UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 30, 2013

Mr. Michael J. Pacilio
Senior Vice President
Exelon Generation Company, LLC
President and Chief Nuclear Officer
Exelon Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3 - SAFETY EVALUATION IN SUPPORT OF REQUEST FOR RELIEF ASSOCIATED WITH THE FIFTH 10-YEAR INSERVICE INSPECTION INTERVAL PROGRAM (TAC NOS. ME9682, ME9683, ME9684, ME9685, ME9686, ME9687, ME9688, ME9689, ME9690, ME9691, ME9692, ME9693, ME9694, ME9695, ME9696, AND ME9697)

Dear Mr. Pacilio:

By letter dated September 28, 2012, as supplemented by letters dated November 19, 2012, January 10, 2013, January 24, 2013, and June 3, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML12275A070, ML123250319, ML13010A456, ML13025A161, and ML13154A248, respectively) Exelon Generation Company, LLC (the licensee) submitted Relief Requests I5R-01, I5R-02, I5R-03, I5R-04, I5R-05, I5R-07, I5R-10, and I5R-11, to the U.S. Nuclear Regulatory Commission (NRC). The licensee proposed alternatives to or requested relief from certain inservice inspection requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," Dresden Nuclear Power Station (DNPS), Units 2 and 3, for the fifth 10-year ISI interval program; which commenced on January 20, 2013, and will end on January 19, 2023

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(i), the licensee requested to use the proposed alternatives in Relief Requests I5R-05, I5R-07, I5R-10, and I5R-11, on the basis that the alternatives provide an acceptable level of quality and safety. Pursuant to 10 CFR Part 50, Section 50.55a(a)(3)(ii), the licensee requested to use the proposed alternatives in I5R-01, I5R-02, I5R-03, and I5R-04 on the basis that the proposed alternative will provide reasonable assurance of quality and safety of the subject component and compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff has reviewed the subject requests and concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i) for requests I5R-02, I5R-05, I5R-07, I5R-10, and I5R-11, and in CFR 50.55a(a)(3)(ii) for requests I5R-01, I5R-03, and I5R-04, and is in compliance with the ASME Code requirements. Therefore, the NRC staff authorizes alternative requests I5R-01, I5R-02, I5R-03, I5R-04, I5R-05, I5R-07, I5R-10, and I5R-11 at DNPS, Units 2 and 3, for the fifth 10-year ISI interval program. All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third party review by the Authorized Nuclear Inservice Inspector.

If you have any questions on this action, please contact the NRC Senior Project Manager, Brenda Mozafari, at (301) 415-2020.

Sincerely,

Travis L. Tate

Travis L. Tate, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

Enclosures:
Enclosure 1 - Safety Evaluation – Relief Request I5R-01
Enclosure 2 - Safety Evaluation – Relief Request I5R-02
Enclosure 3 - Safety Evaluation – Relief Request I5R-03
Enclosure 4 - Safety Evaluation – Relief Request I5R-04
Enclosure 5 - Safety Evaluation – Relief Request I5R-05
Enclosure 6 - Safety Evaluation – Relief Request I5R-07
Enclosure 7 - Safety Evaluation – Relief Request I5R-10
Enclosure 8 - Safety Evaluation – Relief Request I5R-11

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

ON THE FIFTH 10-YEAR INSERVICE INSPECTION INTERVAL

REQUEST FOR RELIEF I5R-01 REGARDING

INSPECTION OF THE STANDBY LIQUID CONTROL NOZZLE INNER RADIUS

EXELON GENERATION COMPANY, LLC.

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated September 28, 2012 (Agencywide Documents Access and Management System (ADAMS) at Accession No. ML12275A069), Exelon Generation Company, LLC (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, for Dresden Nuclear Power Station (DNPS) , Units 2 and 3. By supplemental letter dated November 19, 2012, (ADAMS Accession No. ML123250319) the licensee superseded it's version of Relief Request I5-01 in its entirety. The request for alternative covers the inspection of the standby liquid control (SBLC) nozzle inner radius where the licensee was unable to achieve essentially 100 percent inspection coverage due to the design of the subject component. Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee requested alternative requirements, for inservice examination on the basis that the ASME Code requirement would result in a hardship without a compensating increase in quality and safety. The licensee provided additional information for Relief Request I5R-01 in a supplemental letter dated June 3, 2013 (ADAMS Accession No. ML13154A248).

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR Section 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 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, which was incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

In 10 CFR 50.55a(a)(3) it states that proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section, or portions thereof, may be used when authorized by the Director of the Office of Nuclear Reactor Regulation or the Director of the Office of New Reactors, as appropriate. Any proposed alternatives must be submitted and authorized prior to implementation.

In 10 CFR 50.55a(a)(3) it further states that alternatives to the requirements of Paragraph (g) may be used, when authorized by the NRC, if: (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.

The licensee has requested an alternative from ASME Code requirements pursuant to 10 CFR 50.55a(a)(3)(ii). The ASME Code of record for DNPS, Unit 2 and 3, fifth 10-year inservice inspection (ISI) interval program is the 2007 Edition, through the 2008 Addenda, of Section XI, of the ASME Code. Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the Commission to authorize the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION FOR RELIEF REQUEST I5R-01

The information provided by the licensee in support of the alternative to the ASME Code requirements has been evaluated and the bases for disposition are documented below.

Request for Relief I5R-1, Revision 1, ASME Code, Section XI, Examination Category B-D, Item B3.100 ASME Code Requirement for Full Penetration Welded Nozzles in Vessels

ASME Code Component

ASME Code Class:	1
Reference:	IWB-2500, Table IWB-2500-1, Figure IWB-2500-7
Examination Category:	B-D
Item Number:	B3.100
Description:	Inspection of Standby Liquid Control Nozzle Inner Radius
Component Numbers:	DNPS, Unit 2: N12-1 DNPS, Unit 3: N12-1
Drawing Numbers:	DNPS, Unit 2: ISI-128, Sheets DNPS, Unit 3: ISI-125, Sheets

ASME Code Requirement

ASME Code, Section XI, Table IWB-2500-1, Figure IWB-2500-7 requires a volumetric examination to be performed on the inner radius section of all reactor pressure vessel nozzles each inspection interval. This includes nozzles with full penetration welds-to-vessel shell (head) cast nozzles, but excludes man ways and holes either welded to or integrally cast in vessel.

Licensee's Basis for Relief Request (As stated)

> Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested on the basis that conformance with the [ASME] Code requirements impose hardship without a compensating increase in the level of quality and safety.
>
> The Standby Liquid Control (SBLC) nozzle, as shown in Figure I5R-01.1, is designed with an integral socket to which the boron injection piping is fillet welded. The SBLC nozzle is located near the bottom of the vessel in an area which is inaccessible for ultrasonic examinations from the inside of the vessel. Therefore, ultrasonic examinations would need to be performed from the outside diameter of the vessel. As shown in Figure I5R-01.1, the ultrasonic scan would need to travel through the full thickness of the vessel into a complex cladding/socket configuration. These geometric and material reflectors inherent in the design prevent a meaningful examination from being performed on the inner radius of the SBLC nozzle.
>
> In addition, the inner radius socket attaches to piping which injects boron at locations far removed from the nozzle. Therefore, the SBLC nozzle inner radius is not subjected to turbulent mixing conditions that are concern at other nozzles.
>
> Compliance with the applicable [ASME] Code requirements would require an ultrasonic examination to be performed on the outside diameter of the reactor pressure vessel. Geometric and material reflectors would prevent a meaningful examination, resulting in inaccurate data. Based on this, the [ASME] Code requirements impose hardship without a compensating increase in the level of quality and safety in accordance with 10 CFR 50.55a(a)(3)(ii).

The NRC staff requested that the licensee discuss how the proposed alternative provides reasonable assurance of structural integrity or leak tightness of the subject components. The licensee provided the information to the NRC staff in its letter dated June 3, 2013.

The licensee's Response (As stated)

> The proposed alternative looks for and verifies structural integrity and leak tightness by ensuring no leakage at nominal operating pressure during the Class 1 System Leak test. This test is a [ASME] Code required exam that verifies no through wall leakage for the entire [ASME Code,] Class 1 boundary. It is performed prior to start-up and unit operation every outage to provide confidence there is no leakage. The Standby Liquid Control nozzle is included as part of this test. Passing of this test with no identified through wall leakage at the nozzle provides reasonable assurance of structural integrity and leak tightness of the Standby Liquid Control nozzle.
>
> During normal plant operation, the Standby Liquid Control nozzle is inaccessible for direct visual observation based on its location in the drywell. This does not preclude the nozzle from being monitored. Leakage from the nozzle would be collected in the drywell sumps, prompting action from the station. No leakage from this nozzle has ever been observed at [DNPS], Units 2 and 3 from this location.

Licensee's Proposed Alternative Examination (As stated)

> As an alternate examination, Exelon Generation Company, LLC will perform a VT-2 visual examination of the subject nozzles at DNPS, Units 2 and 3 each refueling outage in conjunction with the [ASME Code,] Class 1 System Leakage Test.

NRC Staff Evaluation

The ASME Code requires 100 percent volumetric examination to be performed on the inner radius section of all reactor pressure vessel nozzles each inspection interval. This test is an ASME Code required examination that verifies no through-wall leakage for the entire ASME Code Class 1 boundary. This includes nozzles with full penetration welds-to-vessel shell (head) cast nozzles, but excludes man ways and holes either welded to or integrally cast in vessel. However, the NRC staff determined design configurations of the subject welds and the proximity of surrounding area limit access for ultrasonic test (UT) scanning. In order to effectively increase the examination coverage, the nozzle-to-vessel welds would require design modifications and removal of adjacent components. Thus, the NRC staff finds that 100 percent of the ASME Code-required volumetric examinations are considered a hardship.

The SBLC nozzle, as shown in Figure I5R-01.1, provided in the licensee's submittal, is designed with an integral socket to which the boron injection piping is fillet welded. The subject SBLC nozzle is located near the bottom of the vessel in an area which is inaccessible for UT examinations from the internal side of the vessel. The UT examinations would need to be performed from the outside diameter of the vessel. As a result of the configuration the NRC staff determined that in order for the licensee to perform a UT examination, UT scans would be required to travel through the full thickness of the vessel into a complex cladding/socket configuration. The NRC staff finds the licensee would not be able to perform a meaningful examination on the on the inner radius of the SBLC nozzle due to these geometric and material reflectors inherent in the design.

