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RS-16-072

10 CFR 50.55a(z)

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Byron Station, Units 1 and 2
Renewed Facility Operating License Nos. NPF-37 and NPF-66
NRC Docket Nos. STN 50-454 and STN 50-455

Subject: Relief Requests Associated with the Fourth Inservice Inspection Interval

In accordance with 10 CFR 50.55a, "Codes and standards," paragraphs (z)(1) and (z)(2), Exelon Generation Company, LLC (EGC), hereby requests NRC approval of the attached relief requests associated with the fourth Inservice Inspection (ISI) Interval for Byron Station, Units 1 and 2. The fourth interval of the Byron ISI Program is currently scheduled to begin on July 16, 2016, and end on July 15, 2025, subject to the allowable changes for inspection intervals in IWA-2430, "Inspection Intervals," and will comply with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 2007 Edition with the 2008 Addenda. The latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b)(2) is the 2007 Edition with the 2008 Addenda. EGC proposes the following relief requests for the Byron fourth 10-year ISI interval:

- I4R-01 requests approval for alternate risk-informed selection and examination criteria for Category B-F, B-J, C-F-1, and C-F-2 pressure retaining piping welds in accordance with ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B, Section XI, Division 1."
- I4R-05 requests approval of alternative repair criteria for Class 2 and 3 moderate energy carbon steel raw water piping systems in accordance with ASME Code Case N-789, "Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping for Raw Water Service, Section XI, Division 1."
- I4R-06 requests approval of alternative repair criteria for ASME Class 2 and 3 moderate energy carbon steel piping systems in accordance with ASME Code Case N-786, "Alternative Requirements for Sleeve Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping Section XI, Division 1."
- I4R-08 requests approval to defer Code-required volumetric examination of specific Byron Station, Unit 2 reactor vessel full penetration pressure retaining Examination Category B-A and B-D welds.

The bases for these relief requests are provided in Attachments 1 through 4, respectively.

Relief requests similar or identical to I4R-01, I4R-05, I4R-06, and I4R-08 have previously been approved for use at Byron Station. EGC requests approval of these requests by April 17, 2017, to support implementation of the Byron Station fourth 10-year ISI interval.

There are no regulatory commitments contained within this letter.

Should you have any questions concerning this letter, please contact Mr. Mitchel A. Mathews at (630) 657-2819.

Respectfully,

David M. Gullott
Manager – Licensing
Exelon Generation Company, LLC

Attachments:

1. 10 CFR 50.55a Request Number I4R-01
2. 10 CFR 50.55a Request Number I4R-05
3. 10 CFR 50.55a Request Number I4R-06
4. 10 CFR 50.55a Request Number I4R-08

ATTACHMENT 1
10 CFR 50.55a Request Number I4R-01
Proposed Alternative In Accordance with 10 CFR 50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--

1. ASME Code Component(s) Affected

Code Class:	1 and 2
Reference:	Table IWB-2500-1, Table IWC-2500-1
Examination Category:	B-F, B-J, C-F-1, and C-F-2
Item Number:	B5.10, B5.40, B5.70, B9.11, B9.21, B9.22, B9.31, B9.32, B9.40, C5.11, C5.21, C5.30, C5.41, C5.51, C5.61, C5.70, and C5.81
Description:	Alternate Risk-Informed Selection and Examination Criteria for Examination Category B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds
Component Number:	Unit 1 and Unit 2 Pressure Retaining Piping

2. Applicable Code Edition and Addenda

The fourth 10-year interval of the Byron Station, Units 1 and 2 Inservice Inspection (ISI) Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition with the 2008 Addenda.

3. Applicable Code Requirement

Table IWB-2500-1, Examination Category B-F, requires volumetric and surface examinations on all welds for Item Numbers B5.10, B5.40, and B5.70.

Table IWB-2500-1, Examination Category B-J, requires volumetric and surface examinations on a sample of welds for Item Numbers B9.11 and B9.31, volumetric examinations on a sample of welds for Item Number B9.22, and surface examinations on a sample of welds for Item Numbers B9.21, B9.32, and B9.40. The weld population selected for inspection is specified in Note (2) and Note (3).

Note (2) Examinations shall include the following:

(a) All terminal ends in each pipe or branch run connected to vessels.

(b) All terminal ends and joints in each pipe or branch run connected to other components where the stress levels exceed either of the following limits under loads associated with specific seismic events and operational conditions:

(1) primary plus secondary stress intensity range of $2.4S_m$ for ferritic steel and austenitic steel.

(2) cumulative usage factor U of 0.4.

(c) All dissimilar metal welds not covered under Examination Category B-F.

(d) Additional piping welds so that the total number of circumferential butt welds (or branch connection or socket welds) selected for examination equals 25% of the circumferential butt welds (or branch connection or socket welds) in the reactor coolant piping system. This total does not include welds exempted by Paragraph IWB-1220 or welds in Item Number B9.22. These additional welds may be located as follows:

(1) For PWR plants
  (a) one hot-leg and one cold-leg in one reactor coolant piping loop,
  (b) one branch, representative of an essentially symmetric piping configuration among each group of branch runs that are connected to reactor coolant loops and that perform similar system functions, and
  (c) each piping and branch run exclusive of the categories of loop and runs that are part of system piping of (a) and (b) above.

Note (3) A 10% sample of PWR high pressure safety injection system circumferential welds in piping ≥ NPS 1½ and < NPS 4 shall be selected for examination. This sample shall be selected from locations determined by the Owner as most likely to be subject to thermal fatigue. Thermal fatigue may be caused by conditions such as valve leakage or turbulence effects.

Table IWC-2500-1, Examination Categories C-F-1 and C-F-2 require volumetric and surface examinations on a sample of welds for Item Numbers C5.11, C5.21, C5.51, and C5.61; and surface examinations on a sample of welds for Item Numbers C5.30, C5.41, C5.70, and C5.81. The weld population selected for inspection is specified in Note (2) for both Examination Categories.

Note (2) The welds selected for examination shall include 7.5%, but not less than 28 welds, of all dissimilar metal, austenitic stainless steel and high alloy welds (Examination Category C-F-1) or of all carbon and low alloy steel welds (Examination Category C-F-2) not exempted by IWC-1220. (Some welds not exempted by IWC-1220 are not required to be nondestructively examined per Examination Categories C-F-1 and C-F-2. These welds, however, shall be included in the total weld count to which the 7.5% sampling rate is applied.) The examinations shall be distributed as follows:

(a) the examinations shall be distributed among the Class 2 systems prorated, to the degree practicable, on the number of nonexempt dissimilar metal, austenitic stainless steel and high alloy welds (Examination Category C-F-1) or carbon and low alloy welds (Examination Category C-F-2) in each system;

(b) within a system, the examinations shall be distributed among terminal ends, dissimilar metal welds, and structural discontinuities prorated, to the degree practicable, on the number of nonexempt terminal ends, dissimilar metal welds, and structural discontinuities in the system; and

(c) within each system, examinations shall be distributed between line sizes prorated to the degree practicable.

4. Reason for Request

In accordance with 10 CFR 50.55a(z)(1), relief is requested on the basis that the proposed alternative utilizing Electric Power Research Institute (EPRI) Topical Report (TR) 112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Rev. B-A (Reference 1) along with two enhancements from ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B, Section XI, Division 1," (Reference 4) will provide an acceptable level of quality and safety. As stated in *Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)* (Reference 2):

> *"The staff concludes that the proposed RI-ISI Program as described in EPRI TR-112657, Revision B, is a sound technical approach and will provide an acceptable level of quality and safety pursuant to 10 CFR 50.55a for the proposed alternative to the piping ISI requirements with regard to the number of locations, locations of inspections, and methods of inspection."*

The initial Byron Station Risk-Informed Inservice Inspection (RI-ISI) Program was submitted during the second period of the second ISI interval for Unit 1 and during the first period of the second interval for Unit 2. This initial RI-ISI Program was developed in accordance with EPRI TR-112657, Revision B-A, as supplemented by ASME Code Case N-578-1. The initial program was approved for use by the Nuclear Regulatory Commission (NRC) via a safety evaluation (SE) as transmitted to Exelon Generation Company, LLC (EGC) on February 5, 2002 (Reference 5).

The Byron Station RI-ISI Program was resubmitted using the same approach during the third ISI interval for Units 1 and 2. The program was approved for use by the NRC via SE as transmitted to EGC on September 25, 2007 (Reference 6).

