November 2, 2001

Mr. Oliver D. Kingsley, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 - RELIEF REQUEST NO. PR-20 (TAC NOS. MB1488 AND MB1489)

Dear Mr. Kingsley:

By letter dated March 14, 2001, Exelon Generation Company, LLC (EGC or the licensee) submitted a request for relief from certain American Society of Mechanical Engineers (ASME) Code Section XI requirements for Dresden Nuclear Power Station, Units 2 and 3. By letter dated August 23, 2001, EGC revised their relief request. Specifically, the licensee requested relief for removing bolting when leakage is detected at bolted connections for control rod drive housings during system pressure tests. In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed an alternative to existing ASME Code Section XI requirements.

The Nuclear Regulatory Commission (NRC) staff has reviewed the licensee's submittals and determined that the proposed alternative contained in Relief Request PR-20 will provide an acceptable level of quality and safety. Therefore, the licensee's proposed alternative is authorized in accordance with 10 CFR 50.55a(a)(3)(i) for the third 10-year inservice inspection interval for Dresden Nuclear Power Station, Units 2 and 3.

The enclosed safety evaluation contains the basis for this determination. This completes the staff's effort for TAC Nos. MB1488 and MB1489.

Sincerely,

/RA/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

Enclosure: Safety Evaluation

cc w/encl: See next page

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**Concurred by Safety Evaluation dated 10/17/01
* see previous concurrence
ADAMS Accession Number: ML012950242

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. PR-20

EXELON GENERATION COMPANY, LLC

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

Inservice inspection (ISI) of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (Code) and applicable addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(6)(g)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the licensee demonstrates that (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code Section XI, “Rules for Inservice Inspection of Nuclear Power Plant Components,” to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) twelve months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. For Dresden Nuclear Power Station, Units 2 and 3 (DNPS), the applicable edition of Section XI of the ASME Code for the third 10-year ISI interval is the 1989 Edition.

By letter dated March 14, 2001, Exelon Generation Company, LLC, the licensee, submitted Relief Request PR-20 for relief from certain ASME Code, Section XI requirements for ISI. The licensee submitted a supplement to this relief request on August 23, 2001, based on discussions with the NRC staff. The information provided by the licensee in support of the request for relief from Code requirements has been evaluated and the basis for disposition is documented below.

Enclosure

2.0 RELIEF REQUEST NO. PR-20

2.1 Components for which Relief is Requested

Code Class:	1
References:	Table IWA-5250(a)(2)
Examination Category:	B-P
Item Number:	B15.ST and B15.OT
Description:	Bolting Removal when Leakage is Detected at Bolted Connection for Control Rod Drive (CRD) Housing during System Pressure Test

2.2 Code Requirement from which Relief is Requested

IWA-5250(a)(2) requires that if leakage occurs at a bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100.

2.2 Licensee's Proposed Alternative to Code

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee proposed conducting annual UT training in accordance with 10 CFR 50.55a(b)(2)(xiv) in lieu of Subsubarticle VII-4240 of Section XI of ASME Code, 1995 Edition with 1996 Addenda, Appendix VII. The annual ultrasonic training would require that all personnel qualified for performing ultrasonic examinations in accordance with Section XI of the ASME Code, Appendix VIII, receive 8 hours of annual hands-on training on specimens that contain cracks. This training must be completed no earlier than 6 months prior to performing ultrasonic examinations at a licensee's facility.

2.3 Licensee's Bases for Requesting Relief and Justification for Granting Relief

The licensee states that CRD housing leakage has been observed at DNPS when the primary system was pressurized prior to reaching normal operating temperature range during system pressure testing.

There are 177 drives in each unit with 8 bolts in each drive. The licensee states that it is current maintenance practice to discard existing bolting and replace with new bolting which has been examined using MT and VT-1. The licensee also states that bolting removed from exchanged drives has not revealed significant degradation caused by inservice corrosion. A number of bolts were rejected during the course of these examinations because of linear indications found in the head-to-shank region. A root cause evaluation concluded that the defects were caused by the manufacturing process at a maximum depth of 0.036 inches with no sign of crack initiation or propagation.

General Electric Co. conducted an analysis for Commonwealth Edison Company in 1991 which concluded that as few as three uniformly distributed and unflawed bolts can support all imposed loads and maintain the applicable ASME Code stress limits. Alternatively, the applicable ASME Code stress limits can be maintained with eight bolts having defects that are 0.157 inches deep and extending 360° around each bolt shank.

The licensee is requesting relief from the requirements of IWA-5250(a)(2) of the 1989 ASME Section XI for CRD bolts on the basis that significant degradation of CRD bolting has not been experienced at DNPS, that significant margin to failure has been shown, and that the replacement of existing bolting with new bolting, or VT-1 examination of existing bolting prior to reinstallation as required by Table IWB-2500-1 on CRD bolting when disassembled for maintenance, will provide an acceptable level of quality and safety. Leakage from CRD housings will be monitored by unidentified drywell leakage and appropriate actions will be taken in accordance with DNPS Technical Specifications. CRD bolts will not be removed for a VT-3 examination when leakage is detected at the CRD housing.

2.4 Proposed Alternative Examination

As an alternative examination, DNPS will replace existing bolting with new bolting, or VT-1 examine existing bolting prior to reinstallation during scheduled drive exchanges. The licensee proposes that the CRD bolts will not be removed for a VT-3 examination when leakage is detected at the CRD housing flange connection during the conduct of the system pressure tests in accordance with IWB-5000.

3.0 STAFF EVALUATION

The discovery of crack indications in CRD bolts (cap screws) was reported by General Electric (GE) in RICSIL No. 019, “CRD Cap Screw Crack Indications,” issued May 19, 1988. The cracks were circumferential in the cap screw shank directly below the cap screw head. GE issued two additional service information letters (SILs) concerning the crack indications in CRD bolts, SIL No. 483, “CRD Cap Screw Crack Indications” on March 17, 1989, and SIL No. 483, Revision 1, “CRD Cap Screw Crack Indications.” Significant degradation of CRD bolting has not been experienced at DNPS, most likely because any leakage that occurs is non-borated water which is non-corrosive. In addition, significant margin to failure has been shown, and the replacement of existing bolting with new bolting, or VT-1 examination of existing bolting prior to reinstallation as required by Table IWB-2500-1 on CRD bolting when disassembled for maintenance, will provide an acceptable level of quality and safety.

4.0 CONCLUSION

Based on the above evaluation, the staff concludes that the removal of bolting when leakage is detected is not required since the leakage is non-corrosive and will be monitored by unidentified drywell leakage, and appropriate actions will be taken in accordance with DNPS Technical Specifications. In addition, significant degradation of CRD bolting has not been experienced at DNPS, significant margin to failure has been shown, and the replacement of existing bolting with new bolting, or VT-1 examination of existing bolting, prior to reinstallation as required by Table IWB-2500-1 on CRD bolting when disassembled for maintenance, will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative PR-20 is authorized for the third 10-year ISI interval for Dresden Nuclear Power Station, Units 2 and 3.

Principal Contributor: J. Davis, EMCB

Date: November 2, 2001