1. INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

This document is a safety evaluation report (SER) on the application to renew the operating licenses for the Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, as filed by Exelon Generation Company (EGC or the applicant). By letter dated January 3, 2003, EGC submitted its application to the U.S. Nuclear Regulatory Commission (NRC or the Commission) for renewal of the DNPS and QCNPS operating licenses for up to an additional 20 years. The NRC received the application on January 3, 2003. The NRC staff (the staff) reviewed the DNPS/QCNPS license renewal application (LRA) for compliance with the requirements of Title 10, Part 54, of the *Code of Federal Regulations* (10 CFR Part 54), "Requirements for Renewal of Operating Licenses for Nuclear Power Plants," and prepared this report to document the results of its safety review. The NRC license renewal project managers for the DNPS and QCNPS safety review are Mr. Rajender Auluck and Mr. T. J. Kim. Mr. Auluck may be contacted by telephone at (301) 415-1025 or by electronic mail at RCA@nrc.gov. Alternatively, written correspondence can be sent to the following address:

License Renewal and Environmental Impacts Program
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001
Attention: R. Auluck, Mail Stop O-11F1

In its January 3, 2003, submittal letter, the applicant requested renewal of the operating licenses issued under Section 104b of the Atomic Energy Act of 1954, as amended, for DNPS Unit 2 (License No. DPR-19), DNPS Unit 3 (License No. DPR-25), QCNPS Unit 1 (License No. DPR-29), and QCNPS Unit 2 (License No. DPR-30) for a period of 20 years beyond the current license expirations of midnight, December 22, 2009, January 12, 2011, December 14, 2012, and December 14, 2012, respectively. The DNPS is located in Grundy County, Illinois, on the shore of a man-made cooling lake, with the Illinois River to the north and the Kankakee River to the east. The QCNPS is located in Rock Island County, Illinois, on the east bank of the Mississippi River opposite the mouth of the Wapsipinicon River, and about 3 miles north of Cordova, Illinois. Units 2 and 3 of the DNPS and Units 1 and 2 of the QCNPS each consist of a General Electric boiling-water reactor (BWR/3) authorized to individually operate at a steady-state reactor power level not to exceed 2957 megawatts-thermal, or approximately 850 megawatts-electric. Details concerning the plant and the site are found in the updated final safety analysis report (UFSAR) for DNPS/QCNPS.

The license renewal process proceeds along two tracks, which include both a technical review of safety issues and an environmental review. The requirements for these two reviews are specified in NRC regulations 10 CFR Parts 54 and 51, respectively. The safety review for the DNPS and QCNPS license renewals is based on the applicant's LRA, docketed correspondence, and the answers to requests for additional information (RAIs) from the NRC staff. In meetings and docketed correspondence, the applicant has also supplemented its answers to the RAIs. Unless otherwise noted, the staff reviewed and considered information submitted through June 22, 2004. The public can review the LRA and all pertinent information and material, including the UFSAR, at the NRC Public Document Room, 11555 Rockville Pike,

Rockville, Maryland 20852-2738. In addition, the DNPS/QCNPS LRA and significant information and material related to the license renewal review are available on the NRC's web page at http://www.nrc.gov

This SER summarizes the findings of the staff's safety review of the DNPS/QCNPS LRA and delineates the scope of the technical details considered in evaluating the safety aspects of the proposed operation of the plants for up to an additional 20 years beyond the term of the current operating licenses. The staff reviewed the LRA in accordance with NRC regulations and the guidance presented in NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR), which the NRC issued in July 2001.

Sections 2 through 4 of the SER document the staff's review and evaluation of license renewal issues that it considered during the review of the LRA. Section 5 is reserved for the report of the Advisory Committee on Reactor Safeguards (ACRS). The conclusions of this report are in Section 6 of the SER.

Appendix A is a list of commitments made by EGC. Appendix B is a chronology of the principal correspondence between the NRC and the applicant related to the review of the LRA. Appendix C is a list of the principal NRC staff reviewers and its contractors for this project. Appendix D is a list of the major references used in support of this SER.

In accordance with 10 CFR Part 51, the staff prepared plant-specific supplements to the generic environmental impact statement (GEIS). These supplements discuss the environmental considerations related to renewing the licenses for DNPS and QCNPS. The plant-specific supplements to the GEIS were issued separately. The NRC staff issued Supplement 17 to NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants Regarding the Dresden Nuclear Power Station, Units 2 and 3," on June 29, 2004, and Supplement 16 to NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants Regarding the Quad Cities Nuclear Power Station, Units 1 and 2," on June 30, 2004.

1.2 License Renewal Background

Pursuant to the Atomic Energy Act of 1954, as amended, and NRC regulations, operating licenses for commercial power reactors are issued for up to 40 years. These licenses can be renewed for up to 20 additional years. The original 40-year license term was selected on the basis of economic and antitrust considerations, rather than on technical limitations. However, some individual plant and equipment designs may have been engineered on the basis of an expected 40-year service life.

In 1982, the NRC anticipated interest in license renewal and held a workshop on nuclear power plant aging. The NRC team then established a comprehensive program plan for nuclear plant aging research (NPAR). On the basis of the results of that research, a technical review group concluded that many aging phenomena are readily manageable and do not pose technical issues that would preclude extending the life of nuclear power plants. In 1986, the NRC published a request for comment on a policy statement that would address major policy, technical, and procedural issues related to license renewal for nuclear power plants.

In 1991, the NRC published the license renewal rule in 10 CFR Part 54 (the Rule). The NRC participated in an industry-sponsored demonstration program to apply the Rule to a pilot plant and develop experience to create implementation guidance. To establish a scope of review for license renewal, the Rule defined age-related degradation unique to license renewal. However, during the demonstration program, the NRC found that many aging mechanisms occur and are managed during the period of the initial license. In addition, the NRC found that the scope of the review did not allow sufficient credit for existing aging management programs (AMPs), particularly for the implementation of the Maintenance Rule (10 CFR 50.65), which also manages plant aging phenomena.

As a result, in 1995, the NRC amended 10 CFR Part 54. The amended license renewal rule establishes a regulatory process that is simpler, more stable, and more predictable than the previous license renewal rule. In particular, 10 CFR Part 54 was amended to focus on managing the adverse effects of aging rather than on identifying age-related degradation unique to license renewal. The Rule changes were intended to ensure that important systems, structures, and components (SSCs) within the scope of the Rule will continue to perform their intended functions during the period of extended operation. In addition, the integrated plant assessment (IPA) process was clarified and simplified to be consistent with the revised focus on passive, long-lived structures and components (SCs).

In parallel with these efforts, the NRC pursued a separate rulemaking effort to amend 10 CFR Part 51 to focus the scope of the review of environmental impacts of license renewal and to fulfill, in part, the NRC's responsibilities under the National Environmental Policy Act of 1969 (NEPA).

1.2.1 Safety Reviews

License renewal requirements for power reactors are based on two principles:

(1) The regulatory process is adequate to ensure that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety, with the possible exception of the detrimental effects of aging on the functionality of certain SSCs during the period of extended operation, as well as a few other safety-related issues.

(2) The plant-specific licensing basis must be maintained during the renewal term in the same manner and to the same extent as during the original licensing term.

In implementing these two principles, 10 CFR 54.4 defines the scope of license renewal as including those plant SSCs (1) that are safety-related, (2) whose failure could affect safety-related functions, and (3) that are relied on to demonstrate compliance with the NRC's regulations for fire protection, environmental qualification (EQ), pressurized thermal shock (PTS), anticipated transients without scram (ATWS), and station blackout (SBO).

Pursuant to 10 CFR 54.21(a), the applicant for a license renewal must review all SSCs that are within the scope of the Rule to identify SCs that are subject to an aging management review (AMR). The SCs that are subject to an AMR are those that perform an intended function without moving parts, or without a change in configuration or properties, and that are not subject to replacement based on a qualified life or specified time period. As required by

10 CFR 54.21(a)(3), an applicant for a license renewal must demonstrate that the effects of aging will be managed in such a way that the intended function or functions of the SCs that are within the scope of license renewal will be maintained, consistent with the current licensing basis (CLB), for the period of extended operation. Active equipment, however, is considered to be adequately monitored and maintained by existing programs. In other words, the detrimental effects of aging that may affect active equipment are more readily detectable and will be identified and corrected through routine surveillance, performance monitoring, and maintenance activities. The surveillance and maintenance programs for active equipment, as well as other aspects of maintaining the plant design and licensing basis, are required to continue throughout the period of extended operation.

Pursuant to 10 CFR 54.21(d), the LRA is required to include a supplement to the updated final safety analysis report (UFSAR). This UFSAR Supplement must contain a summary description of the applicant's programs and activities for managing the effects of aging.

Another requirement for license renewal is the identification and updating of time-limited aging analyses (TLAAs). During the design phase for a plant, certain assumptions are made about the initial length of time the plant will be operated, and these assumptions are incorporated into design calculations for several of the plant's SSCs. In accordance with 10 CFR 54.21(c)(1), these calculations must be shown to be valid for the period of extended operation or must be projected to the end of the period of extended operation, or the applicant must demonstrate that the effects of aging on these SSCs will be adequately managed for the period of extended operation.

In July 2001, the NRC issued Regulatory Guide (RG) 1.188, "Standard Format and Content for Applications to Renew Nuclear Power Plant Operating Licenses"; NUREG-1800, "Standard Review Plan for the Review of License Renewal Applications for Nuclear Power Plants" (SRP-LR); and NUREG-1801, "Generic Aging Lessons Learned (GALL) Report." These documents describe methods acceptable to the NRC staff for implementing the license renewal rule and techniques used by the NRC staff in evaluating applications for license renewal. The RG endorses an implementation guideline prepared by the Nuclear Energy Institute (NEI) as an acceptable method of implementing the license renewal rule. The NEI guideline, NEI 95-10, "Industry Guideline for Implementing the Requirements of 10 CFR Part 54—The License Renewal Rule," Revision 3, was issued in March 2001.

Exelon Generation Company is the fifth license renewal applicant to fully utilize the process defined in NUREG-1801, otherwise known as the GALL Report. The purpose of the GALL Report is to provide the staff with a summary of staff-approved AMPs for the aging of most SCs that are subject to an AMR. If an applicant commits to implementing these staff-approved AMPs, the time, effort, and resources used to review an applicant's LRA will be greatly reduced, thereby improving the efficiency and effectiveness of the license renewal review process. The GALL Report summarizes the aging management evaluations, programs, and activities credited for managing aging for most of the SCs used throughout the industry. The report also serves as a reference for both applicants and staff reviewers to quickly identify those AMPs and activities that the staff has determined will provide adequate aging management during the period of extended operation.

