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10 CFR 50.90

RS-05-129

November 18, 2005

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

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. STN 50-456 and STN 50-457

Byron Station, Units 1 and 2
Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Application for Technical Specification Improvement Regarding Steam Generator Tube Integrity

References:
(1) Letter from G. F. Dick (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Braidwood Station, Units 1 and 2 – Issuance of Exigent Amendments RE: Revision of Scope of Steam Generator Inspections for Unit 2 Refueling Outage 11," dated April 25, 2005

(2) Letter from J. B. Hopkins (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station, Unit 2 – Issuance of Amendment," dated September 19, 2005

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) is requesting an amendment to Appendix A, Technical Specifications (TS), of Facility Operating License Nos. NPF-72, NPF-77, NPF-37, and NPF-66 for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, respectively.

The proposed amendment would revise the TS requirements related to steam generator tube integrity. The change is consistent with NRC-approved Revision 4 to Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," with two exceptions as discussed below. The availability of this TS improvement was announced in the Federal Register on May 6, 2005 (i.e., 70 FR 21426) as part of the consolidated line item improvement process (CLIIP).

EGC proposes to revise the TSTF-449 version of TS 5.5.9, Steam Generator Program, to exclude the portion of the tube below 17 inches from the top of the hot leg tubesheet in the

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Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators. This proposed license amendment request, in effect, redefines the Braidwood Station, Unit 2, and Byron Station, Unit 2, primary pressure boundary from the hot leg tube end weld to 17 inches below the top of the hot leg tube sheet. This change is supported by Westinghouse Electric Company, LLC, LTR-CDME-05-32, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2," Revision 2, dated August 2005. The NRC has previously granted similar amendments, on a one-time basis, for Braidwood Station, Unit 2, and Byron Station, Unit 2, in References 1 and 2, respectively.

EGC also proposes to delete Westinghouse laser welded sleeves as a steam generator tube repair method.

LTR-CDME-05-32 contains information proprietary to Westinghouse Electric Company, LLC; it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, it is respectfully requested that the information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR 2.390.

The attached amendment request is subdivided as shown below.

Attachment 1 provides an evaluation of the proposed changes.

Attachments 2-A and 2-B include the marked-up TS pages with the proposed changes indicated for Braidwood Station and Byron Station, respectively.

Attachments 3-A and 3-B include the associated typed TS pages with the proposed changes incorporated for Braidwood Station and Byron Station, respectively.

Attachments 4-A and 4-B include the associated revised TS bases for information only.

Attachment 5 provides an affidavit for withholding signed by Westinghouse Electric Company, LLC, the owner of proprietary information provided in Attachment 7. Also enclosed are a Westinghouse authorization letter, CAW-05-2047, Proprietary Information Notice and Copyright Notice.

Attachment 6 provides a non-proprietary version of Westinghouse LTR-CDME-05-32.

Attachment 7 provides a proprietary version of Westinghouse LTR-CDME-05-32.

EGC requests that this proposed change be approved by September 16, 2006, to support the preparations for Braidwood, Unit 2, Refueling Outage 12. Once approved, the change will be implemented within 60 days.

The proposed amendment has been reviewed by the Braidwood Station and the Byron Station Plant Operations Review Committees and approved by their respective Nuclear Safety Review Boards in accordance with the requirements of the EGC Quality Assurance Program.

EGC is notifying the State of Illinois of this application for a change to the TS by sending a copy of this letter and its attachments to the designated State Official.

Should you have any questions about this letter, please contact J. A. Bauer at (630) 657-2801.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 18th day of November 2005.

Respectfully,

Joseph A. Bauer

Joseph A. Bauer
Manager, Licensing

Attachment 1: Evaluation of Proposed Changes
Attachment 2-A: Markup of Proposed Technical Specifications Page Changes for Braidwood Station
Attachment 2-B: Markup of Proposed Technical Specifications Page Changes for Byron Station
Attachment 3-A: Typed Pages for Technical Specification Changes for Braidwood Station
Attachment 3-B: Typed Pages for Technical Specification Changes for Byron Station
Attachment 4-A: Revised Technical Specification Bases Pages for Braidwood Station
Attachment 4-B: Revised Technical Specification Bases Pages for Byron Station
Attachment 5: Application for Withholding and Affidavit
Attachment 6: Non-proprietary Version of Westinghouse LTR-CDME-05-32, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2," Revision 2, dated August 2005
Attachment 7: Proprietary Version of Westinghouse LTR-CDME-05-32, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2," Revision 2, dated August 2005

ATTACHMENT 1
Evaluation of Proposed Changes

INDEX

1.0 DESCRIPTION

The proposed amendment revises the requirements in Technical Specification (TS) related to steam generator tube integrity for Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2. The changes are consistent with NRC-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4, except as discussed below. The availability of this TS improvement was announced in the Federal Register on May 6, 2005 (i.e., Reference 1), as part of the consolidated line item improvement process (CLIIP).

The proposed amendment also revises the TSTF-449 version of TS 5.5.9, Steam Generator Program, to exclude the portion of the tube below 17 inches from the top of the hot leg tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators from TS 5.5.9.d, "Provisions for SG tube inspections." This proposed license amendment request, in effect, redefines the Braidwood Station, Unit 2, and Byron Station, Unit 2, primary pressure boundary from the hot leg tube end weld to 17 inches below the top of the hot leg tube sheet. This change is supported by Westinghouse Electric Company, LLC, LTR-CDME-05-32 (i.e., Reference 2). In addition, the proposed amendment deletes the current TS 5.5.9.e.6 and TS 5.5.9.e.10 allowance to use Westinghouse laser welded sleeves as a SG tube repair method.

2.0 PROPOSED CHANGE

Consistent with the NRC-approved Revision 4 of TSTF-449, the proposed TS changes include:

- Revised TS definition of LEAKAGE,
- Revised TS 3.4.13, "RCS Operational LEAKAGE,"
- New TS 3.4.19, "Steam Generator (SG) Tube Integrity,"
- Revised TS 5.5.9, "Steam Generator (SG) Program," and
- Revised TS 5.6.9, "Steam Generator Tube Inspection Report."

Proposed revisions to the TS Bases are also included in this application. As discussed in the NRC's model safety evaluation, adoption of the revised TS Bases associated with TSTF-449, Revision 4, is an integral part of implementing this TS improvement. The changes to the affected TS Bases pages will be incorporated in accordance with the TS Bases Control Program.

EGC proposes to add the following alternate repair criteria (ARC) to the proposed TS 5.5.9.c:

> "For Unit 2 only, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair."

EGC proposes to revise the TS 5.5.9.d, "Provisions for SG tube inspections," to exclude the portion of the tube below 17 inches from the top of the hot leg tubesheet from inspections for Braidwood Station, Unit 2, and Byron Station, Unit 2. The following is added to the TS 5.5.9.d description of the extent of the tube inspections:

"For Unit 2, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded."

EGC proposes to revise the existing requirements of TS 5.5.9.e.6 and TS 5.5.9.e.10 by deleting the Westinghouse laser welded sleeving repair methodology. The requirements for the proposed TS 5.5.9.c and TS 5.5.9.f do not allow the use of Westinghouse laser welded sleeves.

3.0 BACKGROUND

The background for this application is adequately addressed by the NRC Notice of Availability published on May 6, 2005 (i.e., 70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (i.e., 70 FR 10298), and TSTF-449, Revision 4, except as discussed below.

Braidwood Station, Unit 2, and Byron Station, Unit 2, each contain four Westinghouse Model D5 recirculating, pre-heater type SGs. Each SG contains 4,570 thermally treated Alloy-600 U-tubes that have an outer diameter of 0.750 inch with a 0.043 inch nominal wall thickness. The support plates are 1.12 inch thick stainless steel and have quatrefoil broached holes. The tubing within the tubesheet is hydraulically expanded throughout the full thickness of the tubesheet. The tubesheet is approximately 21 inches thick. The low row U-bend region, up through row nine, received additional thermal stress relief following tube bending. The units operate on approximately 18-month fuel cycles.

The SG inspection scope is governed by: Braidwood Station TS 5.5.9; Byron Station TS 5.5.9; the Electric Power Research Institute (EPRI) Pressurized Water Reactor (PWR) SG Examination Guidelines; regulatory documents and commitments; Exelon ER-AP-420 procedure series (Steam Generator Management Program Activities); and the results of Braidwood Station, Unit 2, and Byron Station, Unit 2, degradation assessments. The inspection techniques and equipment are capable of reliably detecting the known and potential specific degradation mechanisms applicable to the Braidwood Station, Unit 2, and Byron Station, Unit 2, SGs. The inspection techniques, essential variables and equipment are qualified to Appendix H, "Performance Demonstration for Eddy Current Examination," of the EPRI PWR SG Examination Guidelines.

Indications of cracking were reported based on the results from the nondestructive, eddy current examination of the SG tubes during the fall 2004 outage at Catawba Nuclear Station, Unit 2, as described in Reference 3. Tube indications were reported approximately seven inches from the top of the hot leg tubesheet in one tube, and just above the tube-to-tubesheet welds in a region of the tube known as the tack expansion in several other tubes. Finally, indications were also reported in the tube-end welds (TEWs), also known as tube-to-tubesheet welds, joining the tube to the tubesheet.

Catawba Nuclear Station, Unit 2, has Westinghouse designed Model D5 SGs similar to those in service at Braidwood Station, Unit 2, and Byron Station, Unit 2. Model D5 SGs were fabricated with Alloy 600TT (i.e., thermally treated) tubes. Thus, there is a potential for tube indications similar to those reported at Catawba Nuclear Station, Unit 2, within the hot leg tubesheet region to be identified in the Braidwood Station, Unit 2, and Byron Station, Unit 2, SGs if similar inspections were to be performed. It is noted that the fabrication technique used for the installation of the SG tubes at Braidwood Station, Unit 2, would be expected to lead to a much lower likelihood for crack-like indications to be present in the region known as the tack expansion relative to Catawba Nuclear Station, Unit 2. The Braidwood Station, Unit 2, fabrication technique results in lower residual stress. The fabrication technique used for the installation of the SG tubes at Byron Station, Unit 2, was similar to that used in the Catawba Nuclear Station, Unit 2, SGs. Therefore, the residual stress in the Byron Station, Unit 2, tack expansion region is expected to be similar to the residual stress in the Catawba Nuclear Station, Unit 2, tack expansion region.

Potential inspection plans for the tubes and the welds underwent intensive industry discussions in March 2005. The findings in the Catawba Nuclear Station, Unit 2, SG tubes present three distinct issues with regard to the SG tubes at Braidwood Station, Unit 2, and Byron Station, Unit 2:

1) indications in internal bulges and overexpansions within the hot leg tubesheet;
2) indications at the elevation of the tack expansion transition; and
3) indications in the tube-to-tubesheet welds and propagation of these indications into the adjacent tube material.

In order to preclude unnecessarily plugging tubes in the Braidwood Station, Unit 2, and Byron Station, Unit 2, SGs, an analysis was performed to identify the portion of the tube within the hot leg tubesheet necessary to maintain structural and leakage integrity for both normal operating and accident conditions. Tube inspections will be limited to identifying and repairing degradation in this portion of the tubes. The technical justification for the inspection and repair methodology is provided in Reference 2. The limited hot leg tubesheet inspection criteria were developed for the hot leg tubesheet region of Model D5 SGs considering the most stringent loads associated with plant operation, including transients and postulated accident conditions. The limited hot leg tubesheet inspection criteria were selected to prevent tube burst and axial separation due to axial pullout forces acting on the tube and to ensure that the steam line break (SLB) leakage limits are not exceeded. Reference 2 provides technical justification for allowing tubes with indications that are below 17 inches from the top of the hot leg tubesheet (i.e., within approximately four inches of the tube end) to remain in-service.

Constraint provided by the hot leg tubesheet precludes tube burst for cracks within the tubesheet. The criteria for tube burst described in Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," Revision 1 dated January 2001 (i.e., Reference 4), and NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," dated August 1976 (i.e., Reference 5), are satisfied due to the constraint provided by the tubesheet. Through application of the limited hot leg tubesheet inspection scope described herein, the existing operating leakage limit provides assurance that excessive leakage (i.e., greater than accident analysis assumptions) will not occur during a postulated SLB event.