The transition from the 2001 Edition through the 2003 Addenda to the 2007 Edition with the 2008 Addenda of ASME Section XI for Byron Station's fourth ISI interval does not impact the currently approved RI-ISI evaluation methods and process used in the third ISI interval, and the requirements of the new Code Edition/Addenda will be implemented as detailed in the Byron Station ISI Program Plan. Therefore, with the exception of specific weld locations that may have changed due to maintenance or modification activities (e.g., Fukushima FLEX modification) and the addition of an Alloy 600 Augmented Examination Program, the proposed alternative RI-ISI Program for the fourth ISI interval is the same program methodology as approved in Reference 6 for the third ISI interval.

The Risk Impact Assessment completed as part of the original baseline RI-ISI Program was an implementation/transition check on the initial impact of converting from a traditional ASME Section XI Program to the new RI-ISI methodology. For the fourth interval ISI update, there is no transition occurring between two different methodologies, but rather, the previously approved RI-ISI methodology and evaluation will be maintained for the new interval. The original methodology of the evaluation has not changed, and the change in risk was simply re-assessed using the initial 1989 Edition, no Addenda ASME Section XI Program prior to RI-ISI and the new element selection for the fourth ISI interval RI-ISI Program. This same process has been maintained in each revision to the Byron Station RI-ISI assessment that has been performed to date.

Based on the fourth ISI interval update of this risk impact assessment, the change in risk from the pre-RI-ISI Section XI Program to the Fourth Interval RI-ISI Program was within the 1.00E-06 and 1.00E-07 acceptance criteria for delta-core damage frequency (Delta-CDF) and delta-large early release frequency (Delta-LERF) as described in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis." The Delta-CDF and Delta-LERF values for Byron Station, Units 1 and 2 are listed in the following table.

Change in Risk from Byron Station Pre-RI-ISI Section XI Program to Fourth Interval RI-ISI Program		
Unit No.	**Delta-CDF**	**Delta-LERF**
Unit 1	1.77E-08	3.95E-10
Unit 2	-1.52E-08*	-1.54E-10*

* Negative values represent an improvement in plant risk.

The actual "evaluation and ranking" procedure including the Consequence Evaluation and Degradation Mechanism Assessment processes of the currently approved (Reference 6) RI-ISI program remain unchanged and are continually applied to maintain the Risk Categorization and Element Selection methods of EPRI TR-112657, Revision B-A. These portions of the RI-ISI Program have been and will continue to be reevaluated and revised as major revisions of the site Probabilistic Risk Assessment (PRA) occur and modifications to plant configuration are made. The Consequence Evaluation, Degradation Mechanism Assessment, Risk Ranking, Element Selection, and Risk Impact Assessment steps encompass the complete *living program* process applied under the Byron Station RI-ISI Program.

5. <u>Proposed Alternative and Basis for Use</u>

The proposed alternative originally implemented in the Byron Station, Units 1 and 2, "Risk Informed Inservice Inspection Evaluation," (Reference 3), along with the two enhancements noted below, provide an acceptable level of quality and safety as required by 10 CFR 50.55a(z)(1). This same program, along with these enhancements, was resubmitted and is currently approved for Byron Station's third ISI interval as documented in Reference 6.

The fourth ISI interval RI-ISI Program will be a continuation of the current application and will continue to be a living program as described in Section 4 of this relief request. No changes to the evaluation methodology as currently implemented under EPRI

TR-112657, Revision B-A, are required as part of this interval update. The following two enhancements will continue to be implemented.

a. In lieu of the evaluation and sample expansion requirements in Section 3.6.6.2, "RI-ISI Selected Examinations" of EPRI TR-112657, Byron Station will utilize the requirements of Paragraph -2430, "Additional Examinations," contained in ASME Code Case N-578-1 (Reference 4). The alternative criteria for additional examinations contained in ASME Code Case N-578-1 provide a more refined methodology for implementing necessary additional examinations. The reason for this selection is that the guidance discussed in EPRI TR-112657 includes requirements for additional examinations at a high level, based on service conditions, degradation mechanisms, and the performance of evaluations to determine the scope of additional examinations, whereas ASME Code Case N-578-1 provides more specific and clearer guidance regarding the requirements for additional examinations that is structured similar to the guidance provided in ASME Section XI, Paragraphs IWB-2430 and IWC-2430. Additionally, similar to the current requirements of ASME Section XI, Byron Station intends to perform additional examinations that are required due to the identification of flaws or relevant conditions exceeding the acceptance standards, during the outage the flaws are identified.

b. To supplement the requirements listed in EPRI TR-112657, Table 4-1, "Summary of Degradation-Specific Inspection Requirements and Examination Methods," Byron Station will utilize the provisions listed in Table 1, Examination Category R-A, "Risk-Informed Piping Examinations," contained in ASME Code Case N-578-1 (Reference 4). To implement Note 10 of this table, paragraphs and figures from the 2007 Edition with the 2008 Addenda of ASME Section XI (i.e., Byron Station's Code of Record for the fourth ISI interval) will be utilized which parallel those referenced in the code case. Table 1 of ASME Code Case N-578-1 will be used as it provides a detailed breakdown for examination method and categorization of parts to be examined. Based on these methods and categorization, the examination figures specified in EPRI TR-112657, Section 4 will then be used to determine the examination volume/area based on the degradation mechanism and component configuration. For elements not subject to a degradation mechanism, ASME Code Case N-578-1, Table 1, Note 1 will be applied using the expanded examination volume.

   In addition, Alloy 600/82/182 elements potentially subject to primary water stress corrosion cracking (PWSCC) with full structural weld overlay applied will be removed from the RI-ISI Program and examined under the Byron Station Alloy 600 Augmented Examination Program under ASME Code Case N-770-x; and non-Alloy 600/82/182 elements with full-structural weld overlay applied will be removed from the RI-ISI Program and examined under ASME Section XI, Non-Mandatory Appendix Q. Alloy 600/82/182 elements potentially subject to PWSCC that have been mitigated with a mechanical stress improvement process will remain in the RI-ISI Program and will also be addressed under the Byron Station Alloy 600 Augmented Examination Program under ASME Code Case N-770-x.

Piping examinations under this augmented examination program are currently performed in accordance with the criteria below. This augmented examination program is subject to change and will be maintained in accordance with the latest regulations relative to PWSCC and the Byron Station Alloy 600 Augmented Examination Program. For elements evaluated under RI-ISI to only be subject to the PWSCC degradation mechanism and all elements with full structural weld overlay applied, the requirements incorporated in the augmented examination program are:

1. Weld Locations with Full-Structural Overlays
   Alloy 600/82/182 locations with applied full-structural weld overlays where the degradation mechanism assessment of the overlaid weld identified PWSCC, or PWSCC and another degradation mechanism as determined by the RI-ISI Program, will be removed from the RI-ISI Program and administered solely under the Byron Station Alloy 600 Augmented Examination Program. These locations will receive examinations as specified under ASME Code Case N-770-x separate from the RI-ISI Program in order to maintain compliance with 10 CFR 50.55a(g)(6)(ii)(F).

   Non Alloy 600/82/182 locations with applied full-structural weld overlays will be removed from the RI-ISI Program and treated solely under the requirements of ASME Section XI, 2007 Edition with the 2008 Addenda, Non-Mandatory Appendix Q.

2. Weld Locations Mitigated with Mechanical Stress Improvement Process
   For Alloy 600/82/182 locations where the PWSCC degradation mechanism has been mitigated with a Mechanical Stress Improvement Process, the elements will remain in the RI-ISI Program and are subject to the normal RI-ISI element selection process. These welds will also be governed by the Byron Station Alloy 600 Augmented Examination Program under ASME Code Case N-770-x. The selection and examination of these welds will comply with both RI-ISI and the ASME Code Case N-770-x requirements. The examinations in the fourth ISI interval may be credited to both programs as applicable.

The Byron Station RI-ISI program, as developed in accordance with EPRI TR-112657, Rev. B-A (Reference 1), requires that 25% of the elements that are categorized as "High" risk (i.e., Risk Category 1, 2, and 3) and 10% of the elements that are categorized as "Medium" risk (i.e., Risk Categories 4 and 5) be selected for inspection. For this application, the guidance for the examination volume for a given degradation mechanism is provided by the EPRI TR-112657 while the guidance for the examination method and categorization of parts to be examined are provided by the EPRI TR-112657 as supplemented by ASME Code Case N-578-1.

For NRC staff consideration in the evaluation of this alternative RI-ISI Program, Enclosure, "BY-PRA-031, Rev. 0, 'Byron Nuclear Generating Station Units 1 and 2, PRA Capability Assessment for RI-ISI, (Summary: PRA Capability Assessment for Risk-Informed Inservice Inspection Applications),' dated December 2015," to this relief request contains a summary of the Regulatory Guide 1.200, Revision 2 (Reference 7), evaluation performed on the Byron/Braidwood BB PRA-014, Revision BB011b, Quantification Notebook (Reference 8), and the impact of the identified gaps on the technical adequacy of the Byron Station PRA Model to support this RI-ISI application (see Attachment 1, Table 1).

