secondary pressure differential and, except for flaws addressed through application of the alternate repair criteria discussed in Specification 6.19.c.4,..."**

   Response

   Based on a July 13, 2006 teleconference with NRC staff, it is understood that proposed wording related to the 1.4 safety factor could be misconstrued to mean that proposed BVPS-2 TS 6.19.c.4 contains an alternative safety factor requirement. Therefore, proposed TS 6.19.b.1 has been revised to clarify that the reference to TS 6.19.c.4 pertains to flaws rather than to the safety factor by stating, "...except for flaws addressed through application of..." instead of "...except as permitted through application of..."

   In addition, further clarification needs to be made regarding which indications are limited to a probability of burst under postulated main steam line break conditions less than $1X10^{-2}$. As currently written, once the alternate repair criteria in 6.19.c.4 is implemented, the probability of burst for all indications (even those that are not axially oriented outside diameter stress corrosion cracking at tube support plate locations) is limited to $1X10^{-2}$. For example, "When alternate repair criteria discussed in Specification 6.19.c.4 are applied to axially oriented outside diameter stress corrosion cracking indications at tube support plate locations, the probability that one or more of these indications in a SG will burst under postulated main steam line break conditions shall be less than $1X10^{-2}$."

Response

Proposed TS 6.19.b.1 has been more precisely worded as suggested above.

3. **TS Section 6.19.c.2 and TS Section 6.19.c.3 appear to contradict each other. TS Section 6.19.c.2 provides a specific plugging limit for a sleeve, whereas TS Section 6.19.c.3 indicates that a tube will be plugged regardless of flaw depth if there is a flaw in the sleeve at the sleeve-to-tube joint. Please discuss your plans to modify TS 6.19.c.3 to remove reference to flaws in the sleeve portion of the sleeve-to-tube joint. Alternatively, discuss your plans to modify your TS to reflect current industry practice. Namely remove 6.19.c.2 and modify 6.19.c.3 to indicate that tubes with a flaw in a sleeve or in the original tube wall of the sleeve-to-tube joint shall be plugged.**

Response

Proposed BVPS-2 TS 6.19.c.2 and 6.19.c.3 do not contradict each other; however, they do provide two differing criteria that each result in plugging of a tube based on flaws found in a sleeve at the sleeve-to-tube joint. Since only the most restrictive criterion needs to be specified, the scope of proposed TS 6.19.c.2 has been revised to limit application of the percent through-wall repair criterion to the non-joint portion of a sleeve. To avoid possible confusion due to wordiness, proposed TS 6.19.c.3 has been revised to more concisely describe applicability to the joint only.

4. **In the Applicable Safety Analysis section associated with the Bases for 3/4.4.5, "Steam Generator (SG) Tube Integrity," it was indicated that the analysis for most design basis accident and transients, other than a SG tube rupture, assume that the SG tubes retain their structural integrity. This section further goes on to indicate that an exception to this assumption that tubes retain structural integrity is applied to the Unit 2 steam line break analysis. The basis for such a statement is not clear.**

Implementation of the voltage-based repair criteria limits the likelihood that a tube will burst under steam line break conditions by imposing a limit of $1X10^{-2}$ on the probability of burst. As a result of this limit, the accident induced leakage methodology assumes that tubes retain their structural integrity during design basis accidents. Please discuss your plans to modify your Bases to make it consistent with the staff's original approval of the voltage-based repair criteria discussed in Generic Letter 95-05.

Response

Terminology used in the proposed BVPS-2 TS Bases has been revised to avoid inconsistency with the concept of structural integrity described in the staff's approval for the use of voltage based repair criteria.

In addition, the staff notes that you indicated that you would be deleting the technical basis for the voltage-based repair criteria from your TS Bases. The reason for this is not clear. Please discuss your plans for including the technical basis for the voltage-based tube repair criteria in your TS Bases.

Response

Proposed BVPS-2 TS Bases statements regarding voltage-based repair criteria would be removed because TS 3/4.4.5 would no longer contain requirements associated with the criteria. Instead, administrative TS 6.19 would contain requirements for a steam generator program, including program requirements for the application of voltage-based repair criteria. This approach is consistent with the philosophy of the TSTF-449 model. Administrative technical specifications do not have a corresponding TS Bases section.

In your Bases, you indicated that "accident induced leakage" adds 2.1 gallons per minute (gpm) to the total leakage assumed in the Unit 2 steam line break analysis. You further indicate that you assume there is 150 gallons per day (gpd) (approximately 0.1 gpm) operational leakage (from each of the three SGs) which when added to the 2.1 gpm "accident induced leakage" results in a total assumed leakage of 2.4 gpm. Since the Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, indicates the accident induced leakage rate includes any primary-to-secondary leakage existing prior to the accident in addition to primary-to-secondary leakage induced during the accident, it appears that your statements are inconsistent with TSTF-449. Please discuss your plans to modify the terminology in your basis to be consistent with the definition of accident induced leakage in TSTF-449. In addition, discuss your plans to re-incorporate this definition into your Bases (i.e., it was deleted in your most recent submittal). For example, based on the staff's understanding of your accident analysis (as written in your current proposed TS Bases), a statement such as the following would be considered consistent with TSTF-449: "In support of voltage based repair criteria, analyses were performed pursuant to Generic Letter 95-05 to determine the maximum main steam line break primary-to-secondary leak rate that could occur without offsite doses exceeding the limits of Title 10 of the *Code of Federal Regulations* Part 50.67 as supplemented by Regulatory Guide 1.183 and without control room doses exceeding General Design Criteria-19. This analyses requires the leakage from the faulted SG to be limited to 2.2 gpm and the leakage from the non-faulted SGs to be limited to 150 gpd (approximately 0.2 gpm)." In other words, the accident induced leakage for the faulted SG is limited to 2.2 gpm and the accident induced leakage from each of the two non-faulted SGs is limited to 0.1 gpm (or more precisely to 150 gpd per SG).

Please note that there are several places in the Bases that discuss accident induced leakage. Appropriate modifications should be made to all applicable areas (e.g., Page B 3/4 4-4f; Applicable Safety Analysis section associated with the Bases for 3/4.4.5).

Response

The example wording provided above indicates that the NRC staff has correctly interpreted the leakage assumptions described in the proposed BVPS-2 TS Bases. However, use of the phrase "accident induced" throughout the proposed TS Bases is not always consistent with the concept of "accident induced leakage" as applied in the TSTF-449 model. Therefore, the proposed TS Bases have been revised throughout to avoid

conflicts with this concept. Deleted wording that previously described the meaning of accident induced leakage rate has been reinstated.

In your Bases under Surveillance Requirement (SR) 4.4.6.2.b, Note 2 is discussed twice. Please discuss your plans to remove this redundancy.

Response

The second statement regarding Note 2 has been determined to be unnecessary. This statement has been removed from both the BVPS-1 and BVPS-2 proposed TS Bases.