Implementation of this proposed methodology involves limited inspection of the tubes within the hot leg tubesheet to depths of 17 inches from the top of the tubesheet using specialized rotating eddy current probes. The limited tubesheet inspection length of tubing must be demonstrated to be non-degraded below the top of the tubesheet interface. If cracks are found within the top of hot leg tubesheet to 17 inches below the top of tubesheet, the tube must be repaired or removed from service.

The NRC has previously granted similar amendments, on a one-time basis, for Braidwood Station, Unit 2, and Byron Station, Unit 2, in References 6 and 7, respectively.

4.0 TECHNICAL ANALYSIS

Exelon Generation Company, LLC (EGC) has reviewed the safety evaluation (SE) published on March 2, 2005 (i.e., 70 FR 10298), as part of the CLIIP Notice for Comment. This included the NRC staff's SE, the supporting information provided to support TSTF-449, and the changes associated with Revision 4 to TSTF-449. EGC has concluded that the justifications presented in the TSTF proposal and the SE prepared by the NRC staff are applicable to Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, and justify this amendment for the incorporation of the changes to their respective TS, except as discussed below.

A technical justification has been developed to identify the safety significant portion of the tube within the tubesheet. This justification (i.e., Reference 2) has been reviewed and approved in accordance with the requirements of Exelon Generation Company, LLC (EGC) procedures and is provided as Attachment 7. The safety significant portion of the tube is the length of tube that is engaged in the tubesheet from the secondary face that is required to maintain structural and leakage integrity over the full range of steam generator operating conditions, including the most limiting accident conditions. The evaluation determined that degradation in tubing below the safety significant portion of the tube does not require repair and serves as the basis for the tubesheet inspection program.

The bases for determining the safety significant portion of the tube within the tubesheet is based upon analyses and testing programs that quantified the tube-to-tubesheet radial contact pressure for bounding plant conditions as described in Reference 2. The tube-to-tubesheet radial contact pressure provides resistance to tube pull-out and resistance to leakage during plant operation and transients. Temperature effects and upward bending of the tubesheet due to primary and secondary differential pressure during normal and transient conditions, result in the tube-to-tubesheet contact pressure increasing with distance from the top of the tubesheet. Due to these effects, the tubesheet bore tends to dilate near the top of the tubesheet and constricts the tube near the bottom of the tubesheet. Testing and analyses have shown that tube-to-tubesheet engagement lengths of approximately three inches to 8.6 inches were sufficient to maintain structural integrity (i.e., resist tube pull-out resulting from loading considering differential pressures of three times the normal operating pressure difference and considering differential pressures of 1.4 times the limiting accident pressure difference). The variation of the required engagement length is a function of the radial tube location within the tube bundle. EGC has decided to add additional conservatism to the minimum structural distances of three inches to 8.6 inches by performing inspections to depths of 17 inches below the top of the hot leg tubesheet. The increase in contact pressure at this depth significantly increases the tube structural strength and resistance to leakage.

Since the proposed 17-inch tube inspection depth traverses below the mid-plane of the hot leg tubesheet, the tube-to-tubesheet contact pressure significantly aids in restricting primary-to-secondary leakage as differential pressure increases. Based on engineering judgment, given that there is no significant primary-to-secondary leakage during normal operation, there will be no significant leakage during postulated accident conditions from indications located below the mid-plane of the tubesheet (i.e., greater than approximately 10.5 inches below the top of the tubesheet). The rationale for this conclusion based on engineering judgment is the interaction of temperature and tubesheet bending effects that increase the contact pressure between the tube and the tubesheet, thereby increasing the resistance to primary-to-secondary leakage during normal operating or accident conditions.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., SLB) is limited by flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications. The primary-to-secondary leak rate during postulated SLB accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet (i.e., including indications in the tube end welds). This conclusion is based on the observation that while the driving pressure causing leakage increases by approximately a factor of two, the flow resistance associated with an increase in the tube-to-tubesheet contact pressure, during a SLB, increases by up to approximately a factor of three. While such a leakage decrease is logically expected, the postulated accident leak rate could be conservatively bounded by twice the normal operating leak rate if the increase in contact pressure is ignored. Since normal operating leakage is limited to less than 0.104 gpm (i.e., 150 gpd) per TS 3.4.13, "RCS Operational Leakage," the associated accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by approximately 0.2 gpm. This value is well within the assumed faulted SG accident leakage rate of 0.5 gpm discussed in Byron/Braidwood Updated Final Safety Analysis Report, Table 15.1-3, "Parameters Used in Steam Line Break Analyses." Hence it is reasonable to omit any consideration of inspection of the tube, tube end weld, bulges/overexpansions or other anomalies below 17 inches from the top of the hot leg tubesheet.

The proposed inspection sampling length of 17 inches from the top of the hot leg tubesheet provides a high level of confidence that the structural and leakage criteria are maintained during normal operating and accident conditions.

Degradation found in the portion of the tube below 17 inches from the top of the hot leg does not require repair or plugging as described in Reference 2.

In summary:

- Reference 2 notes that the structural integrity requirements of NEI 97-06 (i.e., Reference 4), and RG 1.121 (i.e., Reference 5), are met by sound tube engagement lengths ranging from approximately three to 8.6 inches from the top of the hot leg tubesheet. The region of the tube below those elevations, including the tube-to-tubesheet weld, is not needed for structural integrity during normal operation or accident conditions. Inspections will be performed to a depth of 17 inches from the top of the hot leg tubesheet.
- NEI 97-06 defines the tube as extending from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, but specifically excludes the tube-to-tubesheet weld from the definition of the tube.
- The welds were originally designed and analyzed as the primary pressure boundary in accordance with the requirements of Section III of the 1971 edition of the American

Society of Mechanical Engineers (ASME) Code, Summer 1972 Addenda and selected paragraphs of the Winter 1974 Addenda for the Braidwood Station, Unit 2 and Byron Station, Unit 2, SGs. This proposed license amendment request, in effect, redefines the primary pressure boundary from the tube end weld to 17 inches below the top of the hot leg tube sheet.

- Section XI of the ASME Code deals with the in-service inspection of nuclear power plant components. The ASME Code (i.e., Editions 1971 through 2004) specifically recognizes that the SG tubes are under the purview of the NRC through the implementation of the requirements of the TS as part of the plant operating license.

Deleting the allowance to use the Westinghouse laser welded sleeving repair methodology results in more conservative requirements for repair of degraded SG tubes. No Westinghouse laser-welded sleeves are currently installed in the Braidwood Station, Unit 2, and Byron Station, Unit 2, SGs. EGC has been informed by Westinghouse Electric Company, LLC, that they (i.e., Westinghouse) no longer possess the technology associated with laser welded sleeves. Reference to Westinghouse laser welded sleeving as a repair methodology is no longer appropriate and is therefore deleted.

Following implementation of this proposed amendment, Braidwood Station, Unit 2, and Byron Station, Unit 2, TS 5.5.9.f will allow use of ABB Combustion Engineering (CE), Inc., TIG welded sleeving methodology for repair of degraded SG tubes. CE Licensing Report CEN-627-P, "Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," dated January 1996 provides limitations on the use of TIG welded sleeves for use in Westinghouse designed Model D SGs. Specifically, only two types of TIG welded sleeves, the roll transition zone sleeve and tube support plate sleeve, are considered for installation in Braidwood Station, Unit 2, and Byron Station, Unit 2, SGs. The tube support plate sleeve is approximately centered at either the first and/or second support plate and does not extend into the tubesheet. The upper end of the roll transition zone sleeve is located above the secondary face of the tubesheet, while the sleeve lower end is located near the neutral axis of the tubesheet. Therefore, the sleeve joint would be located in the portion of the SG tube that is within the proposed inspection length (i.e., in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet).

5.0 REGULATORY ANALYSIS

A description of this proposed change and its relationship to applicable regulatory requirements and guidance was provided in the NRC Notice of Availability published on May 6, 2005 (70 FR 10298), and TSTF-449, Revision 4.

5.1 VERIFICATION AND COMMITMENTS

The following information is supplied to support the NRC staff's review of this amendment application:

Plant	**Braidwood, Unit 1**
Steam Generator Model(s)	Babcock & Wilcox feedring replacement steam generators
Effective Full Power Years (EFPY) of	Approximately 5.64 EFPY at time of last inspection in

Plant	Braidwood, Unit 1
service for currently installed SGs	October 2004 (i.e., Refueling Outage A1R11)
Tubing Material	Alloy 690
Number of tubes per SG	6,633
Number and percentage of tubes plugged in each SG	SG A: 10 (~0.15%) SG B: 14 (~0.21%) SG C: 5 (~0.08%) SG D: 1 (~0.02%)
Number of tubes repaired in each SG	None
Degradation mechanisms identified	Fan Bar Wear, Lattice Grid Wear, and Foreign Object Wear
Current primary-to-secondary leakage limits: per SG: Total: Leakage evaluated at:	 150 gallons per day 600 gallons per day Room Temperature
Approved Alternate Tube Repair Criteria (ARC): 1. None	Not Applicable
Approved SG Tube Repair Methods: 1. None	Not Applicable
Performance criteria for accident leakage:	
1. Main Steamline Break	Faulted SG – 0.5 gpm at Room Temperature Each intact SG – 0.218 gpm at Room Temperature
2. Locked Rotor	Total for all SGs – 1.0 gpm at Room Temperature
3. Locked Rotor with Failed Open SG Power Operated Relief Valve (PORV)	SG with failed open PORV – 0.5 gpm at Room Temperature Each intact SG – 0.218 gpm at Room Temperature
4. Rod Cluster Control Assembly Ejection	Total for all SGs – 1.0 gpm at Room Temperature
5. Steam Generator Tube Rupture (SGTR)	Total for intact SGs – 1.0 gpm at Room Temperature

Plant	Braidwood, Unit 2
Steam Generator Model(s)	Westinghouse Model D5
Effective Full Power Years (EFPY) of service for currently installed SGs	Approximately 14.16 EFPY at time of last inspection in April 2005 (i.e., Refueling Outage A2R11)
Tubing Material	Alloy 600TT
Number of tubes per SG	4,570
Number and percentage of tubes plugged in each SG	SG A: 57 (~1.25%) SG B: 48 (~1.05%) SG C: 56 (~1.23%) SG D: 25 (~0.55%)
Number of tubes repaired in each SG	None

ATTACHMENT 1
Evaluation of Proposed Changes

Plant	**Braidwood, Unit 2**
Degradation mechanisms identified	Anti-Vibration Wear, Pre-Heater/Tube Support Plate Wear, Foreign Object Wear, and Outside Diameter Stress Corrosion Cracking (ODSCC) at the tube support plate intersections
Current primary-to-secondary leakage limits: per SG: Total: Leakage evaluated at:	 150 gallons per day 600 gallons per day Room Temperature
Approved Alternate Tube Repair Criteria (ARC): 1. Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.	 Amendment 135, dated April 25, 2005, approved this allowance on a one-time basis for Refueling Outage 11 and the subsequent operating cycle.
Approved SG Tube Repair Methods: 1. Laser welded sleeving as described in a Westinghouse Technical Report and subject to the limitations and restrictions as approved by the NRC. (Laser welded sleeving methodology is being deleted by this amendment request) 2. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and Unit 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, "Operating Performance of the ABB CENO Steam Generator Tube Sleeve for use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.	 Amendment No. 46 dated March 4, 1994, approved use of WCAP-13698, Revision 1, "Laser Welded Sleeves for 3/4-Inch Diameter Tube Feeding-Type and Westinghouse Preheater Steam Generators" Applicability limits: Elevated laser welded tubesheet sleeves were not approved for use. Amendment No. 113 dated May 4, 2001, approved use of WCAP-13698, Revision 4 for sleeve plugging criteria Sleeve plugging criteria: 38.7% of the nominal wall thickness. Amendment No. 75 dated April 12, 1996 Applicability limits: 1) A single tube may contain a maximum of one tubesheet sleeve and two support plate sleeves; 2) Post Weld Heat Treatment is required on all freespan welds; and 3) Expanded inspections are required for verification of tube cleaning prior to welding. Sleeve plugging criteria: 32% of the nominal wall thickness.