In addition to this risk-informed evaluation, selection, and examination procedure, all ASME Section XI piping components, regardless of risk classification, will continue to receive Code-required system pressure testing as part of the current ASME Section XI Program. VT-2 visual examinations are scheduled in accordance with the Byron Station System Pressure Testing program, which remains unaffected by the RI-ISI Program.

6. Duration of Proposed Alternative

Relief is requested for the fourth ISI interval for Byron Station, Units 1 and 2.

7. Precedents

- Byron Station, Units 1 and 2 Third ISI Interval Relief Request I3R-02 was authorized by NRC Safety Evaluation (SE) dated September 25, 2007, ADAMS Accession No. ML072610510. The Fourth Inspection Interval relief request utilizes a similar RI-ISI methodology to the one that was previously approved.
- Relief Request RR-7 was authorized for St. Lucie Plant, Unit 2 by NRC SE dated August 10, 2015, ADAMS Accession No. ML15196A623.
- Relief Request 4RR-01 was authorized for Susquehanna Steam Electric Station, Units 1 and 2 by NRC SE dated April 28, 2015, ADAMS Accession No. ML15098A478.
- Relief Request I5R-02 was authorized for Dresden Station Units 2 and 3 by NRC SE dated September 30, 2013, ADAMS Accession No. ML13260A585.
- Relief Request I5R-02 was authorized for Quad Cities Station Units 1 and 2 by NRC SE dated September 30, 2013, ADAMS Accession No. ML13267A097.

8. References

1. Electric Power Research Institute (EPRI) Topical Report (TR) 112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Rev. B-A, dated December 1999.
2. Letter from W. H. Bateman (NRC) to G. L. Vine (EPRI), "Safety Evaluation Report Related to EPRI Risk-Informed Inservice Inspection Evaluation Procedure (EPRI TR-112657, Revision B, July 1999)," dated October 28, 1999, ADAMS Accession Nos. ML993190460 and ML993190474.
3. Initial Risk-Informed Inservice Inspection Evaluation - Byron Nuclear Power Station Units 1 and 2, dated August 2000.

4. ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B, Section XI, Division 1," dated March 28, 2000.
5. Letter from A. J. Mendiola (NRC) to O. D. Kingsley (EGC), "Approval of Relief Request I2R-40 for Application of Risk-Informed Inservice Inspection Program as an Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and Class 2 Piping Welds for Byron Station, Units 1 and 2 (TAC Nos. MB0567 and MB0568)," dated February 5, 2002, ADAMS Accession No. ML020030027.
6. Letter from R. Gibbs (NRC) to C. M. Crane (EGC), "Byron Station, Unit Nos. 1 and 2 - Evaluation of Proposed Risk-Informed Request for an Inservice Inspection Program for the Third 10-Year Inservice Inspection Interval (TAC Nos. MD3855 and MD3856)," dated September 25, 2007, ADAMS Accession No. ML072610510.
7. NRC Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, dated March 2009, ADAMS Accession No. ML090410014.
8. BB PRA-014, "Byron and Braidwood Stations, Quantification Notebook," Rev. BB011b, dated September 2012.

9. Enclosure

BY-PRA-031, Rev. 0, "Byron Nuclear Generating Station Units 1 and 2, PRA Capability Assessment for RI-ISI, (Summary: PRA Capability Assessment for Risk-Informed Inservice Inspection Applications)," dated December 2015.

ENCLOSURE

BY-PRA-031, Rev. 0, "Byron Nuclear Generating Station Units 1 and 2, PRA Capability Assessment for RI-ISI, (Summary: PRA Capability Assessment for Risk-Informed Inservice Inspection Applications)," dated December 2015

1. Introduction

Exelon Generation Company, LLC (EGC) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the Probabilistic Risk Analysis (PRA) models for all operating EGC nuclear generation sites. This approach includes both a proceduralized PRA maintenance and update process, and the use of self-assessments and independent peer reviews. The following information describes this approach as it applies to the Byron Station PRA.

2. PRA Maintenance and Update

The EGC risk management process ensures that the applicable PRA model remains an accurate reflection of the as-built and as-operated plants. This process is defined in the EGC Risk Management program, which consists of a governing procedure ER-AA-600, "Risk Management," and subordinate implementation procedures. EGC procedure ER-AA-600-1015, "[Full Power Internal Events (FPIE)] PRA Model Update," delineates the responsibilities and guidelines for updating the FPIE PRA models at all operating EGC nuclear generation sites. The overall EGC Risk Management program, including ER-AA-600-1015, defines the process for implementing regularly scheduled and interim PRA model updates, for tracking issues identified as potentially affecting the PRA models (e.g., due to changes in the plant, errors or limitations identified in the model, industry operating experience), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built and as-operated plants, the following activities are routinely performed:

- Design changes and procedure changes are reviewed for their impact on the PRA model.
- New engineering calculations and revisions to existing calculations are reviewed for their impact on the PRA model.
- Maintenance unavailabilities are captured, and their impact on Core Damage Frequency (CDF) is trended.
- Plant specific initiating event frequencies, failure rates, and maintenance unavailabilities for equipment that can have a significant impact on the PRA model are updated approximately every four years.

In addition to these activities, EGC risk management procedures provide the guidance for particular risk management and PRA quality and maintenance activities. This guidance includes:

- Documentation of the PRA model, PRA products, and bases documents.
- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the FPIE PRA models for EGC nuclear generation sites.
- Guidance for use of quantitative and qualitative risk models in support of the On-Line Work Control Process Program for risk evaluations for maintenance tasks (e.g., corrective maintenance, preventive maintenance, minor maintenance, surveillance tests, and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10 CFR 50.65(a)(4)).

In accordance with this guidance, regularly scheduled PRA model updates nominally occur on an approximately 4-year cycle; longer intervals may be justified if it can be shown that the PRA continues to adequately represent the as-built, as-operated plant.

The most recent update of the Byron Station PRA model (i.e., designated the BB011b model) (Reference 6) was completed in November 2012 as a result of a regularly scheduled update to the previous 6E PRA model (Reference 8). The BB011b model is the most recent evaluation of the risk profile at Byron Station for internal event challenges, including internal flooding. The Byron Station PRA modeling is highly detailed, including a wide variety of initiating events, modeled systems, operator actions, and common cause events. The PRA model quantification process used for the Byron Station PRA is based on the event tree / fault tree methodology, which is a well-known methodology in the industry.

3. <u>PRA Peer Review</u>

Several assessments of technical capability have been made, and continue to be planned for the Byron Station PRA model. A chronological list of the assessments performed includes the following:

- The Braidwood Station PRA was subjected to a Westinghouse Owners Group (WOG) Peer Review in September 1999.
- The Byron Station PRA was subjected to a separate Peer Review in July 2000.
- The Byron and Braidwood Stations PRA BB011b version of the model was Peer Reviewed in July 2013 for internal events and internal flooding (Reference 7). That Peer Review of the Byron and Braidwood Stations PRA was against the requirements of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA standard (Reference 1) and any Clarifications and Qualifications provided in the Nuclear Regulatory Commission (NRC) endorsement of the Standard contained in Revision 2 to Regulatory Guide (RG) 1.200 (Reference 2). Of the 320 supporting requirements reviewed 302 were considered met at Capability Category I/II or greater. Ten were considered Capability Category I and only six Not-Met. According to EPRI

TR-1021467-A (Reference 5), the ten that meet Capability Category I are sufficient for RI-ISI. The six Not-Met include the following items with a synopsis of the Peer Review finding:

- Initiating Events – it was not shown or documented that candidate-initiating events that took place other than at power were considered as occurring at power.
- Data – it was not shown or documented that short term, repeated equipment failures were classified as a single failure.
- Data – it was not shown or documented that post-maintenance tests occurring before equipment was declared operable was eliminated from equipment counts.
- LERF Model – there was a lack of documentation on the limitations of the Large Early Release Frequency (LERF) model for use in applications.
- Internal Flooding – there are two topics, which are not relevant since the internal flooding analysis is not used in RI-ISI analysis.

The Peer Review findings, including both those items not-met as well as those that do not meet Capability Category I, have negligible impact on the RI-ISI analysis. Most of the findings relate to missing documentation rather than shortcomings in the PRA analysis.

