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10 CFR 50.90

RS-01-011

February 9, 2001

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

Subject: Request for Technical Specifications Change
Braidwood Station, Unit 1, Steam Generator Inspection Frequency Revision for the Fall 2001 Refueling Outage

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," we are proposing a change to the Technical Specifications (TS) of Facility Operating License Nos. NPF-72 and NPF-77 for the Braidwood Station, Units 1 and 2. Although this TS change is only applicable to Braidwood Station, Unit 1, we are submitting the TS change on the Braidwood Station, Units 1 and 2, dockets since the affected TS page is common to both Units. The proposed one-time change revises the Steam Generator (SG) inspection frequency requirements in TS 5.5.9.d.2, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," for the Braidwood Station, Unit 1, fall 2001 refueling outage, to allow a 40 month inspection interval after one SG inspection, rather than after two consecutive inspections resulting in C-1 classification. This one-time change is proposed to eliminate unnecessary SG inspections during the upcoming Unit 1 fall 2001 refueling outage, thus resulting in significant dose, schedule, and cost savings as detailed in the attachment.

Braidwood Station, Unit 1, replaced SGs during a refueling outage that was completed in November 1998. The replacement SGs are Babcock and Wilcox International (BWI) design and incorporate significant improvements, including thermally treated Inconel-690 tubing. During Unit 1 spring 2000 refueling outage, following the first cycle of operation after SG replacement, 100% of the tubing was inspected full-length (i.e., from hot leg tube end to

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cold leg tube end, including the U-bends) with eddy current and showed no defective or degraded tubes. The Unit 1 spring 2000 refueling outage inspection results along with the improved BWI replacement SG design and industry experience regarding BWI SG inspection results, provides the basis for proposing an extension of the inspection interval for the fall 2001 refueling outage from a maximum of 24 calendar months to a maximum of 40 months with one category C-1 inspection result.

We request approval of the proposed change prior to August 9, 2001. This would support postponing SG inspections currently planned for the Unit 1, fall 2001 refueling outage. SG inspections will be performed during the Unit 1 spring 2003 refueling outage in accordance with the requirements in TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program, " and the Electric Power Research Institute Pressurized Water Reactor Steam Generator Examination Guidelines. The Unit 1, fall 2001 refueling outage is currently scheduled to begin September 22, 2001, with SG inspections scheduled to begin on September 25, 2001.

This request is subdivided as follows.

1. Attachment A gives a description and safety analysis of the proposed change.

2. Attachment B-1 includes the marked-up TS page for the proposed change for the Braidwood Station, Unit 1. Attachment B-2 includes the associated TS page with the proposed change incorporated for the Braidwood Station, Unit 1.

3. Attachment C describes our evaluation performed using the criteria in 10 CFR 50.91(a)(1), "Notice for public comment," which provides information supporting a finding of no significant hazards consideration using the standards in 10 CFR 50.92(c), "Issuance of amendment."

4. Attachment D provides information supporting an environmental assessment and a finding that the proposed change satisfies the criteria for a categorical exclusion.

The proposed change has been reviewed by the Braidwood Station Plant Operations Review Committee and the Nuclear Safety Review Board in accordance with the Quality Assurance Program.

Exelon Generation Company, LLC is notifying the State of Illinois of this application for a change to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

Should you have any questions concerning this letter, please contact Ms. Kelly M. Root at (630) 663-7292.

Respectfully,

R. M. Krich
Director - Licensing
Mid-West Regional Operating Group

Affidavit

Attachments:
Attachment A: Description and Safety Analysis of the Proposed Changes
Attachment B-1: Marked-Up TS Page for Proposed Changes for Braidwood Station, Unit 1
Attachment B-2: Incorporated TS Page for Proposed Changes for Braidwood Station, Unit 1
Attachment C: Information Supporting a Finding of No Significant Hazards Consideration
Attachment D: Information Supporting an Environmental Assessment

cc: Regional Administrator - NRC Region III
NRC Senior Resident Inspector - Braidwood Station
Office of Nuclear Facility Safety - Illinois Department of Nuclear Safety

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| **STATE OF ILLINOIS** | ) | |
| **COUNTY OF DUPAGE** | ) | |
| **IN THE MATTER OF** | ) | |
| **EXELON GENERATION COMPANY, LLC** | ) | **Docket Nos.** |
| **BRAIDWOOD STATION - UNITS 1 and 2** | ) | **STN 50-456 and STN 50-457** |

SUBJECT: **Request for Technical Specifications Change**
**Braidwood Station, Unit 1, Steam Generator Inspection Frequency Revision for the Fall 2001 Refueling Outage**

## AFFIDAVIT

I affirm that the content of this transmittal is true and correct to the best of my knowledge, information and belief.

R. M. Krich
Director - Licensing
Mid-West Regional Operating Group

Subscribed and sworn to before me, a Notary Public in and

for the State above named, this 9th day of

February, 2001.

