Exelon Generation
4300 Winfield Road
Warrenville, IL 60555

www.exeloncorp.com

Exelon. Nuclear

10 CFR 50.55a

RS-08-160
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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Braidwood Station, Units 1 and 2
Facility Operating License Nos. NPF-72 and NPF-77
NRC Docket Nos. 50-456 and 50-457

Subject: Third 10-Year Inservice Inspection Interval, Relief Request I3R-01, "Request for Relief for 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 In Accordance with 10 CFR 50.55a(a)(3)(i)"

Reference: Letter from A. J. Mendiola (NRC) to O. D. Kingsley (Exelon), "Braidwood Station, Units 1 and 2 – Interval 2 Inservice Inspection Program - Relief Request I2R-39, Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and Class 2 Piping Welds (TAC Nos. MB0506 and MB0507)," dated February 20, 2002

In accordance with 10 CFR 50.55a(a)(3)(i), Exelon Generation Company, LLC, (EGC) is requesting authorization to use a risk-informed inservice inspection (RI-ISI) program as an alternative to the examination program of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, 2001 Edition through the 2003 Addenda for examination category B-F, B-J, C-F-1, and C-F-2 pressure retaining piping welds.

The RI-ISI program was developed in accordance with Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," December 1999, and was previously approved for use in Braidwood Station's Second Inservice Inspection Interval (Reference).

Attachment 1 contains the Braidwood Station relief request, I3R-01, which provides justification that the use of the RI-ISI program provides an acceptable level of quality and safety. Attachment 2 contains the assessment of PRA technical adequacy. Attachment 3 is a summary of the Regulatory Guide 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," evaluation performed on Revision 6C of the Braidwood Station PRA model and the impact of the identified gaps on technical adequacy of the Braidwood PRA model to support the Braidwood RI-ISI application.

There are no regulatory commitments contained in this letter. EGC requests authorization of this relief request by December 2009. If you have any questions concerning this letter, please contact Ms. Lisa A. Schofield at (630) 657-2815.

Respectfully,

Patrick R. Simpson
Manager – Licensing

Attachments:

1. Braidwood Station 10 CFR 50.55a Relief Request I3R-01
2. General Assessment of PRA Technical Adequacy
3. Summary of Regulatory Guide 1.200, Revision 1, Evaluation Performed on Revision 6C of Braidwood Station PRA model

ATTACHMENT 1

Braidwood Station 10 CFR 50.55a Relief Request I3R-01

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10 CFR 50.55a RELIEF REQUEST I3R-01

Request for Relief for 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 In Accordance with 10 CFR 50.55a(a)(3)(i)

1.0 ASME CODE COMPONENTS AFFECTED:

Code Class:	1 and 2
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:	Pressure Retaining Piping

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The Inservice Inspection program is based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, 2001 Edition through the 2003 Addenda.

3.0 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 includes the following:

1. All terminal ends in each pipe or branch run connected to vessels.
2. 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:
    a. primary plus secondary stress intensity range of $2.4S_m$ for ferritic steel and austenitic steel.
    b. cumulative usage factor U of 0.4.
3. All dissimilar metal welds not covered under Examination Category B-F.

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4. Additional piping welds so that the total number of circumferential butt welds, branch connections, or socket welds selected for examination equals 25% of the circumferential butt welds, branch connection, or socket welds in the reactor coolant piping system. This total does not include welds exempted by IWB-1220 or welds in Item Number B9.22.

5. 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.

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 includes the following:

1. Welds selected for examination shall include 7.5%, but not less than 28 welds, of all dissimilar metal, austenitic stainless steel or 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 or high alloy welds (Examination Category C-F-1) or nonexempt carbon and low alloy steel welds (Examination Category C-F-2) in each system;

    b. within a system, per the applicable Examination Category, 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 that system; and

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

4.0 <u>REASON FOR REQUEST:</u>

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative utilizing Reference 1 along with two enhancements from Reference 4 will provide an acceptable level of quality and safety.