A summary of this assessment of the current Peer Review Not-Met and Capability Category I Supporting Requirements (SR) relative to the RI-ISI relief request is provided in Table 1. All unaddressed gaps will be reviewed for consideration during the next Byron Station model update (i.e., anticipated to be 2016) but are judged to have low impact on the PRA model or its ability to support a full range of PRA applications. These items are documented in the PRA Updating Requirement Evaluation (URE) database so that they can be tracked and their potential impacts accounted for in applications where appropriate. In addition, plant changes made since the last PRA update have been reviewed and determined to not have a significant PRA impact. These items are also documented in UREs for consideration in future PRA updates, as appropriate.

4. Guidance from EPRI Report on PRA Technical Adequacy for RI-ISI

EPRI report TR-1021467-A (Reference 5) provides guidance on the PRA Standard Capability Category necessary to support RI-ISI. This report received a Safety Evaluation (SE) from the NRC in January 2012. RG 1.200 considers it a good practice to have; in general, SRs meet Capability Category II for applications. However, according to the EPRI report not all SRs require Capability Category II to adequately support RI-ISI applications. According to the EPRI report many of the Byron Station gaps listed in Table 1 do not require Capability Category II, but instead only require Capability Category I, the most basic level. Therefore according to EPRI TR-1021467-A and the associated NRC SE, the Byron Station PRA model BB011b is adequate for use in the RI-ISI application.

5. General Conclusion Regarding PRA Capability

The Byron Station PRA maintenance and update processes and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in RI-ISI applications. As specific risk-informed PRA applications are performed, remaining gaps to specific requirements in the PRA standard will be reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

6. Conclusion Regarding PRA Capability for Risk-Informed ISI

The Byron Station PRA model continues to be suitable for use in the RI-ISI application. This conclusion is based on:

- PRA maintenance and update processes in place,
- PRA technical capability evaluations that have been performed and are being planned, and
- RI-ISI process considerations, as noted above, that demonstrate the relatively limited sensitivity of the EPRI RI-ISI process to PRA attribute capability beyond ASME PRA Standard Capability Category I.

In support of the PRA analyses for the Byron Station 10-year interval evaluation using the BB011B PRA model, the remaining gaps to the PRA standard have been reviewed to determine which, if any, would merit RI-ISI-specific sensitivity studies in the presentation of the application results. The result of this assessment concluded that no additional sensitivity studies are merited.

7. References

1. ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," dated March 2009.
2. NRC Regulatory Guide (RG) 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment results for Risk-Informed Activities," Revision 2, dated March 2009.
3. RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, dated November 2002.
4. Reactor Oversight Program MSPI Bases Document, Byron Generating Station, Revision 4, dated February 2008.
5. EPRI TR-1021467-A, "Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-informed Inservice Inspection Programs," dated June 2012.

6. BB-PRA-014, "Quantification Notebook," Rev. BB011b, dated November 2012.
7. "Byron/Braidwood Nuclear Plants RG 1.200 Internal Events and Flooding PRA Peer Review Report," dated December 2013.
8. BB-PRA-014, "Quantification Notebook," Rev. 6E, dated June 2009.

Table 1: Byron / Braidwood Not Met and Capability Category I Supporting Requirements

Supporting Requirement (Note 1)	Capability Category	Risk-Informed ISI Evaluation Impact	EPRI TR-1021467 Requirement
DA-C5 (DA-C5)	Not Met	These SRs are associated with the counting of failures and demands in the development of failure probabilities. Having failed to collapse failures and demands from a single post-maintenance event results in higher (more conservative) component failure rates. Thus the impact on the risk-informed consequence analysis is also conservative and acceptable.	CC I/II/III
DA-C6 (DA-C6)	Not Met	These SRs are associated with the counting of failures and demands in the development of failure probabilities. Having failed to collapse failures and demands from a single post-maintenance event results in higher (more conservative) component failure rates. Thus the impact on the risk-informed consequence analysis is also conservative and acceptable.	CC I/II/III
IE-A7 (IE-A5)	Not Met	This SR deals with not including other than at-power events in the development of Initiating Events. For many of the risk-informed consequence assessments the initiator is assumed to occur with a frequency of 1.0/yr. The remaining consequence calculations are each a difference calculation where the value of the initiator cancels out. Thus, there is no impact on the risk-informed application.	CC I
IFQU-A6	Not Met	The assessment of this SR identifies that certain operator action timing information and the potential impact on cues was not addressed in the flooding assessment. The RI-ISI consequence analysis does not depend on the internal flooding analysis.	RI-ISI analysis does not depend on the internal flooding analysis

Table 1: Byron / Braidwood Not Met and Capability Category I Supporting Requirements

Supporting Requirement (Note 1)	**Capability Category**	**Risk-Informed ISI Evaluation Impact**	**EPRI TR-1021467 Requirement**
IFSN-B3	Not Met	The assessment of this SR is related to the need for additional discussion of the inputs into the flood scenario development. The RI-ISI consequence analysis does not depend on the internal flooding analysis.	RI-ISI analysis does not depend on the internal flooding analysis
LE-G5 (LE-G5)	Not Met	This assessment for this SR relates to a need for increased discussion of limitations in the LERF analysis that could impact applications. There are no state-of-the-art modeling limitations in the LERF analysis that impact this application.	CC I/II/III
HR-E3 (HR-E3)	CC I	This SR relates to having a structured review with Operations and training personnel to confirm the interpretation of the procedure is consistent with training expectations and plant use. CC I is acceptable for RI-ISI applications.	CC I
HR-E4 (HR-E4)	CC I	This SR relates to having a structured review with Operations and training personnel to confirm the interpretation of the procedure is consistent with training expectations and plant use. CC I is acceptable for RI-ISI applications.	CC I
IE-A8 (IE-A6)	CC I	This SR addresses interviews of plant personnel to determine if potential initiating events are missing from the PRA. The components being evaluated here do not have an impact on the modeled Initiating Events. CC I is acceptable for RI-ISI applications.	CC I
IE-A9 (IE-A7)	CC I	This SR addresses review of plant-specific precursors to determine if potential initiating events are missing from the PRA. The components being evaluated here do not have an impact on the modeled Initiating Events. CC I is acceptable for RI-ISI applications.	CC I
IFEV-A6	CC I	The assessment of this SR relates to a lack of inclusion of Braidwood specific OE in the development of internal flooding frequencies. The RI-ISI consequence analysis does not depend on the internal flooding analysis.	RI-ISI analysis does not depend on the internal flooding analysis

Table 1: Byron / Braidwood Not Met and Capability Category I Supporting Requirements

Supporting Requirement (Note 1)	Capability Category	Risk-Informed ISI Evaluation Impact	EPRI TR-1021467 Requirement
IFSN-A6	CC I	The assessment of this SR relates to a lack of inclusion of HELB contributions to internal flooding. The RI-ISI consequence analysis does not depend on the internal flooding analysis.	RI-ISI analysis does not depend on the internal flooding analysis
LE-F1 (LE-F1a)	CC I	These SRs relate to the identification and documentation of significant contributors to LERF. The inclusion of this documentation does not impact the ability of the LERF model to provide accurate results. CC I is acceptable for RI-ISI applications.	CC I
LE-G3 (LE-G3)	CC I	These SRs relate to the identification and documentation of significant contributors to LERF. The inclusion of this documentation does not impact the ability of the LERF model to provide accurate results. CC I is acceptable for RI-ISI applications.	CC I
QU-D4 (QU-D3)	CC I	This SR relates to providing a comparison of quantification results to those from other plants. CC I is acceptable for RI-ISI applications.	CC I
SC-A5 (SC-A5)	CC I	This SR relates to the assessment and development of beyond 24-hour mission times. CC I is acceptable for RI-ISI applications.	CC I

Note:
The EPRI TR-1021467 SR numbering is based on the 2005 ASME/ANS standard. The Byron Station PRA peer review was against the ASME/ANS 2009 standard. The 2005 standard equivalent to the ASME/ANS 2009 standard are listed in parentheses.

ATTACHMENT 2
10 CFR 50.55a Request No. I4R-05
Proposed Alternative In Accordance with 10 CFR 50.55a(z)(2)
--Hardship or Unusual Difficulty Without Compensating Increase
In Level of Quality or Safety--

1. ASME Code Component(s) Affected

All Class 2 and 3 moderate energy carbon steel raw water piping systems. Raw water is defined in ASME Code Case N-789, "Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping for Raw Water Service, Section XI, Division 1," as water such as from a river, lake, or well or brackish/salt water - used in plant equipment, area coolers, and heat exchangers. In many plants it is referred to as "Service Water." This code case applies to Class 2 and 3 moderate energy (i.e., less than or equal to 200°F (93°C) and less than or equal to 275 pounds per square inch gage (psig) (1.9 Mega Pascals (MPa)) maximum operating conditions) carbon steel raw water piping.

2. Applicable Code Edition and Addenda

The fourth 10-year interval of the Byron Station, Units 1 and 2 Inservice Inspection (ISI) Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition with the 2008 Addenda.

