FENOC

FirstEnergy Nuclear Operating Company

Richard G. Mende
Director, Site Operations

724-682-7773

September 1, 2006
L-06-132

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

Subject: Beaver Valley Power Station, Unit No. 2
Docket No. 50-412, License No. NPF-73
Supplement to License Amendment Request No. 183 - Submittal of Final Proposed Technical Specification Changes

By letter dated April 11, 2005, FirstEnergy Nuclear Operating Company (FENOC) submitted License Amendment Request (LAR) No. 183 - Revised Steam Generator Inspection Scope, for Beaver Valley Power Station Unit No. 2 (Letter L-05-061, Reference 1). Revised markups to the proposed Technical Specifications and Bases were provided on January 27, 2006 (Letter L-06-013, Reference 2).

On July 19, 2006, the NRC issued Amendment 156 to the BVPS Unit No. 2 Operating License, authorizing the implementation of an extended power uprate. On August 3, 2006, FENOC provided a Supplement to License Amendment Request Nos. 324 and 196, Steam Generator Tube Integrity, for Beaver Valley Unit Nos. 1 and 2 (Letter L-06-119, Reference 3). Attachment A provides final proposed changes to the BVPS Unit 2 Technical Specifications, reflecting both LAR No. 183 and the changes resulting from the above two licensing activities. Attachment B, which proposes final changes to the Technical Specification Bases, is provided for information only.

The proposed final changes to the Technical Specifications do not affect the conclusions of either the supporting safety analysis or the no significant hazard evaluation provided in Reference 1. 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.

A001

I declare under penalty of perjury that the foregoing is true and correct. Executed on September __1__, 2006.

Sincerely,

Richard G. Mende

Reference:

1. Beaver Valley Unit No. 2 License Amendment Request No. 183 - Revised Steam Generator Inspection Scope, Letter L-05-061 dated April 11, 2005
2. Beaver Valley Unit No. 2 Supplement to License Amendment Request No . 183 Revised Steam Generator Inspection Scope (TAC No. MC6768), Letter L-06-013 dated January 27, 2006
3. Beaver Valley Unit Nos. 1 and 2, Supplement to License Amendment Request Nos. 324 and 196 - Steam Generator Tube Integrity (TAC Nos. MC8861 and MCS862), Letter L-06-119 dated August 3, 2006

Attachments:

A. Proposed Technical Specification Changes – LAR No. 183
B. Proposed Technical Specification Bases Changes – LAR No. 183

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

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

License Amendment Request No. 183

The following is a list of the affected pages. Since changes related to LAR No. 196 (Steam Generator Tube Integrity) are subject to license amendment issuance, the most recent revisions proposed for that LAR (Letter L-06-119, August 3, 2006) have been incorporated in the attached pages. Additional markups related to LAR No. 183 are shown in strike-through/double-underline format.

Page
6-22*
6-22a
6-27*
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6-30
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* This page is not changed and is provided for readability only

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Proposed draft page from Unit 2 LAR 196 (TSTF-449 – SG Tube Integrity)

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.

Markups to proposed draft page from Unit 2 LAR 196 (TSTF-449 – SG Tube Integrity)

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.

4. Report the following information to the NRC within 90 days after achieving Mode 4 following an outage in which the F* methodology was applied:

a. Total number of indications, location of each indication, orientation of each indication, severity of each indication, and whether the indications initiated from the inside or outside surface.

b. The cumulative number of indications detected in the tubesheet region as a function of elevation within the tubesheet.

c. The projected end-of-cycle accident-induced leakage from tubesheet indications.

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

Markups to proposed draft page from Unit 2 LAR 196 (TSTF-449 – SG Tube Integrity)

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 Specifications 6.19.c.4 or 6.19.c.5.

2. Tubes ~~with sleeves~~ found by inservice inspection to contain a flaw~~s~~ in a sleeve (~~that are not in~~ excluding 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. Tube support plate voltage-based repair criteria ~~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

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Proposed draft page from Unit 2 LAR 196 (TSTF-449 – SG Tube Integrity)

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.

Markups to proposed draft page
from Unit 2 LAR 196 (TSTF-449 – SG Tube Integrity)

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.

5. The F* methodology, as described below, may be applied to the expanded portion of the tube in the hot-leg tubesheet region as an alternative to the 40% depth based criteria of Technical Specification 6.19.c.1:

a) Tubes with no portion of a lower sleeve joint in the hot-leg tubesheet region shall be repaired or plugged upon detection of any flaw identified within 3.0 inches below the top of the tubesheet or within 2.2 inches below the bottom of roll transition, whichever elevation is lower. Flaws located below this elevation may remain in service regardless of size.

b) Tubes which have any portion of a sleeve joint in the hot-leg tubesheet region shall be plugged upon detection of any flaw identified within 3.0 inches below the lower end of the lower sleeve joint. Flaws located greater than 3.0 inches below the lower end of the lower sleeve joint may remain in service regardless of size.

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

Markups to proposed draft page from Unit 2 LAR 196 (TSTF-449 – SG Tube Integrity)

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, and d.5 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.

Markups to proposed draft page from Unit 2 LAR 196 (TSTF-449 – SG Tube Integrity)

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.

5. When the F* methodology has been implemented, inspect 100% of the inservice tubes in the hot-leg tubesheet region with the objective of detecting flaws that may satisfy the applicable tube repair criteria of Technical Specification 6.19.c.5 every 24 effective full power months or one interval between refueling outages (whichever is less).

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 B

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

License Amendment Request (LAR) No. 183

The following is a list of the affected pages. Since changes related to LAR No. 196 (Steam Generator Tube Integrity) are subject to license amendment issuance, the most recent revisions proposed for that LAR (Letter L-06-119, August 3, 2006) are shown in strike-through/double underline format. One additional markup related to LAR No. 183 is annotated as such.

Page
B 3/4 4-2*
B 3/4 4-3*
B 3/4 4-3a*
B 3/4 4-3b

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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.~~

<u>3/4.4.5 Steam Generator (SG) Tube Integrity</u>

<u>BACKGROUND</u>

<u>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</u>

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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

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During an SG inspect ... eam Generator Program rep ... ice by plugging. If a tu ... ria but was not plugged ... ube integrity.

In the context of t ... the entire length of the ... irs made to it, between ... and the tube-to-tubesheet ... eet weld is not considered part of the tube.

F* INSERT

The combined projected leak rate from all alternate repair criteria (i.e., voltage based repair criteria and application of F*) must be less than the maximum allowable steamline break leak rate limit in any one steam generator in order to maintain doses within the limits of 10 CFR 50.67 as supplemented by Regulatory Guide 1.183 and within GDC-19 values during a postulated steam line break event.

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).

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.