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As stated in Reference 2:

> The staff concludes that the proposed RI-ISI program as described in EPRI TR112657, 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 Braidwood Station risk-informed inservice inspection (RI-ISI) program was submitted during the First Period of the Second Inspection Interval. This initial RI-ISI program was developed in accordance with EPRI TR-112657, Revision B-A, as supplemented by Code Case N-578-1. The program was approved for use by the NRC on February 20, 2002 (Reference 5).

The transition from the 1989 Edition to the 2001 Edition through the 2003 Addenda of ASME Section XI for Braidwood Station's Third Inspection Interval does not impact the currently approved risk-informed ISI evaluation process used in the Second Inspection Interval, and the requirements of the new Code edition/addenda will be implemented as detailed in the Braidwood Station ISI Program Plan.

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 Third Interval ISI update, there is no transition occurring between two different methodologies, but rather, the currently approved RI-ISI methodology and evaluation will be maintained for the new interval.

As an added measure of assurance, any new systems, portions of systems, or components being included in the RI-ISI Program for the Third Inspection Interval will be added to the Risk Impact Assessment performed during the previous interval. These components will be addressed within the evaluation at the start of the new interval to assure that the new Third Inspection Interval RI-ISI element selection provides an acceptable overall change-in-risk when compared to the old ASME Section XI population of exams which existed prior to the implementation of the first RI-ISI Program.

The "actual evaluation and ranking procedure" including the Consequence Evaluation and Degradation Mechanism Assessment processes of the currently approved (Reference 5) 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 as major revisions of the site PRA occur and modifications to plant configuration are made. If the evaluation concludes that the PRA model or plant configuration change has a significant impact on the model results, appropriate changes are made to the RI-ISI program. The Consequence Evaluation, Degradation Mechanism Assessment, Risk Ranking, and Element Selection steps encompass the complete living program process applied under the Braidwood Station RI-ISI Program.

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5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

The proposed alternative originally implemented in the Initial Risk-Informed Inservice Inspection Evaluation, Braidwood Station (Reference 3), along with the two enhancements noted below, provide an acceptable level of quality and safety as required by 10 CFR 50.55a(a)(3)(i). This original program along with these same two enhancements is currently approved for the Braidwood Station Second Inspection Interval as documented in Reference 5.

The Third Inspection Interval RI-ISI Program will be a continuation of the current application and will continue to be a living program as described in the "Reason For Request" section 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.

In lieu of the evaluation and sample expansion requirements in Section 3.6.6.2, "RI-ISI Selected Examinations" of EPRI TR-112657, Braidwood Station will utilize the requirements of Subarticle -2430, "Additional Examinations" contained in Code Case N-578-1 (Reference 4). The alternative criteria for additional examinations contained in Code Case N-578-1 provides a more refined methodology for implementing necessary additional examinations.

To supplement the requirements listed in Table 4-1, "Summary of Degradation-Specific Inspection Requirements and Examination Methods" of EPRI TR-112657, Braidwood Station will utilize the provisions listed in Table 1, Examination Category R-A, "Risk-Informed Piping Examinations" contained in Code Case N-578-1 (Reference 4). To implement Note 10 of this table, paragraphs and figures from the 2001 Edition through the 2003 Addenda of ASME Section XI (Braidwood Station code of record for the Third Inspection Interval) will be utilized which parallel those referenced in the Code Case for the 1989 Edition. Table 1 of Code Case N-578-1 will be used as it provides a detailed breakdown for examination method and categorization of parts to be examined.

The Braidwood 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 EPRI TR-112657 while the guidance for the examination method and categorization of parts to be examined are provided by EPRI TR-112657, as supplemented by Code Case N-578-1.