3. Applicable Code Requirement

ASME Section XI, IWA-4400 of the 2007 Edition with the 2008 Addenda provides requirements for welding, brazing, metal removal, and installation of repair/replacement activities.

4. Reason for Request

In accordance with 10 CFR 50.55a(z)(2), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Exelon Generation Company, LLC (EGC) is requesting a proposed alternative from the requirement for replacement or internal weld repair of wall thinning conditions resulting from degradation in Class 2 and 3 moderate energy carbon steel raw water piping systems in accordance with IWA-4000 for Byron Station, Units 1 and 2. Such degradation may be the result of mechanisms such as erosion, corrosion, cavitation, and pitting - but excluded are conditions involving flow-accelerated corrosion (FAC), corrosion-assisted cracking, or any other form of cracking. IWA-4000 requires repair or replacement in accordance with the Owner's Requirements and the original or later Construction Code. Other alternative repair or evaluation methods are not always practicable because of wall thinness and/or moisture issues.

The primary reason for this request is to permit installation of technically sound temporary repairs to provide adequate time for evaluation, design, material procurement, planning and scheduling of appropriate permanent repair or replacement of the defective piping, considering the impact on system availability, maintenance rule applicability, and availability of replacement materials.

Performing code repair/replacement in lieu of implementing this relief request would in some cases necessitate extending Technical Specifications (TS) Actions to allow for installation of a permanent repair/replacement, putting the plant at higher safety risks compared with the short time necessary to install a technically sound pad repair. Use of this code case may avoid a plant shutdown in situations where it may be necessary to shut the plant down for a code repair/replacement activity. This could result in an unnecessary plant transient and the loss of safety system availability as compared to maintaining the plant online.

Implementing this relief request during refueling outages will enable a greater number of scheduled corrosion inspections during the outages. The ability to install non-intrusive repair pads rather than scheduling contingency plans for piping replacement will enable longer corrosion inspection windows, increased scope of inspection, and improved overall plant safety.

5. Proposed Alternative and Basis for Use

EGC proposes to implement the requirements of ASME Code Case N-789 as a temporary repair methodology for degradation in Class 2 and 3 moderate energy raw water piping systems resulting from mechanisms such as erosion, corrosion, cavitation, or pitting, but excluding conditions involving flow-accelerated corrosion (FAC), corrosion-assisted cracking, or any other form of cracking. These types of defects are typically identified by small leaks in the piping system or by pre-emptive non-code required examinations performed to monitor the degradation mechanisms.

The alternative repair technique described in ASME Code Case N-789 involves the application of a metal reinforcing pad welded to the exterior of the piping system, which reinforces the weakened area and restores pressure integrity. This repair technique will be utilized when it is determined that this temporary repair method is suitable for the particular defect or degradation being resolved.

This code case requires that the cause of the degradation be determined, and that the extent and rate of degradation in the piping be evaluated to ensure that there are no other unacceptable locations within the surrounding area that could affect the integrity of the repaired piping. The area of evaluation will be dependent on the degradation mechanism present. A baseline thickness examination will be performed for a completed structural pad, attachment welds, and surrounding area, followed by monthly thickness monitoring for the first three months, with subsequent frequency based on the results of this monitoring, but at a minimum of quarterly. Areas containing pressure pads shall be visually observed at least once per month to monitor for evidence of leakage. If the areas containing pressure pads are not accessible for direct observation, then monitoring will be accomplished by visual assessment of surrounding areas or ground surface areas above pressure pads on buried piping, or monitoring of leakage collection systems, if available.

For the pressure pad design, the higher of two (2) times the actual measured corrosion rate or four (4) times the estimated maximum corrosion rate for the system will be used. If the actual measured corrosion rate in the degraded location is unavailable, the estimated maximum corrosion rate for the system assumed in the design will be calculated based on the same degradation mechanism as the degraded location.

The repair will be considered to have a maximum service life of the time until the next refueling outage, when a permanent repair or replacement must be performed. Additional requirements for the design of reinforcement pads, installation, examination, pressure testing, and inservice monitoring are provided in ASME Code Case N-789.

Based on the above justification, the use of ASME Code Case N-789 as a proposed alternative to the requirements of ASME Section XI will provide an acceptable level of quality and safety that does not impose an undue hardship.

All other ASME Section XI requirements for which relief was not specifically requested and authorized by the NRC staff will remain applicable including third party review by the Authorized Nuclear Inservice Inspector.

ASME Code Case N-789 was approved by the ASME Board on Nuclear Codes and Standards on June 25, 2011; however, it has not been incorporated into NRC Regulatory Guide (RG) 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," and thus is not available for application at nuclear power plants without specific NRC approval. Therefore, EGC requests use of this alternative repair technique described in this code case via this relief request. NRC Draft Regulatory Guide DG-1296, "Proposed Revision 18 of Regulatory Guide 1.147," proposes to approve use of ASME Code Case N-789 with two conditions. Both of the proposed conditions are included as requirements in this relief request.

6. Duration of Proposed Alternative

The proposed alternative is requested for the Byron Station, Unit 1 and 2 fourth inservice inspection interval. When ASME Code Case N-789 is approved for use by the NRC, this relief request will no longer be applied and this code case, including Regulatory Guide 1.147 conditions, will be used in lieu of this relief request.

7. Precedents

- Byron Station, Units 1 and 2 Third ISI Interval Relief Request I3R-21 was authorized by NRC Safety Evaluation (SE) (Reference 1). This relief request was part of an EGC fleet-wide submittal. Use of ASME Code Case N-789 was authorized for various stations whose ISI Program was based on ASME Section XI, 2007 Edition with the 2008 Addenda. This relief request for the Byron Station, Units 1 and 2 fourth 10-year ISI interval, utilizes a similar approach to the previously approved relief request.
- Relief Request BV3-N-789 was authorized for Beaver Valley Units 1 and 2 by NRC SE dated June 19, 2015, ADAMS Accession No. ML15163A147.
- Relief Request RR-3-43 was authorized for Indian Point Nuclear Generating Unit No. 3 by NRC SE dated February 22, 2008, ADAMS Accession No. ML080280073.

8. References

1. ASME Code Case N-789, "Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping for Raw Water Service, Section XI, Division 1." An EGC fleet-wide submittal, including Byron Station, Units 1 and 2, was authorized by NRC SE dated May 10, 2012, ADAMS Accession No. ML12121A637.
2. ASME Code Case N-786, "Alternative Requirements for Sleeve Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping Section XI, Division 1." An EGC fleet-wide submittal, including Byron Station, Units 1 and 2, was authorized by NRC SE dated July 31, 2014, ADAMS Accession No. ML14175B593.
3. Draft Regulatory Guide DG-1296, "Proposed Revision 18 of Regulatory Guide 1.147," dated March 2016, ADAMS Accession No. ML15027A202.

ATTACHMENT 3
10 CFR 50.55a Request No. I4R-06
Proposed Alternative In Accordance with 10 CFR 50.55a(z)(2)
--Hardship or Unusual Difficulty Without Compensating Increase
In Level of Quality or Safety--

1. ASME Code Component(s) Affected

All ASME Class 2 and 3 moderate energy (i.e., less than or equal to 200°F (93°C) and less than or equal to 275 psig (1.9 MPa) maximum operating conditions) carbon steel piping systems.

2. Applicable Code Edition and Addenda

The fourth 10-year interval of the Byron Station, Units 1 and 2 Inservice Inspection (ISI) Program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2007 Edition with the 2008 Addenda.

3. Applicable Code Requirement

ASME Section XI, IWA-4400 of the 2007 Edition with the 2008 Addenda provides requirements for welding, brazing, metal removal, and installation of repair/replacement activities.

4. Reason for Request

In accordance with 10 CFR 50.55a(z)(2), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Exelon Generation Company, LLC (EGC) is requesting a proposed alternative from the requirement for replacement or internal weld repair of wall thinning conditions resulting from degradation in Class 2 and Class 3 moderate energy carbon steel piping systems in accordance with IWA-4000 for Byron Station, Units 1 and 2. The degradation may be the result of mechanisms such as localized erosion, corrosion, cavitation, and pitting, but conditions involving any form of cracking are excluded. IWA-4000 requires repair or replacement in accordance with the Owner's Requirements and the original or later Construction Code.

One reason for this request is to permit installation of technically sound temporary repairs, in the form of Type A or partial-structural Type B reinforcing sleeves, to provide adequate time for evaluation, design, material procurement, planning and scheduling of appropriate permanent repair or replacement of the defective piping, considering the impact on system availability, maintenance rule applicability, and availability of replacement materials.