Attachment B-1

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Changes

License Amendment Request No. 324

The following is a list of the affected pages:

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* New page
**Provided for readability only

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

ADMINISTRATIVE CONTROLS

CHANNEL CHECK

1.10 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

LEAKAGE

1.14 LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump 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 LEAKAGE through a steam generator to the secondary system (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

Unidentified LEAKAGE shall be all LEAKAGE (except reactor coolant pump seal water injection or leakoff) that is not Identified LEAKAGE.

c. Pressure Boundary LEAKAGE

Pressure Boundary LEAKAGE shall be LEAKAGE (except ~~steam generator tube~~ primary to secondary LEAKAGE) through a nonisolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

1.15 THROUGH 1.17 (DELETED)

QUADRANT POWER TILT RATIO (QPTR)

1.18 QPTR shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The DOSE EQUIVALENT I-131 is calculated with the following equation:

$$C_{I-131_{D.E.}} = C_{I-131} + \frac{C_{I-132}}{170} + \frac{C_{I-133}}{6} + \frac{C_{I-134}}{1000} + \frac{C_{I-135}}{34}$$

Where "C" is the concentration, in microcuries/gram of the iodine isotopes. This equation is based on dose conversion factors derived from ICRP-30.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals;

~~3/4.4.5 STEAM GENERATORS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.4.5 Each steam generator shall be OPERABLE.~~

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

~~ACTION:~~

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

~~SURVEILLANCE REQUIREMENTS~~

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

~~4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. Steam generator tubes shall be examined in accordance with Article 8 of Section V ("Eddy current Examination of Tubular Products") and Appendix IV to Section XI ("Eddy Current Examination of Nonferromagnetic Steam Generator Heat Exchanger Tubing") of the applicable year and addenda of the ASME Boiler and Pressure Vessel Code required by 10CFR50, Section 50.55a(g). The tubes selected for each inservice inspection shall include at least 3 percent of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:~~

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

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

~~1. All nonplugged tubes that previously had detectable wall penetrations greater than 20 percent, and~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

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

3. ~~A tube inspection pursuant to Specification 4.4.5.4.a.8 shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.~~

c. ~~The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:~~

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

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

~~SURVEILLANCE REQUIREMENTS (Continued)~~

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

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

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

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

~~a. The first inservice inspection of the Model 54F steam generators shall be performed after 6 Effective Full Power Months but within 24 calendar months of initial criticality following steam generator replacement. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.~~

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

~~SURVEILLANCE REQUIREMENTS (Continued)~~

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

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

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

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

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

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

~~4.4.5.4 Acceptance Criteria~~

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

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

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

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

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

~~5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy current inspection probe shall be deemed a defective tube.~~

~~6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging because it may become unserviceable prior to the next inspection. The plugging limit is equal to the 40 percent of the nominal tube wall thickness.~~

~~7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss of coolant accident, or a steamline or feedwater line break as specified in 4.4.5.3.c, above.~~

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

~~b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit) required by Table 4.4-2.~~

~~4.4.5.5 Reports~~

~~a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be submitted in a Special Report in accordance with 10 CFR 50.4.~~

~~b. The complete results of the steam generator tube inservice inspection shall be submitted in a Special Report in accordance with 10 CFR 50.4 within 12 months following the completion of the inspection. This Special Report shall include:~~

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

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

~~3. Identification of tubes plugged.~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.~~

~~TABLE 4.4-1~~

~~MINIMUM NUMBER OF STEAM GENERATORS TO BE~~

~~INSPECTED DURING INSERVICE INSPECTION~~

~~Preservice Inspection~~	~~No~~	~~Yes~~
~~No. of Steam Generators per Unit~~	~~Three~~	~~Three~~
~~First Inservice Inspection~~	~~All~~	~~Two~~
~~Second & Subsequent Inservice Inspections~~	~~One (1)~~	~~One (2)~~

~~Table Notation:~~

~~(1) The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 9 percent of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.~~

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

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 4S tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to Specification 6.6	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s are C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to Specification 6.6.	N/A	N/A

$s = \frac{9}{n}\%$ Where n is the number of steam generators inspected during an inspection.

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 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.

ACTION:

- - - - - - - - - - - - - - - - - - GENERAL NOTE - - - - - - - - - - - - - - -

Separate action statement entry is allowed for each SG tube.

- -

a. With one or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program:

   1. Verify within 7 days that tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.

   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. With Action a not being completed within the specified completion time or if SG tube integrity is not being maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.5.2 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.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 150 gallons per day primary-~~-~~to-~~-~~secondary LEAKAGE through any one steam generator, and

d. 10 gpm identified LEAKAGE.

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

ACTION:

~~b~~a. With any Reactor Coolant System operational LEAKAGE ~~greater than any one of the above~~ not within limits~~, excluding~~ for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE, reduce the LEAKAGE ~~rate~~ to within limits within 4 hours ~~or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours~~.

~~a~~b. With the required action and associated completion time of Action a not met, or with ~~any~~ pressure boundary LEAKAGE, or with primary to secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the ~~next~~ following 30 hours.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the following leakage detection instrumentation at least once per 12 hours:(1)

1. Containment atmosphere gaseous radioactivity monitor.

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(1) Only on leakage detection instrumentation required by LCO 3.4.6.1.

SURVEILLANCE REQUIREMENTS (Continued)

2. Containment atmosphere particulate radioactivity monitor.

3. Containment sump discharge flow monitor.

4. Containment sump narrow range level monitor.

b. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours ~~during steady state operation~~.(2)(3)

c. Verifying primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one steam generator at least once per 72 hours.(2)

(2) Not required to be performed ~~in MODE 3 or 4~~ until 12 hours after establishment of steady state operation.

(3) Not applicable to primary to secondary LEAKAGE.

PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (Continued)

The methodology listed in WCAP-14040-NP-A was used with two exceptions:

a) Use of ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P-T Limits for Section XI, Division 1", and

b) Use of methodology of the 1996 version of ASME Section XI, Appendix G, "Fracture Toughness Criteria for Protection Against Failure".

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

6.9.7 STEAM GENERATOR TUBE INSPECTION REPORT

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 6.19, 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.

6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

(next page is 6-23)

Containment Leakage Rate Testing Program (Continued)

b. Air Lock testing acceptance criteria and required action are as stated in Specification 3.6.1.3 titled "Containment Air Locks."

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

6.18 Technical Specifications (TS) Bases Control Program

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

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

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

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

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

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

d. Proposed changes that meet the criteria of Specification 6.18.b.1 & 2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.19 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

Steam Generator Program (Continued)

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. Provisions for 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 all SGs and leakage rate for an individual SG. Leakage is also not to exceed 1 gpm per SG, except during a SG tube rupture.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.6.2.

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.