Georgene R. Van Duyne
Notary Public

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MY COMMISSIO

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GEORGENE R. VANDUYNE
NOTARY PUBLIC, STATE OF ILLINOIS
MY COMMISSION EXPIRES 9-28-2003

## ATTACHMENT A

## BRAIDWOOD STATION, UNIT 1
## DESCRIPTION AND SAFETY ANALYSIS OF THE PROPOSED CHANGES

### A. SUMMARY OF PROPOSED CHANGES

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," we are proposing a change to the Technical Specifications (TS) of Facility Operating License Nos. NPF-72 and NPF-77 for the Braidwood Station, Units 1 and 2. Although this proposed TS change is only applicable to Braidwood Station, Unit 1, we are submitting the proposed TS change on Braidwood Station, Units 1 and 2, dockets since the affected TS page is common to both Units. The proposed one-time change revises the Steam Generator (SG) inspection interval requirements in TS 5.5.9.d.2, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," for the Braidwood Station, Unit 1, fall 2001 refueling outage, to allow a 40 month inspection interval after one SG inspection, rather than after two consecutive inspections resulting in C-1 category. In accordance with TS 5.5.9.c, "Steam Generator (SG) Tube Surveillance Program, Inspection Results Classification," C-1 category is defined as "< 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective."

The proposed change is described in detail in Section E of this Attachment. Attachment B-1 includes the marked-up TS page for the proposed change for the Braidwood Station, Unit 1. Attachment B-2 includes the associated TS page with the proposed change incorporated for the Braidwood Station, Unit 1.

We request approval of the proposed change prior to August 9, 2001. This would support postponing SG inspections during the Unit 1, fall 2001 refueling outage. SG inspections will be performed during the Unit 1 spring 2003 refueling outage in accordance with the requirements in TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program, " and Electric Power Research Institute (EPRI) "PWR Steam Generator Examination Guidelines," Volume 1, Revision 5, September 1997. This proprietary document provides guidance for developing SG program inspection scope and frequency, identifying degradation mechanisms, and qualification of inspection techniques and personnel. The upcoming Unit 1, fall 2001 refueling outage is currently scheduled to begin September 22, 2001, with SG inspections scheduled to begin on September 25, 2001.

### B. DESCRIPTION OF THE CURRENT REQUIREMENTS

TS 5.5.9.d.1, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," requires that subsequent inservice inspections of SG tubes after the first inservice inspection be performed at intervals ≥ 12 calendar months and ≤ 24 calendar months after the previous inspection. In accordance with the Extension Criteria in TS 5.5.9.d.2, 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.

## C. BASES FOR THE CURRENT REQUIREMENTS

The inspection of the SG tubes ensures that the structural integrity of this portion of the Reactor Coolant System (RCS) will be maintained. Inservice inspection of SG tubes is essential in order to maintain surveillance of the condition of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of SG tubes also provides a means of characterizing the nature and cause of any tube degradation so that timely corrective measures can be taken.

## D. NEED FOR REVISION OF THE REQUIREMENT

TS 5.5.9.d requires two consecutive inspection results in the C-1 category before the inspection interval can be extended from a maximum of 24 calendar months to a maximum of 40 months. This one-time change (i.e., extending the inspection interval for the Unit 1, fall 2001 refueling outage from a maximum of 24 calendar months to a maximum of 40 months) is proposed to eliminate unnecessary SG inspections during the upcoming Unit 1, fall 2001 refueling outage. The one-time elimination of this SG inspection will result in a radiation dose savings of approximately 18 person-REM, a schedule savings of approximately 1 day critical path time, and a cost savings of approximately $2,500,000. These estimates are based on inspecting two of the four SGs.

Braidwood Station, Unit 1, replaced SGs during the refueling outage that was completed in November 1998. The replacement SGs are Babcock and Wilcox International (BWI) design and incorporate significant improvements, including thermally treated Inconel-690 tubing. During the Unit 1 spring 2000 refueling outage (i.e., A1R08), following the first cycle of operation after SG replacement, 100% of the tubing was inspected full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) with eddy current and showed no defective or degraded tubes. These inspection results along with the improved BWI replacement SG design and industry experience regarding BWI SG inspection results, provides the basis for extending the inspection interval that comes due during the upcoming Unit 1, fall 2001 refueling outage from a maximum of 24 calendar months to a maximum of 40 months with one C-1 category inspection result.

## E. DESCRIPTION OF THE PROPOSED CHANGES

The proposed one-time change revises TS 5.5.9.d.2 as follows.

- TS 5.5.9.d.2 currently states, "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 month;"

- TS 5.5.9.d.2 will be revised to state, "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."**

## F. SAFETY ANALYSIS OF THE PROPOSED CHANGES

The inspection of the SG tubes ensures that the structural integrity of this portion of the Reactor Coolant System (RCS) will be maintained. As discussed below, the Unit 1 spring 2000 refueling outage inspection results along with the improved BWI replacement SG design and industry experience regarding BWI SG inspection results, provides the basis for proposing an extension of the inspection interval and demonstrating that the integrity of the RCS will be maintained.

As previously discussed, TS 5.5.9.d requires two consecutive inspections resulting in category C-1 classification before the inspection interval can be extended from a maximum of 24 calendar months to a maximum of 40 months. We are proposing that for the upcoming Braidwood Station, Unit 1, fall 2001 refueling outage, the inspection interval be extended to a maximum of 40 months after only one SG inspection that resulted in all inspection results being classified in the C-1 category based on the following.

### SG Design Improvements

Industry experience with recirculating SGs using mill annealed Inconel-600 tubing has lead to significant design improvements in replacement SG design and fabrication. Problems associated with tube degradation (i.e., stress corrosion cracking (SCC), intergranular attack (IGA), pitting, and wastage) have been addressed through changes in tube materials and stress relief. Problems associated with secondary system fouling and flow-induced vibration and wear have been addressed with changes to the tube bundle support system. These design improvements, along with others, have been incorporated into the BWI replacement SG design and are discussed below.