Plant	Braidwood, Unit 2
Performance criteria for accident leakage:	
1. Main Steamline Break	Faulted SG – 0.5 gpm at Room Temperature Each intact SG – 0.218 gpm at Room Temperature
2. Locked Rotor	Total for all SGs – 1.0 gpm at Room Temperature
3. Locked Rotor with Failed Open SG Power Operated Relief Valve (PORV)	SG with failed open PORV – 0.5 gpm at Room Temperature Each intact SG – 0.218 gpm at Room Temperature
4. Rod Cluster Control Assembly Ejection	Total for all SGs – 1.0 gpm at Room Temperature
5. Steam Generator Tube Rupture (SGTR)	Total for intact SGs – 1.0 gpm at Room Temperature

Plant	Byron, Unit 1			
Steam Generator Model(s)	Babcock & Wilcox feedring replacement steam generators			
Effective Full Power Years (EFPY) of service for currently installed SGs	Approximately 6.684 EFPY at time of last inspection in March 2005 (i.e., Refueling Outage B1R13)			
Tubing Material	Alloy 690			
Number of tubes per SG	6,633			
Number and percentage of tubes plugged in each SG	SG A 1 (~0.02%)	SG B 1 (~0.02%)	SG C 1 (~0.02%)	SG D 5 (~0.08%)
Number of tubes repaired in each SG	None			
Degradation mechanisms identified	Fan Bar Wear, Lattice Grid Wear, and Foreign Object Wear			
Current primary-to-secondary leakage limits: per SG: Total: Leakage evaluated at:	 150 gallons per day 600 gallons per day Room Temperature			
Approved Alternate Tube Repair Criteria (ARC): 1. None	Not Applicable			
Approved SG Tube Repair Methods: 1. None	Not Applicable			

Plant	Byron, Unit 1
Performance criteria for accident leakage:	
1. Main Steamline Break	Faulted SG – 0.5 gpm at Room Temperature Each intact SG – 0.218 gpm at Room Temperature
2. Locked Rotor	Total for all SGs – 1.0 gpm at Room Temperature
3. Locked Rotor with Failed Open SG Power Operated Relief Valve (PORV)	SG with failed open PORV – 0.5 gpm at Room Temperature Each intact SG – 0.218 gpm at Room Temperature
4. Rod Cluster Control Assembly Ejection	Total for all SGs – 1.0 gpm at Room Temperature
5. Steam Generator Tube Rupture (SGTR)	Total for intact SGs – 1.0 gpm at Room Temperature

Plant	Byron, Unit 2			
Steam Generator Model(s)	Westinghouse Model D5			
Effective Full Power Years (EFPY) of service for currently installed SGs	Approximately 15.738 EFPY at time of last inspection in October 2005 (i.e., Refueling Outage B2R12)			
Tubing Material	Alloy 600TT			
Number of tubes per SG	4,570			
Number and percentage of tubes plugged in each SG	SG A 144 (~3.15%)	SG B 123 (~2.69%)	SG C 57 (~1.25%)	SG D 25 (~0.55%)
Number of tubes repaired in each SG	None			
Degradation mechanisms identified	Anti-Vibration Wear, Pre-Heater/Tube Support Plate Wear, and Foreign Object Wear			
Current primary-to-secondary leakage limits: per SG: Total: Leakage evaluated at:	 150 gallons per day 600 gallons per day Room Temperature			
Approved Alternate Tube Repair Criteria (ARC): 1. Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.	 Amendment 144, dated September 19, 2005, approved this allowance on a one-time basis for Refueling Outage 12 and the subsequent operating cycle.			

Plant	Byron, Unit 2
Approved SG Tube Repair Methods:	
1. Laser welded sleeving as described in a Westinghouse Technical Report and subject to the limitations and restrictions as approved by the NRC. (Laser welded sleeving methodology is being deleted by this amendment request)	Amendment No. 58 dated March 4, 1994, approved use of WCAP-13698, Revision 1, "Laser Welded Sleeves for 3/4-Inch Diameter Tube Feeding-Type and Westinghouse Preheater Steam Generators" Applicability limits: Elevated laser welded tubesheet sleeves not approved for use. Amendment No. 119 dated May 4, 2001, approved use of WCAP-13698, Revision 4 for sleeve plugging criteria Sleeve plugging criteria: 38.7% of the nominal wall thickness.
2. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and Unit 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, "Operating Performance of the ABB CENO Steam Generator Tube Sleeve for use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.	Amendment No. 83 dated April 12, 1996 Applicability limits: 1) A single tube may contain a maximum of one tubesheet sleeve and two support plate sleeves; 2) Post Weld Heat Treatment is required on all freespan welds; and 3) Expanded inspections are required for verification of tube cleaning prior to welding. Sleeve plugging criteria: 32% of the nominal wall thickness.
Performance criteria for accident leakage:	
1. Main Steamline Break	Faulted SG – 0.5 gpm at Room Temperature Each intact SG – 0.218 gpm at Room Temperature
2. Locked Rotor	Total for all SGs – 1.0 gpm at Room Temperature
3. Locked Rotor with Failed Open SG Power Operated Relief Valve (PORV)	SG with failed open PORV – 0.5 gpm at Room Temperature Each intact SG – 0.218 gpm at Room Temperature
4. Rod Cluster Control Assembly Ejection	Total for all SGs – 1.0 gpm at Room Temperature
5. Steam Generator Tube Rupture (SGTR)	Total for intact SGs – 1.0 gpm at Room Temperature

5.2 NO SIGNIFICANT HAZARDS CONSIDERATION

Exelon Generation Company, LLC, (EGC) has reviewed the proposed no significant hazards consideration determination published on March 2, 2005 (i.e., 70 FR 10298) as part of the consolidated line item improvement process (CLIIP) item. EGC has concluded that the proposed determination presented in the notice is applicable to Braidwood Station, Units 1 and 2, and Byron Station, Units 1 and 2, and the determination is hereby incorporated by reference to satisfy the requirements of 10 CFR 50.91 (a), except as discussed below.

The proposed amendment also revises the Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4, version of TS 5.5.9, Steam Generator Program, to exclude the portion of the tube below 17 inches from the top of the hot leg tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators from TS 5.5.9.d, "Provisions for SG tube inspections." This proposed license amendment request, in effect, redefines the Braidwood Station, Unit 2, and Byron Station, Unit 2, primary pressure boundary from the hot leg tube end weld to 17 inches below the top of the hot leg tube sheet. This proposed license amendment also deletes the current TS 5.5.9.e.6 and TS 5.5.9.e.10 allowance to use Westinghouse laser welded sleeves as a SG tube repair method.

EGC has evaluated whether or not a significant hazards consideration is involved with the proposed TS change by focusing on the three criteria set forth in 10 CFR 50.92 as discussed below:

Criteria

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed changes that alter the SG inspection criteria and delete the allowance to repair SG tubes using Westinghouse laser welded sleeves do not have a detrimental impact on the integrity of any plant structure, system, or component that initiates an analyzed event. The proposed changes will not alter the operation of, or otherwise increase the failure probability of any plant equipment that initiates an analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   Of the applicable accidents previously evaluated, the limiting transients with consideration to the proposed changes to the SG tube inspection criteria, are the SG tube rupture (SGTR) event and the steam line break (SLB) accident.

   During the SGTR event, the required structural integrity margins of the SG tubes will be maintained by the presence of the SG tubesheet. SG tubes are hydraulically expanded in the tubesheet area. Tube rupture in tubes with cracks in the tubesheet is precluded by the constraint provided by the tubesheet. This constraint results from the hydraulic expansion process, thermal expansion mismatch between the tube and tubesheet and from the differential pressure between the primary and secondary side. Based on this design, the structural margins against burst, discussed in Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR SG Tubes," are maintained for both normal and postulated accident conditions.

   The proposed changes do not affect other systems, structures, components or operational features. Therefore, the proposed changes result in no significant increase in the probability of the occurrence of a SGTR accident.

At normal operating pressures, leakage from primary water stress corrosion cracking (PWSCC) below the proposed limited inspection depth is limited by both the tube-to-tubesheet crevice and the limited crack opening permitted by the tubesheet constraint. Consequently, negligible normal operating leakage is expected from cracks within the tubesheet region. The consequences of an SGTR event are affected by the primary-to-secondary leakage flow during the event. Primary-to-secondary leakage flow through a postulated broken tube is not affected by the proposed change since the tubesheet enhances the tube integrity in the region of the hydraulic expansion by precluding tube deformation beyond its initial hydraulically expanded outside diameter.

The probability of a SLB is unaffected by the potential failure of a SG tube as this failure is not an initiator for a SLB.

The consequences of a SLB are also not significantly affected by the proposed changes. During a SLB accident, the reduction in pressure above the tubesheet on the shell side of the SG creates an axially uniformly distributed load on the tubesheet due to the reactor coolant system pressure on the underside of the tubesheet. The resulting bending action constrains the tubes in the tubesheet thereby restricting primary-to-secondary leakage below the midplane.

Primary-to-secondary leakage from tube degradation in the tubesheet area during the limiting accident (i.e., SLB) is limited by flow restrictions resulting from the crack and tube-to-tubesheet contact pressures that provide a restricted leakage path above the indications and also limit the degree of potential crack face opening as compared to free span indications. The primary-to-secondary leak rate during postulated SLB accident conditions would be expected to be less than that during normal operation for indications near the bottom of the tubesheet (i.e., including indications in the tube end welds). This conclusion is based on the observation that while the driving pressure causing leakage increases by approximately a factor of two, the flow resistance associated with an increase in the tube-to-tubesheet contact pressure, during a SLB, increases by up to approximately a factor of three. While such a leakage decrease is logically expected, the postulated accident leak rate could be conservatively bounded by twice the normal operating leak rate if the increase in contact pressure is ignored. Since normal operating leakage is limited to less than 0.104 gpm (150 gpd) per TS 3.4.13, "RCS Operational Leakage," the associated accident condition leak rate, assuming all leakage to be from lower tubesheet indications, would be bounded by approximately 0.2 gpm. This value is well within the assumed accident leakage rate of 0.5 gpm discussed in Updated Final Safety Analysis Table 15.1-3, "Parameters Used in Steam Line Break Analyses." Hence it is reasonable to omit any consideration of inspection of the tube, tube end weld, bulges/overexpansions or other anomalies below 17 inches from the top of the hot leg tubesheet. Therefore, the consequences of a SLB accident remain unaffected.

Based on the above discussion, the proposed changes do not involve an increase in the consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes do not involve the use or installation of new equipment and the currently installed equipment will not be operated in a new or different manner. No new or different system interactions are created and no new processes are introduced. The proposed changes will not introduce any new failure mechanisms, malfunctions, or accident initiators not already considered in the design and licensing bases.

   Based on this evaluation, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed changes maintain the required structural margins of the SG tubes for both normal and accident conditions. Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," Revision 1 and Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," are used as the bases in the development of the limited hot leg tubesheet inspection depth methodology for determining that SG tube integrity considerations are maintained within acceptable limits. RG 1.121 describes a method acceptable to the NRC for meeting General Design Criteria (GDC) 14, "Reactor coolant pressure boundary," GDC 15, "Reactor coolant system design," GDC 31, "Fracture prevention of reactor coolant pressure boundary," and GDC 32, "Inspection of reactor coolant pressure boundary," by reducing the probability and consequences of a SGTR. RG 1.121 concludes that by determining the limiting safe conditions for tube wall degradation the probability and consequences of a SGTR are reduced. This RG uses safety factors on loads for tube burst that are consistent with the requirements of Section III of the American Society of Mechanical Engineers (ASME) Code.