The SBLC nozzle inner radius is not subjected to turbulent mixing conditions that are a concern at other nozzles due to subject nozzle attaches to piping which injects boron at locations far removed from the nozzle. The NRC staff determined that based on the above, the ASME Code requirement to volumetrically examine the subject nozzle would be a hardship without a compensating increase in safety.

The licensee also noted that during normal plant operation, the SBLC nozzle is inaccessible for direct visual observation based on its location in the drywell. This does not preclude the nozzle from being indirectly monitored. Leakage from the nozzle would be collected in the drywell sumps, prompting action from the station. No leakage from this nozzle has ever been observed at DNPS, Units 2 and 3, from this location.

The NRC staff finds the licensee's proposed alternative to perform ASME Code required system leakage tests prior to start-up and unit operation every outage provides reasonable assurance of structural integrity, leak tightness, and no leakage at nominal operating pressure of the subject nozzle, associated piping, and components.

4.0 CONCLUSION

As set forth above, the NRC staff has determined that authorizing the licensee's proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii) is authorized by law and will not endanger life or property, or the common defense and security, and is otherwise in the public interest given due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. Furthermore, the NRC staff concluded that the proposed alternative will provide reasonable assurance of leak tightness of the subject component. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the licensee's proposed alternative contained in Relief Request I5R-01 for DNPS, Units 2 and 3, fifth 10-year ISI interval, which commenced on January 20, 2013, and will end on January 19, 2023.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST NO. I5R-02 REGARDING A RISK-INFORMED INSERVICE INSPECTION

PROGRAM FOR THE FIFTH 10-YEAR INSERVICE INSPECTION INTERVAL

EXELON GENERATION COMPANY, LLC

DRESDEN NUCLEAR POWER STATION, UNITS, 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated September 28, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12275A069), as supplemented by letter dated June 3, 2013 (ADAMS Accession No. ML13154A248), Exelon Generation Company, LLC (the licensee), requested U.S. Nuclear Regulatory Commission (NRC) authorization to extend the risk-informed inservice inspection (RI-ISI) program plan for Dresden Nuclear Power Station (DNPS), Units 2 and 3, to the fifth 10-year inservice inspection (ISI) interval. The DNPS RI-ISI program was initially submitted to the NRC by letter dated October 18, 2000 (ADAMS Accession No. ML003762371), and was approved by the NRC for use in the third 10-year ISI interval by letter dated September 5, 2001 (ADAMS Accession No. ML012050103). The use of the DNPS RI-ISI program was requested for the fourth 10-year interval by the licensee in a letter dated September 6, 2002 (ADAMS Accession No. ML022610153), and subsequently approved by the NRC staff by letter dated September 4, 2003 (ADAMS Accession No. ML032370480).

The licensee has considered relevant information since the development of the original program, and has reviewed and updated the RI-ISI program. The current licensee submittal proposed the continuation of the updated RI-ISI program during the fifth 10-year ISI interval.

2.0 REGULATORY EVALUATION

Pursuant to title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1, 2, and 3, components (including supports) shall meet the requirements, "except design and access provisions and preservice examination requirements" set forth in the ASME Code to the extent practical within the limitations of design, geometry, and materials of construction of the components. Paragraph 10 CFR 50.55a(g) also states that the ISI of the ASME Code, Class 1, 2, and 3, components is to be performed in accordance with Section XI of the ASME Code and applicable addenda, except where specific relief has been granted by the NRC.

The regulations also require during the first 10-year ISI interval, and during subsequent intervals, that the licensee's ISI program complies with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference into 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the conditions listed therein. DNPS is currently in its fifth 10-year ISI interval.

Pursuant to 10 CFR 50.55a(g), a certain percentage of ASME Code categories B-F, B-J, C-F-1 and C-F-2, pressure retaining piping welds must receive ISI during each 10-year ISI interval. The ASME Code requires 100 percent of all B-F welds and 25 percent of all B-J welds greater than 1-inch nominal pipe size be selected for volumetric or surface examination, or both, on the basis of existing stress analyses. For categories C-F-1 and C-F-2 piping welds, 7.5 percent of non-exempt welds are selected for volumetric or surface examination, or both. According to 10 CFR 50.55a(a)(3), the NRC may authorize alternatives to the requirements of 10 CFR 50.55a(g), if an applicant demonstrates that the proposed alternatives would provide an acceptable level of quality and safety, or that compliance with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The NRC staff finds that there is regulatory basis for the licensee to request, and the NRC to authorize this alternative, pursuant to the technical evaluation that follows. The information provided by the licensee in support of the request has been evaluated by the NRC staff and the bases for disposition are documented below.

The NRC staff has developed the following documents to evaluate proposed RI-ISI programs:

Regulatory Guide (RG) 1.174, "*An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis*" (ADAMS Accession No. ML023240437),

RG 1.178, "*An Approach For Plant-Specific Risk-Informed Decisionmaking - Inservice Inspection of Piping*" (ADAMS Accession No. ML032510128), and

RG 1.200, Revision 1, *An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities* (ADAMS Accession No. ML070240001).

RG 1.174 provides guidance on the use of probabilistic risk analysis findings and risk insights in support of licensee requests for changes to a plant's licensing basis. RG 1.178 describes an RI-ISI program as one that incorporates risk insights that can focus inspections on more important locations, while at the same time maintaining or improving public health and safety. RG 1.200 describes one acceptable approach for determining whether the quality of the probabilistic risk assessment (PRA), in total or the parts that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making.

3.0 TECHNICAL EVALUATION FOR RELIEF REQUEST I5R-02

Licensee's Proposed Alternative to ASME Code

The licensee is proposing to continue use of the DNPS RI-ISI program plan in the fifth 10-year ISI interval as an alternative to the current ASME Code, Section XI, 2007 Edition through the 2008 Addenda, examination requirements for Class 1 examination categories B-F and B-J piping welds and Class 2 examination categories C-F-1 and C-F-2 piping welds. The proposed alternative is sought for the DNPS fifth 10-year ISI interval which began on January 20, 2013, and is scheduled to end January 19, 2023.

The licensee's process used to develop the initial RI-ISI program was based on Electric Power Research Institute, Inc. (EPRI) Topical Report TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Revision B-A (Reference 1). The alternative will continue to use the same two enhancements proposed and approved in the previous intervals RI-ISI program.

The licensee stated that in lieu of the evaluation and sample expansion requirements in Section 3.6.6.2, "RI-ISI Selected Examinations" of Reference 1, the requirements of Subarticle-2430, "Additional Examinations" contained in ASME Code Case N-578-1 (Reference 3), will be used as the first enhancement. The second enhancement proposed by the licensee is to use Table 1, Examination Category R-A, "Risk-Informed Piping Examinations" contained in Code Case N-578-1 as an alternative to the requirements listed in Table 4-1, "Summary of Degradation-Specific Inspection Requirements and Examination Methods" of Reference 1.

NRC Staff Evaluation

The NRC staff has reviewed and evaluated the licensee's proposed RI-ISI program, including those portions related to the applicable methodology and processes, based on guidance and acceptance guidelines provided in RGs 1.174 and 1.178, in Standard Review Plan 3.9.8, and in the EPRI-TR-112657, Revision B-A. An acceptable RI-ISI program plan is expected to meet the five key principles discussed in RGs 1.174 and 1.178, SRP 3.9.8, and the EPRI-TR, as stated below:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins.
4. When proposed changes result in an increase in core damage frequency (CDF) or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
5. The impact of the proposed change should be monitored by using performance measurement strategies.

The NRC staff determined the first principle is met in this relief request because an alternative ISI program may be authorized pursuant to 10 CFR 50.55a(3)(i) and, therefore, an exemption request is not required.

The second and third principles require assurance that the alternative program is consistent with the defense-in-depth philosophy and that sufficient safety margins are maintained, respectively. Assurance that the second and third principles are met is based on the application of the approved methodology and not on the particular inspection locations selected. The licensee stated that they are using the same methodology as the original RI-ISI submittal. The licensee also stated in the June 3, 2013, submittal, that the augmented inspection programs associated with intergranular stress corrosion cracking, service water integrity, flow accelerated corrosion and high energy line breaks remain unaffected by the fifth interval RI-ISI program. The approved methodology was applied to the piping added in accordance with the 2007 Edition through the 2008 Addenda of ASME Code, Section XI. Since the NRC staff determined that the methodology used to develop the RI-ISI program for the fifth 10-year ISI interval is unchanged from the methodology approved for development of the RI-ISI program used in the third and fourth 10-year ISI interval and the augmented programs remain unchanged, the second and third principles are met.

The fourth principle requires that any increase in CDF and risk are small and consistent with the Commission's Safety Goal Policy Statement, needs an estimate of the change in risk. The change in risk estimate is dependent on the location of inspections in the proposed ISI program compared to the location of inspections that would be performed using the requirements of ASME Code, Section XI. The NRC staff has previously determined that it is not necessary to develop a new deterministic ASME program for each new 10-year interval but, instead, it is acceptable to compare the new proposed RI-ISI program with the last deterministic ASME program. The licensee states that a new risk impact analysis was performed. The fifth interval update of the risk impact assessment provided in the response to the NRC request for additional information (RAI) represents a change of 4.62E-09 for Unit 2, and 3.00E-09 for Unit 3, with regards to CDF and 2.03E-09 for Unit 2, and 9.85E-09 for Unit 3, with regards to the large early release frequency. These values satisfy the acceptance criteria of RG 1.174 and EPRI TR-112657 when compared to the last deterministic Section XI inspection program. Thus, the NRC staff finds that the licensee's analysis provides assurance that the fourth key principle is met.