- Thermally Treated Inconel-690 Tubing

  Each of the four Braidwood Station, Unit 1, SGs contains 6633 thermally treated Inconel-690 U-tubes that have a nominal outer diameter of 0.6875" and a thickness of 0.040". The development of thermally treated Inconel-690 tubing was prompted by the significant numbers of mill annealed Inconel-600 tubes being removed from service due to degradation. Thermally treated Inconel-690 tubing is similar to mill annealed Inconel-600 tubing but contains a 13% higher chromium content and correspondingly reduced nickel content. The higher chromium content reduces the degree of sensitization (i.e., the amount of chromium depleted in areas adjacent to the metal grain boundaries), thus increasing

resistance to corrosion attack at the metal grain boundaries. Heat treatment of Inconel-690 for optimum SCC resistance involves mill annealing at temperatures sufficient to put all the carbon into solution, followed by a thermal treatment to precipitate carbides on the metal grain boundaries into the tube metal microstructure. Resistance to SCC is greatest when the metal grain boundaries are fully populated with carbides.

Extensive testing has been performed which demonstrates thermally treated Inconel-690 tubing is superior to mill annealed Inconel-600 tubing in its resistance to both primary and secondary system SCC, pitting, and general corrosion. Examples of this data are given in proceedings from the 1986 EPRI Workshop on Thermally Treated Alloy 690 Tubes for Nuclear Steam Generators (Reference 1). Testing was performed with statically loaded Reverse U-Bend (RUB) specimens, where cracking was observed within approximately 300 hours for mill annealed Inconel-600 tubing and 800 hours for thermally treated Inconel-600 tubing, while cracking was not observed for the thermally treated Inconel-690 tubing even after 12,000 hours. Testing was also performed on statically loaded tensile specimens tested in 680°F primary water. While mill annealed Inconel-600 tubing exhibited cracking within 2,900 hours, thermally treated Inconel-690 did not exhibit cracking even after 7,000 hours of testing. Thermally treated Inconel-690 was also compared to mill annealed Inconel-600 tubing in steam tests to produce accelerated primary water SCC (PWSCC). Steam tests are performed in 760°F steam produced from hydrogenated pure water. These test results showed mill annealed Inconel-600 tubing exhibited cracking within 1,000 hours, while thermally treated Inconel-690 did not exhibit any signs of cracking after 6,000 hours (References 2 and 3).

The tubing procurement specification used in construction of the Braidwood Station, Unit 1, replacement SGs was designed to assure mill production of tubing that achieves the corrosion resistance properties as indicated by industry standards and research. The specification also outlines the physical, mechanical, and extensive inspection and qualification requirements necessary to limit fabrication defects. Cracks, laminations, scratches, draw-marks, pores, seams, laps, or stains are considered defects and are subject to rejection or conditioning in accordance with tested, approved, and controlled methods.

In addition to the thermal treatment process that was performed on all tubing, additional stress relief was performed on all U-bends up to a 12" centerline radius. The smallest centerline radius U-bend in the replacement SG design is 3.6" as compared to 2.25" in the original SGs. This larger radius reduces residual stress in the low row U-bend region. The additional stress relief and larger minimum radius U-bend design provides added assurance that this region will not develop cracking.

Industry data supports the laboratory test results demonstrating the superior performance of thermally treated Inconel-690 as compared to mill annealed Inconel-600 tubing.

- Tube Bundle Support System

  Experience with first generation mill annealed Inconel-600 tubed recirculating SGs has identified the following issues relating to tube bundle support design.

  - Dry-out and deposition in crevices of drilled hole type support plates leading to under deposit IGA.
  - SCC resulting from denting of the tubes due to magnetite development on the carbon steel tube support plates or open crevices at the tubesheet joint.
  - IGA and/or SCC associated with the high residual stress of rolled tube-to-tubesheet joint expansion particularly in combination with the unexpanded crevice design, which encouraged crevice corrosion as sludge concentrated in this critical area.
  - Mechanical wear to the tube from fretting induced by flow-induced vibration.

  As described below, the BWI replacement SG design incorporates features to greatly reduce or eliminate these potential damage mechanisms.

  The replacement SG design uses a Type 410S stainless steel lattice grid tube support structure. The selection of Type 410S stainless steel for the lattice grid design was based on extensive research that evaluated various materials with respect to denting potential, mechanical compatibility, resistance to fretting wear with thermally treated Inconel-690, and weldability.

  Industry experience with both the lattice grid and broached plate support designs concludes that the lattice grid is superior for a recirculating SG in that it provides the following.

  - High circulation rates through lower flow resistance, thus leading to a lower tendency to accumulate deposits than a broached plate.
  - Superior vibration restraint and fretting resistance.
  - Reduced denting potential due to the selection of an optimum stainless material.

  The BWI design lattice supports are positioned within the SG shroud at elevations selected to prevent flow-induced vibration resulting in potential tube fretting, while not creating excessive flow resistance resulting in tube deposits. The tubes are held in position within the diamond-shaped bar openings formed by the lattice grids which provide line support contact. This minimizes the area of crevices between the tubes and bars that could trap corrosion products or encourage dry-out caused by local superheat.