   For axially oriented cracking located within the tubesheet, tube burst is precluded due to the presence of the tubesheet. For circumferentially oriented cracking, Westinghouse letter LTR-CDME-05-32, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron Unit 2 and Braidwood Unit 2," Revision 2, dated August 2005, defines a length of degradation free expanded tubing that provides the necessary resistance to tube pullout due to the pressure induced forces, with applicable safety factors applied. Application of the limited hot leg tubesheet inspection depth criteria will preclude unacceptable primary-to-secondary leakage during all plant conditions. The methodology for determining leakage provides for large margins between calculated and actual leakage values in the proposed limited hot leg tubesheet inspection depth criteria.

Therefore, the proposed changes do not involve a significant hazards consideration under the criteria set forth in 10 CFR 50.92(c).

5.3 APPLICABLE REGULATORY REQUIREMENTS/CRITERIA

The applicable regulatory requirements and guidance associated with this application are adequately addressed by the NRC Notice of Availability published on May 6, 2005 (i.e., 70 FR 24126), the NRC Notice for Comment published on March 2, 2005 (i.e., 70 FR 10298), and TSTF-449, Revision 4, except as discussed below.

Section 182a of the Atomic Energy Act requires applicants for nuclear power plant operating licenses to include technical specifications (TS) as part of the license. The Commission's regulatory requirements related to the content of the TS are contained in Title 10, Code of Federal Regulations (10 CFR), Section 50.36, "Technical specifications." The TS requirements in 10 CFR 50.36 include the following categories: (1) safety limits, limiting safety systems settings and control settings, (2) limiting conditions for operation (LCO), (3) surveillance requirements, (4) design features, and (5) administrative controls. The SG tube inspection requirements are included in the TS in accordance with 10 CFR 50.36(c)(5), "Limiting Conditions for Operation."

As stated in 10 CFR 50.59, "Changes, tests, and experiments," paragraph (c)(1)(i), a licensee is required to submit a license amendment pursuant to 10 CFR 50.90, "Application for amendment of license or construction permit," if a change to the TS is required. Furthermore, the requirements of 10 CFR 50.59 necessitate that the NRC approve the TS changes before the TS changes are implemented. EGC's submittal revising the requirements of TS 5.5.9, Steam Generator Program, as provided in TSTF-449, to exclude the portion of the tube below 17 inches from the top of the hot leg tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators from TS 5.5.9.d, "Provisions for SG tube inspections," and to delete Westinghouse laser welded sleeves as an approved SG tube repair method meets the requirements of 10 CFR 50.59(c)(1)(i) and 10 CFR 50.90.

RG 1.121 margins against burst are maintained for both normal and postulated accident conditions due to the constraint provided by the tubesheet.

NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," dated April 7, 2005, provides additional regulatory insight regarding SG tube degradation.

6.0 ENVIRONMENTAL CONSIDERATION

EGC has reviewed the environmental evaluation included in the model SE published on March 2, 2005 (i.e., 70 FR 10298) as part of the CLIIP. EGC has concluded that the staff's findings presented in that evaluation are applicable to Braidwood Station and Byron Station, Units 1 and 2, and the evaluation is hereby incorporated by reference for this application, except as discussed below.

A review has determined that the proposed amendment revising the Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-449, "Steam Generator Tube Integrity," Revision 4, version of TS 5.5.9, Steam Generator Program, to exclude the portion of the tube below 17 inches from the top of the hot leg tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators from TS 5.5.9.d, "Provisions for SG tube inspections," and to delete Westinghouse laser welded sleeves as an approved SG tube repair method would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for protection against radiation," or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in paragraph (c)(9) of 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review." Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 PRECEDENT

This application is being made in accordance with the CLIIP. EGC is, however, proposing a variation or deviation from the TS changes described in TSTF-449, Revision 4, or the NRC staff's model SE published on March 2, 2005 (i.e., 70 FR 10298). One variation revises the requirements of TS 5.5.9, Steam Generator Program, as provided in TSTF-449, to exclude the portion of the tube below 17 inches from the top of the hot leg tubesheet in the Braidwood Station, Unit 2, and Byron Station, Unit 2, steam generators from TS 5.5.9.d, "Provisions for SG tube inspections." The NRC has previously granted similar amendments, on a one-time basis, for Braidwood Station, Unit 2, and Byron Station, Unit 2, in References 6 and 7, respectively. A second variation deletes the allowance to use Westinghouse laser welded sleeves as a repair method.

8.0 REFERENCES

1. Federal Register Notices:

   Notice for Comment published on March 2, 2005 (i.e., 70 FR 10298).
   Notice of Availability published on May 6, 2005 (i.e., 70 FR 24126).

2. LTR-CDME-05-32, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2," Revision 2, dated August 2005

3. NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," dated April 7, 2005

4. NEI 97-06, "Steam Generator Program Guidelines," Revision 1, dated January 2001

5. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes, " dated August 1976

6. Letter from G. F. Dick (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Braidwood Station, Units 1 and 2 – Issuance of Exigent Amendments RE: Revision of Scope of Steam Generator Inspections for Unit 2 Refueling Outage 11," dated April 25, 2005

7. Letter from J. B. Hopkins (NRC) to C. M. Crane (Exelon Generation Company, LLC), "Byron Station, Unit 2 – Issuance of Amendment," dated September 19, 2005

Attachment 2-A

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

Application for Technical Specification Improvement
Regarding Steam Generator Tube Integrity

Markup of Technical Specifications Pages

1.1 Definitions

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

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

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

3. Reactor Coolant System (RCS) LEAKAGE through a ~~S~~team ~~G~~enerator ~~(SG)~~ to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

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

c. Pressure Boundary LEAKAGE

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

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

~~d. 600 gallons per day total primary to secondary LEAKAGE through all Steam Generators (SGs); and~~

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

INSERT NEW SPECIFICATION 3.4.19

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 Steam Generator (SG) Tube Integrity

LCO 3.4.19 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.

ACTIONS

------------------------------NOTE------------------------------

Separate Condition entry is allowed for each SG tube.

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

INSERT NEW SPECIFICATION 3.4.19

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.19.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.19.2	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

5.5 Programs and Manuals

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

INSERT 5.5 - 7

Each SG shall be demonstrated OPERABLE by performance of an augmented inservice inspection program.

a. SG Sample Selection and Inspection

Each SG shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of SGs specified in Table 5.5.9-1.

b. SG Tube Sample Selection and Inspection

-----------------------------NOTE-----------------------------
When referring to an SG tube, the sleeve shall be considered a part of the tube if the tube has been repaired per Specification 5.5.9.e.10.

The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of SG tubes shall be performed at the frequencies specified in Specification 5.5.9.d and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9.e. When applying the expectations of Specification 5.5.9.b.1 through 5.5.9.b.3, previous defects or imperfections in the area repaired by the sleeve are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include ≥ 3% of the total number of tubes in all SGs. The tubes selected for these inspections shall be selected on a random basis except:

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then ≥ 50% of the tubes inspected shall be from these critical areas;

INSERT 5.5-7

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. 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.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

INSERT 5.5 - 8

2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each SG shall include:

   i. All tubes that previously had detectable tube wall penetrations > 20% that have not been plugged or sleeved in the affected area, and all tubes that previously had detectable sleeve wall penetrations that have not been plugged,

   ii. Tubes in those areas where experience has indicated potential problems,

   iii. A tube inspection (pursuant to Specification 5.5.9.e.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;

3. The tubes selected as the second and third samples (if required by Table 5.5.9-2) during each inservice inspection may be subjected to a partial tube inspection provided:

   i. 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

   ii. The inspections include those portions of the tubes where imperfections were previously found;

INSERT 5.5-8

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 not to exceed a total of 1 gpm for all SGs.

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

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired. For Unit 2 only, degradation identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection. TIG welded sleeves found by inservice inspection to contain flaws with a depth equal to or exceeding 32% of the nominal wall thickness shall be plugged.

The following alternate tube repair criteria may be applied as an alternative to the 40% depth-based criteria:

1. For Unit 2 only, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.

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. For Unit 2, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

4. A random sample of ≥ 20% of the total number of laser welded sleeves and ≥ 20% of the total number of Tungsten Inert Gas (TIG) welded sleeves installed shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection exceeding the repair limit is detected, an additional 20% of the unsampled sleeves shall be inspected and if an imperfection exceeding the repair limit is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve, the tube at the heat treated area, and the tube-to-sleeve joints. The inservice inspection for the sleeves is required on all types of sleeves installed in the SGs to demonstrate acceptable structural integrity;

INSERT 5.5 - 9

5. For Unit 2 during Refueling Outage 11, a 20% minimum sample of all inservice tubes from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be inspected by rotating probe. This sample shall include a 20% minimum sample of the total population of bulges and overexpansions within the SG from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet.

c. Inspection Results Classification

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

NOTE

Previously degraded tubes or sleeves must exhibit significant (> 10% of wall thickness) further wall penetrations to be included in the percentage calculations.

Category	Inspection Results
C-1	< 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but ≤ 1% of the total tubes inspected are defective, or ≥ 5% and ≤ 10% of the total tubes inspected are degraded tubes.
C-3	> 10% of the total tubes inspected are degraded tubes or > 1% of the inspected tubes are defective.

INSERT 5.5-9

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 Unit 1 tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

   Inspect 100% of the Unit 2 tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for Unit 2 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

5.5 Programs and Manuals

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

d. Inspection Frequencies

The inservice inspections of SG tubes (dependent upon inspection results classification) shall be performed at the following frequencies:

1. The first inservice inspection shall be performed after 6 Effective Full Power months but ≤ 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals ≥ 12 calendar months and ≤ 24 calendar months after the previous inspection;

2. Extension Criteria: 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. An exception to this Extension Criteria is that for Braidwood Unit 1 a one-time inspection interval extension of a maximum of once per 40 months is allowed for the inspection performed immediately following the A1R08 inspection. This is an exception to the Extension Criteria in that the inspection interval extension is based on the result of only one inspection result falling into the C-1 category;

3. If the results of the inservice inspection of an SG conducted in accordance with Table 5.5.9-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.d.2; the interval may then be extended to a maximum of once per 40 months; and

INSERT 5.5-10

these Specifications, tube plugging is not a repair. All acceptable repair methods are listed below.

1. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, "Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

4. Additional unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:

   i. Reactor to secondary tube leaks (not including leaks originating from tube to tube sheet welds) in excess of the limits of LCO 3.4.13.d and LCO 3.4.13.e, "RCS Operational LEAKAGE",

   ii. A seismic occurrence greater than the Operating Basis Earthquake (OBE),

   iii. A Condition IV Loss Of Coolant Accident (LOCA) requiring actuation of the Engineered Safety Features, or

   iv. A Condition IV main steam line or feedwater line break.

The provisions of SR 3.0.2 are not applicable to SG Tube Surveillance Program inspection frequencies.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

e. Acceptance Criteria

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 < 20% of the nominal tube or sleeve 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 unrepaired imperfections ≥ 20% of the nominal tube or sleeve wall thickness caused by degradation;

4. % 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 or sleeve containing an unrepaired defect is defective;

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.~~ The plugging or repair limit imperfection depth for the tubing is equal to 40% of the nominal wall thickness. ~~The plugging limit imperfection depth for laser welded sleeves is equal to 38.7% of the nominal wall thickness.~~ The plugging limit imperfection depth for TIG welded sleeves is equal to 32% of the nominal wall thickness.

Retained as 5.5.9.c in INSERT 5.5.8. Editorial changes incorporated

For Unit 2 ~~during Refueling Outage 11 and the subsequent operating cycle, this definition does not apply to degradation identified in the portion of the tube below 17 inches from the top of the hot leg tubesheet.~~ Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.