Steam Generator Program (Continued)

d. Provisions for SG Tube Inspections

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

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

2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. During each period 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 intervals between 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 interval between refueling outages (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

Attachment B-2

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Changes

License Amendment Request No. 196

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LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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ADMINISTRATIVE CONTROLS

INDEX

ADMINISTRATIVE CONTROLS

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is or would be subcritical from its present condition assuming all full length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

LEAKAGE

1.14 LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump 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 LEAKAGE through a steam generator to the secondary system (primary to secondary LEAKAGE).

b. Unidentified LEAKAGE

Unidentified LEAKAGE shall be all LEAKAGE (except reactor coolant pump seal water injection or leakoff) that is not Identified LEAKAGE.

c. Pressure Boundary LEAKAGE

Pressure Boundary LEAKAGE shall be LEAKAGE (except ~~steam generator tube~~ primary to secondary LEAKAGE) through a nonisolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

~~3/4.4.5 STEAM GENERATORS~~

~~LIMITING CONDITION FOR OPERATION~~

~~3.4.5 Each steam generator shall be OPERABLE.~~

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

~~ACTION:~~

~~With one or more steam generators inoperable, restore the inoperable generator(s) to OPERABLE status prior to increasing T_{avg} above 200ºF.~~

~~SURVEILLANCE REQUIREMENTS~~

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

~~4.4.5.2 Steam Generator Tube Sample Selection and Inspection The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. Steam generator tubes shall be examined in accordance with Article 8 of Section V ("Eddy Current Examination of Tubular Products") and Appendix IV to Section XI ("Eddy Current Examination of Nonferromagnetic Steam Generator Heat Exchanger Tubing") of the applicable year and addenda of the ASME Boiler and Pressure Vessel Code required by 10CFR50, Section 50.55a(g). When applying the exceptions of 4.4.5.2.a through 4.4.5.2.c, previous defects or imperfections in the area repaired by sleeving are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include at least 3 percent of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:~~

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

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

~~1. All nonplugged tubes that previously had detectable wall penetrations greater than 20 percent, and~~

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

~~3. At least 3 percent of the total number of sleeved tubes in all three steam generators. A sample size less than 3 percent is acceptable provided all the sleeved tubes in the steam generator(s) examined during the refueling outage are inspected. These inspections will include both the tube and the sleeve, and~~

~~4. A tube inspection pursuant to Specification 4.4.5.4.a.8. If any selected tube does not permit the passage of the eddy current probe for a tube or sleeve inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.~~

~~5. Indications left in service as a result of application of the tube support plate voltage-based repair criteria (4.4.5.4.a.10) shall be inspected by bobbin coil probe during all future refueling outages.~~

~~c. The tubes selected as the second and third samples (if required by Table 4.4-2) during each inservice inspection may be subjected to a partial tube inspection provided:~~

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

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

~~d. Implementation of the steam generator tube-to-tube support plate repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

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

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

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

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

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

~~b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.~~

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

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

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

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

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

~~4.4.5.4 Acceptance Criteria~~

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

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

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

~~3. Degraded Tube means a tube or sleeve containing imperfections greater than or equal to 20 percent of the nominal wall thickness caused by degradation.~~

Proposed changes to draft page from Unit 2 LAR 173 (EPU)

~~SURVEILLANCE REQUIREMENTS (Continued)~~

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

~~5. Defect means an imperfection of such severity that it exceeds the plugging or repair limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy current inspection probe shall be deemed a defective tube.~~

~~6. Plugging or Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving in the affected area because it may become unserviceable prior to the next inspection. The plugging or repair limit imperfection depths are specified in percentage of nominal wall thickness as follows:~~

~~a) Original tube wall — 40%~~

~~This definition does not apply to tube support plate intersections for which the voltage based repair criteria are being applied. Refer to 4.4.5.4.a.10 for the repair limit applicable to these intersections.~~

~~b) ABB Combustion Engineering TIG welded sleeve wall — 27%~~

~~c) Westinghouse laser welded sleeve wall — 25%~~

~~7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss of coolant accident, or a steamline or feedwater line break as specified in 4.4.5.3.c, above.~~

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

Proposed changes to draft page from Unit 2 LAR 173 (EPU)

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~9. Tube Repair refers to sleeving which is used to maintain a tube in-service or return a tube to service. This includes the removal of plugs that were installed as a corrective or preventive measure. The following sleeve designs have been found acceptable:~~

~~a) ABB Combustion Engineering TIG welded sleeves, CEN-629-P, Revision 02 and CEN-629-P Addendum 1.~~

~~b) Westinghouse laser welded sleeves, WCAP-13483, Revision 2.~~

~~10. Tube Support Plate Plugging Limit is used for the disposition of an alloy 600 steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging (repair) limit is based on maintaining steam generator tube serviceability as described below:~~

~~a) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.~~

~~b) Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be repaired or plugged, except as noted in 4.4.5.4.a.10.c below.~~

~~c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit[(1)] may remain in service if a rotating pancake coil or acceptable alternative inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit[(1)] will be plugged or repaired.~~

~~d) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.~~

~~The mid-cycle repair limits are determined from the following equations:~~

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left(\frac{CL - \Delta t}{CL}\right)$$

~~(1) The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~where:~~

~~V_{URL}~~	~~=~~	~~upper voltage repair limit~~
~~V_{LRL}~~	~~=~~	~~lower voltage repair limit~~
~~V_{MURL}~~	~~=~~	~~mid cycle upper voltage repair limit based on time into cycle~~
~~V_{MLRL}~~	~~=~~	~~mid cycle lower voltage repair limit based on V_{MURL} and time into cycle~~
~~Δt~~	~~=~~	~~length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented~~
~~CL~~	~~=~~	~~cycle length (the time between two scheduled steam generator inspections)~~
~~V_{SL}~~	~~=~~	~~structural limit voltage~~
~~Gr~~	~~=~~	~~average growth rate per cycle length~~
~~NDE~~	~~=~~	~~95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20-percent has been approved by NRC)[(2)]~~

~~Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.4.5.4.a.10.a, 4.4.5.4.a.10.b, and 4.4.5.4.a.10.c.~~

~~(2) The NDE is the value provided by the NRC in GL 95 05 as supplemented.~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair all tubes exceeding the plugging or repair limit) required by Table 4.4-2.~~

~~4.4.5.5 Reports~~

~~a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged or repaired in each steam generator shall be submitted in a Special Report in accordance with 10 CFR 50.4.~~

~~b. The complete results of the steam generator tube and sleeve inservice inspection shall be submitted in a Special Report in accordance with 10 CFR 50.4 within 12 months following the completion of the inspection. This Special Report shall include:~~

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

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

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

~~c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. The written report shall provide a description of investigations conducted to determine the cause of the tube degradation and corrective measures taken to prevent recurrence.~~

~~d. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the Commission prior to returning the steam generators to service (MODE 4) should any of the following conditions arise:~~