The fourth principle also requires demonstration of the technical adequacy of the PRA. As discussed in RGs 1.178 and 1.200, an acceptable change in risk evaluation (and risk-ranking evaluation used to identify the most risk significant locations) requires the use of a PRA of appropriate technical quality that models the as-built and as-operated plant. A review of the DNPS PRA was conducted under the auspices of the Boiling Water Reactor Owners' Group peer review in January 2001. The licensee states there were no significant level A Facts and Observations (F&Os) from the peer review and all significant level B F&Os were addressed and closed out with the completion of a DNPS PRA model update in 2005. The DNPS PRA was updated in 2005 and 2009 and assessments of the status of gap analysis relative to the new models and the requirements in Addendum B of the ASME Code PRA standard were completed after each update.

In the submittal, the license referred to EPRI TR-1021467, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs" (Reference 2), which received a safety evaluation (SE) from NRC in January 2012 (ADAMS Accession No. ML11325A340). This topical report provides guidance on determining the technical adequacy of PRAs used to develop a RI-ISI program that utilizes the traditional methodology as described in EPRI TR-112657, Revision B-A. The licensee states that the DNPS PRA contains three supporting requirements with gaps to Capability Category (CC) II of the ASME/ANS RA-Sb-2005 but did not indicate the assessed CC. According to NRC's SE for EPRI TR-1021467, the supporting requirements with gaps identified in the submittal only require CC I and do not require CC II; therefore, a CC I is sufficient and acceptable. From the description of the three gaps provided in the submittal, the NRC staff concludes that these gaps are related to those characteristics that are needed to meet CC II requirements, such as using plant specific operational records, and finds that the discussion indicates the supporting requirements are met at CC I. Therefore, consistent with the guidelines in EPRI TR-1021467, the NRC staff finds the DNPS PRA model suitable for use in this RI-ISI application.

The fifth principle of risk-informed decision making requires that the impact of the proposed change be monitored by using performance measurement strategies. The RI-ISI program is a living program and, as such, is subject to periodic reviews. The licensee indicates that the Consequence Evaluation, Degradation Mechanism Assessment, Risk Ranking, Element Selection and Risk Impact Assessment steps encompass the living program process applied to the DNPS RI-ISI program. The June 3, 2013, submittal, stated that examination locations for the fifth interval had changed based on PRA Model revisions, plant modifications and to optimize code examination coverage. Therefore, the NRC staff finds that the licensee's proposed alternative provides assurance the fifth principle is met.

Based on the above discussion, the NRC staff concludes that the five key principles of risk-informed decision making are ensured by the licensee's proposed fifth 10-year RI-ISI program, and, therefore, the proposed program is acceptable.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a, and is in compliance with the ASME Code's requirements. Therefore, the NRC staff authorizes continued use of the RI-ISI program as described in Relief Request I5R-02 at DNPS, Units 2 and 3, for the fifth 10-year ISI interval, which commenced on January 20, 2013, and will end on January 19, 2023. All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including third-party review by Authorized Nuclear Inservice Inspector.

5.0 REFERENCES

1. EPRI TR-112657, Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure, Final Report," December 1999.

2. EPRI TR-1021467, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs," June 18, 2012.

3. ASME Code Case N-578-1, Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B, Section XI Division 1, ASME, New York, New York, March 28, 2000.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST I5R-03 REGARDING PRESSURE TESTING OF

REACTOR PRESSURE VESSEL HEAD FLANGE SEAL LEAK DETECTION SYSTEM

EXELON GENERATION COMPANY, LLC

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated September 28, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12275A069), as supplemented by letter dated January 10, 2013 (ADAMS Accession No. ML13010A456), Exelon Generation Company, LLC (the licensee) submitted a Relief Request I5R-03 for the U.S. Nuclear Regulatory Commission (NRC) approval. The licensee proposed an alternative to certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. Relief Request I5R-03 relates to the system leakage test of the reactor pressure vessel head (RPVH) flange seal leak detection (leakoff) system. Relief Request I5R-03 is requested for the fifth 10-year Inservice Inspection (ISI) interval of the Dresden Nuclear Power Station (DNPS), Units 2 and 3, which commenced on January 20, 2013, and will end on January 19, 2023.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(ii), the licensee proposed alternative system leakage test for the RPVH flange seal leak detection line system, on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATIONS

Section 50 55a(g)(4) requires that ASME Code Class 1, 2, and 3, components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for ISI 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 in-service 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), 12 months prior to the start of the 120-month interval, subject to the conditions listed therein.

Section 10 CFR 50.55a(a)(3) states, in part, that alternatives to the requirements of Paragraph (g) or portions thereof, 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.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the NRC to authorize the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION FOR RELIEF REQUEST I5R-03

ASME Code Components Affected (As stated)

ASME Code Class:	Class 2
Examination Category:	C-H, Table IWC-2500-1
Item No.:	C7.10
Component:	Flange Seal Leak Detection Line Pressure Retaining Components
System:	Reactor Pressure Vessel Head Flange Seal Leak Detection System

The component for which RFA I5R-03 is applicable is listed below:

Plant	Drawing
Dresden, Unit 2	M-26 Sh. 1
Dresden, Unit 3	M-357 Sh. 1

Applicable Code Edition and Addenda (As stated)

The ASME code of record for the fifth 10-year ISI interval at DNPS, Units 2 and 3, is the 2007 Edition through 2008 Addenda of the ASME Code, Section XI.

Applicable Code Requirement (As stated)

The ASME Code, Section XI, Table IWC-2500-1, Examination Category C-H, Item No. C7.10, requires that all Class 2 pressure retaining components are subjected to a system leakage test with a visual (VT-2) examination in accordance with IWC-5220. This pressure test is to be conducted once each inspection period.

Reason for Request (As stated)

The RPVH flange leak detection line is separated from the reactor pressure boundary by one passive membrane, a silver plated O-ring located on the vessel flange. A second O-ring is located on the opposite side of the tap in the vessel flange. This line is required during plant operation in order to indicate failure of the inner flange seal O-ring. Failure of the O-ring would result in the annunciation of a high level alarm in the control room. On this annunciation, control room operators would quantify the leakage rate from the O-ring and then isolate the leak detection line from the drywell sump by closing the valve. This action is taken in order to prevent steam cutting the O-ring and the vessel flange. Failure of the inner O-ring is the only condition under which this line is pressurized.

The configuration of this system precludes manual testing while the vessel head is removed, because the odd configuration of the vessel tap combined with the small size of the tap and the high test pressure requirement (1000 pounds per square inch gage (psig) minimum) prevents the tap in the flange from being temporarily plugged. The opening in the flange is only 3/16 inch in diameter and smooth walled, making a high pressure temporary seal very difficult. Failure of this seal could possibly cause ejection of the device used for plugging into the vessel.
A pneumatic test performed with the head installed is precluded due to the configuration of the top head. The top head of the vessel contains two grooves that hold the O-rings. The O-rings are held in place by a series of retainer clips spaced 15 degrees apart. The retainer clips are contained in a recessed cavity in the top head. If a pressure test was performed with the head on, the inner O-ring would be pressurized in a direction opposite to what it would see in normal operation. This test pressure would result in a net inward force on the O-ring that would tend to push it into the recessed cavity that houses the retainer clips. The O-ring material is only 0.050 inch thick with a silver plating thickness of 0.004 inch to 0.006 inch, and could very likely be damaged by this deformation into the recessed areas on the top head.

In addition to the problems associated with the O-ring design that preclude this testing, it is also questionable whether a pneumatic test is appropriate for this line. Although the line will initially contain steam if the inner O-ring leaks, the system actually detects leakage rate by measuring the level of condensate in a collection chamber. This would make the system medium water at the level switch. Finally, the use of a pneumatic test performed at a minimum of 1000 psig would represent an unnecessary risk in safety for the inspectors and test engineers in the unlikely event of a test failure, due to the large amount of stored energy contained in air pressurized to 1000 psig.

Proposed Alternative and Basis for Use (As stated)

The VT-2 examination will be performed on the RPVH flange seal leak detection line during vessel flood-up during a refueling outage. The static head developed due to the water above the vessel flange will allow for the detection of any gross leakage in the line. This examination will be performed with the frequency specified in Table IWC-2500-1 for the system leakage test (i.e., once each inspection period).

Duration of Relief (As stated)

The Relief Request I5R-03 is applicable for the fifth 10-year ISI interval of DNPS, Units 2 and 3, which commenced on January 20, 2013, and will end on January 19, 2023.

4.0 NRC STAFF EVALUATION

The NRC staff has evaluated Relief Request I5R-03 pursuant to 10 CFR 50.55a(a)(3)(ii). The NRC staff evaluation focuses on whether the compliance with the specified requirements of 10 CFR 50.55a(a)(g), or portions thereof, would result in hardship or unusual difficulty, and if there is a compensating increase in the level of quality and safety despite the hardship. The

ASME Code, Section XI, Table IWC-2500-1, Examination Category C-H, requires that all Class 2 pressure retaining components are subjected to a system leakage test and a VT-2 examination in accordance with IWC-5220 each inspection period. For the system leakage test of the RPVH flange seal leak detection (leakoff) line system, the licensee proposed an alternative (i.e., pressure) to the above requirements (i.e., testing at system operating pressure) on the basis that the specified requirements would result in hardship or unusual difficulty. The proposed alternative is to perform the system leak test using static pressure head developed by water above the vessel flange when the vessel is flooded for refueling during the outage. The frequency of examination will be once each inspection period in accordance with the Table IWC-2500-1 requirement.

Within the context of Relief Request I5R-03, the licensee specified the limitations that precluded the IWC-5220 system leak test. The licensee stated that the leak off line is design such that the inner O-ring seal has to be failed in order to perform the required leak test. The licensee has considered performing a pneumatic pressure test with the RPVH on. The licensee stated the issue with this type of test is that the inner O-ring is pressurized in the direction that is opposite to the direction that is pressurized during the normal operation. The resulting net inward force from the pressure tends to push the O-ring into the recessed cavity that houses the retainer clips holding the O-ring in place. The NRC staff determined that the unusual forced deformation of the O-ring into the recessed cavity would very likely damage the O-ring and the examination would, in all likelihood, be unsuccessful. The licensee has also considered temporarily plugging the tap in the flange and pressurizing the line manually with the RPVH off. The licensee stated the issue with this type of test is that the leakoff line opening in the flange is smooth walled making a high pressure temporary seal very challenging. The NRC staff determined that the temporary seal would very likely fail causing ejection of the plugging device into the vessel. In addition, the NRC staff determined the use of a pneumatic test would expose the personnel conducting the leak test to an unnecessary risk in the unlikely event of a test failure due to presence of large amount of stored energy in the pressurized air. The NRC staff finds that the limitations provided by the licensee and discussed above constitute a justifiable hardship if the leak test were to be performed in accordance with the pressure requirement of IWC-5220.