  The U-bends are supported using a Type 410S stainless steel flat bar U-bend restraint (i.e., FUR). This system provides close tolerance support of the U-bend region of the tube bundle to prevent flow-induced vibration, similar to the lattice grid tube support plate system. The potential for tube fretting is reduced due to material compatibility and a relatively long contact length as compared to original SG support designs. By distributing the contact force, the FURs minimize the possibility of fretting. The FURs are designed with all spaces orientated with an upward slope. This promotes continuous sweeping during operation and avoids the potential for creating steam pockets and corresponding dry-out leading to deposition. FURs do not cross the bundle centerline, thereby avoiding creation of horizontal tube contacts where deposits have the potential to collect.

The tube-to-tubesheet joint accomplishes axial load resistance and the physical fastening of the tubing to the vessel. Original SG designs encountered severe corrosion problems with the SG tube-to-tubesheet joint region associated with open (i.e., unexpanded) crevices, and/or SCC at the high residual stress cold worked locations on the surface of the transition zone between the roller expanded and unexpanded tube. The BWI replacement SG design incorporates the following features to address tube-to-tubesheet joint configuration concerns.

- Full depth expansion to eliminate a tubesheet crevice that could encourage denting or accumulation of contaminants against the transition zone.
- Hydraulic expansion that leaves minimal residual stresses and cold work as compared to mechanical roller expansion techniques. Hydraulic expansions typically produce 20 – 40% less stress than hardrolled expansions.

• Increased Circulation Ratio

Circulation ratio is defined as the ratio of riser mass flow rate to steam outlet flow rate. By maximizing circulation ratio of the SG secondary side fluid, concerns regarding heat transfer performance, generator sludge management, corrosion product transfer, tube dry-out, etc., can be minimized. The replacement SG design has a circulation ratio of greater than five, which is more than double that experienced in original design SGs.

• Loose Parts Potential

The replacement SG design incorporates features to minimize the development of loose parts during operation and maintenance. Specific design efforts have been taken to minimize corrosion potential on small thickness metal parts and incorporate mechanisms for capturing or eliminating fasteners. Overall, the improved design features incorporated in the Braidwood Station, Unit 1, replacement SGs provide reasonable assurance that SG tube integrity will be maintained over the proposed operating period.

**First Outage Inspection Sampling**

TS 5.5.9.d.1 requires that the first inservice inspection of SG tubes be performed after six Effective Full Power (EFP) months but ≤ 24 calendar months of initial criticality. This SG inspection requirement was satisfied during the Braidwood Station, Unit 1, spring 2000 refueling outage. As required by TS 5.5.9.b, "Steam Generator (SG) Tube Surveillance Program, SG Tube Sample Selection and Inspection," TS Table 5.5.9-1, "Minimum Number of Steam Generators to be Inspected During Inservice Inspection," and TS Table 5.5.9-2, "Steam Generator Tube Inspection," the SG Tube Inspection requirements for the first inservice inspection were to inspect 6%/SG of the tubes in at least two SGs. Braidwood Station inspected significantly more than the minimum TS requirement during the Unit 1 spring 2000 refueling outage by performing 100% full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) bobbin inspection of all four SGs. This was in accordance with Section of 3.3.1, "Examination of Tubes," of the EPRI Pressurized Water Reactor (PWR) SG Examination Guidelines. The inspection was performed using ERPI PWR SG Examination Guidelines Appendix H, "Performance Demonstration for Eddy Current Examination," qualified techniques and Appendix G, "Qualification of Nondestructive Examination Personnel for

Analysis of NDE Data," qualified data analysts and equipment . The inspection results showed no degraded or defective tubes, and all SGs were classified as category C-1.

For the second inservice inspection of SG tubes, TS require that a minimum of 12% of the entire unit's tube population be inspected in at least one SG. By performing 100% full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) inspection of all SGs during the first outage after SG replacement, i.e., Unit 1 spring 2000 refueling outage, Braidwood Station inspected significantly more SG tubes than would be required for both the first and second inservice inspections after SG replacement. Therefore, even though we are proposing a one-time extension of the interval between inspections, the scope of the inspections already performed during the Unit 1 spring 2000 refueling outage was significantly expanded from that required by the TS over the first two refueling outages after SG replacement.

First outage inspection sampling results, along with industry experience and the operational assessment discussed below indicate that tube integrity will be maintained over the proposed operating period.

**EPRI PWR SG Examination Guidelines**

The EPRI PWR SG Examination Guidelines base SG inspection frequency on inspection results and performance criteria. We have followed the recommendations of Section 3.3.1 of the ERPI PWR SG Examination Guidelines, which contain the provisions regarding SG inspection frequency based on inspection results.

1. After the first cycle of operation (i.e., a duration not less than six EFP months and not more than 24 EFP months) for either new or replacement SGs, a 100% full-length (i.e., tube end to tube end) examination using general purpose eddy current probes shall be performed on all SGs.
2. During subsequent inservice inspections, if tube degradation (i.e., active damage mechanisms as defined in Appendix F, "Terminology," of the ERPI PWR SG Examination Guidelines) are identified, all SGs shall be examined at the end of each fuel cycle or 24 EFP months, whichever is less, or as necessary to satisfy published regulatory requirements.
3. During subsequent inservice inspections, if active damage mechanisms are not identified, the number of SGs to be examined and/or the frequency of examination, shall be performed as required by Section 3.3.2, "Steam Generators Free from Active Damage Mechanisms," of the EPRI PWR SG Examination Guidelines.
4. 100% of the tubing and 100% of each type of repair shall be inspected within a rolling 60 EFP month time frame. If 60 EFP months occur during an operating cycle, completion of that cycle is acceptable and is within stated requirement.
5. No SG shall operate more than two fuel cycles between inspections.