1.2.2 Environmental Reviews

In December 1996, the staff revised the environmental protection regulations in 10 CFR Part 51 to facilitate environmental reviews for license renewal. The staff prepared the "Generic Environmental Impact Statement (GEIS) for License Renewal of Nuclear Plants" (NUREG-1437, Revision 1) to document its evaluation of the possible environmental impacts associated with renewing licenses of nuclear power plants. For certain types of environmental impacts, the GEIS establishes generic findings that are applicable to all nuclear power plants. These generic findings are identified as Category 1 issues in Subpart A of Appendix B to 10 CFR Part 51. Pursuant to 10 CFR 51.53(c)(3)(i), an applicant for license renewal may incorporate these generic findings in its environmental report. Analyses of the environmental impacts of license renewal that must be evaluated on a plant-specific basis (i.e., Category 2 issues) must be included in an environmental report in accordance with 10 CFR 51.53(c)(3)(ii).

In accordance with NEPA and the requirements of 10 CFR Part 51, the NRC performed a plant-specific review of the environmental impacts of license renewal, including whether new and significant information was not considered in the GEIS. Two public meetings were held, one near QCNPS on December 16, 2003, and one near DNPS on January 14, 2004, as part of the NRC's scoping process to identify environmental issues specific to each plant. The results of the environmental reviews and recommendations on the license renewal actions are documented in the NRC plant-specific Supplements 16 and 17 to the GEIS, which were issued on June 30, 2004 and June 29, 2004, for QCNPS and DNPS, respectively.

1.3 Principal Review Matters

The requirements for renewing operating licenses for nuclear power plants are described in 10 CFR Part 54. The staff performed its technical review of the DNPS/QCNPS LRA in accordance with Commission guidance and the requirements of 10 CFR Part 54. The standards for renewing a license are contained in 10 CFR 54.29. This SER describes the results of the staff's safety review.

In 10 CFR 54.19(a), the Commission requires a license renewal applicant to submit general information. The applicant provided this general information in Chapter 1 of its LRA for DNPS/QCNPS, submitted by letter dated January 3, 2003. The staff finds that the applicant has submitted the information required by 10 CFR 54.19(a) in Section 1 of the LRA.

In 10 CFR 54.19(b), the Commission requires that license renewal applications (LRAs) include "conforming changes to the standard indemnity agreement, 10 CFR 140.92, Appendix B, to account for the expiration term of the proposed renewed license." The applicant stated the following in Section 1.1.10 of its LRA regarding this issue:

> The current indemnity agreement for Dresden and Quad Cities state in Article VII that the agreement shall terminate at the time of expiration of the licenses specified in Item 3 of the Attachment to the agreement. Item 3 of the Attachment to the indemnity agreement, lists license numbers, DPR-19, DPR-25, DPR-29, and DPR-30. Applicant requests that any necessary conforming changes be made to Article VII and Item 3 of the Attachment, and any other sections of the indemnity agreement as appropriate to ensure that the indemnity agreement continues to apply during both the terms of the current licenses and the terms of the renewed licenses. Applicant understands that no changes may be necessary for this purpose if the current license numbers for each of the units are retained.

The staff intends to maintain the original license number upon issuance of the renewed license. Therefore, there is no need to make conforming changes to the indemnity agreement, and the requirements of 10 CFR 54.19(b) have been met.

In 10 CFR 54.21, the Commission requires that each LRA for a nuclear facility contain (a) an IPA, (b) CLB changes during staff review of the LRA, (c) an evaluation of TLAAs, and (d) a UFSAR Supplement. Sections 3 and 4 and Sections A and B of the LRA address the license renewal requirements of 10 CFR 54.21(a), (c), and (d), respectively.

In 10 CFR 54.21(b), the Commission requires that each year following submission of the application, and at least 3 months before scheduled completion of the staff's review, an amendment to the renewal application must be submitted that identifies any changes to the CLB of the facility that materially affect the contents of the LRA, including the UFSAR Supplement. The applicant's update to the LRA was issued on March 5, 2004.

In 10 CFR 54.22, the Commission outlines requirements regarding technical specifications. In Appendix D of the LRA, the applicant stated that no technical specification changes had been identified as being necessary to support issuance of the renewed operating licenses for DNPS/QCNPS. This adequately addresses the requirements of 10 CFR 54.22.

The staff evaluated the technical information required by 10 CFR 54.21 and 10 CFR 54.22 in accordance with the NRC's regulations and the guidance provided by the SRP-LR. The staff's evaluation of the LRA in accordance with 10 CFR 54.21 and 10 CFR 54.22 is contained in Sections 2, 3, and 4 of this SER.

The staff's evaluation of the environmental information required by 10 CFR 54.23 is contained in the final plant-specific supplement to the GEIS, which states the considerations related to renewing the licenses for DNPS/QCNPS. This was prepared by the staff separate from this report. The report of the ACRS, required by 10 CFR 54.25, is incorporated into Section 5 of this SER. The findings required by 10 CFR 54.29 can be found in Section 6 of this SER.

1.4 Interim Staff Guidance

The license renewal program is a living program. The staff, industry, and other interested stakeholders gain experience and develop lessons learned with each renewed license. The lessons learned address the Commission's performance goals of maintaining safety, improving effectiveness and efficiency, reducing regulatory burden, and increasing public confidence. The lessons learned are captured in interim staff guidance (ISG) for use by the staff and interested stakeholders until the improved license renewal guidance documents are revised.

The current set of relevant ISGs and the SER sections in which the issues are addressed by the staff are provided in the following table.

Interim Staff Guidance for License Renewal		
ISG Issue (Approved ISG No.)	**Purpose**	**SER Section**
Station Blackout (SBO) Scoping (ISG-02)	The license renewal rule 10 CFR 54.4(a)(3) includes 10 CFR 50.63(a)(1)—SBO. The SBO rule requires that a plant must withstand and recover from an SBO event. The recovery time for offsite power is much faster than that of emergency diesel generators (EDG)s. The offsite power system should be included within the scope of license renewal.	**2.5.1.5.2** **3.5.2.4.2**
Concrete Aging Management Program (ISG-03)	Lessons learned from the GALL demonstration project indicated that GALL is not clear whether concrete need an AMP.	**3.5.2.2.1** **3.5.2.2.2** **3.5.2.4.1** **3.5.2.4.2**
Fire Protection (FP) System Piping (ISG-04)	This ISG clarifies the staff position on wall thinning of FP piping system in GALL AMPs XI.M26 and XI.M27. The new position is that there is no need to disassemble FP piping, as oxygen can be introduced in the FP piping which can accelerate corrosion. Instead, non-intrusive methods such as volumetric inspection should be used. Testing of sprinkler heads should be performed every 50 years and 10 years after initial service. This ISG eliminated Halon/carbon dioxide system inspections for charging pressure, valve line ups, and automatic mode of operation test from GALL. The staff considers these test verifications to be operational activities.	**3.3.2.3.2** **3.3.2.3.3** **3.3.2.4.6**
Identification and Treatment of Electrical Fuse Holder (ISG-05)	This ISG includes fuse holder AMR and AMP (i.e., same as terminal blocks and other electrical connections). The position includes only fuse holders that are not inside the enclosure of active components (e.g., inside of switchgears and inverters). Operating experience finds that metallic clamps (spring-loaded clips) have a history of age-related failures from aging stressors such as vibration, thermal cycling, mechanical stress, corrosion, and chemical contamination. The staff finds that visual inspection of fuse clips is not sufficient to detect the aging effects from fatigue, mechanical stress, and vibration.	**3.6.2.4.1**

1.5 Summary of Open Items

As a result of its review of the LRA for QCNPS and DNPS, including additional information submitted to the NRC through June 22, 2004, the staff identified the following open items. An issue was considered open if the applicant had not presented a sufficient basis for resolution. Each open item (OI) has been assigned a unique identifying number. The items identified in this section have been properly closed by the technical staff.

OI-2.1-1: (Section 2.1.3.1.2 - Application of the Scoping Criteria in 10 CFR 54.4(a)(2))

The staff determined that the applicant did not provide a sufficient basis for limiting consideration of fluid spray interactions to only those non-safety-related SSCs located within 20 feet of an active safety-related SSC. In particular, the staff required additional clarification regarding the capability of active and passive safety-related SSCs located greater than 20 feet from a potential spray source to tolerate wetting, the specific operating experience that was relied upon to determine that it was not credible for fluid sprays to affect equipment greater than 20 feet from a failure location, specific methods to detect leakage in normally accessible and inaccessible areas, and justification for use of exposure duration in limiting the scope of potential failure mechanisms considered during scoping. This issue was identified as Open Item 2.1-1.

The applicant responded to Open Item 2.1-1 by letters dated April 9, 2004 and May 18, 2004 (ADAMS Accession Nos. ML041070456, and ML041480178). In addressing this open item, the applicant revised the scoping methodology for nonsafety-related moderate energy piping systems that have the potential to spatially interact with safety-related systems. Specifically, the applicant eliminated the 20 foot separation criterion and credit for the early detection of leakage that was previously used to exclude certain moderate energy nonsafety-related piping and components from the scope of License Renewal. The revised methodology assumes that all safety-related components, active as well as passive, could be adversely affected by spray or wetting from a non-safety moderate energy system located in the same general area of the plant. Therefore, the applicant stated that all components from moderate energy nonsafety-related systems located in the same general area as a safety-related component (active or passive) would be included within the scope of license renewal. The applicant defined “General area" as the same floor (elevation) of a major building with no barrier walls between the fluid source and the safety-related component. Barrier walls were defined as barriers that form the boundary of a room on the same elevation of a major building separating the safety-related components from a spray or leak generated by a non-safety-related component located on the other side of the barrier wall. The applicant stated that all barrier walls credited for protection of safety-related components were previously included within the scope of license renewal during structural scoping and subject to aging management review.

In accordance with the revised methodology, the applicant expanded the license renewal boundaries of seventeen systems previously determined to be within the scope of license renewal and identified an additional intended function for the main condenser at Quad Cities. Additionally, the applicant added the following five nonsafety-related systems to the scope of license renewal that were previously excluded from the scope of license renewal: circulating water (Dresden and Quad Cities), laundry (Dresden), zinc addition (Dresden and Quad Cities), extraction steam (Quad Cities), and feedwater heater vents and drains (Quad Cities) . In its

May 18, 2004 response to Open Item 2.1-1, the applicant identified LRA revisions, scoping results changes, and aging management program changes required as a result of the scoping methodology revision. The staff review of these revised scoping results and associated aging management programs are described in Sections 2.3 and 3.0.5 of this report.

On the basis of the above, the staff concludes that the applicant adequately resolved the issues identified in Open Item 2.1-1. Specifically, the elimination of the twenty foot limitation on spray interactions, consideration of potential adverse effects for both active and passive safety-related equipment, and elimination of credit for early detection of leakage adequately addressed the staff's methodology concerns. Furthermore, the staff determined that the applicant's revised methodology considered a reasonable spectrum of potential nonsafety-related spatial interactions with safety-related equipment. Therefore, the staff concludes that the revised methodology for scoping nonsafety-related equipment provides reasonable assurance that the applicant considered nonsafety-related SSCs whose failure could prevent satisfactory accomplishment of a safety-related intended function within the scope of license renewal. On this basis, Open Item 2.1-1 is resolved.