The other reason for this request is to permit installation of long-term repairs, in the form of full-structural Type B reinforcing sleeves, for locally degraded portions of piping systems. The design, construction, and inservice monitoring of such sleeves provide a technically sound equivalent replacement for the segment of degraded piping that is encompassed.

5. Proposed Alternative and Basis for Use

EGC proposes to implement the requirements of ASME Code Case N-786, "Alternative Requirements for Sleeve Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping Section XI, Division 1," for repair of degradation in Class 2 and 3 moderate energy carbon steel piping systems resulting from mechanisms such as localized erosion, corrosion, cavitation, or pitting, but excluding conditions involving any form of cracking. These types of defects are typically identified by small leaks in the piping system or by pre-emptive non-code required examinations performed to monitor the degradation mechanisms.

This code case invokes the design requirements of the original Construction Code or ASME Section III. Reconciliation and use of editions and addenda of ASME Section III will be in accordance with ASME Section XI, IWA-4220, and only editions and addenda of ASME Section III that have been accepted by 10 CFR 50.55a may be used. The Code of Record, 2007 Edition with the 2008 Addenda, will be used when applying the various Subsection IWA paragraphs of ASME Section XI unless specific regulatory relief to use other editions or addenda is approved.

The alternative repair technique described in ASME Code Case N-786 involves the application of Type A and Type B full encirclement sleeve halves welded together with full penetration longitudinal seam welds to reinforce structural integrity in the degraded area. In the case of Type B reinforcing sleeves, the ends are also welded to the piping in order to restore pressure integrity. This repair technique will be utilized when it is determined that this repair method is suitable for the particular defect or degradation being resolved without flaw removal. Use of this repair method will be limited to pipe and fittings; as a result the following condition shall apply to the application of ASME Code Case N-786:

Reinforcing sleeves may not be applied to pumps, valves, expansion joints, vessels, heat exchangers, tubing, or flanges; and may not be applied over flanged joints, socket welded or threaded joints, or branch connection welds.

The code case requires that the cause of the degradation be determined and that the extent and rate of degradation in the piping be evaluated to ensure that there are no other unacceptable locations within the surrounding area that could affect the integrity of the repaired piping. Surrounding areas showing signs of degradation shall be identified and included in the Owner's plan for thickness monitoring for full-structural reinforcing sleeves. The area of evaluation will be dependent on the degradation mechanism present, but shall extend at least $0.75 \sqrt{RT_{nom}}$ beyond the edge of any sleeve attachment weld. If the cause of the degradation is not determined, the maximum permitted service life of any reinforcing sleeve shall be the time until the next refueling outage. In addition, the following condition shall apply to the application of ASME Code Case N-786:

The initial degradation rate selected for design of all sleeves shall be equal to or greater than two (2) times the maximum rate observed at the location of the repair. If the degradation rate for that location is unknown, four (4) times the estimated maximum degradation rate for that or a similar system at the same *plant site for the same degradation mechanism shall be applied. If both the degradation rate for that location and the cause of the degradation are not conclusively determined, four (4) times the maximum degradation rate observed for all degradation mechanisms for that or a similar system at the same plant site shall be applied.*

"Full-structural Type B" means that the sleeve and attachment welds alone maintain full capability to withstand structural (mechanical) and pressure loading for which the piping is presently designed without need for additional support or reinforcement, and without reliance on any piping that is encased by the sleeve. Type A and partial-structural Type B sleeves rely on the encased underlying piping to provide some structural (mechanical) and/or pressure retaining integrity.

Type B reinforcing sleeves may be applied to leaking systems by installing a gasket or sealant between the sleeve and the pipe as permitted by the code case, and then clamping the reinforcing sleeve halves to the piping prior to welding. Residual moisture is then removed by heating prior to welding. If welding of any type of sleeve occurs on a wet surface, the maximum permitted life of the sleeve shall be the time until the next refueling outage.

The code case requires that the Owner shall prepare and implement a plan for thickness monitoring by inspection of full-structural reinforcing sleeves and their attachment welds. To accomplish this, a baseline thickness examination will be performed for completed full-structural Type B reinforcing sleeves, partial penetration attachment welds, and surrounding areas, followed by similar thickness monitoring inspections performed at a minimum of every refueling outage for the life of the repair. More frequent thickness monitoring examinations will be scheduled based on the maximum degradation rates observed during these inspections, such that the required design thicknesses will not be infringed upon before each subsequently scheduled thickness monitoring examination.

Partial-structural Type B reinforcing sleeves and Type A reinforcing sleeves completely encompass the degraded areas. These sleeves are designed to accommodate predicted maximum degradation and must be removed at the next refueling outage. Accordingly, the code case does not require inservice monitoring for these sleeves. However, because of NRC concerns discussed in the NRC Safety Evaluation (SE) for the EGC sites concerning the approval to apply ASME Code Case N-789, the following condition shall apply to the application of ASME Code Case N-786:

Type A reinforcing sleeves and partial-structural Type B reinforcing sleeves shall be visually observed at least once per month to monitor for evidence of leakage. If the areas containing these sleeves are not accessible for direct observation, then monitoring will be accomplished by visual assessment of surrounding areas or ground surface areas above such sleeves on buried piping, or monitoring of leakage collection systems, if available.

When used on buried piping, the area of full-structural Type B reinforcing sleeves will need to be physically accessible for the required examinations, which could necessitate installation of removable barriers at the repair location in lieu of backfilling the pipe at that location. For Type A and partial-structural Type B reinforcing sleeves installed on buried piping, the monitoring will be based on visual assessment as discussed above.

Type A reinforcing sleeves and partial-structural Type B reinforcing sleeves shall have a maximum permitted service life of the time until the next refueling outage, when a permanent repair or replacement must be performed. Neither the Type A nor the partial-structural Type B reinforcing sleeve may remain in service beyond the end of the next refueling outage after they are installed, unless specific regulatory relief is obtained. This means that if such a repair is performed in mid-cycle (e.g., one month before the scheduled refueling outage) the reinforcing sleeve would be removed no later than the upcoming refueling outage (e.g., in one month) unless specific regulatory relief is obtained. Even if removal during the next scheduled refueling outage becomes challenging (e.g., it is installed on a system required to be functional during the refueling outage), it would still need to be removed when the system is not required to be functional and prior to the conclusion of the next scheduled refueling outage after it was installed.

A similar situation exists with common cooling lines that require a dual unit outage in order to remove them from service. Unless a full-structural Type B reinforcing sleeve is installed, specific regulatory approval would need to be obtained in order to defer removal of a Type A or partial-structural Type B reinforcing sleeve beyond the next upcoming refueling outage of either unit.

Full-structural Type B reinforcing sleeves will be removed and an IWA-4000 repair or replacement will be performed prior to the time that inservice monitoring indicates that pressure integrity (leak tightness) or structural integrity could be impaired based on measured degradation between monitoring activities. Additional requirements for design, installation, examination (including volumetric examination in accordance with ASME Section III, NC-5200 and NC-5300, or ND-5200 and ND-5300), pressure testing, and inservice examination of reinforcing sleeves are provided in ASME Code Case N-786, with the following additional condition:

> *Branch connections may be installed on reinforcing sleeves only for filling or venting purposes during installation or leakage testing of the sleeve, and shall be limited to Nominal Pipe Size (NPS) 1 or smaller in size.*

All other ASME Section XI requirements for which relief was not specifically requested and authorized by the NRC staff will remain applicable including third party review by the Authorized Nuclear Inservice Inspector.

Performing code repair/replacement in lieu of implementing this relief request would in some cases necessitate extending Technical Specifications (TS) Actions to install a permanent repair/replacement, putting the plant at higher safety risks than warranted compared with the short time necessary to install a technically sound sleeve repair. Without the use of this code case in some situations, it may be necessary to shut the plant down in order to perform a code repair/replacement activity; however, this results in an unnecessary plant transient and the loss of safety system availability as compared to maintaining the plant online.

Implementing this relief request during refueling outages will enable a greater number of scheduled corrosion inspections during the outages. The ability to install non-intrusive repair sleeves rather than scheduling contingency plans for piping replacement, will enable longer corrosion inspection windows, increased scope of inspection, and improved overall plant safety.

Based on the above, the use of ASME Code Case N-786 for full-structural Type B reinforcing sleeves and for Type A and partial-structural Type B reinforcing sleeves will apply when compliance with the specified Code requirements of ASME Section XI would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

ASME Code Case N-786 was approved by the ASME Board on Nuclear Codes and Standards on March 24, 2011; however, it has not been incorporated into NRC Regulatory Guide (RG) 1.147, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," and thus is not available for application at nuclear power plants without specific NRC approval. Therefore, EGC requests use of the alternative repair techniques described in the code case via this relief request. NRC Draft Regulatory Guide DG-1296, "Proposed Revision 18 of Regulatory Guide 1.147," proposes to approve use of ASME Code Case N-786 without conditions.