In its response dated January 10, 2013 to the NRC staff's request for additional information, the licensee provided an estimate for personnel radiation exposure if the above mentioned manual pressurization of the RPVH flange leak detection line would be performed. The licensee stated that for performance of the examination and resulting corrective maintenance, the personnel would be exposed up to 75 millirem per hour radiation field with total station exposure estimated to be up to 7.3 rem radiation dose. The licensee stated that this estimate to radiation exposure is based on a single RPVH disassembly and reassembly for refueling in the previous outages. The licensee stated that exposure to the above mentioned levels of radiation is not in accordance with industry principles of maintaining personnel radiation exposure as low as reasonably achievable (ALARA). The NRC staff notes that a large part of this radiation dose would be incurred during a routine refueling outage, if the test was conducted prior to initial removal of the reactor vessel head, and that all of the dose would be incurred if the test were conducted after the reactor was refueled. If the test was conducted prior to initial head removal, significant personnel safety issues exist while the plant is at normal operating temperature. The NRC staff finds that personnel exposure to levels of radiation above ALARA and potential personnel safety impacts constitute a justifiable hardship.

In addition, the NRC staff notes that during normal operating conditions over the life of the plant, the likelihood that the RPVH flange leakoff line piping components be exposed to environments and operational conditions that cause potential materials degradation is very low, because this line would have only been used in the event of inner O-ring seal failure and leak.

In summary, the NRC staff finds that the licensee's proposed alternative leak test (i.e., using static pressure head developed by water above the vessel flange when the vessel is flooded for refueling) accompanied with a VT-2 examination each inspection period is adequate to detect leakage in the RPVH flange seal leak detection line system and provides reasonable assurance of structural integrity or leak tightness of the system. The NRC staff has also determined that complying with the requirements would result in hardship or unusual difficulty due to potential personnel safety, equipment damage, and personnel exposure to radiation. Therefore, the NRC staff authorizes RFA I5R-03.

5.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity or leak tightness of the subject components and complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the use of Relief Request I5R-03 for the fifth 10-year ISI interval of DNPS, Units 2 and 3, which commenced on January 20, 2013, and will end on January 19, 2023.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third-party review by the Authorized Nuclear In-service Inspector.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
REQUEST FOR ALTERNATIVE I5R-04 REGARDING SYSTEM LEAKAGE TEST FOR ISOLATION CONDENSER SHELL SIDE AND ASSOCIATED PIPING
EXELON GENERATION COMPANY, LLC
DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3
DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated September 28, 2012 (Agencywide Documents Access and Management Systems (ADAMS) Accession No. ML12275A069), as supplemented by letter dated January 24, 2013 (ADAMS Accession No. ML13025A161), Exelon Generation Company (the licensee) submitted for the U.S. Nuclear Regulatory Commission (NRC) approval the Relief Request I5R-04. The licensee superseded Relief Request I5R-04. The licensee proposed an alternative to certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI. Relief Request I5R-04 is related to system leakage test of the isolation condenser (IC) shell side and associated piping. The licensee submitted Relief Request I5R-04 for the fifth 10-year inservice inspection (ISI) interval of the Dresden Nuclear Power Station (DNPS), Units 2 and 3, which commenced on January 20, 2013, and will end on January 19, 2023.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(ii), the licensee proposed an alternative system leakage test for the IC system (i.e., IC shell side and associated piping), on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

2.0 REGULATORY EVALUATION

In 10 CFR 50.55a(g)(4) it specifies that ASME Code Class 1, 2, and 3, components (including supports) must meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for In-service Inspection (ISI) 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), 12 months prior to the start of the 120-month interval, subject to the condition listed therein.

Section10 CFR 50.55a(a)(3) states, in part, that alternatives to the requirements of Paragraph (g) may be used, when authorized by the NRC, if the licensee demonstrates (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.

3.0 TECHNICAL EVALUATION FOR RELIEF REQUEST

ASME Code Components Affected (As stated)

ASME Code Class:	Class 3
Examination Category:	D-B, Table IWD-2500-1
Item No.:	D2.10
Component:	Isolation Condenser Shell Side and associated piping
System:	The Isolation Condenser System

The component for which an alternative is proposed is listed below.

Dresden, Unit No.	Drawing	Test Block No.
2	M-28, M-39	2ICO1, 2IC02
3	M-359, M-369	3ICO1, 3IC02

Applicable Code Edition and Addenda (As stated)

The ASME code of record for the fifth 10-year ISI interval at DNPS, Units 2 and 3, is the 2007 Edition and 2008 Addenda of the ASME Code, Section XI.

Applicable Code Requirement (As stated)

The ASME Code, Section XI, Table IWD-2500-1, Examination Category D-B, Item No. D2.10, requires all Class 3 pressure retaining components be subject to a system leakage test with a visual (VT-2) examination in accordance with IWD-5220. This pressure test is to be conducted once each inspection period.

In IWD-5221, "Pressure," it states that system leakage test shall be conducted at the system pressure obtained while the system, or portion of the system, is in service performing its normal operating function or at the system pressure developed during a test conducted to verify system operability (e.g., to demonstrate system safety function or satisfy technical specification surveillance requirements).

Reason for Request (As stated)

The IC is not normally in service. It is normally in a standby alignment with its shell side vented to the atmosphere through a non-isolable vent line. For the ISI purposes, the IC is divided into two test blocks (e.g., 2IC01 and 2IC02 for the Unit 2 IC upper and lower portions, respectively). The system is normally aligned with the IC shell side water level greater than or equal to 6 feet in accordance with DNPS, Units 2 and 3, technical specifications (TSs). However, the shell side water level of the IC is administratively maintained between 7 feet and 7.4 feet in accordance with plant procedures.

The IC system operability is verified through performance of TS surveillance requirement (SR) 3.5.3.4, "Isolation Condenser System Heat Removal Capability Verification Test," every 60 months (i.e., during two of the three inspection periods for the fifth 10-year ISI interval). The system pressures developed during the performance of this TS SR meet the requirements of IWD-5221. However, the 60-month frequency does not meet the Table IWD-2500-1 examination frequency requirement of once per inspection period.

Performance of an additional IC heat removal capability test solely for the purposes of performing a system leakage test requires a minimum of a 25 percent reduction in reactor power to perform the examination. This presents an unnecessary transient on the affected DNPS reactor and a challenge to station operators. Additionally, as previously stated, the IC shell cannot be isolated and pressurized to meet IWD-5221 examination pressure requirements when in a standby alignment. It would be an abnormal activity to fill the IC to the top simply to achieve a slight increase in static head for the additional system leakage test. As an additional complication, any water added to the IC shell to raise the level above the normal operating level would subsequently have to be drained and processed as radwaste radioactive waste.

Imposing a large transient on the reactor plant to verify the performance of the IC heat removal capability at a greater frequency than required by the DNPS TS and filling the IC to the top to perform a system leakage test presents a hardship without a compensating increase in quality and safety.

Proposed Alternative and Basis for Use (As stated)

As an alternative, the licensee proposed the performance of a system leakage test with a VT-2 examination of all DNPS, Units 2 and 3, IC Test Blocks (i.e., Test Blocks 2(3)IC01 and 2(3)IC02)) every 60 months during the performance of TS SR 3.5.3.4 versus once per inspection period as required by Table IWD-2500-1.

During inspection periods where TS SR 3.5.3.4 is not performed on the unit's isolation condenser, an additional VT-2 examination will be performed for Test Blocks 2(3)IC02 (i.e., the lower portion of the ICs and associated piping). This leakage test will be performed with IC shell side water level at the normal standby level versus at the normal pressure when the system is in service performing its normal operating function or at the system pressure developed during a test conducted to verify system operability as discussed in IWD-5221.

Duration of Relief (As stated)

The Relief Request I5R-04 is applicable for the fifth 10-year ISI interval of DNPS, Units 2 and 3, which commenced on January 20, 2013, and will end on January 19, 2023.

NRC Staff Evaluation

The NRC staff has evaluated Relief Request I5R-04 pursuant to 10 CFR 50.55a(a)(3)(ii). The NRC staff evaluation focuses on whether compliance with the specified requirements of 10 CFR 50.55a(g), or portions thereof, would result in hardship or unusual difficulty, and if there is a compensating increase in the level of quality and safety despite the hardship. According to the ASME Code, Section XI, (i.e., Table IWD-2500-1, Examination Category D-B, Item No. D2.10), all Class 3 pressure retaining components are to be subjected to the system leakage test and the VT-2 examinations each inspection period. The pressure requirement for conducting the system leakage test is specified in IWD-5221. The licensee proposed to conduct the system leakage test of the IC shell and associated piping (i.e., Test Blocks 2IC01 and 2IC02 for the DNPS, Unit 2, IC upper and lower portions, respectively, and Test Blocks 3IC01 and 3IC02 for the DNPS, Unit 3, IC upper and lower portions, respectively) during performance of TS SR 3.5.3.4 "Isolation Condenser System Heat Removal Capability Verification Test" every 60 months instead of the required each inspection period. The licensee performs the TS SR 3.5.3.4 test to verify the IC system operability when in service performing its normal operating function. The NRC staff notes that the performance of TS SR 3.5.3.4 test every 60 months meets the requirements of IWD-5221, however, the frequency of test does not meet the requirements of Table IWD-2500-1.

The NRC staff acknowledges that the current request, Relief Request I5R-04 (submitted for the fifth 10-year ISI interval of DNPS, Units 2 and 3), is a continuance of the request I4R-06 that was authorized by the NRC on September 4, 2003, (ADAMS Accession No. ML032370480) for the fourth 10-year ISI interval of DNPS, Units 2 and 3. Within context of Relief Request I5R-04, the NRC staff has determined that the licensee provided adequate description and technical information to support the basis for a hardship or unusual difficulty. The basis includes causing an unnecessary reactor transient if the TS SR 3.5.3.4 test was performed each inspection period or creating excess radioactive waste if the IC was filled to the top each inspection period. The NRC staff finds that imposing an unnecessary transient on the reactor or creating excess radioactive waste constitutes a justifiable hardship or unusual difficulty.