Retained as 5.5.9.c.1 in INSERT 5.5.8. Editorial changes incorporated

For Unit 2 ~~during Refueling Outage 11 and the subsequent operating cycle,~~ degradation identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection;

Retained as 5.5.9.c in INSERT 5.5.8. Editorial changes incorporated

5.5 Programs and Manuals

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

~~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 OBE, LOCA, or a steam line or feedwater line break as specified in Specification 5.5.9.d.4;~~

~~8. Tube Inspection means an inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection shall include the sleeved portion of the tube.~~

Retained in 5.5.9.d in INSERT 5.5-8

For Unit 2 ~~during Refueling Outage 11 and the subsequent operating cycle~~, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded;

~~9. Preservice Inspection means an inspection of the full length of each tube in each SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial MODE 1 operation using the equipment and techniques expected to be used during subsequent inservice inspections;~~

~~10. Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following processes:~~

~~i. Laser welded sleeving as described in a Westinghouse Technical Report and subject to the limitations and restrictions as approved by the NRC, or~~

Retained in 5.5.9.f in INSERT 5.5-9

ii. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, " Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

Tube repair includes the removal of plugs that were previously installed as a corrective or preventative measure. A tube inspection per Specification 5.5.9.e.8 is required prior to returning previously plugged tubes to service;

11. The SG shall be determined OPERABLE after completing the corresponding actions (plug or repair in the affected area all tubes exceeding the plugging or repair limit) required by Table 5.5.9-2; and

12. For Unit 2 during Refueling Outage 11 and the subsequent operating cycle:

<u>Bulge</u> refers to a tube diameter deviation within the tubesheet of 18 volts or greater as measured by bobbin coil probe; and

<u>Overexpansion</u> refers to a tube diameter deviation within the tubesheet of 1.5 mils or greater as measured by bobbin coil probe.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

Table 5.5.9-1 (page 1 of 1)
Minimum Number of Steam Generators to be Inspected During Inservice Inspection

Preservice Inspection	Yes
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[a]

(a) The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the unit) 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. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described above.

Table 5.5.9-2 (page 1 of 1)
Steam Generator Tube Inspection

1st Sample Inspection[a]		2nd Sample Inspection		3rd Sample Inspection	
Result	Action Required	Result	Action Required	Result	Action Required
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 SG.	C-1	None	N/A	N/A
		C-2	Plug or repair defective tubes, and inspect additional 4S tubes in this SG.	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 SG, plug or repair defective tubes, and inspect 2S tubes in each other SG.	All other SGs C-1	None	N/A	N/A
		Any other SG C-2 but no other SG C-3	Perform action for C-2 result of second sample.	N/A	N/A
		Any other SG C-3	Inspect all tubes in each SG and plug or repair defective tubes.	N/A	N/A

(a) Sample size shall be a minimum of S tubes per SG:

$$S = 3 \frac{N}{n} \%$$

Where:
N = The number of SGs in the unit (4), and
n = the number of SGs inspected during an inspection.

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI, 1992 Edition with the 1992 Addenda.

5.6.9 Steam Generator (SG) Tube Inspection Report~~s~~

INSERT 5.6-6

a. Following each inservice inspection of SG tubes, the number of tubes plugged or repaired in each SG shall be reported to the NRC within 15 days.

b. The complete results of the SG tube inservice inspection shall be submitted to the NRC within 12 months following the completion of the inspection. The report shall include:

1. Number and extent of tubes inspected,
2. Location and percent of wall thickness penetration for each indication of an imperfection, and
3. Identification of tubes plugged or repaired.

c. Results of SG tube inspections that fall into Category C-3 shall be reported to the NRC within 30 days and prior to resumption of unit operation. The report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

Steam Generator (SG) Tube Inspection Report

INSERT 5.6-6

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, 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.

Attachment 2-B

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455

License Nos. NPF-37 and NPF-66

Application for Technical Specification Improvement
Regarding Steam Generator Tube Integrity

Markup of Technical Specifications Pages

ii
1.1-4
3.4.13-1
3.4.13-2
3.4.19-1 (new page)
3.4.19-2 (new page)
5.5-7
5.5-8
(Byron Unit 1) 5.5-9
(Byron Unit 2) 5.5-9
5.5-10
5.5-11
(Byron Unit 1) 5.5-12
(Byron Unit 2) 5.5-12
(Byron Unit 1) 5.5-13
(Byron Unit 2) 5.5-13
(Byron Unit 1) 5.5-14
(Byron Unit 2) 5.5-14
5.5-26
5.5-27
5.6-6

1.1 Definitions

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

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

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

3. Reactor Coolant System (RCS) LEAKAGE through a ~~S~~team ~~G~~enerator ~~(SG)~~ to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

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

c. Pressure Boundary LEAKAGE

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

MASTER RELAY TEST

A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

~~d. 600 gallons per day total primary to secondary LEAKAGE through all Steam Generators (SGs); and~~

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

S

--------------------NOTE--------------------
Not required to be performed until 12 hours after establishment of steady state operation.
----

2. Not applicable to primary to secondary LEAKAGE.

INSERT NEW SPECIFICATION 3.4.19

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 Steam Generator (SG) Tube Integrity

LCO 3.4.19 SG tube integrity shall be maintained.

<u>AND</u>

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.

ACTIONS

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

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

INSERT NEW SPECIFICATION 3.4.19

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.19.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.19.2	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

5.5 Programs and Manuals

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

INSERT 5.5 - 7

Each SG shall be demonstrated OPERABLE by performance of an augmented inservice inspection program.

a. SG Sample Selection and Inspection

Each SG shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of SGs specified in Table 5.5.9-1.

b. SG Tube Sample Selection and Inspection

-----------------------------NOTE-----------------------------
When referring to an SG tube, the sleeve shall be considered a part of the tube if the tube has been repaired per Specification 5.5.9.e.10.

The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.5.9-2. The inservice inspection of SG tubes shall be performed at the frequencies specified in Specification 5.5.9.d and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.5.9.e. When applying the expectations of Specification 5.5.9.b.1 through 5.5.9.b.3, previous defects or imperfections in the area repaired by the sleeve are not considered an area requiring reinspection. The tubes selected for each inservice inspection shall include ≥ 3% of the total number of tubes in all SGs. The tubes selected for these inspections shall be selected on a random basis except:

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then ≥ 50% of the tubes inspected shall be from these critical areas;

Steam Generator Program (Byron)

INSERT 5.5-7

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. 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.

5.5 Programs and Manuals

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

INSERT 5.5 - 8

~~2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each SG shall include:~~

~~i. All tubes that previously had detectable tube wall penetrations > 20% that have not been plugged or sleeved in the affected area, and all tubes that previously had detectable sleeve wall penetrations that have not been plugged,~~

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

~~iii. A tube inspection (pursuant to Specification 5.5.9.e.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;~~

~~3. The tubes selected as the second and third samples (if required by Table 5.5.9-2) during each inservice inspection may be subjected to a partial tube inspection provided:~~

~~i. 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~~

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

INSERT 5.5-8

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 not to exceed a total of 1 gpm for all SGs.

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

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired. For Unit 2 only, degradation identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection. TIG welded sleeves found by inservice inspection to contain flaws with a depth equal to or exceeding 32% of the nominal wall thickness shall be plugged.

The following alternate tube repair criteria may be applied as an alternative to the 40% depth-based criteria:

1. For Unit 2 only, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.

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. For Unit 2, the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to

5.5 Programs and Manuals

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

INSERT 5.5 - 9

4. A random sample of ≥ 20% of the total number of laser welded sleeves and ≥ 20% of the total number of Tungsten Inert Gas (TIG) welded sleeves installed shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection exceeding the repair limit is detected, an additional 20% of the unsampled sleeves shall be inspected and if an imperfection exceeding the repair limit is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve, the tube at the heat treated area, and the tube-to-sleeve joints. The inservice inspection for the sleeves is required on all types of sleeves installed in the SGs to demonstrate acceptable structural integrity.

c. Inspection Results Classification

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

---------------------------NOTE---------------------------
Previously degraded tubes or sleeves must exhibit significant (> 10% of wall thickness) further wall penetrations to be included in the percentage calculations.

Category	Inspection Results
C-1	< 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but ≤ 1% of the total tubes inspected are defective, or ≥ 5% and ≤ 10% of the total tubes inspected are degraded tubes.
C-3	> 10% of the total tubes inspected are degraded tubes or > 1% of the inspected tubes are defective.

5.5 Programs and Manuals

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

4. A random sample of ≥ 20% of the total number of laser welded sleeves and ≥ 20% of the total number of Tungsten Inert Gas (TIG) welded sleeves installed shall be inspected for axial and circumferential indications at the end of each cycle. In the event that an imperfection exceeding the repair limit is detected, an additional 20% of the unsampled sleeves shall be inspected and if an imperfection exceeding the repair limit is detected in the second sample, all remaining sleeves shall be inspected. These inservice inspections will include the entire sleeve, the tube at the heat treated area, and the tube-to-sleeve joints. The inservice inspection for the sleeves is required on all types of sleeves installed in the SGs to demonstrate acceptable structural integrity;

INSERT 5.5 - 9

5. For Unit 2 during Refueling Outage 12, a 20% minimum sample of all inservice tubes from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be inspected by rotating probe. This sample shall include a 20% minimum sample of the total population of bulges and overexpansions within the SG from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet.

c. Inspection Results Classification

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

-----------------------------NOTE-----------------------------
Previously degraded tubes or sleeves must exhibit significant (> 10% of wall thickness) further wall penetrations to be included in the percentage calculations.

Category	Inspection Results
C-1	< 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective.
C-2	One or more tubes, but ≤ 1% of the total tubes inspected are defective, or ≥ 5% and ≤ 10% of the total tubes inspected are degraded tubes.
C-3	> 10% of the total tubes inspected are degraded tubes or > 1% of the inspected tubes are defective.

INSERT 5.5-9

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 Unit 1 tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

   Inspect 100% of the Unit 2 tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for Unit 2 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

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

INSERT 5.5 - 10

d. Inspection Frequencies

The inservice inspections of SG tubes (dependent upon inspection results classification) shall be performed at the following frequencies:

1. The first inservice inspection shall be performed after 6 Effective Full Power months but ≤ 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals ≥ 12 calendar months and ≤ 24 calendar months after the previous inspection;

2. Extension Criteria: 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;

3. If the results of the inservice inspection of an SG conducted in accordance with Table 5.5.9-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 5.5.9.d.2; the interval may then be extended to a maximum of once per 40 months; and

INSERT 5.5-10

these Specifications, tube plugging is not a repair. All acceptable repair methods are listed below.

1. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, "Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

4. Additional unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.5.9-2 during the shutdown subsequent to any of the following conditions:

   i. Reactor to secondary tube leaks (not including leaks originating from tube to tube sheet welds) in excess of the limits of LCO 3.4.13.d and LCO 3.4.13.e, "RCS Operational LEAKAGE",

   ii. A seismic occurrence greater than the Operating Basis Earthquake (OBE),

   iii. A Condition IV Loss Of Coolant Accident (LOCA) requiring actuation of the Engineered Safety Features, or

   iv. A Condition IV main steam line or feedwater line break.

The provisions of SR 3.0.2 are not applicable to SG Tube Surveillance Program inspection frequencies.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

e. Acceptance Criteria

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 < 20% of the nominal tube or sleeve 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 unrepaired imperfections ≥ 20% of the nominal tube or sleeve wall thickness caused by degradation;

4. % 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 or sleeve containing an unrepaired defect is defective;

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. The plugging or repair limit imperfection depth for the tubing is equal to 40% of the nominal wall thickness. The plugging limit imperfection depth for laser welded sleeves is equal to 38.7% of the nominal wall thickness. The plugging limit imperfection depth for TIG welded sleeves is equal to 32% of the nominal wall thickness;

Retained as 5.5.9.c in INSERT 5.5.8. Editorial changes incorporated

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 OBE, LOCA, or a steam line or feedwater line break as specified in Specification 5.5.9.d.4;

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

e. Acceptance Criteria

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 < 20% of the nominal tube or sleeve 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 unrepaired imperfections ≥ 20% of the nominal tube or sleeve wall thickness caused by degradation;

4. % 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 or sleeve containing an unrepaired defect is defective;

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.~~ The plugging or repair limit imperfection depth for the tubing is equal to 40% of the nominal wall thickness. ~~The plugging limit imperfection depth for laser welded sleeves is equal to 38.7% of the nominal wall thickness.~~ The plugging limit imperfection depth for TIG welded sleeves is equal to 32% of the nominal wall thickness.