The initial Revision 0 of this request for relief was submitted on March 31, 2008 (Reference 6). As part of the NRC acceptance review, supplemental information regarding the Braidwood Station PRA technical adequacy required by Regulatory Guide 1.200, Revision 1, was requested in NRC letter dated May 22, 2008 (Reference 7). The initial request for relief was withdrawn on May 29, 2008 (Reference 8) until the Braidwood Station PRA model evaluation was completed.

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Attachment 2 contains a summary of the Regulatory Guide 1.200, Revision 1, evaluation performed on Revision 6C of the Braidwood Station PRA model and the impact of the identified gaps on technical adequacy of the Braidwood Station PRA model to support the Braidwood Station RI-ISI application.

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 pressure testing as part of the current ASME Section XI program. VT-2 visual examinations are scheduled in accordance with the Braidwood Station pressure testing program, which remains unaffected by the RI-ISI program.

6.0 DURATION OF PROPOSED ALTERNATIVE:

Relief is requested for the Third Ten-Year Inspection Interval for Braidwood Station, Units 1 and 2.

For Braidwood Station, Unit 1, the Third Ten-Year Inspection Interval started on July 29, 2008, and is currently scheduled to end on July 28, 2018. For Braidwood Station, Unit 2, the Third Ten-Year Inspection Interval started on October 17, 2008, and is currently scheduled to end on October 16, 2018.

7.0 PRECEDENTS:

A similar relief request has been approved for:

> Braidwood Station Second Inspection Interval Relief Request I2R-39 was authorized per SER dated February 20, 2002.

The Third Inspection Interval Relief Request utilizes an identical RI-ISI methodology as was previously approved.

8.0 REFERENCES:

1) Electric Power Research Institute (EPRI) Topical Report (TR) 112657, Revision B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," 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

3) Initial Risk-Informed Inservice Inspection Evaluation, Revision 0 - Braidwood Station, Units 1 and 2 dated July 2000 (Letter BW000102 from Timothy Tulon (Commonwealth Edison Company) to the NRC, "Braidwood Station Interval 2 Inservice Inspection Program: Relief Request I2R-39, Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and Class 2 Piping Welds," dated October 16, 2000)

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4) American Society of Mechanical Engineers (ASME) Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, or 3 Piping, Method B"

5) Letter from A. J. Mendiola (NRC) to O. D. Kingsley (Exelon), "Braidwood Station, Units 1 and 2 - Interval 2 Inservice Inspection Program - Relief Request I2R-39, Alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and Class 2 Piping Welds (TAC Nos. MB0506 and MB0507)," dated February 20, 2002

6) Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Third 10-Year Inservice Inspection Interval, Relief Request I3R-01, 'Request for Relief for Alternative Risk-Informed Selection and Examination Criteria for Examination Category B-F, B-J, C-F-1, and C-F-2 Pressure Retaining Piping Welds,'" dated March 31, 2008

7) Letter from M. J. David (U. S. Nuclear Regulatory Commission) to C. G. Pardee (Exelon Generation Company, LLC), "Braidwood Station, Units 1 and 2 – Request for Supplemental Information Regarding Relief Request I3R-01 (TAC Nos. MD8456 and MD8457)," dated May 22, 2008

8) Letter RS-08-072 from P. R. Simpson (Exelon Generation Company, LLC) to U. S. Nuclear Regulatory Commission, "Withdrawal of Third 10-Year Inservice Inspection Interval Relief Request I3R-01," dated May 29, 2008

ATTACHMENT 2

General Assessment of PRA Technical Adequacy

ATTACHMENT 2
GENERAL ASSESSMENT OF PRA TECHNICAL ADEQUACY FOR
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Exelon Generation Company, LLC, (EGC) employs a multi-faceted approach to establishing and maintaining the technical adequacy and plant fidelity of the 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 Braidwood Station, Units 1 and 2, PRA model.