As stated in Section 3.3.2 of the ERPI PWR SG Examination Guidelines, if the SGs are free from active damage mechanisms, some latitude is provided in terms of the number of SGs to be inspected and/or frequency of inspection. For these SGs, any of the following options may be performed.

1. Inspect $\geq$ 20% of the tubes and $\geq$ 20% of each type of repair in each SG at each refueling outage, or

2. Inspect ≥ 40% of the tubes and ≥ 40% of each type of repair in half the number of SGs at each refueling outage, or
3. Inspect ≥ 40% of the tubes and ≥ 40% of each type of repair in each SG at every other refueling outage.

Braidwood Station, Unit 1, does not have an active SG damage mechanism as defined in Appendix F of the ERPI PWR SG Examination Guidelines. Therefore, Option 3 as indicated above will be met without performing SG inspections during the upcoming Unit 1, fall 2001 refueling outage.

Inspection frequency in accordance with Section 5.0 "Steam Generator Assessments," of the EPRI PWR SG Examination Guidelines is also determined by measurement results against performance criteria. As part of our SG Tube Surveillance Program, both a Condition Monitoring Assessment and Operational Assessment are performed after each inspection and the results are compared to performance criteria. The performance criteria against which the results are compared involve SG tube structural integrity, accident-induced leakage, and operational leakage. The performance criteria are contained in Section 2.0 "Performance Criteria," of Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines" (Reference 4). The Condition Monitoring Assessment is retrospective and is intended to confirm that adequate SG integrity has been maintained since the previous inspection. The Operational Assessment is predictive and is intended to provide reasonable assurance that the performance criteria will be met throughout the next operating period. If it is determined that the performance criteria will not be met at the end of the next cycle of operation, then the operational cycle length must be adjusted accordingly.

**Condition Monitoring Assessment**

After completion of the Braidwood Station, Unit 1, spring 2000 refueling outage SG inspections, a Condition Monitoring Assessment was performed in accordance with EPRI "Steam Generator Integrity Assessment Guidelines," Revision 1, March 2000. This proprietary document provides guidelines for evaluating the condition of SG tubes based on inspection results. The results showed that all performance criteria had been met based on full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) bobbin inspection of all of the tubes of all four SGs. One tube showed a distorted indication in the fan bar region of the SG based on bobbin coil inspection. This area was inspected using a Plus Point probe and was confirmed to be minor wear that is < 10% through wall (TW). Although this tube was significantly below the 40% TW TS 5.5.9.e acceptance criteria, it was removed from service by mechanical plugging. Accordingly, based on the Unit 1 spring 2000 refueling outage inspection results of the "as found" condition of the SG, all performance criteria were met.

**Operational Assessment**

An Operational Assessment was performed in accordance with EPRI SG Integrity Assessment Guidelines to evaluate the predicted condition of the SGs after two cycles of operation. The Operational Assessment is summarized below.

Braidwood Station, Unit 1, operated for 1.29 Effective Full Power Years (EFPY) during Cycle 8 (i.e., the first post-SG replacement cycle). The operational assessment evaluated Braidwood Station, Unit 1, for a combined Cycle 9 and Cycle 10 operating period. Both Cycle 9 and Cycle

10 are each estimated to be approximately 1.4 EFPY, therefore the operational assessment conservatively assumed 3.0 EFPY of operation.

Plans call for Braidwood Station, Unit 1, power to be uprated by 5% during this time period prior to the next scheduled SG inspection in the Unit 1 spring 2003 refueling outage. Current plans are to implement a ~2% (i.e., partial) power uprate prior to the end of Cycle 9 (i.e., the current cycle) with full implementation of the 5% power uprate beginning with Cycle 10. The impact of power uprate on the Braidwood Station, Unit 1, SGs has been evaluated in the power uprate license amendment request (Reference 5). The report evaluated the effect of power uprate conditions in the following areas.

- Structural components, including tube plugs and hardware, were evaluated against American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section III, " Rules for Construction of Nuclear Power Plant Components," acceptance criteria and found to be acceptable for normal operating, transient and accident conditions.
- Thermal hydraulic effects including circulation ratio, hydrodynamic stability, SG mass, peak heat flux, pressure drop, and moisture carryover were evaluated and shown to be within acceptable ranges and do not adversely effect the SGs.
- Tube degradation in the form of wastage and/or pitting, Outside Diameter SCC, PWSCC, tube wear, and U-bend fatigue was evaluated and shown to have an insignificant increase. This is mainly due to the superior characteristics of thermally treated Inconel-690 tubing as discussed in the SG Design Improvements section of this attachment.
- In the case of tube wear in the fan bar region, which is the most likely potential damage mechanism in the Unit 1 SGs at this time, wear is predicted to decrease due to a slightly lower circulation ratio and increased secondary side pressure due to the corresponding average temperature increase.

We conservatively assumed the reported fan bar wear indication, discussed above in the Condition Monitoring Section, to be 10% TW. Eddy current uncertainty and growth over two cycles of operation were applied in accordance the EPRI SG Integrity Assessment Guidelines. The largest fan bar wear indication predicted at the end of the proposed operating cycle is < 41% TW. This is based upon a conservative growth rate value that does not take credit for analyses that indicate fan bar wear rates will decrease under power uprate conditions. The < 41% TW value is well below the conservative low average temperature structural limit for fan bar wear, which is 60% TW at power uprated conditions. The structural limit for wear was calculated in accordance with the guidelines contained in Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" (Reference 6).