In addition, the licensee will conduct the VT-2 examinations of the lower portion of the IC system (i.e., Test Blocks 2IC01 and 2IC02 for Dresden, Unit 2, and Test Blocks 3IC01 and 3IC02 for DNPS, Unit 3) with the IC shell side water level at normal standby level during one out of three inspection periods of the fifth 10-year ISI interval. The NRC staff notes that this examination will provide additional assurance that any leakage, if it were to occur, would be detected during inspection periods in which the TS SR 3.5.3.4 test is not performed.

In summary, the NRC staff finds that the licensee's proposed alternative frequency of examination (i.e., every 60 months) with pressure requirement of IWD-5221 for the system leak test is adequate to detect leakage in the subject components and provides reasonable assurance of structural integrity or leak tightness of the IC system. The NRC staff has determined that complying with the requirements would result in hardship to the licensee without a compensating increase in the level of quality and safety. Therefore, the NRC staff authorizes the licensees alternative for Relief Request I5R-04.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity or leak tightness of the subject components and complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(ii). Therefore, the NRC staff authorizes the Relief Request I5R-04 at DNPS, Units 2 and 3, for the fifth 10-year ISI interval, which commenced on January 20, 2013, and will end on January 19, 2023.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized herein by the NRC staff remain applicable, including the third-party review by the Authorized Nuclear In-service Inspector.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

PROPOSED ALTERNATIVE FOR RELIEF REQUEST I5R-05, CONTINUOUS PRESSURE

MONITORING OF THE CONTROL ROD DRIVE SYSTEM ACCUMULATORS

EXELON GENERATION COMPANY, LLC

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated September 28, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12275A069), Exelon Generation Company, LLC (the licensee) submitted proposed alternative I5R-05, "Request for Relief for Continuous Pressure Monitoring of the control rod rrive (CRD) System Accumulators," for U S Nuclear Regulatory Commission (NRC) review and authorization. Specifically, the licensee requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) for performing a system leakage test visual (VT-2) examination of the nitrogen side of the control rod drive (CRD) accumulators, including the attached piping, at Dresden Nuclear Power Station (DNPS), Units 2 and 3. The licensee states that continuous monitoring of CRD accumulator pressure functions as a pressure decay type test and that technical specification surveillance requirements provides greater pressure monitoring than the ASME Code requirement for a VT-2 examination. The licensee requested authorization to use the proposed alternative pursuant to Title 10 of the *Code of Federal Regulations* Part 50 (10 CFR), Section 55a(a)(3)(i), on the basis that the proposed alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), *Inservice Inspection Requirements*, ASME Code Class 1, 2, and 3, components (including supports) shall meet the requirements, except the design and access provisions and the pre-service 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 inspection interval and subsequent 10-year inspection 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) 12 months prior to the start of the 120-month inspection interval, subject to the conditions listed therein.

Section 55a(a)(3) of 10 CFR 50 states, in part, that alternatives to the requirements of 10 CFR 50.55a(g) may be used when authorized by the NRC, if (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.

The NRC staff finds that the regulatory authority exists to authorize the licensee's proposed alternative to the ASME Code requirement on the basis that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff has reviewed and evaluated the licensee's request pursuant to 10 CFR 50.55a(a)(3)(i).

3.0 TECHNICAL EVALUATION FOR RELIEF REQUEST I5R-05

Components for which Relief is Being Requested

Nitrogen side of the CRD accumulators including the attached piping, Examination Category C-H, Item No. C7.10

ASME Code Requirements

The ASME code of record for DNPS, Units 2 and 3, for the fifth 10-year inservice inspection (ISI) interval that is scheduled to commence on January 20, 2013, and scheduled to end on January 19, 2023, is the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI.

Paragraph IWC-2500, Table IWC-2500-1, Examination Category C-H, Item No. C7.10, requires that all Class 2 pressure retaining components be subject to a system leakage test in accordance with IWC-5220 with a VT-2 examination once each inspection period.

Licensee's Proposed Alternative

As an alternate to the VT-2 visual examination requirements, the licensee proposes to perform continuous pressure monitoring of the nitrogen side of the CRD accumulators.

The licensee has cited the following precedents in support of the proposed alternative:

Dresden Nuclear Power Station, Units 2 and 3, Fourth ISI Interval, Relief Request I4R-07. (ADAMS Accession No. ML032370480)

LaSalle County Nuclear Power Station, Units 1 and 2, Third ISI Interval, Relief Request I3R-09. (ADAMS Accession No. ML073610587)

Quad Cities Nuclear Power Station, Units 1 and 2, Fourth ISI Interval, Relief Request I4R-06. (ADAMS Accession No. ML033560386)

Licensee's Basis for Requesting Relief

As required by the TSs at DNPS, Units 2 and 3, the CRD system accumulator pressure must be greater than or equal to 940 pounds per square inch gauge (psig) to be considered operable. During normal operation the accumulators are isolated from the source of make-up nitrogen and the accumulator pressure is continuously monitored by system instrumentation. Should accumulator pressure fall below 1000 psig, an alarm is received in the control room. Since the accumulators are isolated from the source of make-up nitrogen, continuous monitoring of pressure functions as a pressure decay type test.

If an alarm is received in the control room, the pressure drop for the associated accumulator is recorded and the accumulator is recharged in accordance with DNPS procedures. If an accumulator requires charging more than twice in a 30-day period, then a leak check is performed to determine the cause of the pressure loss. When leakage is detected, corrective actions are taken to repair the leaking component, as required by DNPS procedures. The licensee stated that an additional VT-2 examination performed once per inspection period would not provide an increase in quality and safety

The licensee stated that relief is requested from the VT-2 examination requirement specified in Table IWC-2500-1 for the nitrogen side of the CRD system accumulators, including attached piping on the basis that DNPS TS surveillance requirements provides greater pressure monitoring than the ASME Code requirement for a VT-2 examination.

NRC Staff Evaluation

The ASME Code requires that a VT-2 examination is performed on all Class 2 pressure retaining components once during each inspection period. The VT-2 examination is performed at normal operating system pressure with the fluid in the system serving as the pressurizing medium. In order to perform a VT-2 examination of the nitrogen side of the CRD accumulators and the attached piping, it is necessary to apply a soap solution to all surfaces of the subject components and visually examine the surfaces for soap bubbles that would indicate leakage. If a leak occurs between successive visual examinations and continuous pressure monitoring is not employed, the accumulator pressure could drop below the pressure required by the plant TSs for operability for an extended time before it is detected.

As an alternative to the ASME Code-required VT-2 examinations, the licensee has proposed to utilize the continuous on-line pressure monitoring that is currently used to confirm pressure is consistent with required plant TS requirements. If the nominal pressure falls below 1000 psig, an alarm is triggered in the plant control room and plant corrective action is taken to re-charge the specific accumulator to maintain its function. If pressure decay below 1000 psig is observed more than two times in any 30-day period, plant corrective action is to perform a leak check to determine the cause of the pressure loss and, when leakage is detected, perform corrective actions to repair the leaking component.

The proposed alternative requires that the pressure of each accumulator is continuously monitored by system instrumentation. Because each accumulator is isolated from the source of make-up nitrogen, the NRC staff finds that continuous monitoring of the CRD accumulator pressure functions as a pressure decay type test and any leakage from the accumulator would be detected by normal system instrumentation. The NRC staff finds that continuous pressure monitoring ensures immediate detection of large leaks and long-term continuous monitoring results in high-detection sensitivity for even small leaks. Furthermore, continuous monitoring ensures that the CRD system accumulator minimum pressure of 940 psig required plant TSs for operability, is also continuously monitored and maintained. The NRC staff finds that continuous on-line monitoring of the CRD system accumulator pressure and the required corrective actions when leakage is detected provide reasonable assurance that adequate pressure for CRD actuation is maintained. As such, the licensee's proposed alternative provides an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that Relief Request I5R-05, "Request for Relief for Continuous Pressure Monitoring of the Control Rod Drive (CRD) System Accumulators," provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i) and authorizes use of the proposed alternative at DNPS, Units 2 and 3, during the fifth 10-year ISI interval that is scheduled to commence on January 20, 2013, and scheduled to end on January 19, 2023.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and authorized in the subject proposed alternative remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELIEF REQUEST FOR RELIEF I5R-07 REGARDING
BOILING WATER REACTOR VESSEL AND INTERNAL PROJECT GUIDELINES
DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3
DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated September 28, 2012 Agencywide Documents Access and Management System (ADAMS) Accession No. ML12275A069, as supplemented by letter dated November 19, 2012 ADAMS Accession No. ML123250319, Exelon Generation Company, LLC (the licensee) submitted Relief Request I5R-07 for its fifth 10-year inservice inspection (ISI) interval program for its reactor vessel internals (RVI) components at Dresden Nuclear Power Station (DNPS), Units 2 and 3. In this safety evaluation (SE), the term "RVI components" include reactor pressure vessel interior surfaces, attachments, and core support structures. In this request, the licensee proposed to use Boiling Water Reactor Vessel and Internals Project (BWRVIP) guidelines as an alternative to certain requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME Code) for the ISI of reactor pressure vessel interior surfaces, attachments, and core support structures.

2.0 REGULATORY REQUIREMENTS

The ISI of ASME Code Class 1, 2, and 3, components is performed in accordance with Section XI of the ASME Code, as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g), except where specific relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of 10 CFR 50.55a(g) may be used when authorized by the NRC if: (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 pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for ISI of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations states that ISI examination of components and system pressure tests conducted during the first 10-year ISI 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), 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable ASME Codes of record for the fifth 10-year ISI interval for DNPS, Units 2 and 3, is ASME Code, Section XI, 2007 Edition through the 2008 Addenda.