1.0 <u>PRA MAINTENANCE AND UPDATE:</u>

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, "FPIE PRA Model Update," delineates the responsibilities and guidelines for updating the full power internal events 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, etc.), and for controlling the model and associated computer files. To ensure that the current PRA model remains an accurate reflection of the as-built, as-operated plants, the following activities are routinely performed:

- Design changes and revisions to design changes are reviewed for their impact on the PRA model.
- New procedures 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.
- Equipment unavailabilities are captured, and their impact on CDF and LERF 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.

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- The approach for controlling electronic storage of Risk Management (RM) products including PRA update information, PRA models, and PRA applications.
- Guidelines for updating the full power, internal events 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 (corrective maintenance, preventive maintenance, minor maintenance, surveillance tests and modifications) on systems, structures, and components (SSCs) within the scope of the Maintenance Rule (10CFR50.65(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 Braidwood Station PRA model (designated the Revision 6C model) was completed in May 2008.

2.0 PRA SELF ASSESSMENT AND PEER REVIEW

Several assessments of technical capability have been made, and continue to be planned, for the Braidwood Station, Units 1 and 2, PRA models. These assessments are as follows and further discussed in the paragraphs below.

- Scientech conducted an independent self-assessment of the Braidwood Station PRA model in 1999, prior to the Braidwood Station PRA peer review. All significant comments from this review have been addressed.
- An independent PRA peer review of the Braidwood Station PRA model was conducted under the auspices of the PWR Owners Group in 1999, following the Industry PRA Peer Review process (Reference 1). This peer review included an assessment of the PRA model maintenance and update process.
- An independent PRA peer review of the Byron Station PRA model[1] was conducted under the auspices of the PWR Owners Group in 2000, following the Industry PRA Peer Review process (Reference 8). This peer review included an assessment of the PRA model maintenance and update process.
- During 2005 and 2006 the Braidwood Station, Units 1 and 2, PRA model results were evaluated in the PWR Owners Group PRA cross-comparisons study performed in support of implementation of the mitigating systems performance indicator (MSPI) process (Reference 9). Braidwood Station did not have any identified outliers as a result of this review.

[1] Byron and Braidwood use a combined model, with appropriate flags to differentiate between sites/units. Therefore, the Peer review findings for Byron are also applicable to Braidwood.

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- Following the Byron/Braidwood PRA model update in 2007-2008, a self-assessment of the Byron/Braidwood PRA model against the ASME PRA Standard was performed using Regulatory Guide 1.200, Revision 1.

A summary of the disposition of 1999 and 2000 Industry PRA Peer Review facts and observations (F&Os) for the Byron and Braidwood PRA models was documented as part of the statement of PRA capability for MSPI. All significance level A & B F&Os for those peer reviewed were addressed with the completion of the approved PRA model (Revision 6C). After allowing for plant-specific features, there are no MSPI cross-comparison outliers for Braidwood (refer to the third bulleted item above).

In updating the PRA to Revision 6C, changes were made to the PRA to address several Peer Review F&Os, as well as to make other modeling improvements. Following the model update, a capability assessment was performed. This was a self-assessment of the PRA capability of the new model relative to the updated requirements in Addendum B of the ASME PRA Standard (Reference 4) and criteria in RG 1.200, Revision 1 (Reference 3), including the NRC positions stated in Appendix A of Reference 3 and the clarifications in Reference 5, with particular focus on technical elements important to the risk-informed inservice inspection relief request.

A summary of the current open items including the partially resolved items is provided in Attachment 3. These items will be reviewed for consideration during future model updates 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 also documented in the Update Requirements Evaluation (URE) database so that they can be tracked and their potential impacts accounted for in applications where appropriate.

3.0 GENERAL CONCLUSION REGARDING PRA CAPABILITY

The Braidwood Station, Units 1 and 2, PRA maintenance and update process and technical capability evaluations described above provide a robust basis for concluding that the PRA is suitable for use in risk-informed licensing actions. As specific risk-informed PRA applications are performed, remaining gaps to specific requirements in the PRA standard are reviewed to determine which, if any, would merit application-specific sensitivity studies in the presentation of the application results.