~~1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.~~

~~SURVEILLANCE REQUIREMENTS (Continued)~~

~~2. If circumferential crack-like indications are detected at the tube support plate intersections.~~

~~3. If indications are identified that extend beyond the confines of the tube support plate.~~

~~4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.~~

~~5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1 X 10^-2, notify the Commission and provide an assessment of the safety significance of the occurrence.~~

~~TABLE 4.4-1~~

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

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

~~Table Notation~~

1. ~~The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 9% of the tubes if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.~~

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

~~TABLE 4.4-2~~
~~STEAM GENERATOR TUBE INSPECTION~~

~~1ST SAMPLE INSPECTION~~			~~2ND SAMPLE INSPECTION~~		~~3RD SAMPLE INSPECTION~~	
~~Sample Size~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~	~~Result~~	~~Action Required~~
~~A minimum Of S tubes per S.G.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~	~~N/A~~	~~N/A~~
	~~C-2~~	~~Plug or repair defective tubes and inspect additional 2S tubes in this S.G.~~	~~C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~C-2~~	~~Plug or repair defective tubes and inspect additional 4S tubes in this S.G.~~	~~C-1~~	~~None~~
					~~C-2~~	~~Plug or repair defective tubes~~
					~~C-3~~	~~Perform action for C-3 result of first sample~~
			~~C-3~~	~~Perform action for C-3 result of first sample~~	~~N/A~~	~~N/A~~
	~~C-3~~	~~Inspect all tubes in this S.G., plug or repair defective tubes and inspect 2S tubes in each other S.G.~~ ~~Notification to NRC pursuant to Specification 6.6.~~	~~All other S.G.s are C-1~~	~~None~~	~~N/A~~	~~N/A~~
			~~Some S.G.s C-2 but no additional S.G.s are C-3~~	~~Perform action for C-2 result of second sample~~	~~N/A~~	~~N/A~~
			~~Additional S.G. is C-3~~	~~Inspect all tubes in each S.G. and plug or repair defective tubes. Notification to NRC pursuant to Specification 6.6.~~	~~N/A~~	~~N/A~~

~~$S = \frac{9}{n}$ % Where n is the number of steam generators inspected during an inspection.~~

3/4.4.5 STEAM GENERATOR (SG) TUBE INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.5 SG tube integrity shall be maintained

AND

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

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

ACTION:

- - - - - - - - - - - - - - - GENERAL NOTE - - - - - - - - - - - - - - -

Separate action statement entry is allowed for each SG tube.

- -

a. With one or more SG tubes satisfying the tube repair criteria and not plugged or repaired in accordance with the Steam Generator Program:

   1. Verify within 7 days that tube integrity of the affected tube(s) is maintained until the next refueling outage or SG tube inspection.

   2. Plug or repair 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. With Action a not being completed within the specified completion time or if SG tube integrity is not being maintained, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.5.1 Verify SG tube integrity in accordance with the Steam Generator Program.

4.4.5.2 Verify that each inspected SG tube that satisfies the tube repair criteria is plugged or repaired in accordance with the Steam Generator Program prior to entering MODE 4 following a SG tube inspection.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE,

b. 1 gpm unidentified LEAKAGE,

c. 150 gallons per day primary-_to-_secondary LEAKAGE through any one steam generator, and

d. 10 gpm identified LEAKAGE.

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

ACTION:

~~b~~a. With any Reactor Coolant System operational LEAKAGE ~~greater than any one of the above~~ not within limits~~, excluding~~ for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE, reduce the LEAKAGE ~~rate~~ to within limits within 4 hours ~~or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours~~.

~~a~~b. With the required action and associated completion time of Action a not met, or with ~~any~~ pressure boundary LEAKAGE, or with primary to secondary leakage not within limit, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

a. Monitoring the following leakage detection instrumentation at least once per 12 hours:(1)

1. Containment atmosphere gaseous radioactivity monitor.

(1) Only on leakage detection instrumentation required by LCO 3.4.6.1.

SURVEILLANCE REQUIREMENTS (Continued)

2. Containment atmosphere particulate radioactivity monitor.

3. Containment sump discharge flow monitor.

4. Containment sump narrow range level monitor.

b. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours ~~during steady state operation~~.(2)(3)

c. Verifying primary to secondary LEAKAGE is less than or equal to 150 gallons per day through any one steam generator at least once per 72 hours.(2)

---

(2) Not required to be performed ~~in MODE 3 or 4~~ until 12 hours after establishment of steady state operation.

(3) Not applicable to primary to secondary LEAKAGE.

PRESSURE AND TEMPERATURE LIMITS REPORT (continued)

c. The PTLR shall be provided to the NRC upon issuance for each reactor fluence period and for any revision or supplement thereto.

6.9.7 STEAM GENERATOR TUBE INSPECTION REPORT

1. 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 6.19, 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 or repaired during the inspection outage for each active degradation mechanism,

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

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

   h. The effective plugging percentage for all plugging and tube repairs in each SG, and

   i. Repair method utilized and the number of tubes repaired by each repair method.

2. A report shall be submitted within 90 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 6.19, Steam Generator Program, when voltage-based alternate repair criteria have been applied. The report shall include information described in Section 6.b of Attachment 1 to Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

3. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the Commission prior to returning the steam generators to service (MODE 4) should any of the following conditions arise:

   a. If circumferential crack-like indications are detected at the tube support plate intersections.

STEAM GENERATOR TUBE INSPECTION REPORT (continued)

b. If indications are identified that extend beyond the confines of the tube support plate.

c. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

6.10 DELETED

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601 of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiological Work Permit[(1)]. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

(1) Radiation protection personnel, or personnel escorted by radiation protection personnel in accordance with approved emergency procedures, shall be exempt from the RWP issuance requirement during the performance of their radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

TECHNICAL SPECIFICATIONS (TS) BASES CONTROL PROGRAM (Continued)

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

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

d. Proposed changes that meet the criteria of Specification 6.18.b.1 & 2 above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.19 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 or repair of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected, plugged, or repaired to confirm that the performance criteria are being met.

b. Provisions for 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, except for flaws addressed through application of the alternate repair criteria discussed in Specification 6.19.c.4, a safety factor

STEAM GENERATOR PROGRAM (Continued)

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.

When alternate repair criteria discussed in Specification 6.19.c.4 are applied to axially oriented outside diameter stress corrosion cracking indications at tube support plate locations, the probability that one or more of these indications in a SG will burst under postulated main steam line break conditions shall be less than $1x10^{-2}$.

2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Except during a steam generator tube rupture, leakage from all sources excluding the leakage attributed to the degradation described in TS Section 6.19.c.4 is also not to exceed 1 gpm per SG.

3. The operational LEAKAGE performance criterion is specified in LCO 3.4.6.2.

c. Provisions for SG Tube Repair Criteria

1. Tubes found by inservice inspection to contain a flaw in a non-sleeved region with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired except if permitted to remain in service through application of the alternate repair criteria discussed in Specification 6.19.c.4.