6. Duration of Proposed Alternative

The proposed alternative is requested for the Byron Station, Units 1 and 2 fourth ISI Interval. When ASME Code Case N-786 is approved for use by the NRC and incorporated into RG 1.147, this relief request will no longer be required and the code case, including any RG 1.147 conditions, will be utilized in lieu of this relief request. Any Type A and partial-structural Type B reinforcing sleeves installed before the end of the 10-year ISI interval will be removed during the next refueling outage, even if that refueling outage occurs after the end of the 10-year ISI interval.

7. Precedents

Byron Station, Units 1 and 2 Third ISI Interval Relief Request I3R-24 was authorized by NRC SE dated July 31, 2014 (Reference 1). This relief request was part of an EGC fleet-wide submittal. Use of ASME Code Case N-786 was authorized for various stations whose ISI Program was based on ASME Section XI, 2007 Edition with the 2008 Addenda. This relief request for the Byron Station, Units 1 and 2 fourth 10-year ISI interval, utilizes a similar approach to the previously approved relief request.

8. References

1. ASME Code Case N-786, "Alternative Requirements for Sleeve Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping Section XI, Division 1." An EGC fleet-wide submittal, including Byron Station, Units 1 and 2, was authorized by NRC SE dated July 31, 2014, ADAMS Accession No. ML14175B593.
2. ASME Code Case N-789, "Alternative Requirements for Pad Reinforcement of Class 2 and 3 Moderate-Energy Carbon Steel Piping for Raw Water Service, Section XI, Division 1." An EGC fleet-wide submittal, including Byron Station, Units 1 and 2, was authorized by NRC SE dated May 10, 2012, ADAMS Accession No. ML12121A637.
3. Draft Regulatory Guide DG-1296, "Proposed Revision 18 of Regulatory Guide 1.147," dated March 2016, ADAMS Accession No. ML15027A202.

ATTACHMENT 4
10 CFR 50.55a Request No. I4R-08
Proposed Alternative In Accordance with 10 CFR 50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--

1. ASME Code Component(s) Affected

The affected component is the Byron Station, Unit 2 reactor vessel, specifically, the following American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI (Reference 1) examination categories and item numbers covering examinations of the reactor vessel.

These affected Class 1 examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME Section XI.

Examination Category	Item Number	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-A	B1.40	Head-to-Flange Weld
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inside Radius Section

(Throughout this relief request, the above examination categories are referred to as "the subject examinations.")

See Table 1 below for a listing of the applicable examination areas of the Unit 2 reactor vessel.

2. Applicable Code Edition and Addenda

The fourth 10-year interval of the Byron Station, Units 1 and 2 Inservice Inspection (ISI) Program is based on the ASME Section XI, 2007 Edition with the 2008 Addenda.

The applicable Edition of ASME Section XI for subsequent intervals will be implemented in accordance with the requirements of 10 CFR 50.55a.

3. Applicable Code Requirements

ASME Section XI, IWB-2411, "Inspection Program," requires volumetric examination of reactor vessel pressure retaining welds identified in Table IWB-2500-1 once each 10-year inspection interval. The Byron Station, Unit 2 fourth 10-year ISI interval is currently scheduled to end on July 15, 2025.

4. Reason for the Request

In accordance with 10 CFR 50.55a(z)(1), relief is requested on the basis that the proposed alternative provides an acceptable level of quality and safety.

An alternative is requested from the requirement of IWB-2411, "Inspection Program," that volumetric examination of essentially 100% of reactor vessel pressure retaining, Examination Category B-A and B-D welds and Examination Category B-D inside radius sections, be performed once each ten-year inspection interval.

Extension of the interval between examinations of specific Examination Category B-A and B-D welds and Examination Category B-D inside radius sections from ten years to a maximum of twenty years will result in a reduction in person-rem exposure and examination costs.

5. Proposed Alternative and Basis for Use

Exelon Generation Company, LLC (EGC) proposes to defer ASME Section XI required volumetric examination of specific Byron Station, Unit 2 reactor vessel full penetration pressure retaining Examination Category B-A and B-D welds and Examination Category B-D inside radius sections for the fourth ISI interval, currently scheduled to be performed during the fall 2017 refueling outage. EGC proposes to perform these ASME Section XI required volumetric examinations during the fifth ISI interval for Byron Station, Unit 2 no later than 2027. This date is consistent with those provided in Pressurized Water Reactor Owners Group (PWROG) letter OG-06-356, "Plan for Plant-Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval.' MUHP 5097-99, Task 2059," (i.e., Reference 2) and the latest revised implementation plan provided in PWROG letter OG-10-238, "Revision to the Revised Plan for Plant Specific Implementation of Extended In service Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk Informed Extension of the Reactor Vessel In-Service Inspection Interval,' PA-MSC-0120," (i.e., Reference 3).

In accordance with 10 CFR 50.55a(z)(1), an alternate inspection interval is requested on the basis that the current examination interval can be extended based on a negligible change in risk when compared to the risk criteria specified in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1 (Reference 4). The methodology used to conduct this analysis is based on that defined in WCAP-16168-NP-A, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval," Revision 3 (Reference 5). This methodology focuses on risk assessments of materials within the beltline region of the reactor vessel wall. The results of the calculations for Byron Station, Unit 2 were compared to those obtained from the Westinghouse pilot plant evaluated in Reference 5. Appendix A of Reference 5 identifies the parameters to be compared. By demonstrating that the parameters for Byron Station, Unit 2 are bounded by the results of the Westinghouse pilot plant, the methodology qualifies Byron Station, Unit 2 for an ISI interval extension.

Table 1 below lists the applicable examination areas addressed under this relief request for the Byron Station, Unit 2 reactor vessel.

Table 1: Examination Category, Item Number, Component Description, and Component Identification of Applicable Examinations for Byron Station, Unit 2

Examination Category	**Item Number**	**Component Description**	**Component Identification (Exam Area)** Unit 2 Reactor Vessel: 2RC-01-R
Shell Components			
B-A	B1.30	Flange to Nozzle Shell (Upper Shell) Weld	WR-7
B-A	B1.11	Nozzle Shell to Intermediate Shell Weld	WR-34
B-A	B1.11	Intermediate Shell to Lower Shell Weld	WR-18
B-A	B1.11	Lower Shell to Bottom Head Torus Weld	WR-29
B-A	B1.21	Bottom Head Torus to Head Dome Weld	WR-16
B-A	B1.21	Closure Head Dutchman to Dome Weld	RVHC-02
B-A	B1.40	Closure Head Flange to Dutchman Weld	RVHC-01
Nozzle Components			
B-D	B3.90	Outlet Nozzle @ 22° Weld	RPVN-A
B-D	B3.90	Inlet Nozzle @ 67° Weld	RPVN-B
B-D	B3.90	Inlet Nozzle @ 113° Weld	RPVN-C
B-D	B3.90	Outlet Nozzle @ 158° Weld	RPVN-D
B-D	B3.90	Outlet Nozzle @ 202° Weld	RPVN-E
B-D	B3.90	Inlet Nozzle @ 247° Weld	RPVN-F
B-D	B3.90	Inlet Nozzle @ 293° Weld	RPVN-G
B-D	B3.90	Outlet Nozzle @ 338° Weld	RPVN-H
B-D	B3.100	Outlet Nozzle @ 22° Nozzle Inside Radius Section	RPVN-A-NIR
B-D	B3.100	Inlet Nozzle @ 67° Nozzle Inside Radius Section	RPVN-B-NIR
B-D	B3.100	Inlet Nozzle @ 113° Nozzle Inside Radius Section	RPVN-C-NIR
B-D	B3.100	Outlet Nozzle @ 158° Nozzle Inside Radius Section	RPVN-D-NIR
B-D	B3.100	Outlet Nozzle @ 202° Nozzle Inside Radius Section	RPVN-E-NIR
B-D	B3.100	Inlet Nozzle @ 247° Nozzle Inside Radius Section	RPVN-F-NIR
B-D	B3.100	Inlet Nozzle @ 293° Nozzle Inside Radius Section	RPVN-G-NIR
B-D	B3.100	Outlet Nozzle @ 338° Nozzle Inside Radius Section	RPVN-H-NIR

Table 2 below lists the critical parameters investigated in WCAP-16168-NP-A, Revision 3 and compares the results of the Westinghouse pilot plant to those of Byron Station, Unit 2.