4.0 ASSESSMENT OF PRA CAPABILITY NEEDED FOR RISK-INFORMED INSERVICE INSPECTION

In the risk-informed inservice inspection (RI-ISI) program at Braidwood, the EPRI Risk-informed ISI methodology (Reference 6) is used to define alternative inservice inspection requirements. Plant-specific PRA-derived risk significance information is used during the RI-ISI plan development to support the consequence assessment, risk ranking, element selection and risk impact steps.

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The importance of PRA consequence results, and therefore the scope of PRA technical capability, is tempered by three fundamental components of the EPRI methodology.

First, PRA consequence results are binned into one of three conditional core damage probability (CCDP) and conditional large early release probability (CLERP) ranges before any welds are chosen for RI-ISI inspection as illustrated below. Broad ranges are used to define these bins so that the impact of uncertainty is minimized and only substantial PRA changes would be expected to have an impact on the consequence ranking results.

Consequence Results Binning Groups		
Consequence Category	**CCDP Range**	**CLERP Range**
High	CCDP > 1E-4	CLERP > 1E-5
Medium	1E-6 < CCDP ≤ 1E-4	1E-7 < CLERP ≤ 1E-5
Low	CCDP ≤ 1E-6	CLERP ≤ 1E-7

The risk importance of a weld is therefore not tied directly to a specific PRA result. Instead, it depends only on the range in which the PRA result falls. As a consequence, the wide binning would mitigate any PRA modeling uncertainties provided in the methodology. Additionally, conservatism in the binning process (e.g., as would typically be introduced through PRA attributes meeting ASME PRA Standard Capability Category I versus II) will tend to result in a larger inspection population.

Second, the impacts of particular PRA consequence results are further dampened by the joint consideration of the weld failure potential via a non-PRA-dependent damage mechanism assessment. The results of the consequence assessment and the damage mechanism assessment are combined to determine the risk ranking of each pipe segment (and ultimately each element) according to the EPRI Risk Matrix. The Risk Matrix, which equally takes both assessments into consideration, is reproduced below.

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POTENTIAL FOR PIPE RUPTURE PER DEGRADATION MECHANISM SCREENING CRITERIA	CONSEQUENCES OF PIPE RUPTURE IMPACTS ON CONDITIONAL CORE DAMAGE PROBABILITY AND LARGE EARLY RELEASE PROBABILITY			
	NONE	LOW	MEDIUM	HIGH
HIGH FLOW ACCELERATED CORROSION	LOW Category 7	MEDIUM Category 5	HIGH Category 3	HIGH Category 1
MEDIUM OTHER DEGRADATION MECHANISMS	LOW Category 7	LOW Category 6	MEDIUM Category 5	HIGH Category 2
LOW NO DEGRADATION MECHANISMS	LOW Category 7	LOW Category 7	LOW Category 6	MEDIUM Category 4

Third, the EPRI RI-ISI methodology uses an absolute risk ranking approach. As such, conservatism in either the consequence assessment or the failure potential assessment will result in a larger inspection population rather than masking other important components. That is, providing more realism into the PRA model (e.g., by meeting higher capability categories) most likely would result in a smaller inspection population.

These three facets of the methodology reduce the importance and influence of PRA on the final list of candidate welds.

The limited manner of PRA involvement in the RI-ISI process is also reflected in the risk-informed license application guidance provided in Regulatory Guide 1.174 (Reference 7).

Section 2.2.6 of Regulatory Guide 1.174 provides the following insight into PRA capability requirements for this type of application:

> *There are, however, some applications that, because of the nature of the proposed change, have a limited impact on risk, and this is reflected in the impact on the elements of the risk model.*
>
> *An example is risk-informed inservice inspection (RI-ISI). In this application, risk significance was used as one criterion for selecting pipe segments to be periodically examined for cracking. During the staff review it became clear that a high level of emphasis on PRA*

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technical acceptability was not necessary. Therefore, the staff review of plant-specific RI-ISI typically will include only a limited scope review of PRA technical acceptability.