2. Tubes with sleeves found by inservice inspection to contain flaws that are not in the sleeve to tube joint, with a depth equal to or exceeding the following percentages of the nominal sleeve wall thickness, shall be plugged:

ABB Combustion Engineering TIG welded sleeves 27%

Westinghouse laser welded sleeves 25%

3. Tubes with a flaw in a sleeve to tube joint shall be plugged.

4. The following alternate tube repair criteria may be applied as an alternative to the 40% depth based criteria of Technical Specification 6.19.c.1:

Tube Support Plate Voltage-Based Repair Criteria

Tube Support Plate Plugging Limit is used for the disposition of an Alloy 600 steam generator tube for

STEAM GENERATOR PROGRAM (Continued)

continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the plugging (repair) limit is described below:

a) Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

b) Steam generator tubes, with degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts will be repaired or plugged, except as noted in 6.19.c.4.c below.

c) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit (calculated according to the methodology in Generic Letter 95-05 as supplemented) may remain in service if a rotating pancake coil or acceptable alternative inspection does not detect degradation.

d) Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the upper voltage repair limit (calculated according to the methodology in Generic Letter 95-05 as supplemented) will be plugged or repaired.

STEAM GENERATOR PROGRAM (Continued)

e) If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits specified in 6.19.c.4.a, 6.19.c.4.b, 6.19.c.4.c and 6.19.c.4.d.

The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - V_{LRL})\left(\frac{CL - \Delta t}{CL}\right)$$

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by NRC). The NDE is the value provided by the NRC in GL 95-05 as supplemented.

Implementation of these mid-cycle repair limits should follow the same approach as in Specifications 6.19.c.4.a through 6.19.c.4.d.

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

STEAM GENERATOR PROGRAM (Continued)

outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In tubes repaired by sleeving, the portion of the original tube wall between the sleeve's joints is not an area requiring re-inspection. In addition to meeting the requirements of d.1, d.2, d.3, and d.4 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. A degradation assessment shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.

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

2. Inspect 100% of the tubes at sequential periods of 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. No SG shall operate for more than 24 effective full power months or one interval between 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 interval between refueling outages (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.

STEAM GENERATOR PROGRAM (Continued)

4. Indications left in service as a result of application of the tube support plate voltage-based repair criteria (6.19.c.4) shall be inspected by bobbin coil probe during all future refueling outages.

   Implementation of the steam generator tube-to-tube support plate repair criteria requires a 100-percent bobbin coil inspection for hot-leg and cold-leg tube support plate intersections down to the lowest cold-leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold-leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

e. Provisions for monitoring operational primary to secondary LEAKAGE

f. Provisions for SG Tube Repair Methods

   Steam generator tube repair methods shall provide the means to reestablish the RCS pressure boundary integrity of SG tubes without removing the tube from service. For the purposes of these Specifications, tube plugging is not a repair. All acceptable tube repair methods are listed below.

   1. ABB Combustion Engineering TIG welded sleeves, CEN-629-P, Revision 02 and CEN-629-P Addendum 1.

   2. Westinghouse laser welded sleeves, WCAP-13483, Revision 2.

Attachment C-1

Beaver Valley Power Station, Unit No. 1
Proposed Technical Specification Bases Changes

License Amendment Request No. 324

The following is a list of the affected pages:

Page
B-II
B 3/4 4-2
B 3/4 4-2a
B 3/4 4-3d*
B 3/4 4-3e*
B 3/4 4-3f*
B 3/4 4-3g
B 3/4 4-3h
B 3/4 4-3i
B 3/4 4-3j

*Provided for readability only

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TECHNICAL SPECIFICATION BASES INDEX

BASES

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3/4.4.3 SAFETY VALVES (Continued)

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.4 PRESSURIZER

The requirement that (150)kw of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

~~3/4.4.5 STEAM GENERATORS~~

~~One OPERABLE steam generator in a non-isolated reactor coolant loop provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350°F, decay heat is removed by the RHR system.~~

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

~~The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant~~

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~~3/4.4.5 STEAM GENERATORS (Continued)~~

~~operation would be limited by the limitation of steam generator tube leakage between the Primary Coolant System and the Secondary Coolant System (primary-to-secondary LEAKAGE = 150 gallons per day per steam generator). Maintaining a primary-to-secondary LEAKAGE less than this limit helps to ensure adequate margin to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary LEAKAGE of 150 gallons per day per steam generator can readily be detected. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.~~

~~Wastage-type defects are unlikely with proper chemistry of secondary coolant, such as provided by All Volatile Treatment (AVT). However, even if a defect of similar type should develop in-service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect a wastage type defect that has penetrated 20 percent of the original tube wall thickness.~~

~~Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy current inspection, and revision of the Technical Specifications, if necessary.~~

3/4.4.5 Steam Generator (SG) Tube Integrity

BACKGROUND

Steam generator 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 "Reactor Coolant Loop" LCOs 3.4.1.1 (MODES 1 and 2), 3.4.1.2 (MODE 3), and 3.4.1.3 (MODES 4 and 5).

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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. Depending upon materials and design, 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 6.19, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.19, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 6.19. 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 NEI 97-06, "Steam Generator Program Guidelines."

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 SG tube LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.6.2.c, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes that following reactor trip the contaminated secondary fluid is released to the atmosphere via safety valves. Environmental releases before reactor trip are discharged through the main condenser.

The analysis for design basis accidents and transients other than a SGTR assumes the SG tubes retain their structural integrity (i.e., they are assumed not to rupture.) In these analyses, the steam discharge to the atmosphere includes primary to secondary SG tube LEAKAGE equivalent to the operational leakage limit of 150 gpd per SG. 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 3.4.8, "RCS Specific Activity," limits. Pre-accident and concurrent iodine spikes are assumed in accordance with applicable regulatory guidance. 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 10 CFR 50.67 as supplemented by Regulatory Guide 1.183.

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

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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 retain tube integrity.

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

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.19, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

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

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

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the design specification. This includes safety factors and applicable design basis loads based on ASME Code, Section III, Subsection NB and Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.

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 150 gpd per SG. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.6.2, "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 actions 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 subsequently affected SG tubes are governed by subsequent condition entry and application of associated required actions.

a. ACTION 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 4.4.5.1. 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

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determine if a SG tube that should have been plugged has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Action 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, ACTION a 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. If the required actions and associated completion times of ACTION a are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 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 4.4.5.1

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

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

The Steam Generator Program in conjunction with the degradation assessment determines the scope of the inspection and the methods

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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 and the degradation assessment also specify the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 4.4.5.1. The Frequency is determined by the operational assessment and other limits in EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines." The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.19 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 4.4.5.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is removed from service by plugging. The tube repair criteria delineated in Specification 6.19 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). NEI 97-06 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 SR 4.4.5.2 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.

