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10 CFR 50.55a

RS-03-194

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

Quad Cities Nuclear Power Station, Units 1 and 2
Facility Operating License No. DPR-29 and DPR 30
NRC Docket No. 50-254 and 50-265

Subject: Quad Cities Nuclear Power Station, Units 1 and 2, Fourth Interval Inservice Inspection Program Plan

References: (1) Letter from T. J. Tulon (Exelon Generation Company, LLC) to U. S. NRC, "Quad Cities Nuclear Power Station, Units 1 and 2, Fourth Interval Inservice Inspection Program Plan," dated January 17, 2003

(2) Letter from T. J. Tulon (Exelon Generation Company, LLC) to U. S. NRC, "Quad Cities Nuclear Power Station, Units 1 and 2, Fourth Interval Inservice Inspection Program Plan," dated February 7, 2003

(3) Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Response to Request for Additional Information Regarding the Quad Cities Nuclear Power Station, Units 1 and 2, Fourth Interval Inservice Inspection Program Plan," dated August 13, 2003

In References 1 and 2, Exelon Generation Company, LLC (EGC) requested relief from certain requirements of 10 CFR 50.55a, "Codes and standards," for the fourth interval inservice inspection program plan for Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. In Reference 3, additional information was provided to the NRC regarding proposed relief requests I4R-01, I4R-02, and I4R-05. During a teleconference on September 16, 2003, the NRC requested additional information regarding relief request I4R-01. As a result of these discussions, a revised relief request I4R-01 is provided in the attachment to this letter.

If you have any questions or require additional information, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

A047

Attachment: Revised Relief Request I4R-01

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

ATTACHMENT

Revised Relief Request I4R-01

RELIEF REQUEST I4R-01

COMPONENT IDENTIFICATION

Code Class:	1
Reference:	IWB-2500 Table IWB-2500-1
Examination Category:	B-D
Item Number:	B3.100
Description:	Inspection of Standby Liquid Control Nozzle Inner Radius
Component Number:	Unit 1: N10 Unit 2: N10

CODE REQUIREMENT

IWB-2500 states that components shall be examined and tested as specified in Table IWB-2500-1.

Table IWB-2500-1 requires a volumetric examination to be performed on the inner radius section of all reactor pressure vessel nozzles each inspection interval.

BASIS FOR RELIEF

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested on the basis that compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The Standby Liquid Control (SBLC) nozzle, as shown in Figure I4R-01.1, is designed with an integral socket to which the boron injection piping is welded. The SBLC nozzle is located in the upper segment of the lower head of the reactor pressure vessel in an area that is inaccessible for examination from the inside of the reactor vessel. Therefore, the examination must be performed from the outside surface of the lower reactor vessel head.

An examination from the outside surface is conducted from either the nozzle-to-head outer blend radius or the head (plate) surface near the nozzle. Both examination areas provide a unique approach and are discussed separately.

Because of the small diameter of the SBLC nozzle (i.e., nominal 2 inches) and the thickness of the lower reactor vessel head (i.e., nominal 6.125 inches), the ratio of the nozzle diameter to the head thickness make it difficult to perform a meaningful examination from the nozzle-to-head outer blend radius. The inside surface inspection angle, utilizing specialized/contoured wedges or shoes, needs to be a mid-range tangential angle (i.e., 30 degrees - 60 degrees) to adequately detect flaws at the inside radius area of interest while scanning from the outer blend. Given the small diameter nozzle, the tangential angle, utilizing specialized/contoured wedges or shoes, will be in the low-range of 0 degrees – 15 degrees (i.e., essentially a straight beam) when focused to tangentially coincide with the inner radius area of interest. This low angle will not provide sufficient reflectivity for detection of inservice induced defects.

RELIEF REQUEST I4R-01

An examination from the head (plate) surface near the nozzle is also difficult, since the forged nozzle has a stainless steel cladding welded to the inside surface with increased weldment applied at the inner radius area for construction of an engineered socket configuration. After final machining, this engineered socket receives an austenitic internal pipe, which is integrally welded to form a complex cladding/socket configuration. The geometric and change in grain structures at the dissimilar material interface prohibit a timely (i.e., short duration) and meaningful ultrasonic examination of the inside radius section of the nozzle.

The change in grain structures result in mode conversion and angle changes, consequently, significantly longer exam duration is required to resolve a variety of reflective signals. Examination of this unique design is estimated to require a minimum of two hours with a two-person team (i.e., four manhours) while a typical nozzle inner radius examination will require up to 1 manhour with a two-person team.

A review of current ultrasonic techniques was conducted including discussions with the Electric Power Research Institute (EPRI). The long ultrasonic metal path and potential for multiple geometric and dissimilar material reflectors inherent in the nozzle design prevent a meaningful examination from being performed on the inner radius of the SBLC nozzle. Ultrasonic examination of this configuration will require the use of multiple techniques, modes, angles and an extensive amount of time on the component for best effort signal discrimination between geometrical and flaw type signals. This unique design is not typical to designs referenced in the ASME Code Section XI, but is more closely related to a dissimilar metal component. At present, the industry has a qualified technique for dissimilar metal piping welds, but has not addressed this unique design.

In addition, the dose exposure to the technicians performing the examination is a relevant concern. While actions would be taken to provide protection against the inherent radiation field, the large dose rates in the lower head region near the SBLC nozzle present a unique challenge. Dose rates in this region are typically 400 mrem per hour. As such, radiation exposure for this examination could easily exceed 1 rem, based on the general area dose and exam duration.

The proposed relief request will not compromise the level of quality and safety. The inner radius socket attaches to piping that delivers the boron solution far away from the nozzle inner radius. Therefore, the SBLC nozzle inner radius section is not subjected to turbulent mixing conditions that are a concern with other penetrations. In addition, a VT-2 examination of the SBLC nozzle is a part of the Class 1 system leakage test scheduled during each refueling outage. During reactor operation, Technical Specifications (TS) requirements ensure Reactor Coolant System (RCS) leakage is regularly monitored (i.e., every 12 hours), and specify prudent actions, up to and including reactor shutdown, should predetermined limits be exceeded. These actions ensure that the overall level of plant quality and safety will not be compromised.

As such, relief is requested in accordance with 10 CFR 50.55a(a)(3)(ii), since compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

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PROPOSED ALTERNATE EXAMINATION

A VT-2 examination of the SBLC nozzle will be part of the scheduled Class 1 system leakage test performed each refueling. In addition, the TS Surveillance Requirement (SR) for RCS leakage will be satisfied during plant operation (i.e., TS SR 3.4.4.1). These actions ensure that the overall level of plant quality and safety will not be compromised.

APPLICABLE TIME PERIOD

Relief is requested for the fourth ten-year inspection interval of the Inservice Inspection Program for Quad Cities Nuclear Power Station Units 1 and 2.

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FIGURE I4R-01.1

2-INCH STANDBY LIQUID CONTROL NOZZLE