In addition to the above, it is noted that welds determined to be low risk significant are not eliminated from the ISI program on the basis of risk information. For example, the risk significance of a weld may fall from Medium Risk Ranking to Low Risk Ranking, resulting in it not being a candidate for inspection. However, it remains in the program, and if, in the future, the assessment of its ranking changes (either by damage mechanism or PRA risk) then it may again become a candidate for inspection. If it is discovered during the RI-ISI update process that a weld is now susceptible to flow-accelerated corrosion (FAC), inter-granular stress corrosion cracking (IGSCC), or microbiological induced cracking (MIC) in the absence of any other damage mechanism, then it is addressed in an "augmented" program where it is monitored for those special damage mechanisms. That occurs no matter what the Risk Ranking of the weld is determined to be.

5.0 CONCLUSION REGARDING PRA CAPABILITY FOR RISK-INFORMED ISI

The Braidwood Station, Units 1 and 2, PRA models continue to be suitable for use in the risk-informed inservice inspection application. This conclusion is based on:

- The PRA maintenance and update processes in place,
- The PRA technical capability evaluations that have been performed and are being planned, and
- The 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 Braidwood Station, Units 1 and 2 ten-year interval evaluations using the Revision 6C 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.

6.0 REFERENCES

1) Boiling Water Reactors Owners' Group, "BWROG PSA Peer Review Certification Implementation Guidelines," Revision 3, January 1997

2) American Society of Mechanical Engineers ASME RA-S-2002, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," New York, New York, April 2002

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3) U.S. Nuclear Regulatory Commission, Regulatory Guide1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," January 2007

4) American Society of Mechanical Engineers ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," December 2005

5) U.S. Nuclear Regulatory Commission Memorandum from Farouk Eltawila to Michael T. Lesar, "Notice of Clarification to Revision 1 of Regulatory Guide 1.200," dated July 27, 2007

6) EPRI TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure," Revision B-A, December 1999

7) U.S. Nuclear Regulatory Commission, 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

8) Nuclear Energy Institute NEI-00-02, "Industry PRA Peer Review Process," January 2000

9) "Westinghouse Owners' Group Mitigating Systems Performance Index Cross Comparison (PA-RMSC-0209)," WCAP-16464-NP, Revision 0, August 2005

ATTACHMENT 3

**Summary of Regulatory Guide 1.200, Revision 1,
Evaluation Performed on Revision 6C of Braidwood Station PRA Model**

ATTACHMENT 3
SUMMARY OF REGULATORY GUIDE 1.200, REVISION 1, EVALUATION PERFORMED ON REVISION 6C OF BRAIDWOOD STATION PRA MODEL

#	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
1	DOCUMENT the significant contributors (such as initiating events, accident sequences, basic events) to CDF in the PRA results summary. PROVIDE a detailed description of significant accident sequences or functional failure groups.	QU-F3	The PRA Summary notebook and related documentation will provide the types of information required, but the documentation has not been finalized.	None. This is a documentation issue. The PRA models, on which the RI-ISI assessment is based, capture all significant contributors.
2	Include an assessment of the significance of assumptions on the quantitative results.	QU-F4	Open - Identification of key assumptions will be application specific. Also, the QU-F4 SR has been redefined.	See Gap #7.
3	DOCUMENT the quantitative definition used for significant basic event, significant cutset, and significant accident sequence.	QU-F6	Open – Definition of "significant" needs to be added to the quantification documentation. Exception to the RA-S-2002 definition is not taken, but not all significant contributors are explicitly addressed in the documentation.	None. This is a documentation issue. The PRA models, on which the RI-ISI assessment is based, capture all significant contributors.

