April 18, 2005

Mr. Christopher M. Crane, President
and Chief Nuclear Officer
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: BRAIDWOOD STATION, UNITS 1 AND 2 - PUBLIC NOTICE OF APPLICATION FOR AMENDMENTS TO FACILITY OPERATING LICENSES (TAC NO. MC6686 AND MC6687)

Dear Mr. Crane:

The enclosed announcement was forwarded to the Morris Daily Herald and the Joliet Herald News for publication. This announcement relates to Exelon Generation Company, LLC application dated April 11, 2005, for amendment to Facility Operating License Nos. NPF-72 and NPF-77. The proposed amendments would revise Technical Specification 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to incorporate changes in the SG inspection scope for Braidwood Station, Unit 2 only, during refueling outage 11 and the subsequent operating cycle.

Sincerely,
/RA/

George F. Dick, Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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

Enclosure: Public Notice

cc w/encl: See next page

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Braidwood Station Units 1 and 2

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PUBLIC NOTICE

NRC STAFF PROPOSES TO AMEND OPERATING LICENSES AT THE

BRAIDWOOD STATION, UNITS 1 AND 2

The U.S. Nuclear Regulatory Commission (NRC) staff has received an application dated April 11, 2005, from Exelon Generation Company, LLC, for an exigent amendment to the operating licenses for the Braidwood Station, Units 1 and 2, located in Will County, Illinois.

The proposed amendments would revise Technical Specification 5.5.9, "Steam Generator (SG) Tube Surveillance Program," to incorporate changes in the SG inspection scope for Braidwood Station, Unit 2 only, during refueling outage 11, which is scheduled to start in the very near future and the subsequent operating cycle. The amendments would change the inspection scope only for the portions of the SG tubes within the tubesheets.

During a conference call on April 5, 2005, the licensee and NRC personnel discussed the proposed limited tubesheet inspection for the Braidwood Station, Unit 2, SGs. It was concluded that application of a limited tubesheet inspection in areas where degradation could occur warranted a change to the Technical Specification (TS) 5.5.9, "Steam Generator (SG) Tube Surveillance Program." In the absence of such a TS change, the license might have been in violation of Part 50, Appendix B "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," of Title 10 of the *Code of Federal Regulations* (10 CFR), by virtue of the choice of inspection techniques used to inspect the SG tubes. Limiting the inspection scope would obviate this potential violation. Additional guidance is contained in NRC Information Notice 2005-09, "Indications in Thermally Treated Alloy 600 Steam Generator Tubes and Tube-to-Tubesheet Welds," issued on April 7, 2005. Information Notice 2005-09 provided further details of the findings at Catawba Nuclear Station, Unit 2.

Due to the short time interval between identification of the need for a TS change to allow a limited SG inspection scope, the actual performance of the Braidwood, Unit 2, SG inspection in the upcoming refueling outage, and the schedule for restart of the reactor and entry into

Mode 4, time does not permit the Commission to publish a *Federal Register* notice allowing 30 days for prior public comment. Therefore, the licensee requested that this proposed TS change be considered under exigent circumstances as described in 10 CFR 50.91(a)(6).

As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below. The licensee and the NRC staff have evaluated this proposed change with regard to the determination of whether or not a significant hazards consideration is involved.

1. Operation of Braidwood Station, Unit 2, in accordance with the proposed amendments will not involve a significant increase in the probability or consequences of an accident previously evaluated.

> The previously analyzed accidents are initiated by the failure of plant structures, systems, or components. The proposed changes that alter the SG inspection criteria 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., a steam line break (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 approximately a factor of 2.5. 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.

2. The proposed amendments will not create the possibility of a new or different kind of accident from any previously analyzed.

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. The proposed amendment will not involve a significant reduction in a margin of safety.

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-P, "Limited Inspection of the Steam Generator Tube Portion Within the Tubesheet at Braidwood Unit 2 and Byron Unit 2," dated April 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.

Following an initial review of this application, the requested amendments have been evaluated against the standards in 10 CFR 50.92 and the NRC staff has made a proposed determination that the requested amendments involve no significant hazards considerations.

The changes do not significantly increase the probability or consequences of any accident previously considered, nor create the possibility of an accident of a different kind, nor significantly decrease any margin of safety.

If the proposed determination that the requested license amendment involves no significant hazards consideration becomes final, the NRC staff will issue the amendments without first offering an opportunity for a public hearing. An opportunity for a hearing will be published in the *Federal Register* at a later date and any hearing request will not delay the effective date of the amendment.

If the NRC staff decides in its final determination that the amendment does involve a significant hazards consideration, a notice of opportunity for a prior hearing will be published in the *Federal Register* and, if a hearing is granted, it will be held before the amendment is issued.

Comments on the proposed determination of no significant hazards consideration may be (1) telephoned to Gene, Y. Suh, Chief Section 2, Project Directorate III, by collect call, which may be recorded or transcribed, to 301-415-2466, or by facsimile to 301-415-3061, from 7:30 a.m. to 4:15 p.m. on Federal workdays, (2) e-mailed to GYS@nrc.gov, or (3) submitted in writing to the Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555. All comments received by close of business on April 21, 2005, will be considered in reaching a final determination. A copy of the application may be examined electronically through the Agencywide Documents Access and Management System (ADAMS) Public Electronic Reading Room link at the NRC Web site http://www.nrc.gov/reading-rm/adams.html and at the Commission's Public Document Room (PDR), located at One White Flint North, Public File Area O1 F21, 11555 Rockville Pike (first floor), Rockville, Maryland.

Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS, should contact the NRC PDR Reference staff by telephone at 1-800-397-4209, 301-415-4737, or by e-mail to pdr@nrc.gov.