OI-3.5.2.3.2-1: (Section 3.5.2.3.2- ASME Section XI, Subsection IWF (B.1.27))

Some of the Class MC supports discussed by the applicant in the RAI responses regarding Class MC supports seemed to be inaccessible. Therefore, the staff needed to better understand how the applicant is treating these supports. This was identified as Open Item 3.5.2.3.2-1.

To resolve the concerns, the staff requested the applicant to provide the following information:

(a) Identify each type of Class MC support by name and confirm whether the support will be inspected under IWF during the period of extended operation. Provide a technical explanation for those supports that are proposed to be inspected under another program (such as IWE or Structures Monitoring) or for cases where no inspection is planned.

(b) Since Class MC supports are not currently being inspected, provide a commitment to perform a baseline inspection of typical samples of each type of Class MC component support prior to the period of extended operation, to identify and correct any problems affecting performance of intended functions.

(c) Describe how the performance of Class MC component supports in inaccessible areas are currently being managed and how they will be managed during the period of extended operation. Clarify the commitment to the provisions of 10 CFR 50.55(a) covering inaccessible areas.

(d) Review the response to RAI 2.4-2 and identify the aging management program applicable to each item (a) through (k). Also verify the consistency of this RAI response with the response to RAI 2.4-10.

The applicant submitted the responses by letter dated April 9, 2004. After reviewing the applicant's responses, the staff accepts the applicant's proposed use of its Structures Monitoring Program as an alternate AMP to the GALL's ASME IWF program for its Class MC piping supports, with the following modifications.

Modification #1 states that the sample size of the Class MC piping supports should be 15% of the support population, as stipulated in Table IWF-2500-1, because the ten sample supports proposed by the applicant were insufficient.

Modification #2 states that the person who performs the inspection should have demonstrated knowledge of inspection attributes on Class MC piping supports and should be under oversight guidance from the administrator or his designee during the initial inspection activity.

Modification #3 states that a baseline inspection should be performed on the sample supports prior to the period of extended operation.

The applicant submitted its revised responses in a letter dated June 22, 2004. The responses satisfactorily resolve the sample size and inspector's qualification issues. However, the staff was not sure whether the applicant intended to only revise its Structures Monitoring Program prior to the period of extended operation or actually have the MC supports and MC piping sample supports inspected prior to the period of extended operation. In a telephone conference on July 13, 2004, the applicant clarified that a baseline inspection would be performed for these supports prior to the period of extended operation. This is part of Commitment #30 in Appendix A of this SER. The staff considers the Open Item 3.5.2.3.2-1 resolved.

OI-4.2.1(c): (Section 4.2.2.1 - Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement)

In RAI 4.2.1(c), the staff requested the applicant to provide all fluence data for all welds and plates in the beltline and specify which one is bounding with respect to the RPV USE evaluation. In response to RAI 4.2.1(c), in a letter dated October 3, 2003, the applicant provided 54-EFPY surface fluences and 54 EFPY 1/4T fluences for all the beltline material but identified materials that are bounding with respect to the RPV material ART values at 54-EFPY. The applicant also needed to identify the USE for all beltline materials at 54-EFPYs and to identify the limiting materials for each unit. This was identified as Open Item 4.2.1(c).

The applicant's April 9, 2004, letter indicated that all beltline materials, except for the electroslag welds (ESWs) in Quad Cities Unit 2, will have predicted Charpy USE greater than 35 ft-lb, the minimum allowable USE based on the generic BWR equivalent margins analysis documented in BWROG topical report entitled, " 10 CFR Part 50 Appendix G Equivalent Margin Analysis for Low Upper Shelf Energy in BWR/2 Through BWR-6 Vessels." Therefore, all beltline materials, except for the ESW in Quad Cities Unit 2, meet the margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code.

The applicant reevaluated the USE value for Quad Cities Unit 2 ESW using all electroslag weld material surveillance test results from Quad Cities Unit 2, and performed a plant-specific EMA for the Quad Cities Unit 2 ESW. General Electric report GE-NE-0000-0027-0575-01, Revision 0, "The Upper Shelf Energy Evaluation for RPV Electroslag Welds at Quad Cities Unit 2," issued March 5, 2004, included in the applicant's April 9, 2004 letter, contains this analysis. Using the limiting surveillance capsule 18 data and the methodology in RG 1.99, Revision 2, the predicted Charpy USE for the ESW welds is 34.2 ft-lb, which is below the minimum established in the generic BWROG topical report. The applicant's plant-specific EMA was performed using

methods and criteria contained in RG 1.161, “Evaluation of Reactor Pressure Vessels with Charpy Upper-Shelf Energy less than 50 Ft-Lb.” and Appendix K of ASME Code, Section XI. Appendix K and RG 1.161 provide acceptance criteria and evaluation procedures for determining acceptability for operation of a reactor vessel when the vessel metal temperature is in the upper shelf range. The methodology is based on the principles of elastic-plastic fracture mechanics. Flaws will be postulated in the reactor vessel at locations of predicted low upper shelf Charpy impact energy, and the applied J-integral for these flaws will be calculated and compared with the J-integral fracture resistance of the material to determine acceptability. The applicant’s analysis showed that the applied J-integral of the postulated flaws and the J-integral material fracture resistance with a minimum USE of 32.4 ft-lb satisfies the criteria of Appendix K of the ASME Code, Section XI and RG 1.161.

The analysis methods in Appendix K of the ASME Code initially followed the methodology in RG 1.161. The analysis methods in Appendix K of the ASME Code, Section XI were changed in the 1995 Addenda to the 1995 Edition. The analysis method in the 1995 Addenda to the 1995 Edition of the ASME Code changed the method of calculating the contribution to the applied J-integral because of a radial thermal gradient. This change was incorporated into the ASME Code to more accurately represent the contribution to the applied J-integral due to a radial thermal gradient. The applicant’s analysis was performed using the earlier analysis method, i.e., the methods contained in RG 1.161. The staff confirmed the EMA using the analysis methods in both Appendix K to the ASME Code, Section XI, 1995 Addenda to the 1995 Edition, and the earlier analysis method in RG 1.161. This analysis included the effects of the extended power uprate condition. Since the limiting end of extended life USE for Quad Cities Unit 2 ESW exceeds the minimum value of 32.4 ft-lb demonstrated in the applicant’s plant-specific EMA, the staff concludes that all beltline materials, including the ESW in Quad Cities Unit 2 RPV meet the margins of safety against fracture equivalent to those required by Appendix G of Section XI of the ASME Code. Therefore Open Item 4.2.1(c) is closed.

OI-B.1.23-2: (Section 3.0.3.10.2 - One Time Inspection (B.1.23) - Plant Heating System components) and (Section 3.4.2.4.1 Main Steam System-One Time Inspection (B.1.23) - Plant Heating System components)

By RAIs B.1.23-1, B.1.23-2, B.1.23, and B.1.23-2.1 through B.1.23-2.6, the applicant was requested to justify use of the One-Time Inspection program to manage aging effects for various carbon steel, alloy steel, stainless steel, cast iron, and neoprene components in environments such as moist air, steam, water (condensate), and containment atmosphere. By letters dated October 3, 2003, January 26, 2004, and March 25, 2004, the applicant responded to the staff’s RAIs as follows:

1) By RAIs B.1.23-1, B.1.23-2(a), B.1.23-2.3 and B.1.23-2.4, the staff questioned use of the One-Time Inspection program to manage loss of material and cracking for carbon steel, stainless steel, cast iron, brass or bronze, and iron components in lube oil and fuel oil environments. This was identified as Confirmatory Item B.1.23-1. By letters dated October 3, 2003, January 26, 2004 and April 9, 2004, the applicant stated that aging management program, B.2.5, “Lubricating Oil Monitoring Activities,” will be expanded to manage loss of material and cracking for oil coolers and other components in lube oil, turbine electro-hydraulic control (EHC) fluid, and generator hydrogen seal oil environments for the emergency diesel generator system, station blackout diesel generator system, high pressure coolant injection

system, electro-hydraulic control system, reactor core isolation cooling system (Quad Cities), and generator hydrogen seal oil system (Quad Cities). Aging management program, B.1.23, "Fuel Oil Chemistry," will be expanded to manage loss of material for components in a fuel oil environment for the station blackout diesel generator system. The One-Time Inspection program will not be credited to manage the aging effects for these components since periodic inspections will be implemented. The staff considers the Lubricating Oil Monitoring Activities and Fuel Oil Chemistry aging management programs appropriate to manage these aging effects; therefore, staff finds this acceptable.

2) Table 3.3.2 of the LRA identifies components in the Plant Heating System which credit the One-Time Inspection program to manage aging effects for components in a saturated steam or condensate environment. By RAIs B.1.23-1 and B.1.23-2(a) & (b), staff requested the applicant to justify use of one-time inspections to manage the aging effects for these components. By letters dated January 26, 2004 and March 25, 2004, the applicant stated that Plant Heating System components in a saturated steam or condensate environment would be managed by aging management program, B.2.8, "Periodic Inspection of Plant Heating System." The program includes periodic inspections to manage cracking, loss of material, or leakage of selected brass/bronze, carbon steel, cast iron, and stainless steel components. The staff considers the Periodic Inspection of the Plant Heating System program appropriate to manage these aging effects; therefore, staff finds this acceptable.

3) For the main steam system flexible hoses in a containment nitrogen environment, Reference Number 3.4.2.18 of the LRA does not identify any aging effects for these neoprene hoses. By RAIs 3.4.1-3 and B.1.23-2.1, staff requested the applicant to justify with respect to temperature, radiation levels, and time, why neoprene hoses do not require aging management. In responses dated October 3, 2003 and January 26, 2004, further review by the applicant indicated that hoses in Reference Number 3.4.2.18 and 3.4.2.19 of the LRA are not comprised of an elastomer material as earlier reported but are made of stainless steel with an overall stainless steel outer braided jacket. Based on the hose material being stainless steel, the applicant will use the One-Time Inspection program to verify that the hoses are constructed of metal rather than an elastomer material. Any hoses found to be constructed of an elastomer during the one-time inspection will be replaced with metal flexible hoses. The One-Time Inspection program will perform inspections of the installed metal hoses for mechanical damage. This applies to Quad Cities only. The applicant has noted that stainless steel hoses are installed at Dresden. The staff considers use of the One-Time Inspection program acceptable to verify that stainless steel hoses are installed and to inspect the stainless steel hoses for damage.