ATTACHMENT 3
SUMMARY OF REGULATORY GUIDE 1.200, REVISION 1, EVALUATION PERFORMED ON REVISION 6C OF BRAIDWOOD STATION PRA MODEL

#	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
4	The LERF analysis is based on the NUREG/CR-6595 methodology. As such, it represents a generally conservative, simplified approach. The noted SRs meet the Capability Category I criteria.	LE-B1 LE-B2 LE-C1 LE-C2a LE-C2b LE-C3 LE-C4 LE-C8a LE-C9a LE-C10 LE-D1a LE-D1b LE-D2 LE-D4 LE-D5 LE-E2 LE-E3 LE-F1a LE-G3	Open – There are no current plans to upgrade the LERF model.	Not significant. Given the conservative nature of the NUREG/CR-6595 approach used, there is no significant impact on RI-ISI or other Capability Category I applications.
5	… characterize LERF uncertainties consistent with the applicable requirements of Tables 4.5.8-2(d) and 4.5.8-2(e).	LE-F3	Open – A formal evaluation of uncertainties in the LERF model has not been performed. However, since the NUREG/CR-6595 approach has been used, the results are understood to be conservative.	Not significant. Given the conservative nature of the approach used, formal consideration of uncertainties in the LERF modeling has no significant impact on RI-ISI or other Capability Category I applications.

ATTACHMENT 3
SUMMARY OF REGULATORY GUIDE 1.200, REVISION 1, EVALUATION PERFORMED ON REVISION 6C OF BRAIDWOOD STATION PRA MODEL

#	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
6	Addendum B of the ASME PRA Standard added SRs to document the quantitative definition used for significant basic event, significant cutset, significant accident sequence, and significant accident progression sequence in the CDF and LERF analysis.	QU-F6 LE-G6	Open – These new SRs will be addressed during the next full PRA model update, but providing these definitions should not have an impact on the quantitative results from the PRA model.	None. This is a documentation issue. The model is not being changed to address this item.
7	Several SRs associated with treatment of model uncertainty and related model assumptions have been recently redefined. NRC has issued a clarification to its endorsement of the PRA Standard. NRC and EPRI are currently preparing guidance on an acceptable process for meeting these requirements.	QU-E1 QU-E2 QU-E3 QU-E4 QU-F4 IE-C13 IE-D3 AS-C3 SC-C3 SY-C3 HR-G9 HR-I3 DA-D3 DA-E2 DA-E3 IF-F3 LE-E4 LE-F2 LE-F3 LE-G4	Open – These recently redefined SRs will be addressed during the next full PRA model update after the NRC and EPRI guidance becomes available.	To be determined once the new NRC/EPRI guidance is available. However, the EPRI RI-ISI process is defined such that model uncertainties will not unduly influence results.

ATTACHMENT 3
SUMMARY OF REGULATORY GUIDE 1.200, REVISION 1, EVALUATION PERFORMED ON REVISION 6C OF BRAIDWOOD STATION PRA MODEL

#	Description of Gap	Applicable SRs	Current Status / Comment	Importance to RI-ISI
8	Model documentation needs to be revised to reflect the current model (Revision 6C)	AS-C2 DA-C6 DA-C10 DA-D4 IE-C10 IF-C2 IF-C2c IF-D7 IF-E3a IF-F1 IF-F2 QU-F6 SC-A1 SC-B5 SC-C2 SY-A4 SY-C1 SY-C2	Open - Appropriate documentation is being revised to reflect Revision 6C	None. This is a documentation issue. The PRA models, on which the RI-ISI assessment is based has been approved.
9	Plant Specific MOV (or AOV) failure data was not collected for further analysis.	DA-C3	Open – There are no current plans to develop plant specific data for MOVs and AOVs	The Braidwood Risk profile is generally driven by common cause, human error and pump related basic events. Inclusion of plant specific MOV/AOV data is expected to have a negligible impact on the results. As the risk categorization for RI-ISI uses broad bands and is based on re-quantifying the model with surrogate events set to True, this gap is not expected to impact the results of the RI-ISI.