3.0 TECHNICAL EVALUATION FOR RELIEF REQUEST I5-07

The Components for Which an Alternative is Requested

The ASME Code, Section XI, Class 1, examination categories B-N-1 and B-N-2, Code Item Nos. B13.10; Vessel Interior B13.20; Interior Attachments within Beltline Region B13.30; Interior Attachments Beyond Beltline Region; and, Core Support Structure B13.40.

Examination Requirements from Which an Alternative is Requested

The ASME Code, Section XI, requires the visual (VT) examination of certain RVI components. These examinations are included in Table IWB-2500-1, Categories B-N-1 and B-N-2, and identified with the following item numbers:

- B13.10 - Examine accessible areas of the RV interior each period using a technique which meets the requirements for a VT-3 examination, as defined in paragraph IWA-2213 of the ASME Code, Section XI.
- B13.20 - Examine interior attachment welds within the beltline region each interval using a technique which meets the requirements for a VT-1 examination as defined in paragraph IWA-2211 of the ASME Code, Section XI.
- B13.30 - Examine interior attachment welds beyond the beltline region each interval using a technique which meets the requirements for a VT-3 examination, as defined in paragraph IWA-2213 of the ASME Code, Section XI.
- B13.40 - Examine surfaces of the core support structure each interval using a technique which meets the requirements for a VT-3 examination, as defined in paragraph IWA-2213 of the ASME Code, Section XI.

These examinations are performed to assess the structural integrity of the reactor pressure vessel interior surfaces, attachments, and core support structures.

Licensee's Basis for Requesting an Alternative and Justification for Granting Relief

In Relief Request I5R-07, the licensee, in lieu of ASME Code, Section XI, requirements, submitted an alternative inspection program per the BWRVIP guidelines for B-N-1 and B-N-2 reactor pressure vessel interior surfaces, attachments, and core support structures at DNPS, Units 2 and 3. The licensee stated that implementation of the alternative inspection program will maintain an adequate level of quality and safety of the affected welds and components and will not adversely impact the health and safety of the public. As part of its justification for the relief, the licensee stated that boiling-water reactors now examine the reactor pressure vessel interior surfaces, attachments, and core support structures in accordance with BWRVIP guidelines. The proposed alternative includes examination methods, examination volume, frequency, training and successive and additional examinations, flaw evaluations, and reporting. These guidelines have been written to address the examination of safety significant RVI components using appropriate methods and reexamination frequencies. Furthermore, the licensee stated that relief from examinations in Table IWB-2500-1 of the ASME Code, Section XI, are requested pursuant to 10 CFR 50.55a(a)(3)(i). The licensee stated that by letter dated April 30, 2008, the NRC staff issued an SE for the implementation of inspection and evaluation (I&E) guidelines addressed in the relevant BWRVIP reports in lieu of the ASME Code, Section XI, ISI requirements for the reactor pressure vessel interior surfaces, attachments, and core support structures at DNPS units for the licensee's fourth ISI interval.

Alternative Examination

In lieu of the requirements of the applicable edition and addenda of the ASME Code, Section XI, the licensee proposed to examine the DNPS, units' RVI components in accordance with BWRVIP guideline requirements. The licensee included only the RVI components (code components) that are categorized under the jurisdiction of the ASME Code, Section XI. The following BWRVIP reports include I&E guidelines for the ASME Code, Section XI, reactor pressure vessel interior surfaces, attachments, and core support structures. Furthermore, the licensee clarified that not all RVI components listed in the following BWRVIP reports are ASME Code, Section XI components.

BWRVIP-03, "BWRVIP Reactor Pressure Vessel and Internals Examination Guidelines"
BWRVIP-18, Revision 1, "BWRVIP Core Spray Internals Inspection and Flaw Evaluation Guidelines"
BWRVIP-25, "BWRVIP Core Plate Inspection and Flaw Evaluation Guidelines"
BWRVIP-26-A, "BWRVIP Top Guide Inspection and Flaw Evaluation Guidelines"
BWRVIP-27-A, "BWRVIP BWR Standby Liquid Control System/Core Plate Delta P Inspection and Flaw Evaluation Guidelines"
BWRVIP-38, "BWRVIP Shroud Support Inspection and Flaw Evaluation Guideline"
BWRVIP-47-A, "BWR Lower Plenum Inspection and Flaw Evaluation Guidelines"
BWRVIP-48-A, "Vessel ID Attachment Weld Inspection and Flaw Evaluation Guidelines"
BWRVIP-76, Revision 1, "BWR Core Shroud Inspection and Flaw Evaluation Guidelines"
BWRVIP-94, Revision 2, BWRVIP Program Implementation Guide"
BWRVIP-138, Revision 1, "BWRVIP Updated Jet Pump Beam Inspection and Flaw Evaluation Guidelines"
BWRVIP-183, "BWRVIP, Top Guide Grid Beam Inspection and Flaw Evaluation"

The license stated that inspection services by an authorized inspection agency will be applied to the proposed alternative. The licensee further indicated that the BWRVIP has established reporting protocol for examination results and deviation that are consistent with the requirements of BWRVIP-94 report. The licensee clarified that revised version of a BWRVIP report will meet I&E guidelines of its original version, and if it does not meet this criteria, NRC staff approval is mandatory prior to its implementation.

The licensee stated that the RVI code components were conditionally authorized by the NRC staff in letter dated April 30, 2008, for the fourth ISI interval. The licensee further stated that

Inspections that occurred during the 19th outages for DNPS, Units 2 and 3, revealed no indications or flaws in the reactor pressure vessel interior surfaces, attachments, and core support structures.

The licensee, in Table 1 of its submittal dated September 28, 2012, provided a comparison of the ASME Code, Section XI, examination requirements for B-N-1 and B-N-2 categories of the reactor pressure vessel interior surfaces, attachments, and core support structures with the above current BWRVIP I&E guidelines. As an example, in Attachment 1 of the submittal, the licensee provided additional information regarding the BWRVIP inspection requirements for the following welds of the reactor pressure vessel interior surfaces, attachments, and core support structures and their subcomponents representing each of the aforementioned ASME Code, Section XI, category/item numbers (Item Nos. B13.10, B13.20, B13.30, and B13.40, addressed in page 2 of this SE).

Core Spray Piping---B13.10
Jet Pump---B13.20
Core Shroud---B13.30
Core Shroud Support and Core Support Structure---B13.40

The licensee claimed that these examples demonstrated that the inspection techniques that are recommended by the BWRVIP inspection guidelines are superior to the inspection techniques mandated by the ASME Code, Section XI, ISI program. Additionally, these examples showed that the BWRVIP inspection guidelines require more frequent inspections of some RVI components than the corresponding ASME Code, Section XI, ISI program. The licensee claimed that by implementing the BWRVIP inspection guidelines, the aging degradation of the reactor pressure vessel interior surfaces, attachments, and core support structures can be identified in a timely manner so that proper corrective action can be taken to restore the integrity of the applicable component. Therefore, the licensee concluded that implementation of the BWRVIP inspection guidelines for the DNPS, Units 2 and 3, reactor pressure vessel interior surfaces, attachments, and core support structures would provide an acceptable level of quality and safety. The licensee's proposed alternative for the RVI components and subcomponents covered under the scope of this alternative request is summarized in Attachment 1 of this SE.

4.0 NRC STAFF EVALUATION

The NRC staff reviewed the information provided by the licensee in its submittal dated September 28, 2012, as supplemented by letter dated November 28, 2012, regarding its proposed alternatives to the ASME Code, Section XI, ISI requirements and the technical bases for the licensee's proposed alternatives. The NRC staff reviewed the status of each of the referenced BWRVIP guidance documents and found all of the referenced BWRVIP reports to be acceptable, with any additional conditions associated with the implementation of the subject BWRVIP reports outlined in the corresponding NRC staff SE for that report. The NRC staff did, however, identify some issues which required additional clarification by the licensee.

By letter dated June 3, 2013, the licensee responded to NRC staff's requests for additional information (RAIs), the NRC staff found the licensees responses satisfactory with one clarification.

The licensee, in response to RAI-I5R-07-3, stated that shroud structure is not predicted to be exposed to a neutron fluence value greater than 1×10^{21} n/cm^2 (E > 1 MeV) during the fifth ISI interval and that the current estimations of the neutron fluence on shroud structure are not expected to exceed this value during the future. The NRC staff accepts this response, and considers this issue to be closed, however, in the future, if the neutron fluence value exceeds the aforementioned threshold limit, the licensee will take into account the lower fracture toughness values specified in BWRVIP-100-A report for establishing the inspection frequency for the shroud structure.

Attachment 1 of this SE includes attributes related to inspection techniques and frequency of inspections for various RVI components in the DNPS, Units 2 and 3. A comparison of the required ASME Code, Section XI, Category B-N-1 and B-N-2, examination requirements with the current BWRVIP guideline requirements that are applicable to the DNPS, Units 2 and 3, is included in Attachment 1 of this SE.

On April 30, 2008, the NRC staff issued an SE for the DNPS, Units 2 and 3, fourth ISI interval, which allowed the licensee to implement the BWRVIP I&E guidelines in lieu of the ASME Code, Section XI, ISI requirements for reactor pressure vessel interior surfaces, attachments, and core support structures. In this SE, the NRC staff imposed a condition which stated that the licensee is required to continue to implement the ASME Code, Section XI, ISI requirements for the jet pump code components. I&E guidelines for the jet pump code components are addressed in BWRVIP-41, "BWRVIP Jet Pump Assembly Inspection and Flaw Evaluation Guidelines." The NRC staff imposed the condition in the SE dated April 30, 2008, because the licensee did not include BWRVIP-41 in its submittal for the fourth ISI interval. Similarly, for the fifth ISI interval, the licensee in its submittal dated September 28, 2012, did not include the BWRVIP-41 as part of the ISI program for the RVI components. Therefore, the licensee is required to inspect the jet pump code components per ASME Code, Section XI, criteria. According to Appendix A of NUREG 1796, "Safety Evaluation Report, Related to the License Renewal of the Dresden Nuclear Power Station, Units 2 and 3, and Quad Cities Nuclear Power Station Units 1 and 2," the licensee was to implement BWRVIP-41 for the jet pump components. Consistent with this requirement, compliance with BWRVIP-41 and the ASME Code (if any) for jet pump and non-code jet pump components at DNPS, Units 2 and 3, as needed.