4) For non-safety-related (NSR) vents or drains, piping, and valves in various systems, the LRA identifies loss of material due to corrosion for carbon steel, stainless steel, brass, or bronze in an environment of air, moisture, humidity, and leaking fluid. By RAI 3.3-2, the staff requested the applicant to describe the types of corrosion expected and to provide criteria for selecting one-time sample locations for these types of corrosion. The applicant stated in its letter dated October 3, 2003 that general, crevice, and pitting corrosion are expected in these components. The applicant compiled a list of the in-scope NSR vents and drains for the various systems throughout the plants. The One-Time Inspection program will inspect a selected number of NSR vent and drains for the affected systems. The sample population will be representative of all material and environment combinations but may not include components for every system. The criteria used for selection of susceptible inspection locations are as follows: 1)

Corrosiveness of fluid passing through the vent, drain, or piping when in service. Those components servicing more corrosive fluids are given preference. 2) Duration of service when performing venting and draining operations. Those components with higher durations of service are given preference. 3) Frequency of performance of venting and draining operations through the selected components. Those components with higher performance frequencies are given preference. 4) Period that component has been in service. Those components that have been in service longest are given preference. By RAIs B.1.23-2(b) and B.1.23-2.2, staff requested further justification that a one-time inspection is adequate to manage the aging effects for these vent, drains, and valves. By letters dated October 3, 2003 and January 26, 2004, the applicant stated that the NSR vents, drains, valves, and piping are normally outboard of closed safety relief valves or closed isolation valves and are not likely to contain moisture. Any appreciable leakage or condensation inside these vents and drains would be identified in the course of periodic operations or through the daily monitoring of unidentified inputs to radwaste by the operating department. Malfunctioning isolation valves or other degraded conditions are promptly repaired, replaced, or corrected. For the reasons stated above, the applicant considers the rate of material loss due to corrosion to be slow; therefore, one-time inspections will confirm the assumption that loss of material due to corrosion is occurring at a sufficiently slow rate for the subject components. In the event that the results of the one-time inspections fail to provide this confirmation, evaluations will be performed in accordance with the site corrective action process to identify actions, including possible periodic inspections of the vents and drains. Based on the applicant's response, staff concurs that the loss of material due to corrosion for the subject vents, drains, piping, and valves are considered to occur at a sufficiently slow rate such that a one-time inspection is adequate to manage this aging effect; therefore, staff finds this acceptable.

(5) By RAIs B.1.23-2 and B.1.23-2.6, the staff requested the applicant to provide justification for using one-time inspections to manage carbon steel, cast iron, alloy steel, elastomer, and neoprene components in a moist air environment that 1) varies with normal plant conditions, 2) is impractical to monitor or control routinely, and 3) is similar to the environments associated with the Aging Management References listed in part b of RAI B.1.23-2. This was identified as Open Item B.1.23-2. By letter dated March 25, 2004, the applicant concluded by further review that periodic inspections of components in this population would be appropriate. A new aging management program, B.2.9, "Periodic Inspection of Components Subject to Moist Air Environments," was developed for these components. Specifically, the applicant will perform periodic inspections of a representative sample of stainless steel, carbon steel, cast iron, aluminum, copper, brass, and bronze components normally exposed to environments of air and steam; moist air; saturated air; warm moist air; moist containment atmosphere; steam or demineralized water; and hot diesel engine exhaust gases containing moisture and particulates. In addition, the program inspects flexible hoses to detect age-related degradation prior to the loss of function.

The applicant considers a one-time inspection appropriate for managing aging effects for the standby gas treatment system and HVAC systems components with an internal environment of "occasional exposure to moist air" and an external environment of "ambient plant air" or "warm moist air." Components in these systems include doors, closure bolts, equipment frames, piping, fittings, valves, ducts, and filters fabricated of cast iron, carbon steel, brass, bronze, stainless steel, and copper. Based on the materials and environments for these ventilation system components, the applicant believes that either (a) an aging effect is not expected to occur but there is insufficient data to completely rule it out, or (b) an aging effect is expected to

progress very slowly. Based on favorable operating history that revealed no widespread corrosion in the affected system, a limited number of components were selected as representative of the ventilation systems. The worst-case one-time inspection locations will include the following: the air intake ductwork of the standby gas treatment system; main control room HVAC ductwork; emergency diesel generator HVAC air intake ductwork; reactor building HVAC ductwork downstream of the steam coils and chilled water cooling coils; and main control room HVAC drip pan and drainpipe. If the one-time inspection detects corrosion resulting in material loss, results of the examination will be evaluated by engineering to determine the rate of material loss and the need for additional inspections. Unacceptable results will be documented in the corrective action program.

Based on the applicant's response, staff considers the Periodic Inspection of Components Subject to Moist Air Environments acceptable to manage components in a moist air environment and the One-Time Inspection program acceptable to manage ventilation systems components where either (a) an aging effect is not expected to occur but there is insufficient data to completely rule it out, or (b) an aging effect is expected to progress very slowly. Therefore Open Item B.1.23-2 and Confirmatory Item B.1.23-1 are closed.

1.6 Summary of Confirmatory Items

As a result of its review of the LRA for QCNPS and DNPS, including additional information submitted to the NRC through June 22, 2004, the staff identified the following confirmatory items. An issue was considered confirmatory if the staff and the applicant have reached a satisfactory resolution, but the resolution has not yet been formally submitted to the staff. Each confirmatory item (CI) has been assigned a unique identifying number. The items identified in this section have been properly closed by the technical staff.

CI.2.3.4.2-3: (Section 3.1.2.4.1 - Reactor Vessel)

The staff needed the following information from the applicant so that it can evaluate the aging management of the capped CRD nozzles—(1) description of the configuration and location of the capped nozzle including the existing base material for the nozzle, piping (if piping remnants exist) and cap material, and any welds and material type (i.e., 82/182), (2) description of how these welds and caps are managed (e.g., the applicability of the BWRVIP-75 inspection requirements); and (3) discussion on whether the event at Pilgrim (leaking weld at capped nozzle, September 30, 2003) is applicable to Dresden and Quad Cities. A description of the Pilgrim event is discussed in LER 2003-006-00, dated November 24, 2003, which states that the cracking was in an 82/182 weld metal that was repaired extensively. The applicant also needed to include in the discussion the past inspection techniques applied, the results obtained, mitigative strategies followed, weld repairs carried out, and any other relevant information. This was identified as Confirmatory Item 2.3.4.2-3.

In the applicant's letters dated January 26, 2004, and April 9, 2004, the applicant responded to supplementary RAI 2.3.4.2-3. In the applicant's letters, the applicant provided information related to configuration and locations of the capped nozzles for each plant and described how they are managed. At Dresden and Quad Cities, the configuration consists of 304L and 316L SS caps and safe-ends welded to the original carbon steel nozzles. Aging management for these components includes examination in accordance with Section XI of the ASME Code for

the nozzle as stated in AMP B.1.6, “CRD Return Line Nozzle,” and one-time inspection in accordance with AMP B.1.23, “One-Time Inspections” for the remaining portion (safe-end, cap and welds). AMP B.1.2, “Water Chemistry” is also credited for these components.

In addition, the applicant stated that the Pilgrim event does not apply to Dresden and Quad Cities because (1) Pilgrim used an Alloy 600 cap welded directly to the nozzle whereas D/QCNPS used a SS cap and installed a SS safe-end between the cap and the nozzle, (2) Pilgrim used Alloy 82/182 welds whereas D/QCNPS used 308L and 309L SS welds, and (3) Pilgrim had initial weld defects (lack of fusion) that required repair, whereas D/QCNPS welds were completed without requiring any repair. D/QCNPS further stated that their nozzles and caps had radiographic and penetrant testing performed during installation, and had subsequent ultrasonic inspection of the nozzle-to-safe end welds and safe end-to-cap welds in response to the Pilgrim event with no reportable indications. Also, per the D/QCNPS ISI programs, penetrant testing had been performed on these welds with no recordable indications. In addition, Dresden and Quad Cities have placed their capped lines (small bore piping-less than 4 inches) in the One-Time Inspection Program, B.1.23. The staff finds the applicant's response acceptable because it uses low carbon stress corrosion resistant stainless steel safe-ends, caps, and weld material in lieu of Alloy 600, which has been known to be susceptible to stress corrosion cracking based on operating experience. In addition, the caps were welded using low carbon stainless steel weld metal (308L and 309L) with no weld repairs or recordable defects. Pilgrim used Inconel 82/182 and had initial weld defects that required weld repairs, which may have contributed to the cracking. Therefore, Dresden and Quad Cities capped return line nozzle configuration is not similar to Pilgrim and the use of AMPs B.1.2, B.1.6 and B.1.23 is acceptable for managing the aging of these components. Therefore, Confirmatory Item 2.3.4.2-3 is closed.

CI.3.0.3.14.2-1: (Section 3.0.3.14.2- Structures Monitoring Program (B.1.30))

The additional information provided by the applicant in its response to RAI B.1.30 sufficiently answers the questions posed by the staff, with two exceptions. It was not clear whether the category “Piping Component Supports including immediately adjacent piping/tubing,” listed in the response to item (a) of the RAI is meant to include non-ASME piping supports. It also was not clear as to why the Structures Monitoring Program does not include “standard components such as snubbers, struts and spring cans.” In order to completely resolve the response to this RAI, the staff requested that the applicant confirm the following:

(a) the B.1.30 program covers non-ASME piping supports
(b) there are no snubbers, struts and spring cans on non-ASME piping and components

This issue was identified as Confirmatory Item 3.0.3.14.2-1.

In its response to Confirmatory Item 3.0.3.14.2-1, dated December 5, 2003, the applicant stated:

> Exelon has reviewed the supplemental Information Request and provides the following clarification and confirmation.
>
> 1)The Structure Monitoring Program, B.1.30, includes non-ASME piping supports for aging management. The selection of component supports includes a representation of supports throughout the plant, considering environmental conditions as well as configuration.

2)There are standard components such as snubbers, struts, and spring cans on non-ASME piping and components that are in-scope of the License Renewal, which are required to be managed for aging. The Structural Monitoring Program, B.1.30, will inspect the non-ASME component supports including the standard components. The in-scope non-ASME component supports are addressed in LRA Section 2.4.15, Table 2.4-15 under the Component Groups "Support Members" with a "Non-S/R Structural Support" component intended function. Aging Management Reference 3.5.1.29 discusses the aging management of the non-ASME component supports.

The staff finds the applicant's response to Confirmatory Item 3.0.3.14.2-1 to be acceptable, because it clarified that the "Structural Monitoring Program," B.1.30, will inspect non-ASME piping and component supports, including snubbers, struts, and spring cans. This commitment is stated in the enhancements as "The program will provide for inspection of a sample of non-insulated indoor piping external surfaces at locations immediately adjacent to periodically inspected piping supports and inspection of standard components such as snubbers, struts, and spring cans." under B.1.30, Structures Monitoring Program, in the applicant's response to OI-3.5.2.3.2 1: (Section 3.5.2.3.2- ASME Section XI, Subsection IWF (B.1.27)), dated June 22, 2004. Therefore Confirmatory Item 3.0.3.14.2-1 is resolved. This is part of Commitment #30 in Appendix A of this SER.