Retained as 5.5.9.c in INSERT 5.5.8. Editorial changes incorporated

For Unit 2 during ~~Refueling~~ Outage 12 and the subsequent ~~operating~~ cycle, this ~~definition~~ does not ~~apply to~~ degradation ~~identified~~ in the portion ~~of the~~ tube below ~~17 inches~~ from the top of the ~~hot leg tubesheet.~~ Degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.

Retained as 5.5.9.c.1 in INSERT 5.5.8. Editorial changes incorporated

For Unit 2 ~~during Refueling Outage 12 and the subsequent operating cycle,~~ degradation identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection;

Retained as 5.5.9.c in INSERT 5.5.8. Editorial changes incorporated

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

8. Tube Inspection means an inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection shall include the sleeved portion of the tube;

9. Preservice Inspection means an inspection of the full length of each tube in each SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial MODE 1 operation using the equipment and techniques expected to be used during subsequent inservice inspections;

10. Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following processes:

i. Laser welded sleeving as described in a Westinghouse Technical Report and subject to the limitations and restrictions as approved by the NRC, or

Retained in 5.5.9.f

ii. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.

Tube repair includes the removal of plugs that were previously installed as a corrective or preventative measure. A tube inspection per Specification 5.5.9.e.8 is required prior to returning previously plugged tubes to service; and

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

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 OBE, LOCA, or a steam line or feedwater line break as specified in Specification 5.5.9.d.4;

8. Tube Inspection means an inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg. For a tube that has been repaired by sleeving, the tube inspection shall include the sleeved portion of the tube.

Retained in 5.5.9.d in INSERT 5.5-8

For Unit 2 ~~during Refueling Outage 12 and the subsequent operating cycle~~ the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded;

9. Preservice Inspection means an inspection of the full length of each tube in each SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial MODE 1 operation using the equipment and techniques expected to be used during subsequent inservice inspections;

10. Tube Repair refers to a process that reestablishes tube serviceability. Acceptable tube repairs will be performed by the following processes:

   i. Laser welded sleeving as described in a Westinghouse Technical Report and subject to the limitations and restrictions as approved by the NRC, or

Retained in 5.5.9.f in INSERT 5.5-9

   ii. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, " Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

11. The SG shall be determined OPERABLE after completing the corresponding actions (plug or repair in the affected area all tubes exceeding the plugging or repair limit) required by Table 5.5.9-2.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Tube Surveillance Program (continued)

Tube repair includes the removal of plugs that were previously installed as a corrective or preventative measure. A tube inspection per Specification 5.5.9.e.8 is required prior to returning previously plugged tubes to service;

11. The SG shall be determined OPERABLE after completing the corresponding actions (plug or repair in the affected area all tubes exceeding the plugging or repair limit) required by Table 5.5.9-2; and

12. For Unit 2 during Refueling Outage 12 and the subsequent operating cycle:

    Bulge refers to a tube diameter deviation within the tubesheet of 18 volts or greater as measured by bobbin coil probe; and

    Overexpansion refers to a tube diameter deviation within the tubesheet of 1.5 mils or greater as measured by bobbin coil probe.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

Table 5.5.9-1 (page 1 of 1)
Minimum Number of Steam Generators to be
Inspected During Inservice Inspection

Preservice Inspection	Yes
No. of Steam Generators per Unit	Four
First Inservice Inspection	Two
Second & Subsequent Inservice Inspections	One[(a)]

(a) The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3 N % of the tubes (where N is the number of steam generators in the unit) 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. Each of the other two steam generators not inspected during the first inservice inspections shall be inspected during the second and third inspections. The fourth and subsequent inspections shall follow the instructions described above.

Table 5.5.9-2 (page 1 of 1)
Steam Generator Tube Inspection

1st Sample Inspection[a]		2nd Sample Inspection		3rd Sample Inspection	
Result	Action Required	Result	Action Required	Result	Action Required
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 SG.	C-1	None	N/A	N/A
		C-2	Plug or repair defective tubes, and inspect additional 4S tubes in this SG.	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 SG, plug or repair defective tubes, and inspect 2S tubes in each other SG.	All other SGs C-1	None	N/A	N/A
		Any other SG C-2 but no other SG C-3	Perform action for C-2 result of second sample.	N/A	N/A
		Any other SG C-3	Inspect all tubes in each SG and plug or repair defective tubes.	N/A	N/A

(a) Sample size shall be a minimum of S tubes per SG:

$$S = 3 \frac{N}{n} \%$$

Where:
N = The number of SGs in the unit (4), and
n = the number of SGs inspected during an inspection.

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI, 1992 Edition with the 1992 Addenda.

5.6.9 Steam Generator (SG) Tube Inspection Report~~s~~

INSERT 5.6-6

a. ~~Following each inservice inspection of SG tubes, the number of tubes plugged or repaired in each SG shall be reported to the NRC within 15 days.~~

b. ~~The complete results of the SG tube inservice inspection shall be submitted to the NRC within 12 months following the completion of the inspection. The report shall include:~~

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

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

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

c. ~~Results of SG tube inspections that fall into Category C-3 shall be reported to the NRC within 30 days and prior to resumption of unit operation. The report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.~~

Steam Generator (SG) Tube Inspection Report

INSERT 5.6-6

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, 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.

Attachment 3-A

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

Application for Technical Specification Improvement
Regarding Steam Generator Tube Integrity

Typed Technical Specifications Pages

ii
1.1-4
3.4.13-1
3.4.13-2
3.4.19-1
3.4.19-2
5.5-7
5.5-8
5.5-9
5.5-10
5.5-11
5.5-12
5.5-13
5.5-14
5.5-15
5.5-16
5.5-17
5.5-18
5.5-19
5.5-20
5.5-21
5.6-6

1.1 Definitions

LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except Reactor Coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE); b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 Steam Generator (SG) Tube Integrity

LCO 3.4.19 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.

ACTIONS

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

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.19.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.19.2	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

5.5 Programs and Manuals

5.5.9 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. 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.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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 not to exceed a total of 1 gpm for all SGs.

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

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired. For Unit 2 only, degradation identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection. TIG welded sleeves found by inservice inspection to contain flaws with a depth equal to or exceeding 32% of the nominal wall thickness shall be plugged.

The following alternate tube repair criteria may be applied as an alternative to the 40% depth-based criteria:

1. For Unit 2 only, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.

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. For Unit 2 the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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 Unit 1 tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

   Inspect 100% of the Unit 2 tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for Unit 2 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

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

these Specifications, tube plugging is not a repair. All acceptable repair methods are listed below.

1. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, "Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR.

5.5 Programs and Manuals

5.5.11 <u>Ventilation Filter Testing Program (VFTP)</u> (continued)

a. Demonstrate for each of the ESF filter systems that an inplace test of the High Efficiency Particulate Air (HEPA) filters shows a penetration specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Penetration
Control Room Ventilation (VC) Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm	< 0.05%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (after structural maintenance of the HEPA filter housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train, and ≥ 20,070 cfm and ≤ 24,530 cfm per bank	< 1%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (for reasons other than structural maintenance of the HEPA filter housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train	< 1%
Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm	< 1%

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF filter systems that an inplace test of the charcoal adsorber shows a bypass specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Bypass
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm	< 0.05%
VC Filtration System (recirculation, charcoal bed after complete or partial replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 0.1%
VC Filtration System (recirculation for reasons other than complete or partial charcoal bed replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 2%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (after structural maintenance of the charcoal adsorber housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train, and ≥ 20,070 cfm and ≤ 24,530 cfm per bank	< 1%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (for reasons other than structural maintenance of the charcoal adsorber housings)	≥ 60,210 cfm and ≤ 73,590 cfm per train	< 1%
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm per train	< 1%

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

c. Demonstrate for each of the ESF filter systems that a laboratory test of a sample of the charcoal adsorber, when obtained as described in Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, ANSI N510-1980, and ASTM D3803-1989, with any exceptions noted in Appendix A of the UFSAR, at a temperature of 30°C and a Relative Humidity (RH) specified below:

ESF Ventilation System	Penetration	RH
VC Filtration System (makeup)	0.5%	70%
VC Filtration System (recirculation)	4%	70%
Nonaccessible Area Exhaust Filter Plenum Ventilation System	4.5%	70%
FHB Ventilation System	10%	95%

d. Demonstrate for each of the ESF filter systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is < 6 inches of water gauge when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm
Nonaccessible Area Exhaust Filter Plenum Ventilation System	≥ 60,210 cfm and ≤ 73,590 cfm per train
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

e. Demonstrate for each of the ESF filter systems that a bypass test of the combined HEPA filters and damper leakage shows a total bypass specified below at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.12.4 and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Bypass
Nonaccessible Area Exhaust Filter Plenum Ventilation System	≥ 60,210 cfm and ≤ 73,590 cfm per train	≤ 1%
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm	≤ 1%

f. Demonstrate that the heaters for each of the ESF filter systems dissipate the value specified below when tested in conformance with ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR.

ESF Ventilation System	Wattage
VC Filtration System	≤ 29.9 kW and ≥ 24.5 kW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas system, the quantity of radioactivity contained in gas decay tanks or fed into the off gas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with the ODCM.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the waste gas system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas decay tank and fed into the offgas treatment system is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits, and

   3. a clear and bright appearance with proper color or a water and sediment content within limits;

b. Other properties of new fuel oil are within limits within 30 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5 Programs and Manuals

5.5.14 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 UFSAR 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 UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14.b 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) as modified by approved exemptions.

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

b. Air lock testing acceptance criteria are:

1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$; and

2. For each door, seal leakage rate is:

i. $< 0.0024\ L_a$, when pressurized to ≥ 3 psig, and

ii. $< 0.01\ L_a$, when pressurized to ≥ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.17 Battery Monitoring and Maintenance Program

This program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead-Acid Batteries For Stationary Applications," or of the battery manufacturer of the following:

a. Actions to restore battery cells with float voltage < 2.13 V, and

b. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI, 1992 Edition with the 1992 Addenda.

5.6.9 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, 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.

Attachment 3-B

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455

License Nos. NPF-37 and NPF-66

Application for Technical Specification Improvement
Regarding Steam Generator Tube Integrity

Typed Technical Specifications Pages

ii
1.1-4
3.4.13-1
3.4.13-2
3.4.19-1
3.4.19-2
5.5-7
5.5-8
5.5-9
5.5-10
5.5-11
5.5-12
5.5-13
5.5-14
5.5-15
5.5-16
5.5-17
5.5-18
5.5-19
5.5-20
5.5-21
5.6-6

1.1 Definitions

LEAKAGE	LEAKAGE shall be: a. <u>Identified LEAKAGE</u> 1. LEAKAGE, such as that from pump seals or valve packing (except Reactor Coolant pump (RCP) seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or 3. Reactor Coolant System (RCS) LEAKAGE through a Steam Generator to the Secondary System (primary to secondary LEAKAGE); b. <u>Unidentified LEAKAGE</u> All LEAKAGE (except RCP seal water injection or leakoff) that is not identified LEAKAGE; c. <u>Pressure Boundary LEAKAGE</u> LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.
MASTER RELAY TEST	A MASTER RELAY TEST shall consist of energizing each master relay and verifying the OPERABILITY of each relay. The MASTER RELAY TEST shall include a continuity check of each associated slave relay.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

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

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.19 Steam Generator (SG) Tube Integrity

LCO 3.4.19 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.

ACTIONS

------------------------------NOTE------------------------------

Separate Condition entry is allowed for each SG tube.

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.19.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.19.2	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

5.5 Programs and Manuals

5.5.9 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. 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.

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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 not to exceed a total of 1 gpm for all SGs.

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

c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged or repaired. For Unit 2 only, degradation identified in the portion of the tube from the top of the hot leg tubesheet to 17 inches below the top of the tubesheet shall be plugged or repaired upon detection. TIG welded sleeves found by inservice inspection to contain flaws with a depth equal to or exceeding 32% of the nominal wall thickness shall be plugged.

The following alternate tube repair criteria may be applied as an alternative to the 40% depth-based criteria:

1. For Unit 2 only, degradation found in the portion of the tube below 17 inches from the top of the hot leg tubesheet does not require plugging or repair.