During the fabrication of the replacement SGs, it was noted that a number of tubes were in contact with other tubes while the vessels were in a horizontal position. No tube damage was observed in this area during the preservice inspections. The tube-to-tube contact condition was evaluated by BWI and Exelon Generation Company, LLC and the root cause was determined to be due to gravitational effects on the tubes and floating fan bar assembly in the horizontal position. This condition is expected to naturally correct after one or two cycles of operation in the vertical position. Replacement SGs of similar design, including Byron Station, Unit 1, experienced similar tube-to-tube contact during preservice inspection. After one cycle of operation these units reported far fewer tubes in contact and no tube damage.

The Braidwood Station, Unit 1, tube-to-tube contact condition was monitored during the spring 2000 refueling outage. A total of 85 tubes were found to be in contact (i.e., refer to Table "Braidwood Station, Unit 1, Spring 2000 Refueling Outage Tube to Tube Contact," below). There were no indications of tube fretting or degradation associated with these indications. The majority of the tubes in contact were located at the outer periphery of the tube bundle or within one tube of the outermost tube. The table below also lists the number of tubes in contact during baseline inspection while in the horizontal position. As can be seen from the table, tube-to-tube contact appears to be self-correcting and is expected to improve over the next operating cycles.

**Braidwood Station, Unit 1, Spring 2000 Refueling Outage Tube to Tube Contact**

| SGs | A | B | C | D | Total |
|---|---|---|---|---|---|
| Tubes in Contact Pre-Service | 75 | 122 | 122 | 189 | 508 |
| Tubes in Contact Refuel Outage | 2 | 26 | 17 | 40 | 85 |

The tube-to-tube contact condition was evaluated by BWI. The evaluation assessed tube integrity for the potential of fretting/wear damage and corrosion induced degradation due to long term tube-to-tube contact. Based on conservative estimates of wear coefficients and work-rates at the tube-to-tube contact area, it is shown that a maximum tube wall loss of 40% would occur after 60 years of continuous full power operation. Therefore, it is not expected that tube contact fretting would result in exceeding the conservative 60% TW structural criteria during the life of the SGs. The potential for corrosion induced degradation as a result of excessive fouling or deposit bridging compounded by tube contact is bounded by the top of the tubesheet expansion region that has been previously qualified by extensive testing. Therefore, it is concluded that there is not an additional tube degradation risk due to tube contact. The tube contact condition will be monitored over time through the normal inspection program.

Another possible damage mechanism that could affect replacement SG tube integrity is wear from secondary side foreign objects. During the Unit 1 spring 2000 refueling outage, sludge lancing was performed on the secondary side tubesheet region of all four SGs. Upon completion of sludge lancing, a video inspection was performed in this region to identify any foreign objects. No foreign objects detrimental to the SG tubing were identified. Minor debris was removed from the SGs as part of the inspection. At the present time, no foreign objects are known to be present in the SGs. This inspection along with improved SG design and our Foreign Material Exclusion (FME) Program provides confidence that foreign object wear will not occur over the proposed operating period.

During the Unit 1 spring 2000 refueling outage inspection, no forms of degradation other than the minor wear discussed above were identified. The effects of power uprate over the proposed operating cycle (i.e., through Cycle 10) have been evaluated against the NEI 97-06 structural and leakage criteria and found to be acceptable. Therefore, all structural and accident leakage performance criteria in Reference 4 are predicted to be met through the end of Cycle 10.

Braidwood Station meets the recommendations of the recently issued EPRI document, "PWR Primary-to-Secondary Leak Guidelines," Revision 2, dated April 2000. This proprietary document provides guidelines for detecting, monitoring, and mitigating SG primary-to-secondary leakage. Implementation of these guidelines provides assurance that proper monitoring and response will occur in the event primary-to-secondary leakage were to develop over the proposed operating period.

Braidwood Station, Unit 1, implements current industry guidelines with respect to primary and secondary water chemistry. No significant chemistry excursions have occurred which would indicate or lead to increased SG tube degradation.

**Industry Data**

Review of industry data for 47 plants with SGs containing thermally treated Inconel-690 tubing reveals no degradation mechanisms other than mechanical wear. When comparing the operating EFPYs of Braidwood Station, Unit 1, to the other 46 plants with thermally treated Inconel-690 tubing, 36 of these plants have more operating time than Braidwood Station, Unit 1, and have not experienced any degradation other than mechanical wear. Corrosion related degradation is not expected, particularly not early in the life of these SGs, due to the superior corrosion resistant properties of thermally treated Inconel-690 tubing. Review of the wear data from same design BWI replacement SGs indicates that minor wear (i.e., < 20% TW) has been seen in a small number of tubes in some units, and not seen at all in other units. For those wear indications that have a growth rate obtained through multiple inspections, the growth rate has been relatively small (i.e., < 6% TW per cycle). This information provides reasonable assurance that wear indications will not become structurally significant over the proposed cycle of operation.