CI.3.1.2.3.2-1: (Section 3.1.2.3.2 - BWR Vessel ID Attachment Welds Program)

In RAI 4.2-BWRVIPs, the staff requested the applicant to submit the necessary commitments, information, and changes for each of the following applicable BWRVIP reports:

- BWRVIP-05
- BWRVIP-18
- BWRVIP-25
- BWRVIP-26
- BWRVIP-27
- BWRVIP-38
- BWRVIP-41
- BWRVIP-42
- BWRVIP-47
- BWRVIP-48
- BWRVIP-49
- BWRVIP-74
- BWRVIP-75
- BWRVIP-76
- BWRVIP-78
- BWRVIP-86
- Other BWRVIP reports applicable to license renewal

In response to RAI 4.2-BWRVIPS, in a letter dated October 3, 2003, the applicant summarized the NRC's request for information in the seven elements listed below and presented its response to each of those elements.

(1) Verify that Dresden and Quad Cities are bounded by the conditions (materials configuration and inspection methodologies) specified in the applicable BWRVIP documents.

Response: The BWRVIP documents were assembled with participation from the NSSS

supplier and a wide representation from the BWR Owners Group, providing a level of confidence in accuracy and bounding conditions of these documents. However, during a preliminary review when preparing this response, some material differences were noted. Exelon will perform a detailed review of the applicable BWRVIP documents and verify that Dresden and Quad Cities are bounded by the conditions specified or identify and evaluate any exceptions noted.

(2) Provide a commitment to implement programs consistent with the applicable BWRVIP documents or identify the applicable exceptions.

Response: At the completion of the review noted in item 1 above, Exelon will provide a list of commitments to the applicable BWRVIP documents or identify specific exceptions taken.

(3) Describe how the commitments will be tracked.

Response: The commitments, once identified, will be placed in the site implementing procedures with traceability back to the license renewal commitment being made.

(4) Summarize a program description of the applicable BWRVIP documents in the LRA Appendix A, UFSAR Supplement.

Response: Several of the BWRVIP programs are identified in the LRA Appendix A, such as BWRVIP-75, A.1.7; BWRVIP-27, A.1.8; BWRVIP-48, A.1.4; BWRVIP-49, A.1.8; BWRVIP-78, A.1.22; and BWRVIP-86, A.1.22. Once the comprehensive list of commitments is identified in item 2 above, Exelon will update the LRA Appendix A to provide a summary program description to address each applicable BWRVIP document.

(5) Verify that technical specification changes needed to support implementation of the applicable BWRVIP documents have been identified and processed.

Response: There are no additional technical specification changes anticipated. However, once the detailed review summarized in item 1 above is complete, Exelon will confirm that no technical specification changes are needed or identify the needed changes to be processed prior to the start of the extended term of operation.

(6) Identify and evaluate any potential TLAA issue identified by the applicable BWRVIP documents and/or commitments to perform future inspections when inspection tooling is made available.

Response: All applicable TLAAs are discussed in Section 4 of the LRA.

(7) Address items 1 through 6 above for the 16 specific BWRVIP documents listed in the RAI and identify and address other BWRVIP documents applicable to license renewal.

Response: Based on a preliminary review, there appears to be several other BWRVIP documents applicable to license renewal, such as BWRVIP-07 and BWRVIP-63 for core shroud repairs, and BWRVIP-26 for Water Chemistry. Once the detailed review is completed, Exelon will provide an amended response addressing items 1 through 6 for all BWRVIP documents applicable to license renewal.

The staff found the applicant's response incomplete. The response committed to perform a detailed review of the BWRVIP documents applicable to license renewal, prepare an amended response addressing items 1 through 7 for all of those documents applicable to license renewal, and submit it to the staff for review and approval. Therefore, this response was incomplete until an amended response was submitted and approved by the staff. This was identified as Confirmatory Item 3.1.2.3.2-1.

In a letter dated April 9, 2004, the applicant submitted the following amended response to RAI 4.2-BWRVIPs addressing the seven items, which were listed in the initial response to RAI 4.2-BWRVIPs, for all of the BWRVIP documents applicable to license renewal.

1. Verify that Dresden and Quad Cities are bounded by the conditions (materials, configuration and inspection methodologies) specified in the applicable BWRVIP documents.

Amended Response: The site-specific procedures at D/QCNPS implemented all of the inspection methodologies contained in the applicable BWRVIP documents. Additionally, the materials and configurations at D/QCNPS are similar to those specified in the BWRVIP documents with an exception related to the steam dryer hold-down bracket attachment weld (addressed in response to Supplementary RAI B.1.4). Regarding inspection methodologies, the applicant has identified two exceptions related to BWRVIP-74: (1) use of risk-informed ISI to supplement the ISI and GL 88-01 programs for reactor pressure vessel nozzles and safe ends, and (2) use of an NRC-approved code case for the inspection of the reactor vessel leak detection line. The first exception is evaluated in SER Section 3.1.2.4.1 and the second one in SER Section 3.1.2.2.4.

2. Provide a commitment to implement programs consistent with the applicable BWRVIP documents or identify the applicable exceptions.

Amended Response: D/QCNPS provided a commitment for implementing the programs consistent with the applicable BWRVIP documents and identified several exceptions. These exceptions are associated with BWRVIP-38, BWRVIP-41, BWRIP-74, and BWRVIP-75 and are described in SER Sections 3.1.2.3.6 and 3.1.2.4 as appropriate. In addition, the applicant has committed to implement several BWRVIP reports that are being reviewed by the NRC, and will identify any exceptions associated with these reports after the staff's reviews are completed. See amended response 7 for the several BWRVIP reports being reviewed by the NRC. This is part of Commitment #9 in Appendix A of this SER.

3. Describe how the commitments will be tracked.

Amended Response: All license renewal commitments are controlled by the Exelon commitment management process described in LS-AA-110, Commitment Management. Commitment tracking files will be generated for each individual activity credited to implement the requirements of the AMP. In addition, steps in site procedures that implement the various activities specified in the BWRVIP documents are annotated as NRC commitments and are referenced to commitment tracking files that contain sufficient documentation describing the source of the commitment.

4. Summarize a program description of the applicable BWRVIP documents in the LRA Appendix A, UFSAR Supplement.

Amended Response: The FSAR Supplement (LRA Appendix A) Programs A.1.1, A.1.2, A.1.4, A.1.8, A.1.9, and A.1.22 have been updated to reflect the applicable BWRVIP documents, and exceptions as noted in response to Item 2 above. A revised FSAR supplement incorporating these changes was submitted to NRC in the attachment to Exelon transmittal letter dated March 5, 2004 as part of the annual update required by 10 CFR 54.21(b).

5. Verify that technical specification changes needed to support implementation of the applicable BWRVIP documents have been identified and processed.

Amended Response: The only Technical Specification change required for both sites involves revision to the site pressure temperature (P-T) curves. The existing P-T curves will be revised for 54 EFPY prior to the extended term of operation.

6. Evaluate any potential TLAA issue identified by the applicable BWRVIP documents and/or commitments to perform future inspections when inspection tooling is made available.

Amended Response: All applicable TLAA's were discussed in Section 4 of the LRA. The applicant also committed to perform future inspections, as recommended by the BWRVIP documents, when inspection tooling is made available. This commitment is discussed in SER Section 3.1.2.3.6.

7. Address Items 1 through 6 above for the 16 specific BWRVIP documents listed in the RAI and identify and address other BWRVIP documents applicable to license renewal.

Amended Response: In addition to the 16 specific BWRVIP documents listed in RAI 4.2-BWRVIPs, the applicant has identified four additional documents applicable to license renewal: BWRVIP-29, BWRVIP-79, BWRVIP-104, and BWRVIP-116. NRC has issued a safety evaluation report for the first document (BWRVIP-29) but not for the remaining three. However, the applicant has provided an amended response in their letter dated April 9, 2004, addressing items 1 through 6 for all 20 BWRVIP documents applicable to license renewal and has committed to implement these 20 BWRVIP documents as discussed in the amended response to Item 2.

The staff found the responses to RAI 4.2-BWRVIPs acceptable because they addressed all the license renewal applicant action items as identified in the applicable BWRVIP reports, which are listed in the response. In addition, the exceptions identified by the applicant are approved by the staff. The staff has reviewed the updated FSAR Supplement programs and found that they include adequate summary descriptions of the applicable BWRVIP documents. Thus the responses are consistent with the BWRVIP reports applicable to license renewal. Therefore, Confirmatory Item 3.1.2.3.2-1 is closed.

CI.3.1.2.3.6-1: (Section 3.1.2.3.6 - BWR Vessel Internals Program)

The staff issued RAI B.1.9-b requesting the applicant to confirm whether D/QCNPS follows the BWRVIP-25 guidelines for managing aging of the rim hold-down bolts and, if so, to identify and evaluate whether the projected stress relaxation in the rim hold-down bolts is a TLAA issue. In response to RAI B.1.9-b, in a letter dated October 3, 2003, the applicant stated that D/QCNPS follows the BWRVIP-25 guidelines for management of the hold-down bolts. However, the

D/QCNPS core plates had wedges installed along with the repair of their shrouds with tie rods. The applicant further stated that BWRVIP-25 does not recommend inspection of rim hold-down bolts if wedges are installed. The staff reviewed BWRVIP-25 and confirmed the accuracy of the applicant's statements made in this response. The staff finds the applicant's response acceptable because it follows the recommendations of BWRVIP-25, which is approved by the staff. However, the applicant did not identify whether stress relaxation in the rim hold-down bolts is a TLAA. This was identified as Confirmatory Item 3.1.2.3.6-1.

In response, the applicant stated that the stress relaxation of the rim hold-down bolts is not a TLAA for Dresden or Quad Cities. Dresden and Quad Cities have installed wedge retainers, which structurally replace the lateral load resistance provided by the rim hold-down bolts. As such, the failure of the bolts due to stress relaxation is no longer a concern and inspection of the bolts is not required. Therefore the stress relaxation of the rim hold-down bolts does not meet the TLAA Criterion 5 - "involve conclusions or provide the basis for conclusion related to the capability of the core plate to perform its intended function." Additionally, neither the rim hold-down bolts, nor the wedges meet TLAA Criterion 3 - "time-limited assumptions defined by the current operating term." The staff finds this response acceptable because the rim hold-down bolts no longer provide structural load and do not meet the definition of a TLAA as defined in 10 CFR 54.3(a)(3) and (5). In a letter dated January 26, 2004, the applicant submitted the information described above. Therefore, Confirmatory Item 3.1.2.3.6-1 is closed.

CI.3.1.2.3.8-1: (Section 3.1.2.3.8 - Reactor Vessel Surveillance program)

In response to Part 2 of Supplemental RAI B.1.22, in a letter dated November 21, 2003, the applicant stated that if staff does not approve the proposed BWRVIP-116, the applicant will provide a plant-specific surveillance plan for the license renewal period in accordance with 10 CFR Part 50, Appendices G and H, prior to entering the renewed license period. This is Commitment #22 in Appendix A of this SER. This is considered Confirmatory Item 3.1.2.3.8-1.