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. For Unit 2 the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

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 Unit 1 tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.

   Inspect 100% of the Unit 2 tubes at sequential periods of 120, 90, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 48 effective full power months or two refueling outages (whichever is less) without being inspected.

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

e. Provisions for monitoring operational primary to secondary LEAKAGE.

f. Provisions for Unit 2 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

5.5 Programs and Manuals

5.5.9 Steam Generator (SG) Program (continued)

these Specifications, tube plugging is not a repair. All acceptable repair methods are listed below.

1. TIG welded sleeving as described in ABB Combustion Engineering Inc., Technical Reports: Licensing Report CEN-621-P, Revision 00, "Commonwealth Edison Byron and Braidwood Unit 1 and 2 Steam Generators Tube Repair Using Leak Tight Sleeves, FINAL REPORT," April 1995; and Licensing Report CEN-627-P, "Operating Performance of the ABB CENO Steam Generator Tube Sleeve for Use at Commonwealth Edison Byron and Braidwood Units 1 and 2," January 1996; subject to the limitations and restrictions as noted by the NRC Staff.

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

a. Identification of a sampling schedule for the critical variables and control points for these variables;

b. Identification of the procedures used to measure the values of the critical variables;

c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser inleakage;

d. Procedures for the recording and management of data;

e. Procedures defining corrective actions for all off control point chemistry conditions; and

f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR.

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

a. Demonstrate for each of the ESF filter systems that an inplace test of the High Efficiency Particulate Air (HEPA) filters shows a penetration specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Penetration
Control Room Ventilation (VC) Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm	< 0.05%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (after structural maintenance of the HEPA filter housings)	≥ 55,669 cfm and ≤ 68,200 cfm per train, and ≥ 18,556 cfm and ≤ 22,733 cfm per bank	< 1%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (for reasons other than structural maintenance of the HEPA filter housings)	≥ 55,669 cfm and ≤ 68,200 cfm per train	< 1%
Fuel Handling Building Exhaust Filter Plenum (FHB) Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm	< 1%

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

b. Demonstrate for each of the ESF filter systems that an inplace test of the charcoal adsorber shows a bypass specified below when tested in conformance with Regulatory Guide 1.52, Revision 2, and ANSI N510-1980, with any exceptions noted in Appendix A of the UFSAR, at the system flow rate specified below. Verification of the specified flow rates may be accomplished during the performance of SRs 3.7.10.4, 3.7.12.4, and 3.7.13.5, as applicable:

ESF Ventilation System	Flow Rate	Bypass
VC Filtration System (makeup)	≥ 5400 cfm and ≤ 6600 cfm	< 0.05%
VC Filtration System (recirculation, charcoal bed after complete or partial replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 0.1%
VC Filtration System (recirculation for reasons other than complete or partial charcoal bed replacement)	≥ 44,550 cfm and ≤ 54,450 cfm	< 2%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (after structural maintenance of the charcoal adsorber housings)	≥ 55,669 cfm and ≤ 68,200 cfm per train, and ≥ 18,556 cfm and ≤ 22,733 cfm per bank	< 1%
Nonaccessible Area Exhaust Filter Plenum Ventilation System (for reasons other than structural maintenance of the charcoal adsorber housings)	≥ 55,669 cfm and ≤ 68,200 cfm per train	< 1%
FHB Ventilation System	≥ 18,900 cfm and ≤ 23,100 cfm per train	< 1%

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the waste gas system, the quantity of radioactivity contained in gas decay tanks or fed into the off gas treatment system, and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure." The liquid radwaste quantities shall be determined in accordance with the ODCM.

The program shall include:

a. The limits for concentrations of hydrogen and oxygen in the waste gas system and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);

b. A surveillance program to ensure that the quantity of radioactivity contained in each gas decay tank and fed into the offgas treatment system is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and

c. A surveillance program to ensure that the quantity of radioactivity contained in all outdoor liquid radwaste tanks that are not surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system is less than the amount that would result in concentrations less than the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area, in the event of an uncontrolled release of the tanks' contents.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5 Programs and Manuals

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

    1. an API gravity or an absolute specific gravity within limits,

    2. a flash point and kinematic viscosity within limits, and

    3. a clear and bright appearance with proper color or a water and sediment content within limits;

b. Other properties of new fuel oil are within limits within 30 days following sampling and addition to storage tanks; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 31 days.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5 Programs and Manuals

5.5.14 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 UFSAR 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 UFSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14.b 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) as modified by approved exemptions.

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, September 1995 and NEI 94-01, Revision 0.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 42.8 psig for Unit 1 and 38.4 psig for Unit 2

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.10% of containment air weight per day.

Leakage Rate acceptance criteria are:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests and $< 0.75\ L_a$ for Type A tests; and

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

b. Air lock testing acceptance criteria are:

1. Overall air lock leakage rate is ≤ 0.05 L_a when tested at ≥ P_a; and

2. For each door, seal leakage rate is:

i. < 0.0024 L_a, when pressurized to ≥ 3 psig, and

ii. < 0.01 L_a, when pressurized to ≥ 10 psig.

The provisions of SR 3.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.

5.5.17 Battery Monitoring and Maintenance Program

This program provides for restoration and maintenance, based on the recommendations of IEEE Standard 450, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Vented Lead - Acid Batteries For Stationary Applications," or of the battery manufacturer of the following:

A. Actions to restore battery cells with float voltage < 2.13 V, and

B. Actions to equalize and test battery cells that had been discovered with electrolyte level below the minimum established design limit.

5.6 Reporting Requirements

5.6.8 Tendon Surveillance Report

Any abnormal degradation of the containment structure detected during the tests required by the Pre-Stressed Concrete Containment Tendon Surveillance Program shall be reported in the Inservice Inspection Summary Report in accordance with 10 CFR 50.55a and ASME Section XI, 1992 Edition with the 1992 Addenda.

5.6.9 Steam Generator (SG) Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 5.5.9, 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.

Attachment 4-A

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457

License Nos. NPF-72 and NPF-77

Application for Technical Specification Improvement
Regarding Steam Generator Tube Integrity

Typed Technical Specifications Bases Pages
(Information Only)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

RCS Loops-MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

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

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG.

BASES

LCO (continued)

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops-MODES 1 and 2";
LCO 3.4.6, "RCS Loops-MODE 4";
LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level" (MODE 6).

BASES

BACKGROUND (continued)

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). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSIS

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analyses for the Main Steamline Break and the Locked Rotor with Failed Open PORV, base the radioactive discharge to the atmosphere on primary to secondary LEAKAGE from the faulted SG of 0.5 gallon per minute and primary to secondary LEAKAGE from the intact SGs of 0.218 gallon per minute per intact SG. For the Locked Rotor and Rod Cluster Control Assembly Ejection, the radioactive discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a Locked Rotor with a Concurrent Steam Generator (SG) Power Operated Relief Valve (PORV) Failure accident because such leakage contaminates the secondary fluid. Other accidents or transients involve secondary steam release to the atmosphere, such as a Steam Generator Tube Rupture (SGTR). The SGTR is more limiting than the Locked Rotor with a Concurrent SG PORV Failure for site radiation releases.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released for a limited time via the SG PORV. After a tube rupture occurs, reactor coolant immediately begins flowing from the primary system into the secondary side of the ruptured SG causing the RCS pressure to decrease until a reactor trip occurs on low pressurizer pressure. The analysis assumes a Loss of Offsite Power occurs coincident with the reactor trip

BASES

APPLICABLE SAFETY ANALYSES (continued)

causing the Reactor Coolant Pumps to trip and the main condenser to become unavailable when the circulating water pumps are lost.

After the reactor trips, the core power quickly decreases to decay heat levels. The steam dump system cannot be used to dissipate the core decay heat due to the unavailable condenser. Therefore, the secondary pressure increases in the SGs until the SG PORVs open at which time the ruptured SG PORV is assumed to fail in the open position. The ruptured SG failed PORV is isolated when the block valve is manually closed twenty minutes after the PORV first opened. The 1 gpm primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential to the results of this analysis.

The dose consequences resulting from the Locked Rotor with a Concurrent SG PORV Failure accident are well within the limits defined in 10 CFR 100.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

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, valve seats, and gaskets is not pressure boundary LEAKAGE.

BASES

LCO (continued)

b. Unidentified LEAKAGE

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

c. Identified LEAKAGE

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

BASES

LCO (continued)

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). 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.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual Pressure Isolation Valve (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 as identified LEAKAGE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greater due to RCS pressure.

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

ACTIONS

A.1

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

BASES

ACTIONS (continued)

B.1 and B.2

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limit, or if unidentified or 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 unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

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

The RCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. The Surveillance is modified by two Notes. Note 1 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 since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure (≥ 2150 psig), 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. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

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 3.4.19, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. 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 guidelines (Ref. 5).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, Chapter 15.
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.19 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

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

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

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

Specification 5.5.9, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.9. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSIS

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 total initial primary to secondary LEAKAGE of 1.0 gpm for the intact SGs plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via the SG Power Operated Relief Valves (PORVs).

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.) For the Main Steamline Break and the Locked Rotor with Failed Open PORV, the radioactive discharge to the atmosphere is based on primary to secondary LEAKAGE from the faulted SG of 0.5 gallon per minute and primary to secondary LEAKAGE from the intact SGs of 0.218 gallon per minute per intact SG. For the Locked Rotor and Rod Cluster Control Assembly Ejection, the radioactive discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

BASES

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 have 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. For Unit 2 the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, "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 (i.e., primary to secondary 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

BASES

LCO (continued)

significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

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

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident induced leakage requirement of 1 gpm for all SGs, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage, bounds the accident analysis assumptions for primary to secondary LEAKAGE. 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 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

BASES

APPLICABILITY

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

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

ACTIONS

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

A.1 and A.2

Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged or repaired in accordance with the Steam Generator Program as required by SR 3.4.19.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged 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, whichever occurs first. 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, Condition B applies.

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

BASES

ACTIONS (continued)

If the evaluation determines that the affected tube(s) have tube integrity, Required Action A.2 allows plant operation to continue until the next refueling outage or SG inspection, whichever occurs first, 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, whichever occurs first. This Completion Time is acceptable since operation until the next inspection is supported by the operational assessment.

B.1 and B.2

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.19.1

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

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.19.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.19.2

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 5.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

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 the Surveillance 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.

BASES

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 100.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

Attachment 4-B

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455

License Nos. NPF-37 and NPF-66

Application for Technical Specification Improvement
Regarding Steam Generator Tube Integrity

Typed Technical Specifications Bases Pages
(Information Only)

ii
B 3.4.4-3
B 3.4.5-4
B 3.4.13-2
B 3.4.13-3
B 3.4.13-4
B 3.4.13-5
B 3.4.13-6
B 3.4.13-7
B 3.4.13-8
B 3.4.19-1
B 3.4.19-2
B 3.4.19-3
B 3.4.19-4
B 3.4.19-5
B 3.4.19-6
B 3.4.19-7
B 3.4.19-8

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

RCS Loops-MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

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

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG.

BASES

LCO (continued)

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG, which has the minimum water level specified in SR 3.4.5.2. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with the Rod Control System capable of rod withdrawal. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the Rod Control System not capable of rod withdrawal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops-MODES 1 and 2";
LCO 3.4.6, "RCS Loops-MODE 4";
LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level" (MODE 6).

BASES

BACKGROUND (continued)

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). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSIS

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analyses for the Main Steamline Break and the Locked Rotor with Failed Open PORV, base the radioactive discharge to the atmosphere on primary to secondary LEAKAGE from the faulted SG of 0.5 gallon per minute and primary to secondary LEAKAGE from the intact SGs of 0.218 gallon per minute per intact SG. For the Locked Rotor and Rod Cluster Control Assembly Ejection, the radioactive discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute. The LCO requirement to limit primary to secondary LEAKAGE through any one SG to less than or equal to 150 gallons per day is significantly less than the conditions assumed in the safety analysis.