3/4.4.6.1 LEAKAGE DETECTION INSTRUMENTATION (Continued)

SURVEILLANCE REQUIREMENTS (SR)

SR 4.4.6.1.a

SR 4.4.6.1.a requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 4.4.6.1.a requires the performance of a CHANNEL FUNCTIONAL TEST on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 31 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 4.4.6.1.a also requires the performance of a CHANNEL CALIBRATION on the required containment atmosphere radioactivity monitor. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

SR 4.4.6.1.b

SR 4.4.6.1.b requires the performance of a CHANNEL CALIBRATION on the required containment sump monitor. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

3/4.4.6.2 OPERATIONAL LEAKAGE

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.

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3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

BACKGROUND (Continued)

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, requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 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 percent 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).

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

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)
APPLICABLE SAFETY ANALYSES (Continued)

affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes a 450 gpd (150 gpd per steam generator) primary-to-secondary LEAKAGE.

Primary-to-secondary LEAKAGE is a factor in the dose assessment of accidents or transients that involve secondary steam release to the atmosphere, such as a main steam line break (MSLB), a locked rotor accident (LRA), a Loss of AC Power (LACP), a Control Rod Ejection Accident (CREA) and to a lesser extent, a Steam Generator Tube Rupture (SGTR). The leakage contaminates the secondary fluid. The limit on the primary-to-secondary leakage ensures that the dose contribution at the site boundary from tube leakage following such accidents are limited to appropriate fractions of the 10 CFR 50.67 limit of 25 Rem TEDE as allowable by Regulatory Guide 1.183. The limit on the primary-to-secondary leakage also ensures that the dose contribution from tube leakage in the control room is limited to the 10 CFR 50.67 limit of 5 Rem TEDE. Among all of the analyses that release primary side activity to the environment via tube leakage, the MSLB is of particular concern because the ruptured main steam line provides a pathway to release the primary-to-secondary leakage directly to the environment without dilution in the secondary fluid.

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

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3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

LCO (Continued)

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. Should pressure boundary LEAKAGE occur through a component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

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. Primary-_to-_Secondary LEAKAGE through Any One SG

~~Operating experience at PWR plants has shown that sudden increases in leak rate are often precursors to larger tube failures. Maintaining an operating LEAKAGE limit of 150 gpd per steam generator will minimize the potential for a large LEAKAGE event at power. This operating LEAKAGE limit is more restrictive than the operating LEAKAGE limit in standardized technical specifications. This provides additional margin to accommodate a tube flaw which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. This reduced LEAKAGE limit, in conjunction with a leak rate monitoring program, provides additional assurance that this precursor LEAKAGE will be detected and the plant shut down in a timely manner.~~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". 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.

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3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

d. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the 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.

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 3.4.6.3, "RCS Pressure Isolation Valve (PIV)," 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

~~b~~a. 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. ~~If the unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE.~~

~~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.~~

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

ACTIONS (Continued)

~~a~~b. If any pressure boundary LEAKAGE exists or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within ~~36~~ the following 30 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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

SURVEILLANCE REQUIREMENTS (SR)

SR 4.4.6.2.a

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the systems that monitor the containment atmosphere radioactivity and the containment sump level. The 12 hour monitoring of the leakage detection system is sufficient to provide an early warning of increased RCS LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Leakage Detection Instrumentation." Note (1) states that the 12 hour surveillance is required only on leakage detection instrumentation required by LCO 3.4.6.1. This Note allows the 12 hour monitoring to be suspended on leakage detection instrumentation which is inoperable or not required to be operable per LCO 3.4.6.1.

SR 4.4.6.2.b

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.~~

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The RCS water inventory balance must be met with the reactor at steady state operating conditions and near operating pressure. ~~Therefore, this SR is not required to be performed in MODES 3 and 4 until 12 hours of steady state operation near operating pressure have been established.~~The SR is modified by two notes. Note 2 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

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BASES

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

SURVEILLANCE REQUIREMENTS (SR) (Continued)

~~SR 4.4.6.2 (Continued)~~

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

~~An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the systems that monitor the containment atmosphere radioactivity and the containment sump level. The 12 hour monitoring of the leakage detection system is sufficient to provide an early warning of increased RCS LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Leakage Detection Instrumentation."~~

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. ~~Note (1) states that the 12 hour surveillance is required only on leakage detection instrumentation required by LCO 3.4.6.1. This Note allows the 12 hour monitoring to be suspended on leakage detection instrumentation which is inoperable or not required to be operable per LCO 3.4.6.1. Note (2) states that this SR is required to be performed during steady state operation.~~

Note 3 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.

SR 4.4.6.2.c

This SR verifies that primary to secondary LEAKAGE is less than 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 3.4.5, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature (25°C) as described in EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines." 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 EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

3/4.4.6.3 PRESSURE ISOLATION VALVE LEAKAGE

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

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

Attachment C-2

Beaver Valley Power Station, Unit No. 2
Proposed Technical Specification Bases Changes

License Amendment Request No. 196

The following is a list of the affected pages:

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TECHNICAL SPECIFICATION BASES INDEX

BASES

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3/4.4.2 (This Specification number is not used.)

3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 345,000 lbs. per hour of saturated steam at the valve set point.

During shutdown conditions (MODE 4 with any RCS cold leg temperature below the enable temperature specified in 3.4.9.3) RCS overpressure protection is provided by the Overpressure Protection Systems addressed in Specification 3.4.9.3.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

Safety valves similar to the pressurizer code safety valves were tested under an Electric Power Research Institute (EPRI) program to determine if the valves would operate stably under feedwater line break accident conditions. The test results indicated the need for inspection and maintenance of the safety valves to determine the potential damage that may have occurred after a safety valve has lifted and either discharged the loop seal or discharged water through the valve. Additional action statements require safety valve inspection to determine the extent of the corrective actions required to ensure the valves will be capable of performing their intended function in the future.

3/4.4.4 PRESSURIZER

The requirement that 150 kw of pressurizer heaters and their associated controls and emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain natural circulation at HOT STANDBY.