In response to Part 2 of Supplemental RAI B.1.22, in a letter dated November 21, 2003, the applicant stated that if the staff does not approve the proposed BWRVIP-116, Exelon will provide a plant-specific surveillance plan for the license renewal period in accordance with 10 CFR Part 50, Appendices G and H, prior to entering the renewed license period. This is part of Commitment #22 of Appendix A of this SER. This was identified as Confirmatory Item 3.1.2.3.8-1. The staff finds the response acceptable because the applicant commits to provide a plant-specific surveillance program for the license renewal period in accordance with 10 CFR Part 50, Appendices G and H, if the staff does not approve the proposed BWRVIP-116. In a letter dated April 9, 2004, the applicant concurred with Commitment #22. Therefore, Confirmatory Item 3.1.2.3.8-1 is closed.

CI.3.1.2.4.2-1: (Section 3.1.2.4.2 - Reactor Vessel Internals (Including Fuel Assemblies and Control Blades))

The response to RAI 3.1.7b states that Dresden and Quad Cities will implement the BWRVIP recommendations and manage the effects of aging of IASCC through AMPs B.1.2 (Water Chemistry) and B.1.9 (BWR Vessel Internals). AMP B.1.9 is consistent with NUREG-1801 which references the use of BWRVIP-26 for the inspection of the reactor vessel internals, including the top guide, and BWRVIP-76 for the inspection of the shroud. However, according

to Table 2-1 of BWRVIP-76, when fluences exceed 5 x 10^{20} n/cm^2, a plant-specific analysis is required to be submitted to the NRC. This issue was identified as Confirmatory Item 3.1.2.4.2-1.

In response to Confirmatory Item 3.1.2.4.2-1, in a letter dated April 9, 2004, the applicant states that Table 2-1 of BWRVIP-76 provides inspection guidance for welds in un-repaired core shrouds. Note 4 of Table 2-1 indicates that for plants where fluence at the shroud exceeds 5x10^{20} n/cm^2, a plant-specific analysis is required to be submitted to the NRC. However, this analysis is only for un-repaired core shrouds. Since the core shrouds at Dresden and Quad Cities have been repaired and the repairs structurally replace the horizontal welds, the plant-specific analysis suggested by Table 2-1 is not applicable to these shrouds. The applicant further states that the inspection frequencies for the D/QCNPS shrouds are determined using the guidance contained in Section 3 of BWRVIP-76. The applicant inspects the vertical core shroud welds in accordance with BWRVIP-76, Section 3. The staff finds the response consistent with BWRVIP-76. Since the applicant has committed to implement BWRVIP-76 when the staff SER is issued, this completes our review of this issue. Therefore, Confirmatory Item 3.1.2.4.2-1 is closed.

CI.4.2.1: (Section 4.2.2.1 - Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement)

The data for copper content in the limiting beltline plate and limiting beltline weld material presented in LRA Section 4.2.1 appear to be different from the data presented in Appendix F to the Dresden UFSAR. For example, LRA Table 4.2.1-2 lists 0.24 percent copper for the Dresden Unit 2 limiting beltline weld material, whereas Table 22 in Appendix F lists a maximum copper content of 0.21 percent for Dresden Unit 2. In RAI 4.2.1(b), the staff requested the applicant to resolve this apparent discrepancy. This was identified as Confirmatory Item 4.2.1.

In response to RAI 4.2.1(b), in a letter dated October 3, 2003, the applicant provided the following explanation:

> For the beltline region, Table 21 (Shell Course 57—Lower Shell) and Table 22 (Shell Course 58—Lower-Intermediate Shell) of the Dresden FSAR gives values actual chemical analysis of these materials. Tables 21 and 22 contain the chemical analysis for electroslag welds contained in the original FSAR. Since the original publication of the FSAR, the accepted best estimate chemistry for Electroslag Weld (ESW) materials used in B&W vessels accepted by the NRC staff is 0.24% Cu and 0.37% Ni. These values are reported in BAW-2258, "Evaluation of RT_{NDT}, USE and Chemical Composition of Core Region Electroslag Welds for Dresden Units 2 and 3," Framatome Technologies, January 1996, and were previously accepted by the NRC in its review of pressure temperature (P-T) limit curve report GE-NE-B13-02057-04R1a. Exelon submitted reactor vessel chemistry values to the NRC in July 1998 in response to Generic Letter 92-01, Supplement 1. The information provided in that response is included in NRC database RVID.

The staff accepts the applicant's response because the staff has verified the percentage of copper content given in LRA Tables 4.2.1-1 to 4.2.1-8 for the limiting beltline USE materials with the corresponding data in RVID. Therefore, Confirmatory Item 4.2.1 is closed.

CI.4.2.1(a): (Section 4.2.2.1 Reactor Vessel Materials Upper-Shelf Energy Reduction Due to Neutron Embrittlement)

The peak EPU fluence on the vessel is located at approximately 82 inches above the bottom of active fuel and is applied to the lower-intermediate shell and axial welds. Additionally, axial flux

distribution factors are applied to different elevations (by shell) in the beltline region. For the lower shell, the peak fluence is adjusted by the axial flux distribution factor based on an elevation approximately 42 inches above the bottom of active fuel, which represents the lower to lower-intermediate girth weld. The axial flux distribution factor for this location is 0.71. The applicant stated that it applied this factor for calculating the peak pre-EPU fluence for the lower to lower-intermediate shell girth weld and all lower shell materials. In a followup question to RAI 4.2.1(a), the staff requested the applicant to describe how the pre-EPU axial flux profile compares with the EPU axial flux profile. The staff also requested that the applicant submit information about how the axial flux distribution factor was used in calculating the peak-EPU fluence for the lower to lower-intermediate shell girth weld and all lower shell materials. This was identified as Confirmatory Item 4.2.1(a).

In a letter dated April 9, 2004, the applicant referred to Figure 2 in a letter from Exelon to NRC, "Additional Information Regarding Request for License Amendment for Pressure-Temperature Limits," dated July 31, 2003. This figure shows the pre-EPU and EPU axial flux distribution at the inside surface of the reactor pressure vessel. The pre-EPU and EPU axial flux distribution profiles are different, since the pre-EPU flux peaks at an elevation higher than the mid-plane, whereas the EPU flux peaks at the mid-plane. The applicant stated that for determining the peak 54-EFPY surface fluences at the lower shell plate material, lower shell welds and the lower to lower-intermediate shell girth weld, the axial flux distribution factor of 0.71 is applied for pre-EPU and 0.74 is applied for EPU conditions. The staff has independently verified the axial flux distribution factors using the data presented in the figure mentioned above and also verified the peak surface fluences for the lower shell and associated welds as calculated by the applicant. The staff finds the response acceptable because the applicant has used appropriate axial flux distribution factors for calculating the peak 54-EFPY surface fluence for the lower to lower-intermediate shell girth weld and all lower shell materials when determining the limiting materials. Therefore, Confirmatory Item 4.2.1(a) is closed.

CI.4.2.1.6: (Section 4.2.1.6 - Reactor Vessel Circumferential Weld Examination Relief)

The applicant was required to submit an update to LRA Section 4.2.6 to include the circumferential weld examination relief analysis for Quad Cities in accordance with 10 CFR 54.3(a) upon staff's approval of the May 16, 2003, relief request. This issue was identified as Confirmatory Item 4.2.1.6.

In response to Confirmatory Item 4.2.1.6, in a letter dated March 5, 2004, the applicant submitted a revision to the UFSAR Supplement for the reactor vessel circumferential weld examination relief TLAA. The revised supplement refers to the documents related to RPV circumferential weld relief request extension for the license renewal term. The staff reviewed this supplement and found that it provides an adequate summary description regarding the evaluation of this TLAA. Therefore, Confirmatory Item 4.2.1.6 is closed.

CI.4.2.2: (Section 4.2.2.7 - Reactor Vessel Axial Weld Failure Probability)

This axial weld failure probability analysis is required to be performed as a license renewal action item in accordance with the staff FSER of EPRI report TR-113596 (BWRVIP-74) and compliance with the license renewal rule (10 CFR Part 54) enclosed in an October 18, 2001, letter from Mr. C.I. Grimes to Mr. C. Terry. This action item, as stated in the staff's March 7,

2000, letter to Mr. C. Terry, requires the license renewal applicant to monitor axial beltline weld embrittlement. One acceptable method is to determine that the mean RT_{NDT} of the limiting axial beltline weld at the end of the extended period of operation is less than the values specified in Table 1 of this FSER. Therefore this evaluation applies to Dresden Units 2 and 3, as well as to Quad Cities Units 1 and 2. In addition, Dresden and Quad Cities have the same mean RT_{NDT}, because the initial RT_{NDT}, chemical composition, and 54-EFPY surface fluence are the same for the limiting beltline axial welds at Quad Cities and Dresden. Therefore, for Quad Cities and Dresden plants, the mean RT_{NDT} for the limiting beltline axial welds at 54-EFPYs is equal to 19 °C (67 °F). A comparison of the mean RT_{NDT} value of 33 °C (91 °F) for the Clinton axial weld from Table 4.2-1 of this SER with the Dresden and Quad Cities value of 19 °C (67 °F) shows that the NRC analysis of the Clinton axial welds bounds the Dresden and Quad Cities welds. The applicant should confirm that Quad Cities Units 1 and 2 have a mean value of 19 °C (67 °F) and address this TLAA of the axial welds for Quad Cities in the UFSAR Supplement. This was identified as Confirmatory Item 4.2.2.

In response to Confirmatory Item 4.2.2, in a letter dated March 25, 2004, the applicant compared the limiting axial weld 54-EFPY properties for Quad Cities 1 and 2 against the corresponding limiting values calculated by the NRC in the SER for BWRVIP-05 at 64 EFPY and the limiting Clinton values taken from Table 2.6-5 in the March 7, 2000, supplement to the SER. The applicant confirmed that the limiting axial welds at Quad Cities Units 1 and 2 have a mean 54 EFPY RT_{NDT} of 19ºC (67ºF), which is less than the value of 33ºC (91ºF) for Clinton. The comparison also shows that the conditional vessel failure probabilities for Quad Cities Units 1 and 2 are equal to 2.08×10^{-7} and 5.27×10^{-7}, respectively. These failure probabilities are less than the corresponding value for Clinton listed in Table 4.2-1 of this SER. The staff finds the applicant's evaluation for this TLAA acceptable because the conditional probability of failure of Quad Cities Unit 1 and 2 limiting axial welds at 54 EFPY is smaller than the corresponding values calculated by the NRC staff in the SER for BWRVIP-05 at 64 EFPY and the limiting Clinton values found in the March 7, 2000, supplement to the SER.

In a letter dated March 5, 2004, the applicant submitted a revision to the UFSAR supplement for the reactor vessel axial weld failure probability. The staff reviewed this supplement and found that it provides an adequate summary description regarding the evaluation of this TLAA. Therefore, Confirmatory Item 4.2.2 is closed.