Byron Station, Unit 1, performed SG replacement during the fall 1997 refueling outage. Byron Station, Unit 1, replacement SG design and operating conditions are essentially identical to those at Braidwood Station, Unit 1. Since the time of SG replacement, Byron Station, Unit 1, has performed two refueling outages. During the first refueling outage after SG replacement, Byron Station performed 100% full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) bobbin inspection of three of the four SGs and found no defective or degraded tubes. During the second refueling outage after SG replacement, i.e., during the Unit 1 fall 2000 refueling outage, Byron Station performed 100% full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) bobbin inspection of the one SG not inspected during the previous refueling outage and found no defective or degraded tubes. The operating conditions and inspection techniques used for the Byron Station, Unit 1, inspections are essentially identical to those used at Braidwood Station, Unit 1. The fact that no defective or degraded tubes were identified during the first or second inspection after SG replacement at Byron Station, Unit 1, provides reasonable assurance that SG tube integrity requirements will be met for the proposed operating period for Braidwood Station, Unit 1.

**Dose, Schedule, and Cost Impact**

If the proposed change is not approved for the Unit 1, fall 2001 refueling outage, our current plan would be to perform 100% full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) bobbin inspection of two of the four SGs. Assuming this scope, the

following dose, schedule, and cost impacts are predicted based on the Braidwood Station, Unit 1, spring 2000 refueling outage and Byron Station, Unit 1, fall 2000 refueling outage inputs.

- Accumulated personnel dose including SG platform setup, manway removal, eddy current inspection, and tube plugging is estimated to be approximately 18 person-REM.
- The approximate time required to perform 100% full-length (i.e., from hot leg tube end to cold leg tube end, including the U-bends) bobbin inspection of two SGs is 7 days from removal of the first manway to reinstallation of the last manway after completion of the inspection. This would add approximately one day of critical path time to the Unit 1, fall 2001 refueling outage schedule.
- The approximate cost associated with inspecting two of the four SGs including contractor craft support is $2,500,000.

Based on improved SG design, first inservice inspection results, and industry experience, we have concluded that SG inspections are not necessary to meet the SG inspection objectives.

## G. IMPACT ON PREVIOUS SUBMITTALS

We have reviewed the proposed changes regarding their impact on any previous submittals and have determined that there is no impact on any previous submittals.

## H. SCHEDULE REQUIREMENTS

We request approval of the proposed change prior to August 9, 2001. This would support postponing SG inspections during the Unit 1, fall 2001 refueling outage. SG inspections will be performed during the spring 2003 refueling outage in accordance with the requirements in TS 5.5.9, "Steam Generator (SG) Tube Surveillance Program, " and EPRI PWR SG Examination Guidelines. The upcoming Unit 1, fall 2001 refueling outage is currently scheduled to begin September 22, 2001, with SG inspections scheduled to begin on September 25, 2001.

## I. REFERENCES


1. T. Yonezawa, "Evaluation of the Corrosion Resistance of Alloy 690," EPRI NP-4665S-SR Proceedings: Workshop on Thermally Treated Alloy 690 Tubes for Nuclear Steam Generators, Pittsburgh, PA, June 26 - 28, 1986, paper No. 12.

2. R.G. Aspeden, T.F. Grand and D.L. Harrod, "Corrosion Performance of Alloy 690," EPRI NP-6750-M Proceedings: 1989 EPRI Alloy 690 Workshop, New Orleans, LA, April 12 - 14, 1989.

3. G. Santarini, "Alloy 690: Recent Corrosion Results," EPRI NP-6750-M Proceedings: 1989 EPRI Alloy 690 Workshop, New Orleans, LA, April 12 - 14, 1989.

4. Letter from D. Modeen (Nuclear Energy Institute (NEI)) to S. Collins (U.S. NRC), "Revised Industry Steam Generator Program Generic License Change Package," Enclosure 9, NEI 97-06, "Steam Generator Program Guidelines," draft Revision 1, dated December 11, 2000.

5. Letter from R.M. Krich (ComEd) to U.S. NRC, "Request for a License Amendment to Permit Uprated Power Operations at Byron and Braidwood Stations," July 5, 2000.

6. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes (for comment)," August 1976.

**ATTACHMENT B-1**

**PROPOSED TS CHANGES FOR BRAIDWOOD STATION, UNIT 1**

MARKED-UP TS PAGE

5.5-10

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;

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

• 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

## ATTACHMENT B-2

## PROPOSED TS CHANGES INCORPORATED FOR BRAIDWOOD STATION, UNIT 1

TS PAGE

5.5-10

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

## ATTACHMENT C

## INFORMATION SUPPORTING A FINDING OF NO SIGNIFICANT HAZARDS CONSIDERATION

According to 10 CFR 50.92(c), "Issuance of amendment," a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

Exelon Generation Company, LCC is proposing changes to the Technical Specifications (TS) of Facility Operating License Nos. NPF-72 and NPF-77 for the Braidwood Station, Units 1 and 2. The proposed changes revise the Steam Generator (SG) inspection frequency requirements in TS 5.5.9.d.2, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," for Braidwood Station, Unit 1, fall 2001 refueling outage, to allow a 40 month inspection frequency after one inspection, rather than after two consecutive inspections resulting in C-1 classification. This change is necessary to eliminate SG inspections during the Unit 1, fall 2001 refueling outage, thus resulting in significant dose, schedule, and cost savings.

Information supporting the determination that the criteria set forth in 10 CFR 50.92 are met for this amendment request is indicated below.