~~3/4.4.5 STEAM GENERATORS~~

~~One OPERABLE steam generator in a non isolated reactor coolant loop provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two OPERABLE steam generators, combined with other requirements of the Limiting Conditions for Operation ensures adequate~~

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Proposed changes to draft page from Unit 2 LAR 173 (EPU)

~~3/4.4.5 STEAM GENERATORS (Continued)~~

~~decay heat removal capabilities for RCS temperatures greater than 350°F if one steam generator becomes inoperable due to single failure considerations. Below 350°F, decay heat is removed by the RHR system.~~

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

~~The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Primary Coolant System and the Secondary Coolant System (primary to secondary LEAKAGE = 150 gallons per day per steam generator). Axial cracks having a primary to secondary LEAKAGE less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary LEAKAGE of 150 gallons per day per steam generator can readily be detected. LEAKAGE in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged or repaired by sleeving. The technical bases for sleeving are described in the approved vendor reports listed in Surveillance Requirement 4.4.5.4.a.9, as supplemented by Westinghouse letter FENOC-02-304.~~

~~Wastage-type defects are unlikely with the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging or repair will be required of all tubes with imperfections exceeding the plugging or repair limit. Degraded steam generator tubes may be repaired by the installation of sleeves which span the degraded tube section. A steam generator tube with a sleeve installed meets the structural~~

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~~3/4.4.5 STEAM GENERATORS (Continued)~~

~~requirements of tubes which are not degraded, therefore, the sleeve is considered a part of the tube. The surveillance requirements identify those sleeving methodologies approved for use. If an installed sleeve is found to have through wall penetration greater than or equal to the plugging limit, the tube must be plugged. The plugging limit for the sleeve is derived from R. G. 1.121 analysis which utilizes a 20 percent allowance for eddy current uncertainty in determining the depth of tube wall penetration and additional degradation growth. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20 percent of the original tube wall thickness.~~

~~The voltage based repair limits of these surveillance requirements (SR) implement the guidance in GL 95-05 and are applicable only to Westinghouse designed steam generators (SGs) with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The guidance in GL 95-05 will not be applied to the tube-to-flow distribution baffle plate intersections. The voltage based repair limits are not applicable to other forms of SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no NDE detectable cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.~~

~~Implementation of these SRs requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).~~

~~The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95-percent prediction interval curve reduced to account for the lower 95/95-percent tolerance bound for tubing material properties at 650°F (i.e., the 95-percent LTL curve). The voltage structural limit must be adjusted downward to account for potential degradation growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit;~~ $\cancel{V_{URL}}$, ~~is determined from the structural voltage limit by applying the following equation:~~

$$\cancel{V_{URL} = V_{SL} - V_{Gr} - V_{NDE}}$$

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Proposed changes to draft page from Unit 2 LAR 173 (EPU)

~~3/4.4.5 STEAM GENERATORS (Continued)~~

~~where V_{GR} represents the allowance for degradation growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.~~

~~Safety analyses were performed pursuant to Generic Letter 95-05 to determine the maximum MSLB induced primary-to-secondary leak rate that could occur without offsite doses exceeding a small fraction of 10 CFR 50.67 guidelines (considering a concurrent iodine spike), 10 CFR 50.67 (pre accident iodine spike), and without control room doses exceeding 10 CFR 50.67. The current value of the maximum MSLB-induced leak rate and a summary of the analyses are provided in Section 15.1.5 of the UFSAR.~~

~~The mid-cycle equation in SR 4.4.5.4.a.10.d should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.~~

~~SR 4.4.5.5 implements several reporting requirements recommended by GL 95-05 for situations which the NRC wants to be notified prior to returning the SGs to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as found voltage distribution rather than the projected end-of-cycle (EOC) voltage distribution (refer to GL 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL section 6.b (c) criteria.~~

~~Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission pursuant to Specification 6.6 prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy current inspection, and revision of the Technical Specifications, if necessary.~~

3/4.4.5 Steam Generator (SG) Tube Integrity

BACKGROUND

Steam generator 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 "Reactor Coolant Loop" LCOs 3.4.1.1 (MODES 1 and 2), 3.4.1.2 (MODE 3), and 3.4.1.3 (MODES 4 and 5).

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. Depending upon materials and design, 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 6.19, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 6.19, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 6.19. 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 NEI 97-06, "Steam Generator Program Guidelines."

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 SG tube LEAKAGE rate equal to the operational LEAKAGE rate limits in LCO 3.4.6.2.c, "RCS Operational LEAKAGE," plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes that following reactor trip the contaminated secondary fluid is released to the atmosphere via safety valves. Environmental releases before reactor trip are discharged through the main condenser.

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 3.4.8, "RCS Specific Activity," limits. Pre-accident and concurrent iodine spikes are assumed in accordance with applicable regulatory guidance. 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 10 CFR 50.67 as supplemented by Regulatory Guide 1.183 and within GDC-19 values.

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) and the steam discharge to the atmosphere is assumed to include primary to secondary SG tube LEAKAGE equivalent to the operational leakage limit of 150 gpd per SG. However, an increased leakage assumption is applied in the Unit 2 MSLB analysis. In support of voltage based repair criteria pursuant to Generic Letter 95-05, analyses were performed to determine the maximum permissible main steam line break (MSLB) primary to secondary leak rate that could occur without offsite doses exceeding the limits of 10 CFR 50.67 as supplemented by Regulatory Guide 1.183 and without control room doses exceeding GDC-19. An additional 2.1 gpm leakage is assumed in the Unit 2 MSLB analysis resulting from accident conditions. Therefore, in the MSLB analysis, the steam discharge to the atmosphere includes primary to secondary SG tube LEAKAGE equivalent to the operational leakage limit of 150 gpd per SG and an additional 2.1 gpm which results in a total assumed accident induced leakage of 2.4 gpm.

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 or repaired in accordance with the Steam Generator Program.

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. If a tube was determined to satisfy the repair criteria but was not plugged or repaired, the tube may still retain tube integrity.

In the context of this Specification, a SG tube is defined as 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.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 6.19, "Steam Generator Program," and describe acceptable SG tube performance. The Steam Generator Program also provides the evaluation process for determining conformance with the SG performance criteria.

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

The structural integrity performance criterion provides a margin of safety against tube burst or collapse under normal and accident conditions, and ensures structural integrity of the SG tubes under all anticipated transients included in the design specification.

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

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 as described in the Applicable Safety Analyses section. The accident induced leakage rate includes any primary to secondary LEAKAGE existing prior to the accident in addition to primary to secondary LEAKAGE induced during the accident.

The operational LEAKAGE performance criterion provides an observable indication of SG tube conditions during plant operation. The limit on operational LEAKAGE is contained in LCO 3.4.6.2, "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.

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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 actions 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 subsequently affected SG tubes are governed by subsequent condition entry and application of associated required actions.

a. ACTION 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 or repaired in accordance with the Steam Generator Program as required by SR 4.4.5.1. 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 or repaired has tube integrity, an evaluation must be completed that demonstrates that the SG performance criteria will continue to be met until the next refueling outage or SG tube inspection. The tube integrity determination is based on the estimated condition of the tube at the time the situation is discovered and the estimated growth of the degradation prior to the next SG tube inspection. If it is determined that tube integrity is not being maintained, Action 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, ACTION a 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 or repaired 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. If the required actions and associated completion times of ACTION a are not met or if SG tube integrity is not being maintained, the reactor must be brought to HOT STANDBY within 6 hours and COLD SHUTDOWN within the following 30 hours.