CI.B.1.2-1: (Section 3.0.3.2 -Water Chemistry Program (B.1.2))

The staff noted that due to a potential difference in the concentration of sodium pentaborate in the system (the tank and suction piping are typically at a much higher concentration from the remainder of the system), that the proposed chemistry inspections may not provide information on the condition of the tank and pump suction piping. The staff requested in RAI B 1.2 -1 that the applicant provide supplemental information regarding how aging degradation of the SBLC tank and suction piping will be managed, since sampling chemistry downstream of the tank and receipt inspection of the chemicals used in the tank will not provide adequate assurance that degradation is not occurring in this section of the system. This issue was identified as Confirmatory Item B.1.2-1.

The applicant responded in letters dated December 22, 2003 and March 25, 2004, that it will perform an ultrasonic examination of portions of the SBLC tank. This is part of Commitment

#23 of Appendix A of this SER. The ultrasonic examinations will be used to identify potential loss of material and stress corrosion cracking. The applicant will perform one ultrasonic inspection in each quadrant, near the bottom of the tank. The applicant considers this location to be the most susceptible location for degradation. The UT examinations will include a portion of the tank shell and vertical seam weld and, if accessible, a portion of a circumferential weld in accordance with the applicant's NDE procedures. If necessary, the exam results will be addressed by the applicant's corrective action program.

The staff finds that the applicant will adequately manage aging in the SBLC system through the combined use of inspection of the pump casing, ultrasonic inspection of the SBLC tank, and control of addition chemicals according to the applicant's receipt inspection program. Therefore, Confirmatory Item B.1.2-1 is closed.

CI.B.1.17: (Section 3.3.2.3.2 - BWR Reactor Water Cleanup System (B.1.17))

Since the applicant stated that the entire RWCU system piping was replaced with IGSCC-resistant piping in accordance with NRC GL 89-10, the staff subsequently requested the applicant to provide the following information to be verified by the NRC Audit-Inspection Team:

(i) Clarify whether the entire RWCU system piping was replaced with IGSCC-resistant material or whether only portions of the RWCU system piping for each plant were replaced.

(ii) Confirm that, if the entire RWCU system piping was replaced, the piping system includes all the RWCU welds inboard and outboard of the second isolation valves. Confirm whether the selection of material of the replaced piping and weld metal meet the material compositions as described in GALL AMP XI.M25.

(iii) Verify that, if only portions of the RWCU system piping were replaced, the entire RWCU system piping meets the screening criteria, 1(a), (b), and (c) in GALL AMP XI.M25 program element 1, Scope of the Program, as well as the material specifications in GALL AMP XI.M25 program element 2, Preventive Actions.

This was identified as Confirmatory Item B.1.17.

During the audit, the team confirmed the following technical information relating to this AMP with the applicant at the request of the NRC's technical staff:

- All in-scope portions of the RWCU system piping outboard of the second isolation valves were replaced with ASTM SA312 or SA376 Gr. TP316L with a carbon content of less than 0.035%. RWCU piping inboard of the second isolation valve have not been replaced with IGSCC resistant piping.

- The RWCU piping that is inboard of the second isolation valve is Class 1 and 2 piping that is managed by the ISI Program (B.1.1), ASME Section XI, Subsections IWB, IWC, and IWD. All of the RWCU piping and welds on the in-scope portion outboard of the second isolation valves were replaced. The replacement piping and weld metal meets the material compositions as described in NUREG-1801, AMP XI.M25.

- All the RWCU piping and welds on the in-scope portion outboard of the second isolation valves were replaced with piping and weld material that meet the material compositions as described in NUREG-1801, AMP XI.M25. Screening criteria 1(a), (b), and (c) of NUREG-1801, AMP XI.M25 do not apply to this piping.

- In a letter to the NRC, dated August 20, 1993, the licensee committed to replace all RWCU IGSCC susceptible outboard supply and return line piping and the regenerative heat exchangers with IGSCC resistant materials at both Dresden and Quad Cities.

License Renewal Boundary Drawings LR-DRE-M-30 and LR-QDC-M-47-1 show the piping design table for the in-scope portion of the RWCU piping outboard of the second isolation valves to be table AQ. Dresden specification K-4080 and Quad Cities specification R-4411 provide the material specification for piping design table AQ. Design table AQ provides the information on the piping material: all in-scope portions of the RWCU system piping outboard of the second isolation valves were replaced with ASTM SA312 or SA376 Gr. TP316L with a carbon content of less than 0.035%.

Based on the above information, the applicant confirmed that the ten elements of the GALL program, "BWR Reactor Water Cleanup System," as specified in NUREG-1801, AMP XI.M25 (with the exception of the Water Chemistry Program as noted in the LRA) are applicable to Dresden and Quad Cities, and that the applicant's program B.1.17 is consistent with GALL AMP XI.M25, with the exception as noted in the LRA.

On the basis of its review of this AMP, GALL AMP XI.M25, AMR topical report M.05, and the ISI Program plan, the audit team determined that this AMP is consistent with GALL AMP XI.M25, with the exception as noted in the LRA. Therefore, Confirmatory Item B.1.17 is closed.

CI.B.1.23-1: (Section 3.0.3.10 - One Time Inspection (B.1.23))

See response for Open Item B.1.23-2.

CI.B.1.23-2.5: (Section 3.3.2.3.7 Periodic Inspection of Plant Heating System (B.2.8))

The applicant concluded that the Periodic Inspection of Plant Heating System program provides assurance that plant heating system components are routinely inspected for deterioration and leakage, and will adequately manage the components aging effects. The applicant stated that the program provides reasonable assurance that intended functions are maintained consistent with the current licensing basis during the period of extended operation. The staff compared the program against BTP RLSB-1. This issue was identified as Confirmatory Item B.1.23-2.5.

By letter dated March 25, 2004, the applicant described its program to manage loss of material, cracking, and leakage in selected plant heating system components for Dresden and Quad Cities exposed to an environment of saturated steam and condensate. The staff reviewed this program using the guidance in Branch Technical Position RLSB-1 in Appendix A of the SRP-LR and focused on how the program manages aging effects through the effective incorporation of the following 10 elements: program scope, preventive actions, parameters monitored or inspected, detection of aging effects, monitoring and trending, acceptance criteria, corrective actions, confirmation process, administrative controls, and operating experience. The applicant

indicated that the corrective actions, confirmation process, and administrative controls are part of the site-controlled Quality Assurance Program. The staff's evaluation of the Quality Assurance Program is provided separately in Section 3.0.4 of this SER. The remaining seven elements are discussed below. The staff also reviewed the UFSAR supplements for Dresden and Quad Cities to determine whether they provide an adequate description of the program.

[Program Scope] The program will manage age related degradation of plant heating system components such as filter/strainer housings, piping and fittings, pump casings, sight glasses, tanks, thermowells, traps, tubing, and valves. The staff finds that the scope is acceptable because it includes those components that rely on the program for aging management.

[Preventive or Mitigative Actions] The plant heating system periodic inspections do not provide any preventative actions. The inspections provide for condition monitoring to detect degradation prior to a loss of function. Preventative or mitigative actions are not needed for this condition monitoring program; therefore, the staff finds this acceptable.

[Parameters Monitored or Inspected] Visual inspections will be performed on a representative sample of brass or bronze valves, carbon steel piping and fittings, cast iron filter housings, pump casings and valves, and stainless steel thermowells and tubing used in the plant heating systems to determine if aging degradation is occurring. The components are inspected to ensure they are free of cracking, loss of material, and leakage. The inspection will consist of a visual inspection on the internal surface of components for the presence of general, crevice, galvanic, and pitting corrosion. The staff concludes that the applicant is inspecting the appropriate parameters to identify the aging effects; therefore, the staff finds this acceptable.

[Detection of Aging Effects] The plant heating inspections are performed a periodic intervals, and they detect aging prior to the equipment leaking so as to prevent spatial interaction with safety-related equipment. Inspections will be performed in accordance with ASME Code requirements and certified NDE examiners will conduct a VT-3 visual examination. The staff finds this acceptable because the inspections will identify the aging effects managed by this program.

[Monitoring and Trending] The condition of the components used in plant heating systems are monitored at intervals of approximately every 5 years, but not trended. Components are replaced if damage or unacceptable leakage is detected. Operating experience states that leaks were identified and corrected in a timely manner and did not result in a loss of function of any safety-related component. Staff finds that monitoring of these components periodically every 5 years is adequate to identify aging degradation; therefore staff finds this acceptable .

[Acceptance Criteria] The applicant stated that components are inspected for cracking, loss of material, and leakage. The components are replaced if a degraded condition is found. Inspections will be performed in accordance with ASME Code requirements and corrective actions state that evaluations are performed for inspection results that do not satisfy established criteria. The staff finds that the applicant's proposal to perform inspections in accordance with ASME Code requirements and use of engineering evaluations of degradation components will provide acceptance criteria against which the need for corrective actions will be evaluated; therefore, staff finds this acceptable.

[Operating Experience] The applicant stated that Dresden and Quad Cities have experienced leaks in the plant heating systems, but that these leaks were identified and corrected in a timely manner and did not result in a loss of function of any safety-related system, structures, or components. The staff notes that the plant heating system is in scope of license renewal due to the potential for spatial interactions. The staff finds that the operating experience of timely correction of system leaks plus the additional periodic visual inspections supports the applicant's conclusion that the program will be effective in managing aging of the components in the scope of this program; therefore, the staff finds this acceptable. Therefore Confirmatory Item B.1.23-2.5 is closed.

CI.B.1.25-1: (Section 3.0.3.12 - Buried Piping and Tanks Inspection)

The staff had additional concerns regarding concrete asbestos piping and buried carbon steel piping and requested clarifying information. Specifically, with regard to concrete asbestos piping, the applicant's operating history indicated that failures of the piping have occurred. On the basis of this experience, the staff requested justification for why a one-time inspection was not warranted, as well as confirmation that the soil environment for the piping was not aggressive. With regard to buried carbon steel piping, the applicant indicated that much of the piping may not be coated. Given that some of the piping may not be coated, the staff questioned why this was not identified as an exception to the GALL program. This was identified as Confirmatory Item B.1.25-1.