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

The proposed one-time change revises the Steam Generator (SG) inspection interval requirements in Technical Specifications (TS) 5.5.9.d.2, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," for the Braidwood Station, Unit 1, fall 2001 refueling outage, to allow a 40 month inspection frequency after one inspection, rather than after two consecutive inspections results that are within the C-1 category. C-1category is defined as "< 5% of the total tubes inspected are degraded tubes and none of the inspected tubes are defective."

The proposed one-time extension of the Unit 1 SG tube inservice inspection interval does not involve changing any structure, system, or component, or affect reactor operations. It is not an initiator of an accident and does not change any existing safety analysis previously analyzed in the Byron/Braidwood Stations' Updated Final Safety Analysis Report (UFSAR). As such, the proposed changes do not involve a significant increase in the probability of an accident previously evaluated.

Since the proposed change does not alter the plant design, there is no direct increase in SG leakage. Industry experience indicates that the probability of increased SG tube degradation would not go undetected. Additionally, steps described below will further minimize the risk associated with this extension. For example, the scope of inspections performed during the last Braidwood Station, Unit 1, refueling outage (i.e., the first refueling outage following SG replacement) exceeded the TS requirements for the first two refueling outages after SG replacement. That is, more tubes were inspected than were required by TS. Currently, Braidwood Station, Unit 1, does not have an active SG damage mechanism, and will meet the current industry examination guidelines without performing SG inspections during the next refueling outage. Additionally, as part of our SG Tube Surveillance Program, both a Condition Monitoring Assessment and an Operational Assessment are performed after each inspection and compared to the Nuclear Energy Institute (NEI) 97-06, "Steam Generator Program Guidelines," performance criteria. The results of the Condition Monitoring Assessment demonstrated that all performance criteria were met during the Braidwood Station, Unit 1, spring 2000 refueling outage, and the results of the Operational Assessment show that all performance criteria will be met over the proposed operating period. Considering these actions, along with the improved SG design and reliability of Babcock and Wilcox International (BWI) replacement SGs, extending the SG tube inspection frequency does not involve a significant increase in the consequences of an accident previously evaluated.

Therefore, the proposed change does not involve a significant increase in the probability or 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?**

The proposed change revises the SG inspection frequency requirements in TS 5.5.9.d.2, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," for the Braidwood Station, Unit 1, fall 2001 refueling outage, to allow a 40 month inspection interval after one inspection, rather than after two consecutive inspections with inspection results within the C-1 category.

The proposed change will not alter any plant design basis or postulated accident resulting from potential SG tube degradation. The scope of inspections performed during the last Braidwood Station, Unit 1, refueling outage (i.e., the first refueling outage following SG replacement) significantly exceeded the TS requirements for the scope of the first two refueling outages after SG replacement.

Primary to secondary leakage that may be experienced during all plant conditions is expected to remain within current accident analysis assumptions. The proposed change does not affect the design of the SGs, the method of SG operation, or reactor coolant chemistry controls. No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. The proposed change involves a one-time extension to the SG tube inservice inspection frequency, and therefore will not give rise to new failure modes. In addition, the proposed change does not impact any other plant system or components.

Therefore, 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?**

The SG tubes are an integral part of the Reactor Coolant System (RCS) pressure boundary that are relied upon to maintain the RCS pressure and inventory. The SG tubes isolate the radioactive fission products in the reactor coolant from the secondary system. The safety function of the SGs is maintained by ensuring the integrity of the SG tubes. In addition, the SG tubes comprise the heat transfer surface between the primary and secondary systems such that residual heat can be removed from the primary system.

SG tube integrity is a function of the design, environment, and current physical condition. Extending the SG tube inservice inspection frequency by one operating cycle will not alter the function or design of the SGs. SG inspections conducted during the first refueling outage following SG replacement demonstrated that the SGs do not have an active damage mechanism, and the scope of those inspections significantly exceeded those required by the TS. These inspection results were comparable to similar inspection results for the same model of replacement SGs installed at other plants, and subsequent inspections at those plants yielded results that support this extension request. The improved design of the replacement SGs also provides reasonable assurance that significant tube degradation is not likely to occur over the proposed operating period.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

## ATTACHMENT D

## INFORMATION SUPPORTING AN ENVIRONMENTAL ASSESSMENT

Exelon Generation Company, LLC has evaluated the proposed changes against the criteria for identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21, "Criteria for and identification of licensing and regulatory actions requiring environmental assessments." ComEd has determined that the proposed changes meet the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9), "Criteria for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b), "Issuance of amendment." This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," which changes 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 which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria.

**(i) The amendment involves no significant hazards consideration.**

As demonstrated in Attachment C, the proposed changes do not involve any significant hazards consideration.

**(ii) There is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite.**

The proposed changes revise the Steam Generator (SG) inspection frequency requirements in TS 5.5.9.d.2, "Steam Generator (SG) Tube Surveillance Program, Inspection Frequencies," for Braidwood Station, Unit 1, fall 2001 refueling outage, to allow a 40 month inspection frequency after one inspection, rather than after two consecutive inspections resulting in C-1 classification. The proposed changes do not allow for an increase in the unit power level, do not increase the production, nor alter the flow path or method of disposal of radioactive waste or by-products. The proposed changes do not affect actual unit effluents. Therefore, the proposed changes do not change the types or increase the amounts of any effluents released offsite.

**(iii) There is no significant increase in individual or cumulative occupational radiation exposure.**

The proposed changes will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from the proposed changes.