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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 4.4.5.1

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

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

The Steam Generator Program in conjunction with the degradation assessment 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 and the degradation assessment also specify the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, nondestructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 4.4.5.1. The Frequency is determined by the operational assessment and other limits in EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines." The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 6.19 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 4.4.5.2

During an SG inspection, any inspected tube that satisfies the Steam Generator Program repair criteria is repaired or removed from service by plugging. The tube repair criteria delineated in Specification 6.19 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).

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NEI 97-06 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

Steam generator tube repairs are only performed using approved repair methods as described in the Steam Generator Program.

The Frequency of "prior to entering MODE 4 following a SG inspection" ensures that SR 4.4.5.2 has been completed and all tubes meeting the repair criteria are plugged or repaired prior to subjecting the SG tubes to significant primary to secondary pressure differential.

3/4.4.6.1 LEAKAGE DETECTION INSTRUMENTATION (Continued)

SURVEILLANCE REQUIREMENTS (SR)

SR 4.4.6.1.a

SR 4.4.6.1.a requires the performance of a CHANNEL CHECK of the required containment atmosphere radioactivity monitor. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 4.4.6.1.a requires the performance of a CHANNEL FUNCTIONAL TEST on the required containment atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 31 days considers instrument reliability, and operating experience has shown that it is proper for detecting degradation.

SR 4.4.6.1.a also requires the performance of a CHANNEL CALIBRATION on the required containment atmosphere radioactivity monitor. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

SR 4.4.6.1.b

SR 4.4.6.1.b requires the performance of a CHANNEL CALIBRATION on the required containment sump monitor. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Again, operating experience has proven that this Frequency is acceptable.

3/4.4.6.2 OPERATIONAL LEAKAGE

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.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

BACKGROUND (Continued)

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, requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 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 percent 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).

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 150 gpd per steam generator primary-to-secondary LEAKAGE as the initial condition. An exception to the primary-to-secondary LEAKAGE is described below for the main steamline break (MSLB) analyzed in support of voltage-based steam generator tube repair criteria.

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3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

APPLICABLE SAFETY ANALYSES (Continued)

Primary-to-secondary LEAKAGE is a factor in the dose assessment of accidents or transients that involve secondary steam release to the atmosphere, such as a main steam line break (MSLB), a locked rotor accident (LRA), a Loss of AC Power (LACP), a Control Rod Ejection Accident (CREA) and to a lesser extent, a Steam Generator Tube Rupture (SGTR). The leakage contaminates the secondary fluid. The limit on the primary-to-secondary leakage ensures that the dose contribution at the site boundary from tube leakage following such accidents are limited to appropriate fractions of the 10 CFR 50.67 limit of 25 Rem TEDE as allowable by Regulatory Guide 1.183. The limit on the primary-to-secondary leakage also ensures that the dose contribution from tube leakage in the control room is limited to the 10 CFR 50.67 limit of 5 Rem TEDE. Among all of the analyses that release primary side activity to the environment via tube leakage, the MSLB is of particular concern because the ruptured main steam line provides a pathway to release the primary to secondary leakage directly to the environment without dilution in the secondary fluid.

Due to adoption of the voltage based steam generator tube repair criteria per guidance provided by Generic Letter 95-05, the safety analysis for an event resulting in steam discharge to the atmosphere conservatively assumes a 450 gpd primary-to-secondary LEAKAGE (150 gpd per steam generator) for all accidents other ~~that~~ than the MSLB. The dose consequences associated with the MSLB addresses an ~~accident-induced~~ additional 2.1 gpm leakage, which, per Generic Letter 95-05, is postulated to occur (via pre-existing tube defects) as a result of the rapid depressurization of the secondary side due to the MLSB, and the consequent high differential pressure across the faulted steam generator. The maximum allowed total accident induced leakage is ~~2.1~~ 2.4 gpm.

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. Should pressure boundary LEAKAGE occur through a

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

LCO (Continued)

component which can be isolated from the balance of the Reactor Coolant System, plant operation may continue provided the leaking component is promptly isolated from the Reactor Coolant System since isolation removes the source of potential failure.

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. Primary-_to-_Secondary LEAKAGE through Any One SG

~~Operating experience at PWR plants has shown that sudden increases in leak rate are often precursors to larger tube failures. Maintaining an operating LEAKAGE limit of 150 gpd per steam generator will minimize the potential for a large LEAKAGE event at power. This operating LEAKAGE limit is more restrictive than the operating LEAKAGE limit in standardized technical specifications. This provides additional margin to accommodate a tube flaw which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. This reduced LEAKAGE limit, in conjunction with a leak rate monitoring program, provides additional assurance that this precursor LEAKAGE will be detected and the plant shut down in a timely manner.~~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. 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.

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

LCO (Continued)

d. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the 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.

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 3.4.6.2, "RCS Pressure Isolation Valve (PIV)," 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.

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3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

ACTIONS

~~b~~a. 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. ~~If the unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE.~~

~~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.~~

~~a~~b. If any pressure boundary LEAKAGE exists or primary to secondary LEAKAGE is not within limit, or if unidentified or identified LEAKAGE cannot be reduced to within limits within 4 hours, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within ~~36~~ the following 30 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

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

3/4.4.6.2 OPERATIONAL LEAKAGE (Continued)

SURVEILLANCE REQUIREMENTS (SR)

SR 4.4.6.2.a

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the systems that monitor the containment atmosphere radioactivity and the containment sump level. The 12 hour monitoring of the leakage detection system is sufficient to provide an early warning of increased RCS LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Leakage Detection Instrumentation." Note (1) states that the 12 hour surveillance is required only on leakage detection instrumentation required by LCO 3.4.6.1. This Note allows the 12 hour monitoring to be suspended on leakage detection instrumentation which is inoperable or not required to be operable per LCO 3.4.6.1.

SR 4.4.6.2.b

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 and near operating pressure. ~~Therefore, this SR is not required to be performed in MODES 3 and 4 until 12 hours of steady state operation near operating pressure have been established.~~The SR is modified by two notes. Note 2 states that this SR is not required to be performed until 12 hours after establishing steady state operation. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable plant conditions are established.

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

~~An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the systems that monitor the containment atmosphere radioactivity and the containment sump level. The 12 hour monitoring of the leakage detection system is sufficient to provide an early~~

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~~warning of increased RCS LEAKAGE. These leakage detection systems are specified in LCO 3.4.6.1, "Leakage Detection Instrumentation."~~

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. ~~Note (1) states that the 12 hour surveillance is required only on leakage detection instrumentation required by LCO 3.4.6.1. This Note allows the 12 hour monitoring to be suspended on leakage detection instrumentation which is inoperable or not required to be operable per LCO 3.4.6.1. Note (2) states that this SR is required to be performed during steady state operation.~~

Note 3 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.

SR 4.4.6.2.c

This SR verifies that primary to secondary LEAKAGE is less than 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 3.4.5, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature (25°C) as described in EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines." 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 EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."