Primary to secondary LEAKAGE is a factor in the dose releases outside containment resulting from a Locked Rotor with a Concurrent Steam Generator (SG) Power Operated Relief Valve (PORV) Failure accident because such leakage contaminates the secondary fluid. Other accidents or transients involve secondary steam release to the atmosphere, such as a Steam Generator Tube Rupture (SGTR). The SGTR is more limiting than the Locked Rotor with a Concurrent SG PORV Failure for site radiation releases.

The UFSAR (Ref. 3) analysis for SGTR assumes the contaminated secondary fluid is released for a limited time via the SG PORV. After a tube rupture occurs, reactor coolant immediately begins flowing from the primary system into the secondary side of the ruptured SG causing the RCS pressure to decrease until a reactor trip occurs on low pressurizer pressure. The analysis assumes a Loss of Offsite Power occurs coincident with the reactor trip

BASES

APPLICABLE SAFETY ANALYSES (continued)

causing the Reactor Coolant Pumps to trip and the main condenser to become unavailable when the circulating water pumps are lost.

After the reactor trips, the core power quickly decreases to decay heat levels. The steam dump system cannot be used to dissipate the core decay heat due to the unavailable condenser. Therefore, the secondary pressure increases in the SGs until the SG PORVs open at which time the ruptured SG PORV is assumed to fail in the open position. The ruptured SG failed PORV is isolated when the block valve is manually closed twenty minutes after the PORV first opened. The 1 gpm primary to secondary LEAKAGE safety analysis assumption is relatively inconsequential to the results of this analysis.

The dose consequences resulting from the Locked Rotor with a Concurrent SG PORV Failure accident are well within the limits defined in 10 CFR 100.

The RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. <u>Pressure Boundary LEAKAGE</u>

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, valve seats, and gaskets is not pressure boundary LEAKAGE.

BASES

LCO (continued)

b. Unidentified LEAKAGE

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

c. Identified LEAKAGE

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

BASES

LCO (continued)

d. Primary to Secondary LEAKAGE through Any One SG

The limit of 150 gallons per day per SG is based on the operational LEAKAGE performance criterion in NEI 97-06, Steam Generator Program Guidelines (Ref. 4). 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.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through each individual Pressure Isolation Valve (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 as identified LEAKAGE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greater due to RCS pressure.

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

ACTIONS

A.1

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

BASES

ACTIONS (continued)

B.1 and B.2

If any pressure boundary LEAKAGE exists, or primary to secondary LEAKAGE is not within limit, or if unidentified or 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 unit must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit 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.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

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

The RCS water inventory balance must be performed with the reactor at steady state operating conditions and near operating pressure. The Surveillance is modified by two Notes. Note 1 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 since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure (≥ 2150 psig), 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. It should be noted that LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.13.2

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 3.4.19, "Steam Generator Tube Integrity," should be evaluated. The 150 gallons per day limit is measured at room temperature as described in Reference 5. 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 guidelines (Ref. 5).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, Chapter 15.
4. NEI 97-06, "Steam Generator Program Guidelines."
5. EPRI, "Pressurized Water Reactor Primary-to-Secondary Leak Guidelines."

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.19 Steam Generator (SG) Tube Integrity

BASES

BACKGROUND

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

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

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

Specification 5.5.9, "Steam Generator (SG) Program," requires that a program be established and implemented to ensure that SG tube integrity is maintained. Pursuant to Specification 5.5.9, tube integrity is maintained when the SG performance criteria are met. There are three SG performance criteria: structural integrity, accident induced leakage, and operational LEAKAGE. The SG performance criteria are described in Specification 5.5.9. Meeting the SG performance criteria provides reasonable assurance of maintaining tube integrity at normal and accident conditions.

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSIS

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 total initial primary to secondary LEAKAGE of 1.0 gpm for the intact SGs plus the leakage rate associated with a double-ended rupture of a single tube. The accident analysis for a SGTR assumes the contaminated secondary fluid is released to the atmosphere via the SG Power Operated Relief Valves (PORVs).

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.) For the Main Steamline Break and the Locked Rotor with Failed Open PORV, the radioactive discharge to the atmosphere is based on primary to secondary LEAKAGE from the faulted SG of 0.5 gallon per minute and primary to secondary LEAKAGE from the intact SGs of 0.218 gallon per minute per intact SG. For the Locked Rotor and Rod Cluster Control Assembly Ejection, the radioactive discharge to the atmosphere is based on the total primary to secondary LEAKAGE from all SGs of 1 gallon per minute. For accidents that do not involve fuel damage, the primary coolant activity level of DOSE EQUIVALENT I-131 is assumed to be equal to the LCO 3.4.16, "RCS Specific Activity," limits. For accidents that assume fuel damage, the primary coolant activity is a function of the amount of activity released from the damaged fuel. The dose consequences of these events are within the limits of GDC 19 (Ref. 2), 10 CFR 100 (Ref. 3) or the NRC approved licensing basis (e.g., a small fraction of these limits).

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

BASES

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 have 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. For Unit 2 the portion of the tube below 17 inches from the top of the hot leg tubesheet is excluded. The tube-to-tubesheet weld is not considered part of the tube.

A SG tube has tube integrity when it satisfies the SG performance criteria. The SG performance criteria are defined in Specification 5.5.9, "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 (i.e., primary to secondary 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

BASES

LCO (continued)

significant when the addition of such loads in the assessment of the structural integrity performance criterion could cause a lower structural limit or limiting burst/collapse condition to be established." For tube integrity evaluations, except for circumferential degradation, axial thermal loads are classified as secondary loads. For circumferential degradation, the classification of axial thermal loads as primary or secondary loads will be evaluated on a case-by-case basis. The division between primary and secondary classifications will be based on detailed analysis and/or testing.

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

The accident induced leakage performance criterion ensures that the primary to secondary LEAKAGE caused by a design basis accident, other than a SGTR, is within the accident analysis assumptions. The accident induced leakage requirement of 1 gpm for all SGs, except for specific types of degradation at specific locations where the NRC has approved greater accident induced leakage, bounds the accident analysis assumptions for primary to secondary LEAKAGE. 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 3.4.13, "RCS Operational LEAKAGE," and limits primary to secondary LEAKAGE through any one SG to 150 gallons per day. This limit is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line break. If this amount of LEAKAGE is due to more than one crack, the cracks are very small, and the above assumption is conservative.

BASES

APPLICABILITY	Steam generator tube integrity is challenged when the pressure differential across the tubes is large. Large differential pressures across SG tubes can only be experienced in MODE 1, 2, 3, or 4. RCS conditions are far less challenging in MODES 5 and 6 than during MODES 1, 2, 3, and 4. In MODES 5 and 6, primary to secondary differential pressure is low, resulting in lower stresses and reduced potential for LEAKAGE.
ACTIONS	The ACTIONS are modified by a Note clarifying that the Conditions may be entered independently for each SG tube. This is acceptable because the Required Actions provide appropriate compensatory actions for each affected SG tube. Complying with the Required Actions may allow for continued operation, and subsequent affected SG tubes are governed by subsequent Condition entry and application of associated Required Actions. <u>A.1 and A.2</u> Condition A applies if it is discovered that one or more SG tubes examined in an inservice inspection satisfy the tube repair criteria but were not plugged or repaired in accordance with the Steam Generator Program as required by SR 3.4.19.2. An evaluation of SG tube integrity of the affected tube(s) must be made. Steam generator tube integrity is based on meeting the SG performance criteria described in the Steam Generator Program. The SG repair criteria define limits on SG tube degradation that allow for flaw growth between inspections while still providing assurance that the SG performance criteria will continue to be met. In order to determine if a SG tube that should have been plugged 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, whichever occurs first. 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, Condition B applies. A Completion Time of 7 days is sufficient to complete the evaluation while minimizing the risk of plant operation with a SG tube that may not have tube integrity.

BASES

ACTIONS (continued)

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

B.1 and B.2

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.19.1

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

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

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

Program also specifies the inspection methods to be used to find potential degradation. Inspection methods are a function of degradation morphology, non-destructive examination (NDE) technique capabilities, and inspection locations.

The Steam Generator Program defines the Frequency of SR 3.4.19.1. The Frequency is determined by the operational assessment and other limits in the SG examination guidelines (Ref. 6). The Steam Generator Program uses information on existing degradations and growth rates to determine an inspection Frequency that provides reasonable assurance that the tubing will meet the SG performance criteria at the next scheduled inspection. In addition, Specification 5.5.9 contains prescriptive requirements concerning inspection intervals to provide added assurance that the SG performance criteria will be met between scheduled inspections.

SR 3.4.19.2

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 5.5.9 are intended to ensure that tubes accepted for continued service satisfy the SG performance criteria with allowance for error in the flaw size measurement and for future flaw growth. In addition, the tube repair criteria, in conjunction with other elements of the Steam Generator Program, ensure that the SG performance criteria will continue to be met until the next inspection of the subject tube(s). Reference 1 provides guidance for performing operational assessments to verify that the tubes remaining in service will continue to meet the SG performance criteria.

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 the Surveillance 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.

BASES

REFERENCES

1. NEI 97-06, "Steam Generator Program Guidelines."
2. 10 CFR 50 Appendix A, GDC 19.
3. 10 CFR 100.
4. ASME Boiler and Pressure Vessel Code, Section III, Subsection NB.
5. Draft Regulatory Guide 1.121, "Basis for Plugging Degraded Steam Generator Tubes," August 1976.
6. EPRI, "Pressurized Water Reactor Steam Generator Examination Guidelines."

Attachment 5

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457
License Nos. NPF-72 and NPF-77

and

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455
License Nos. NPF-37 and NPF-66

Application for Technical Specification Improvement
Regarding Steam Generator Tube Integrity

Application for Withholding and Affidavit

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Our ref: CAW-05-2047

September 2, 2005

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-CDME-05-32-P, Rev. 2, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2," dated August 2005 (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-05-2047 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Exelon Generation Company, LLC.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-05-2047, and should be addressed to B. F. Maurer, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: B. Benney, NRC
L. Feizollahi, NRC

bcc: B. F. Maurer (ECE 4-7) 1L
R. Bastien, 1L (Nivelles, Belgium)
C. Brinkman, 1L (Westinghouse Electric Co., 12300 Twinbrook Parkway, Suite 330, Rockville, MD 20852)
RCPL Administrative Aide (ECE 4-7A) 1L, 1A (letter and affidavit only)
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J. M. Bunecicky, ECE 560E
D. W. Alexander, ECE 561B

CAW-05-2047

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared B. F. Maurer, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 2nd day
of September, 2005

Sharon L. Fiori
Notary Public

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Acting Manager, Regulatory Compliance and Plant Licensing, Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse "Application for Withholding" accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in LTR-CDME-05-32-P, Rev. 2, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2," dated August 2005 (Proprietary). The information is provided in support of a submittal to the Commission, being transmitted by Exelon Generation Company, LLC and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Byron 2 and Braidwood 2 is expected to be applicable to other licensee submittals in support of implementing a limited inspection of the tube joint with a rotating probe within the tubesheet region of the steam generators.

This information is part of that which will enable Westinghouse to:

(a) Provide documentation of the analyses, methods, and testing for the implementation of the limited inspection length of the steam generator tube joint.

(b) Provide a primary-to-secondary side leakage evaluation for Byron 2 and Braidwood 2 during all plant conditions.

(c) Assist the customer to respond to NRC requests for information.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of meeting NRC requirements for licensing documentation.

(b) Westinghouse can sell support and defense of this information to its customers in the licensing process.

(c) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar licensing support documentation and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Attachment 6

BRAIDWOOD STATION
UNITS 1 AND 2

Docket Nos. STN 50-456 and STN 50-457
License Nos. NPF-72 and NPF-77

and

BYRON STATION
UNITS 1 AND 2

Docket Nos. STN 50-454 and STN 50-455
License Nos. NPF-37 and NPF-66

Application for Technical Specification Improvement
Regarding Steam Generator Tube Integrity

Non-proprietary Version of Westinghouse LTR-CDME-05-32, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Byron 2 and Braidwood 2," Revision 2, dated August 2005