In request I5R-07 for DNPS, Units 2 and 3, the licensee stated that no indications or flaws were detected in RVI Code components during the 19th refueling outages at DNPS, Units 2 and 3. Based on this information, the NRC staff determined that the active aging degradation mechanisms in the ASME Code, Section XI, RVI components may be stabilized or arrested. BWRVIP I&E guidelines require more frequent inspections than ASME Code, Section XI, criteria for RVI components that are susceptible to aging degradation mechanisms. Therefore, subsequent inspections of the RVI components per the relevant BWRVIP I&E guidelines will provide adequate assurance that any emerging aging effects will be identified in a timely manner. In addition, frequent inspections per these guidelines will enable the licensee to effectively monitor existing aging degradation in reactor pressure vessel interior surfaces, attachments, and core support structures.

Consistent with the determination made in the NRC staff's SEs that approved each of the cited BWRVIP inspection requirements, as supplemented by the NRC staff-approved inspection guidelines for the feedwater nozzle and sparger welds, the licensee's proposed alternative will identify aging degradation of the RVI components in a timely manner. Therefore, the NRC staff concludes that the implementation of the inspection requirements specified in the licensee's proposed alternative will ensure that the integrity of the RVI components will be maintained with an acceptable level of quality and safety.

The NRC staff notes that if the licensee intends to take exception to or deviations from, the NRC NRC staff-approved BWRVIP inspection guidelines (i.e. specifically, those inspection requirements listed in Attachment 1 of this SE) at any time in the future, prior NRC staff approval is required.

5.0 CONCLUSION

Based on the information provided in the licensee's submittals, the NRC staff concludes that the alternatives proposed by the licensee as summarized in the Attachment 1 to this SE, will ensure that the integrity of the reactor pressure vessel interior surfaces, attachments, and core support structures is maintained with an acceptable level of quality and safety. However, this does not include the requested alternatives which apply to the inspection of jet pump assembly components based on the provisions of BWRVIP-41. Alternatives based on this topical report are not approved.

Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative for the DNPS, Units 2 and 3, are authorized, with the condition stated above, for the fifth 10-year ISI intervals. All other requirements of the ASME Code, Section XI, for which an alternative has not been specifically requested remain applicable, including third-party review by the Authorized

Nuclear Inservice Inspector. Any ASME Code, Section XI, RVI components that are not included in this request for alternative will continue to be inspected in accordance with the ASME Code, Section XI requirements. Consistent with the requirements addressed in Appendix A of NUREG 1796 for DNPS, Units 2 and 3, I&E guidelines addressed in the relevant BWRVIP reports should be implemented for the non-ASME Code, Section XI, RVI components.

Attachment:
Comparison of ASME Category B-N-1
and B-N-2 Requirements with BWRVIP
Guidance Requirements

ATTACHMENT
(to Enclosure 6)

Comparison of ASME Category B-N-1 and B-N-2 Requirements
With BWRVIP Guidance Requirements [1]

ASME Item No. Table IWB-2500-1	Component	ASME Exam Scope	ASME Exam	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam	BWRVIP Frequency
B13.10	Reactor Vessel Interior	Accessible Areas (Non-specific)	VT-3	Each period	BWRVIP-18-A, 25, 26-A, 27-A, 38, 47-A, 48-A, 76 Revision 1, and 138, Revision 1	BWRVIP examinations satisfy ASME Code, Section XI, VT-3 inspection requirements.		

B13.20	Interior Attachments Within Beltline – Jet Pump Riser Braces	Accessible Welds	VT-1	Each 10-year Interval	BWRVIP-48-A Table 3-2	Riser Brace Attachment	EVT-1 (enhanced visual testing)	100% in first 12 years (with 50% to be inspected in the first 6 years); 25% during each subsequent 6 years
	Lower Surveillance Specimen Holder Brackets				BWRVIP-48-A Table 3-2	Bracket Attachment	VT-1	Each 10-year Interval

ASME Item No. Table IWB-2500-1	Component	ASME Exam Scope	ASME Exam	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam	BWRVIP Frequency
B13.30	Guide Rod Brackets	Accessible Welds	VT-3	Each 10-year Interval	BWRVIP-48-A Table 3-2	Bracket Attachment	VT-3	Each 10-year Interval
	Steam Dryer Support Brackets				BWRVIP-48-A Table 3-2	Bracket Attachment	EVT-1	Each 10-year Interval
	Feedwater Sparger Brackets				BWRVIP-48-A Table 3-2	Bracket Attachment	EVT-1	Each 10-year Interval
	Core Spray Piping Brackets				BWRVIP-48-A Table 3-2	Bracket Attachment	EVT-1	Each 10-year Interval
	Upper Surveillance Specimen Holder Brackets				BWRVIP-48-A Table 3-2	Bracket Attachment	VT-3	Each 10-year Interval
	Shroud Support (Weld H9)				BWRVIP-38, 3.1.3.2 Figures 3-2 and 3-5	Weld H9 (2)	EVT-1 or UT	Maximum of 6 years for one sided EVT-1, Maximum of 10 years for UT
	Weld H12 Shroud Support Legs	Rarely Accessible			BWRVIP-38, 3.2.3	Weld H12	Per BWRVIP-38 NRC SER (7-24-2000), inspect with appropriate method (4)	When accessible

ASME Item No. Table IWB-2500-1	Component	ASME Exam Scope	ASME Exam	ASME Frequency	Applicable BWRVIP Document	BWRVIP Exam Scope	BWRVIP Exam	BWRVIP Frequency
B13.40	Shroud Support Weld H10	Accessible Surfaces	VT-3	Each 10-year Interval	BWRVIP-38 3.1.3.2, Figures 3-2 and 3-5	Shroud Support H10 weld and Leg welds	EVT-1 or UT	Based on as found conditions, to a Maximum 6 years for one-sided EVT-1, 10 years for UT where accessible
	Shroud Vertical Welds				BWRVIP-76 R1, 3.3, Figures 3-1 and 3-3	Vertical and Ring Segment Welds as applicable	EVT-1 or UT	Maximum 6 years for one-sided EVT-1, 10 years for UT
	Shroud Repairs (3)				BWRVIP-76, R1 Section 3.5	Tie-Rod Repair	VT-3	Per designer recommendations per BWRVIP-76 R1

Note (1) This Table provides only an overview of the requirements. For more details, refer to the ASME Code, Section XI, Table IWB-2500-1, and the appropriate BWRVIP document.
Note (2) In accordance with Appendix A of BWRVIP-38, a site specific evaluation will determine the minimum required weld length to be examined.
Note (3) Shroud repairs are currently installed at DNPS, units 2 and 3.
Note (4) When inspection tooling and methodologies are available, they will be utilized to establish a base line inspection of these welds.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF I5R-10

REGARDING USE OF ASME CODE CASE N-532-4

EXELON GENERATION COMPANY, LLC

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

DOCKET NOS. 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated September 28, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML12275A069), Exelon Generation Company, LLC (the licensee) requested an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2007 Edition through the 2008 Addenda, Subsection IWA-2441(b) for Dresden Nuclear Power Station (DNPS), Units 2 and 3. Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(a)(3)(i), the licensee requested approval of Relief Request I5R-10, to allow the use of ASME Code Case N-532-4 for the fifth 10-year inservice inspection (ISI) interval at DNPS.

2.0 REGULATORY EVALUATION

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 In-service Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of components. Paragraph 50.55a(g)(4)(i) requires that inservice examination of components and system pressure tests conducted during the first 10-year ISI interval comply with the requirements of the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b)(2) 12 months prior to the issuance of the operating license, subject to the conditions listed therein. Section 50.55a(g)(4)(ii) requires that inservice examination of components and system pressure tests conducted during subsequent 10-year ISI intervals comply with the requirements of the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b)(2) 12 months prior to the start of the 120-month inspection interval, subject to the conditions listed therein. Section10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if (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. 10 CFR 50.55a(g)(5)(iii) states that if the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the NRC and submit, as specified in 10 CFR 50.4, to support the determinations.

The NRC staff finds that there is regulatory basis for the licensee to request, and the NRC to authorize, this alternative pursuant to the technical evaluation that follows. The information provided by the licensee in support of the request has been evaluated by the NRC staff and the bases for disposition are documented below.

3.0 TECHNICAL EVALUATION FOR RELIEF REQUEST I5R-10

Licensee's Request for Alternative

Code Requirements

ASME Section XI IWA-2441(b) requires Code Cases to be used in an ISI program to be applicable to the edition and addenda specified in the inspection plan.

The Applicability Index for Section XI Code Cases shows ASME Code Case N-532-4 to be the 1981 Edition with the Winter 1983 Addenda through the 2004 Edition with the 2005 Addenda.

The DNPS ISI program is based on the ASME Code, Section XI, 2007 Edition with the 2008 Addenda.

Licensee's Proposed Alternative

EGC proposes to utilize ASME Code Case N-532-4 with the 2007 Edition through the 2008 Addenda for the fifth 10-year ISI interval at DNPS.

Basis for Proposed Alternative

The NRC has accepted ASME Code Case N-532-4 in Regulatory Guide (RG) 1.147, Revision 16 (Reference 2), as an acceptable alternative to the repair/replacement activity documentation requirements and inservice summary report preparation and submission requirements of Section XI.

NRC Staff Evaluation

The licensee's proposed alternative would allow DNPS to utilize ASME Code Case N-532-4 with the 2007 Edition through the 2008 Addenda of ASME Section XI. Code Case N-532-4 is listed as acceptable in Table 1 of Reference 2. This indicates that the NRC staff has found the alternative requirements of the code case acceptable for licensees to utilize in lieu of the requirements of ASME Section XI. Therefore, the repair/replacement activity documentation requirements and inservice summary report preparation and submission requirements of N-532-4 are acceptable to the NRC staff.