Table 2: Critical Parameters for the Application of Bounding Analysis for Byron Station, Unit 2

Parameter	Pilot Plant Basis	Plant-Specific Basis Evaluation	Additional Evaluation Required?
Dominant Pressurized Thermal Shock (PTS) Transients in the NRC PTS Risk Study are Applicable	NRC PTS Risk Study (Reference 6)	PTS Generalization Study (Reference 7)	No
Through-Wall Cracking Frequency (TWCF)	1.76E-08 Events per year (Reference 5)	3.75E-16 Events per year (Calculated per Reference 5)	No
Frequency and Severity of Design Basis Transients	7 heatup/cooldown cycles per year (Reference 5)	Bounded by 7 heatup/cooldown cycles per year	No
Cladding Layers (Single/Multiple)	Single Layer (Reference 5)	Single Layer	No

Table 3 below provides a summary of the latest RV inspection results for Byron Station, Unit 2 and evaluation of the recorded indications. This information confirms that satisfactory examinations have been performed on Byron Station, Unit 2 reactor vessel.

Table 3: Additional Information Pertaining to Reactor Vessel Inspection for Byron Station, Unit 2

Inspection methodology:	The latest Unit 2 inservice inspection was conducted in accordance with the ASME Section XI and ASME Section V, 2001 Edition through the 2003 Addenda. Examinations of Examination Category B-A and B-D welds and Examination Category B-D nozzle inside radius sections were performed to ASME Section XI, Appendix VIII, 2001 Edition through the 2003 Addenda, as modified by 10 CFR 50.55a(b)(2)(xiv, xv, and xvi). Future inservice inspections will be performed to the applicable ASME Section XI, Appendix VIII and 10 CFR 50.55a requirements.
Number of past inspections:	Two Ten-Year inservice inspections have been performed.

Table 3: Additional Information Pertaining to Reactor Vessel Inspection for Byron Station, Unit 2

<table>
<tr><td rowspan="2">Number of indications found:</td><td>There were two indications identified in the beltline region during the most recent inservice inspection. These indications are in the forging material and are acceptable per Table IWB-3510-1 of Section XI of the ASME Code. Both indications are within the inner 1/10th or 1" of the reactor vessel thickness and are acceptable per the requirements of the Alternate PTS Rule, 10 CFR 50.61a (Reference 8) since the number of actual flaws is less than the allowable number of flaws for each flaw size increment. A disposition of these two flaws against the limits of the Alternate PTS Rule is shown in the table below.</td></tr>
<tr><td>
<table>
<tr><th colspan="2">Through-Wall Extent, TWE (in.)</th><th rowspan="2">Scaled Maximum number of forging flaws</th><th rowspan="2">Number of forging Flaws (Axial/Circ.)</th></tr>
<tr><th>TWE_MIN</th><th>TWE_MAX</th></tr>
<tr><td>0.075</td><td>0.375</td><td>76</td><td>2 (0/2)</td></tr>
<tr><td>0.125</td><td>0.375</td><td>30</td><td>2 (0/2)</td></tr>
<tr><td>0.175</td><td>0.375</td><td>8</td><td>1 (0/1)</td></tr>
</table>
</td></tr>
<tr><td rowspan="2">Proposed inspection schedule balance of plant life:</td><td>The third inservice examination for Unit 2 is currently scheduled for 2017. The alternate-schedule examination will be performed no later than 2027.</td></tr>
<tr><td>This date is consistent with those provided in PWROG letter OG-06-356 (Reference 2) and the latest revised implementation plan in PWROG letter OG-10-238 (Reference 3).</td></tr>
</table>

Table 4 summarizes the inputs and outputs for the calculation of through-wall cracking frequency (TWCF) for Byron Station, Unit 2.

Table 4: Details of TWCF Calculation for Unit 2 at 57 Effective Full Power Years (EFPY)

Inputs								
Reactor Coolant System Temperature, T_C [°F]:				N/A	**T_{WALL} [inches]:**			8.625
No.	**Region and Component Description**	**Material Heat No.**	**Cu[a] [wt%]**	**Ni[a] [wt%]**	**R.G. 1.99 Pos.**	**CF[a] [°F]**	**RT_{NDT}(u) [a] [°F]**	**Fluence [10^{19} Neutron /cm^2, E > 1.0 MeV]**
1	Nozzle Shell Forging	4P-6107	0.05	0.74	1.1	31.0	10	1.10
2	NS Forging to IS Forging Circ. Weld WF-562	442011	0.03	0.67	2.1	26.1	40	1.10
3	Intermediate Shell Forging	49D329-1/ 49C297-1	0.01	0.70	1.1	20.0	-20	3.19
4	IS Forging to LS Forging Circ. Weld WF-	442002	0.04	0.63	2.1	66.5	10	3.07
5	Lower Shell Forging	49D330-1/ 49C298-1	0.06	0.73	2.1	18.9	-20	3.19

Outputs						
Methodology Used to Calculate ΔT_{30}:				Regulatory Guide 1.99, Revision 2[b]		
	Controlling Material Region No. (From Above)	**RT_{MAX-XX} [°F]**	**Fluence [10^{19} Neutron/cm^2, E > 1.0 MeV]**	**FF (Fluence Factor)**	**ΔT_{30} [°F]**	**$TWCF_{95-XX}$**
Limiting Forging - FO	1	41.83	1.10	1.027	31.83	1.50E-16
Limiting Circumferential Weld-CW	4	96.20	3.07	1.296	86.20	0.00E+00
$TWCF_{95-TOTAL}(\alpha_{FO}TWCF_{95-FO} + \alpha_{CW}\ TWCF_{95-CW})$:						3.75E-16

<u>**Notes:**</u>
(a) Data obtained from Reference 9.
(b) Reference 10.

6. Duration of Proposed Alternative

Relief is requested for the Fourth ISI Interval for Byron Station, Unit 2.

7. Precedents

- Byron Station, Unit 1 Third ISI Interval Relief Request I3R-23 was authorized by NRC Safety Evaluation (SE) dated December 10, 2014, ADAMS Accession No. ML14303A506. This Fourth ISI Interval relief request for Unit 2 utilizes a similar approach that was previously approved.
- Relief Request IR-3-27 was authorized for Millstone Power Station, Unit No. 3 by NRC SE dated February 16, 2016, ADAMS Accession No. ML16038A001
- Relief Request RPV-U1 was authorized for Diablo Canyon Power Plant, Unit No. 1 by NRC SE dated June 19, 2015, ADAMS Accession No. ML15168A024
- Relief Request I3R-17 was authorized for Callaway Plant, Unit 1 by NRC SE dated for February 10, 2015, ADAMS Accession Number ML15035A148.
- Relief Request I3R-14 was authorized for Shearon Harris Nuclear Power Plant, Unit 1 by NRC SE dated January 5, 2015, ADAMS Accession No. ML14353A324.
- Relief Request I3R-09 was authorized for Wolf Creek Generating Station by NRC SE dated December 10, 2014, ADAMS Accession No. ML14321A864.
- Relief Request 13-ISI-01 was authorized for Watts Bar Nuclear Plant, Unit 1 by NRC SE dated November 28, 2014, ADAMS Accession No. ML14314A987.

8. References

1. ASME Boiler and Pressure Vessel Code, Section XI and Section 2007 Edition with the 2008 Addenda, American Society of Mechanical Engineers, New York.
2. OG-06-356, "Plan for Plant-Specific Implementation of Extended Inservice Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval.' MUHP 5097-99, Task 2059," October 31, 2006.
3. OG-10-238, "Revision to the Revised Plan for Plant Specific Implementation of Extended In service Inspection Interval per WCAP-16168-NP, Revision 1, 'Risk Informed Extension of the Reactor Vessel In-Service Inspection Interval.' PA-MSC-0120," July 12, 2010.
4. NRC Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," November 2002.
5. WCAP-16168-NP-A, Revision 3, "Risk-Informed Extension of the Reactor Vessel In-service Inspection Interval," October 2011.

6. NUREG-1874, "Recommended Screening Limits for Pressurized Thermal Shock (PTS)," March 2010, ADAMS Accession No. ML070860156.

7. NRC Letter Report, "Generalization of Plant-Specific Pressurized Thermal Shock (PTS) Risk Results to Additional Plants," December 14, 2004, ADAMS Accession No. ML042880482.

8. Code of Federal Regulations, 10 CFR Part 50.61a, "Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," U.S. Nuclear Regulatory Commission, Washington D. C., Federal Register, Volume 75, No. 1, dated January 4, 2010 and No. 22 with corrections to part (g) dated February 3, 2010, March 8, 2010, and November 26, 2010.

9. WCAP-17606-NP, Revision 0, "Byron Station, Units 1 and 2 Reactor Vessel Integrity Evaluation to Support License Renewal Time-Limited Aging Analysis," December 2012.

10. NRC Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," May 1988.