The applicant responded to the staff's request for supplemental information in letters dated December 12, 2003 and March 25, 2004. The applicant indicated that the buried concrete piping likely failed as a result of ground shifting or heavy loads transported in the vicinity of the piping. This piping is located in a soil and ground water environment which is not aggressive to concrete based on pH values between 7 to 9, chlorides 5 to 30 ppm, and sulfates 10 to 30 ppm. These values are within the NUREG 1801 criteria (chlorides less than 500 ppm, sulfates less than 1500 ppm and pH greater than 5.5). The applicant indicated that buried carbon steel and ductile iron piping in the Fire Suppression System are externally coated with coal tar wrapping; however, there was some question regarding use of coating on other carbon steel buried piping. The applicant provided supplemental information after a detailed review of plant documents that indicates that all carbon steel buried piping at Dresden and Quad Cities was externally coated. The applicant further indicated that the installation specification required an inspection of the coating integrity prior to burial. The applicant provided operating experience from a recent plant modification that required excavation of some Fire Suppression System piping at Quad Cities. A section of 10 inch schedule 40 carbon steel piping was recently excavated. The applicant indicated that the piping was coated with coal tar wrapping and had been buried in the early 1970's. The nominal wall thickness of this piping is 0.365 inches. The measured minimum and maximum wall thicknesses were 0.320 inches and 0.400 inches respectively. The applicant concluded that there was little effect of aging on this piping after burial for approximately 30 years. The applicant also surveyed the craft personnel who performed the work to assess the condition of the external pipe coating. The applicant provided qualitative information that the coating was "generally in good condition" based on the craft personnel. The staff finds the applicant's response acceptable because the applicant provided information regarding the cause of the concrete piping degradation, provided information consistent with NUREG 1801 that indicates the environment is not aggressive to concrete, that there is reasonable assurance that the buried piping is coated and provided

additional operating experience that indicates there is limited aging degradation of buried piping. Therefore, Confirmatory Item B.1.25-1 is closed.

<u>CI.B.2.5-1</u>: (Section 3.0.3.16 Lube Oil Monitoring Activities)

In its October 3, 2003, response to RAI B.1.23-2(a), the applicant committed to include the following additional components in the scope of this program: components in the reactor core isolation cooling (RCIC) system, additional components in the high pressure coolant injection (HPCI) system, additional components in the emergency diesel generator and auxiliaries system, and additional components in the station blackout diesel system. In addition, the applicant committed to add components exposed to EHC oil (main turbine and auxiliary systems) and generator hydrogen seal oil (turbine oil system - Quad Cities only) to the scope of this program. The staff found that adding the above components to the scope of this program is appropriate, since maintaining oil quality is important for preventing aging effects. However, the applicant did not provided updates to the program elements to address the increased scope of the program. The applicant was requested to provide the appropriate revisions to the 10 elements and the UFSAR summary description of this program. This issue was identified as Confirmatory Item B.2.5-1.

In its draft supplemental response dated December 18, 2003, the applicant further committed to add components exposed to EHC oil (main turbine and auxiliary systems) and generator hydrogen seal oil (turbine oil system) to the scope of this program. The applicant added these components to the scope of the program by letter dated January 26, 2004. The staff found that adding the above components to the scope of this program is appropriate, since maintaining oil quality is important for preventing aging effects in these components.

In a letter dated June 22, 2004, the applicant committed to include the following additional component in the scope of this program: components in the reactor recirculation motor generation oil system. The staff found that adding the above component to the scope of this program is appropriate, since maintaining oil quality is important for preventing aging effects in these components.

[Program Scope] The applicant stated that this AMP is applicable to heat exchanger and other components exposed to a lubricating oil environment in the HPCI, emergency diesel generator and auxilaries, (SBO) diesel and auxlilaries, reactor core isolation cooling (RCIC), generator hydrogen seal oil (HSO), main turbine and auxiliaries (electro-hydraulic control (EHC) oil subsystem), and reactor recirculation motor generator systems. The staff finds that the scope is acceptable because it includes those components that rely on the program for aging management.

[Preventive or Mitigative Actions] The applicant's program monitors and controls the oil properties and impurity levels. When the parameters exceed predefined limits, actions are taken to restore the conditions. The staff finds that maintaining the oil parameters mitigates loss of material and cracking in lubricating oil systems; therefore, the staff finds this acceptable.

[Parameters Monitored or Inspected] The applicant stated that the parameters monitored by the program include viscosity, total acid number, total base number, rotary bomb oxidation test, water demulsability, particle count, fuel and combustion byproducts, sediment, water, anti-

foaming characteristics, whole particle counting, air release and emission spectrum. The applicant also stated that the parameters monitored by the program depends on oil type and type of service. The staff notes that loss of material due to general, crevice, and pitting corrosion and cracking are applicable aging effects for lubricating oil cooler components in a lubricating oil environment at locations containing water or contaminants such as chloride ions. By RAI B.2.5(a), the staff asked the applicant to clarify whether water, moisture, and chloride ions are monitored for all type of oil and service. If not, the staff requested the applicant to provide justification for not including these parameters in monitoring. In its response dated October 3, 2003, the applicant stated that water/moisture is monitored as part of the Lubricating Oil Monitoring Activities program. No monitoring for chloride ions is provided in this program. The applicant explained that EPRI 1003056, Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools, Revision 3, Appendices C and G were reviewed in the development of the Lubricating Oil Monitoring Activities program. These appendices address oil environments in general and lubricating oil environments for heat exchangers, respectively. Appendix C identifies damaging effects associated with chlorides in fuel oil environments, but no similar effects are identified for lubricating oil environments. Appendix G does not identify any applicable aging effects associated with chlorides for lubricating oil environments in heat exchanger components. The applicant also stated that there is no site operating experience of failure or degradation in oil environments attributed to the presence of chlorides. Therefore, the applicant concluded that monitoring for chloride ions is not required for the Lubricating Oil Monitoring Activities program. Based on the applicant's operating experience, the staff finds that the applicant's response satisfactorily addresses the staff's concerns and RAI B.2.5(a) is considered closed. The staff concludes that the applicant is monitoring the appropriate oil parameters; therefore, the staff finds this acceptable.

[Detection of Aging Effects] The applicant stated that samples of lubricating oil are taken monthly for EDGS, EHC oil, reactor recirculation motor generator oil, and HSO systems, quarterly for HPCI, SBO diesel generators, semi-annually for the RCIC pump, and every 24 months for the RCIC turbine. Sampling frequency is increased if plant and equipment operating conditions indicate a need to do so. The applicant stated that the sampling would reveal aging degradation because increased impurities and degradation of oil properties indicate degradation of material in lubricating oil systems. The staff finds this acceptable because sampling and analyses are performed periodically, and the analysis is capable of detecting aging degradation.

The staff also notes that the aging effects of the heat exchangers are also managed by the "Closed-Cycle Cooling Water System" and/or "Heat Exchanger Test and Inspection Activities," AMPs B.1.14 and B.2.6, respectively. For other components, the applicant uses the One-Time Inspection Program (B.1.23) to verify the effectiveness of the Lube Oil Monitoring Activities AMP. The inspections and performance testing under these programs provides additional assurance that loss of material and cracking will be detected before the loss of intended function; therefore, the staff finds this acceptable.

[Monitoring and Trending] The Lube Oil Monitoring program monitors the relevant parameters via samples taken monthly for EDGs, quarterly for HPCI, SBO diesel generators, EHC oil, and HSO systems, semi-annually for the RCIC pump, and every 24 months for the RCIC turbine. The oil analysis results are trended and evaluated using computer software and a database. The applicant stated that the lubricating oil analysis results are trended and evaluated using computer software and a database. The staff finds that monitoring through sample analysis is

appropriate and that the frequency is consistent with industry experience; therefore, the staff finds the monitoring and trending to be acceptable.

[Acceptance Criteria] The applicant stated that normal, alert, and fault levels have been established for the various chemical and physical properties, wear metals, additives, and contaminant levels based on information from oil manufacturers, equipment manufacturers, and industry guidelines, for the specific oil type and application. The applicant also stated that the program maintains contaminant and parameter limits within the application-specific limits. By RAI B.2.5(b), the staff asked the applicant to explain the acceptance criteria of water, moisture, and contaminants. In its response dated October 3, 2003, the applicant provided the acceptable limits for water/moisture and contaminants at normal, alert, and fault levels for emergency diesel generator and SBO diesel components with MOBILGARD 450 NC oil and for HPCI turbine components with MOBIL VAPROTEC LIGHT oil. The applicant stated the acceptable limits are based on EPRI 1003056, "Non-Class 1 Mechanical Implementation Guideline and Mechanical Tools," Revision 3, and that any failures to meet these criteria result in a condition evaluation, an identification of root causes, and correction of the adverse condition. The staff finds that the acceptance criteria are consistent with industry guidelines and that the applicant's activities in case of failure to meet these acceptance criteria are reasonable; therefore, the staff finds the acceptance criteria acceptable.

[Operating Experience] The applicant stated that oil sampling and analysis have detected particulate or water contamination (or both) in lubricating oil systems. The operating experience has produced procedure and program changes, which have improved the effectiveness of lubricating oil testing and inspection activities. By RAI B.2.5(c), the staff asked the applicant to describe the corrective actions made and the operating experience since these corrective actions were implemented. In its response dated October 3, 2003, the applicant provided four examples of corrective actions made as a result of operating experience involving lube oil sampling and analysis. In one of the examples, the applicant stated that the 10/28/99 oil analysis of the Unit 1A (1B) SBO diesel engine crankcases indicated high percentage volume for sediment of 0.3 % (upper limit of 0.05% volume). All physical parameters other than sediment were found to be suitable for use. A recommendation was made to continue sampling/trending oil sample results on a quarterly frequency. The sampling procedure was revised to include requirements to perform sampling on a quarterly basis, and trend results. In another example, the applicant stated that a number of Quad Cities oil analysis results for RHRSW pump bearings showed high metal levels. It was determined that the high/increased wear level concentrations could have been indications of pump shaft, housing, rolling element bearing or bearing cage clearance wear. It was determined that the pump bearing oil analysis required large amounts of oil to be collected because smaller sample amounts had a tendency to show high/erratic wear levels. The sampling procedure was revised to include requirements to draw a relatively large sample. The applicant stated that no operating experience involving recurrence of heat exchanger degradations since implementation of the associated corrective actions. The staff finds that the applicant's response satisfactorily addresses the staff's concerns and RAI B.2.5(c) is considered closed. The staff finds that the applicant's operating experience supports the conclusion that the program will be effective in preventing aging of the components in the scope of this program; therefore, the staff finds this acceptable. Therefore Confirmatory Item B.2.5-1 is closed.

1.7 Summary of Proposed License Conditions

As a result of the staff's review of the DNPS/QCNPS application for license renewal, including the additional information and clarifications submitted subsequently, the staff identified three proposed license conditions. The first license condition requires the applicant to include the UFSAR Supplement in the next UFSAR update required by 10 CFR 50.71(e) following issuance of the renewed license. The second license condition requires that the future activities identified in the UFSAR Supplement and Appendix A of this SER to be completed prior to the period of extended operation. The third license condition requires the implementation of the most recent staff-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) as the method to demonstrate compliance with the requirements of 10 CFR Part 50, Appendix H. Any changes to the BWRVIP ISP capsule withdrawal schedule must be submitted for NRC staff review and approval. Any changes to the BWRVIP ISP capsule withdrawal schedule which affects the time of withdrawal of any surveillance capsules must be incorporated into the licensing basis. If any surveillance capsules are removed without the intent to test them, these capsules must be stored in manner which maintains them in a condition which would support re-insertion into the RPV, if necessary.