The NRC staff determined that the Applicability Index shows the latest code applicable to N-532-4 being the 2004 Edition through the 2005 Addenda, is a timing issue. The 2004 Edition through the 2005 Addenda was the latest code approved by the ASME Code committee when they approved N-532-4. Footnote 1 to N-532-4 states, in part, that all references to IWA-4000 and IWA-6000 used in the case refer to the 2004 Edition with 2005 Addenda of Section XI and Table 3 is provided to provide accurate references for earlier ASME Code editions and addenda.

The NRC staff reviewed changes made to the paragraphs/subparagraphs of Section XI referenced in N-532-4 in editions and addenda from 2004 Edition with 2005 Addenda up to the 2008 Addenda. This review found that there were no changes in IWA-6210(c), (d), (e) and (f), IWA-6220, IWA-6230(b), (c) and (d), IWA-6240(b) and IWA-6350 that would impact the requirements spelled out in N-532-4. Therefore, the NRC staff finds that the licensee can utilize Code Case N-532-4 with the 2007 Edition through the 2008 Addenda of Section XI.

Based on the above, the NRC staff has determined that the licensee's proposed alternative will not impact the repair/replacement activity documentation requirements and inservice summary report preparation and submission requirements of Code Case N-532-4. Therefore, the NRC staff finds that the licensee's proposed alternative will provide an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a, and the proposed alternative proposed in Relief Request I5R-10 provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i), and is in compliance with the ASME Code's requirements. Therefore, the NRC staff authorizes use of Relief Request I5R-10 for the fifth 10-year ISI interval at DNPS, Units 2 and 3, which commenced on January 20, 2013 and will end on January 19, 2013. All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the authorized Nuclear Inservice Inspector.

5.0 REFERENCES

1. Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," Revision 16, October 2010. (ADAMS Accession No. ML101800536)

2. American Society of Mechanical Engineers (ASME) Code Case N-532-4, "Repair/Replacement Activity Documentation and Inservice Summary Report Preparation and Submission, Section XI, Division 1," April, 19, 2006.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST I5R-11

REPAIR OF CLASS 2 AND 3 PIPING USING ASME CODE CASE N-661-1

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

EXELON GENERATION COMPANY

DOCKET NUMBERS 50-237 AND 50-249

1.0 INTRODUCTION

By letter dated September 28, 2012 (Agencywide Documents and Access Management System, (ADAMS) Accession No. ML12275A069), Exelon Generation Company, LLC (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, IWA-2441(b). Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(i), the licensee requested to use the proposed alternative (i.e., Relief Request I5R-11) on the basis that the alternative provides an acceptable level of quality and safety.

The licensee submitted for U.S. Nuclear Regulatory Commission's (NRC) review and approval of Relief Request I5R-11 as part of the proposed fifth 10-year inservice inspection (IS) interval program for the Dresden Nuclear Power Station (DNPS), Units 2 and 3, which commenced on January 20, 2013, and is scheduled to be completed by January 19, 2023.

The licensee adopted the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI, as the Code of Record at DNPS, Units 2 and 3, for the fifth 10-year ISI program. Relief Request I5R-11 is related to the use of ASME Code Case N-661-1 to repair ASME Class 2 and 3 carbon steel piping for raw water service. This safety evaluation (SE) is specifically related to Relief Request I5R-11.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2 and 3, components (including supports) must 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 (ISI) 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), 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein.

Pursuant to 10 CFR 50.55a(a)(3) alternatives to requirements may be authorized by the NRC if the licensee demonstrates that: (i) the proposed alternatives 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.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the NRC to authorize the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION FOR RELIEF REQUEST I5R-11

ASME Code Component(s) Affected

The ASME Code Class 2 and 3 piping is applicable under ASME Code Case N-661-1, "Alternative Requirements for Wall Thickness Restoration of Class 2 and 3 Carbon Steel Piping for Raw Water Service."

Applicable Code Edition and Addenda

ASME Code, Section XI, 2007 Edition through the 2008 Addenda.

Applicable Code Requirement

ASME Code, Section XI, IWA-2441(b), requires code cases be applicable to the edition and addenda specified in the inspection plan. ASME Code Case N-661-1 provides requirements that may be used to restore wall thickness for raw water piping systems that have experienced internal wall thinning.

Reason For Request

On January 20, 2013, DNPS, Units 2 and 3, will start the fifth 10-year ISI interval program under the requirements of the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI. When implementing the the ASME Code, Section XI, Paragraph IWA-2441(b), requires code cases be applicable to the edition and addenda specified in the ISI program.

The ASME Code Case N-661-1 has an applicability limited up to the 2004 Edition through the 2005 Addenda. Since ASME Code Case N-661-1 only applies up to the 2004 Edition through the 2005 Addenda, Paragraph IWA-2441(b), does not allow the use of ASME Code Case N-661-1 for the DNPS fifth 10-year interval ISI program.

Proposed Alternative and Basis for Use

The licensee requests the applicability of ASME Code Case N-661-1 be extended to the 2007 Edition through the 2008 Addenda for use in the DNPS fifth 10-year ISI interval ISI program. The NRC has accepted the use of ASME Code Case N-661-1 as an acceptable method for restoring wall thickness for raw water piping systems that have experienced internal wall thinning in Regulatory Guide (RG) 1.147, Revision 16.

The relief request does not propose technical changes to ASME Code Case N-661-1. The licensee submitted the request to extend applicability to permit use of ASME Code Case N-661-1 under the 2007 Edition through the 2008 Addenda of ASME Section XI. The licensee considered that because no technical change is proposed in this request, the proposed alternative provides an acceptable level of quality and safety, and is consistent with provisions of 10 CFR 50.55a(a)(3)(i).

Duration of Proposed Alternative

Relief is requested for the fifth 10-year ISI interval for DNPS, Units 2 and 3.

NRC Staff Evaluation

The ASME Code Case N-661-1 limits its applicability to the 2004 Edition through the 2005 Addenda of the ASME Code, Section XI. The licensee requested relief from IWA-2441(b) of the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI, to permit the use of ASME Code Case N-661-1. As an alternative, the licensee proposed to extend the applicability of ASME Code Case N-661-1 to the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI, for the repair of Class 2 and 3 carbon steel piping of the raw water service system.

The NRC staff notes that 10 CFR 50.55a has incorporated by reference the 2007 Edition through the 2008 Addenda of the ASME Code, Section XI. Therefore, the licensee is permitted to use the 2007 Edition and 2008 Addenda specified in 10 CFR 50.55a. The NRC staff also notes that 10 CFR 50.55a does not impose conditions on ASME Code Case N-661-1 other than the following two conditions imposed in RG 1.147, Revision 16: (1) if the cause of the [piping] degradation has not been determined, the repair [of the degraded pipe] is only acceptable until the next refueling outage, and (2) when through-wall repairs are made by welding on surfaces that are wet or exposed to water, the weld overlay repair is only acceptable until the next refueling outage. The licensee has not asked relief from either of these two conditions or ASME Code Case N-661-1 other than the applicability of the ASME Code edition and addenda.

Although IWA-2441(b) of the ASME Code, Section XI, requires code cases be applicable to the edition and addenda specified in the 10-year ISI program, the NRC staff does not believe the provisions in the 2007 Edition through the 2008 Addenda would in any way invalidate the design, installation, and examination requirements of ASME Code Case N-661-1 or its applicability. Similarly, the NRC staff determines that use of ASME Code Case N-661-1 does not violate the provisions of the 2007 Edition through 2008 Addenda of ASME Code, Section XI. Therefore, there is no conflict between the code case and the 2007 Edition through the 2008 Addenda of ASME Code, Section XI. In addition, the NRC staff finds no conflict in using ASME

Code Case N-661-1 under the 2007 Edition through the 2008 Addenda as opposed to the 2004 Edition through the 2005 Addenda of the ASME Code, Section XI. Therefore, the NRC staff finds that the licensee's proposed alternative is acceptable and the licensee is permitted to use ASME Code Case N-661-1 in the fifth 10-year ISI program. Pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff determines that extending the applicability of ASME Code Case N-661-1 to the 2007 Edition through 2008 Addenda of the ASME Code, Section XI, provides an acceptable level of quality and safety.

The NRC staff notes that the ASME Code committees have approved ASME Code Case N-661-2 which is the latest version of N-661. The NRC has not officially approved ASME Code Case N-661-2, but once the NRC approves ASME Code Case N-661-2 in RG 1.147, Revision 17, via rulemaking, ASME Code Case N-661-2 will supersede N-661-1. The NRC expects that DNPS will use ASME Code Case N-661-2 at that time. Therefore, the NRC's approval of ASME Code Case N-661-1 is limited to the end of the fifth 10-year ISI interval or until the NRC approves ASME Code Case N-661-2, whichever occurs earlier. As soon as the NRC approves ASME Code Case N-661-2, Relief Request I5R-11 becomes obsolete.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides an acceptable level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(a)(3)(i), and is in compliance with the ASME Code's requirements for which relief was not requested. Therefore, the NRC staff authorizes the use of ASME Code Case N-661-1 at DNPS, Units 2 and 3, for the fifth 10-year ISI interval which is scheduled to commence on January 20, 2013, and end on January 19, 2023, or until the NRC approves ASME Code Case N-661-2 in RG 1.147, Revision 17, whichever occurs earlier.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributors: T. McLellan
J. Poehler
M. Reisi-Fard
K. Hoffman
A. Rezai
J. Wallace
J. Tsao

Date: September 30, 2013

requested and authorized herein by the NRC staff remain applicable, including the third party review by the Authorized Nuclear Inservice Inspector.

If you have any questions on this action, please contact the NRC Senior Project Manager, Brenda Mozafari, at (301) 415-2020.

Sincerely,
/ RA /
Travis L. Tate, Chief
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-237 and 50-249

Enclosures:
Enclosure 1 - Safety Evaluation – Relief Request I5R-01
Enclosure 2 - Safety Evaluation – Relief Request I5R-02
Enclosure 3 - Safety Evaluation – Relief Request I5R-03
Enclosure 4 - Safety Evaluation – Relief Request I5R-04
Enclosure 5 - Safety Evaluation – Relief Request I5R-05
Enclosure 6 - Safety Evaluation – Relief Request I5R-07
Enclosure 7 - Safety Evaluation – Relief Request I5R-10
Enclosure 8 - Safety Evaluation – Relief Request I5R-11

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