March 28, 2008

Mr. Charles G. Pardee
Chief Nuclear Officer
and Senior Vice President
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3; AND QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 - REQUEST TO PROVIDE REVISED SAFETY EVALUATION REPORT FOR AMENDMENT NO. 220 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-19 AND AMENDMENT NO. 211 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-25 FOR DRESDEN, UNITS 2 AND 3, AND AMENDMENT NO. 231 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-29 AND AMENDMENT NO. 227 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-30 FOR THE QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 (TAC NOS. MD4866, MD4867, MD4868 AND MD4869)

Dear Mr. Pardee

By letter dated March 9, 2007, as supplemented by letters dated September 10, 2007, and February 18, 2008, Exelon Generation Company, LLC (Exelon, the licensee) requested that the Nuclear Regulatory Commission (NRC) staff revise the safety evaluation (SE) Section 3.3.3.2, "Thermal-Hydraulic Stability," for the April 4, 2006, license amendment to the Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2, operating licenses. In its March 9, 2007, letter, the licensee identified that the NRC SE discussed the backup stability protection (BSP) methodology in a manner that was not consistent with the information provided in Exelon's submittals requesting the amendment. Exelon requested that the NRC revise the SE to define the immediate-scram region as the area in the power-flow map where the decay ratio (DR) is > 0.8, and the exclusion region as the area in the power-flow map where the DR is > 0.8 - cu, where the term "cu" accounts for the cycle uncertainty. Section 3.3.3.2 currently defines the immediate-scram region "by a RAMONA-3 decay ratio > 0.8," the exclusion region (i.e., the controlled entry region), however, is described "as decay ratio > 0.6."

The NRC staff reviewed the information in the NRC staff's April 4, 2006, SE regarding the BSP methodology, and concluded that the discussion of the immediate scram region is consistent with the licensee's March 9, 2007, request and does not need to be revised. The NRC staff acknowledges a difference in the description of the controlled entry region and provides an evaluation of the licensee's proposed revision below.

Exelon states that the BSP is based on two lines in the power-flow map. One line defines the immediate-scram region; this line is defined as decay ratio DR > 0.8. The other line is a buffer region, which is defined as DR > 0.8 – cu, where the cycle uncertainty is a variable that is calculated using cycle-specific core information. The immediate-scram region (DR > 0.8) conforms, with approved methodologies, and reflects the ± 0.2 DR error expected by calculational uncertainties and some degree of inaccuracy in defining the operating conditions.

The value of cu is calculated on a cycle specific basis, to represent possible variations between the actual power shapes, that the reactor will experience during the cycle and the power shapes predicted, for use in the DR analysis. Exelon states that the radial peaking factor (RPF) results in the largest DR sensitivity. In other words, if the stability calculation uses a radial and axial power distribution predicted during the reload process, a DR error is to be expected when operating with the real power distributions during the cycle.

Exelon has evaluated the degree of variability of the RPF during a representative cycle in QCNPS, Unit 2, Cycle 19. The RPF variability observed during that cycle was used to determine the value of cu by performing stability calculations with the predicted and actual RPF. Exelon stated that this procedure results in a controlled entry region with a buffer of approximately 5 percent power (i.e., the controlled entry region has approximately 5 percent lower power than the scram region). Exelon considers this approximate 5 percent buffer sufficient to correct inadvertent entries into the immediate-scram region and it gives operators sufficient time to take corrective actions.

Exelon states that cu is used generically based on the results of QCNPS, Unit 2, Cycle 19. However, the actual plant- and cycle-specific value of cu is confirmed every reload.

The NRC staff reviewed the proposed SE revision discussed in the licensee's March 9, 2007, September 10, 2007, and February 18, 2008, letters, and found that (1) the immediate scram region defined by DR > 0.8 is technically acceptable because it reflects the accepted uncertainty of the approved Westinghouse's stability methodology, (2) the use of a generic cu is technically acceptable because it is confirmed by calculation on a plant- and cycle-specific basis every reload, and (3) the procedure to calculate the cu is technically acceptable because:

1. The Specified Acceptable Fuel Design Limit protection required by General Design Criteria 10 is accomplished by the scram line, which is calculated at DR > 0.8 for the expected cycle operating conditions.

2. The cu factor is used for a defense-in-depth buffer region, which provides two functions: (a) a buffer that allows the operator to reverse the last control action before a scram is required, and, (b) additional protection for unexpected operating conditions (e.g. unexpected power shapes caused by fuel leakers). The use of a generic cu value based on a detailed evaluation of QCNPS, Unit 2, Cycle 19, which is then verified on plant- and cycle-specific basis, is technically acceptable for a defense-in-depth function.

Based on the results of the NRC staff review, the NRC staff concludes that the buffer region should be defined by the largest region that satisfies both: (1) DR > (0.8 – cu), and (2) immediate scram region minus 5 percent power; since the buffer region is approximately 5 percent power, which allows the operator sufficient margin to maneuver, before a scram is required, should the buffer region be entered inadvertently.

Therefore, NRC staff is reissuing the SE associated with the April 4, 2006, amendments to the DNPS and the QCNPS, in its entirety to clarify the NRC staff's position regarding the controlled entry region, as requested by the licensee in its letter dated March 9, 2007. A revised version of the SE is enclosed. Change bars identify the revised portions of the SE.

If you have any questions regarding this letter, please contact me at 301-415-1055.

Sincerely,

/RA/

Christopher Gratton, Senior Project Manager
Plant Licensing Branch III-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-237, 50-249,
50-254, and 50-265

Enclosure:
As stated

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO

AMENDMENT NO. 220 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-19,

AMENDMENT NO. 211 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-25,

AMENDMENT NO. 231 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-29

AND AMENDMENT NO. 227 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-30

EXELON GENERATION COMPANY, LLC

AND

MIDAMERICAN ENERGY COMPANY

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3, AND

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-237, 50-249, 50-254 AND 50-265

1.0 INTRODUCTION

By letter to the Nuclear Regulatory Commission (NRC, Commission) dated June 15, 2005, (Reference 1) as supplemented by letters dated January 26 (Reference 2), January 31 (Reference 3), February 22 (Reference 4), March 3 (Reference 8), and March 23, 2006 (Reference 9), Exelon Generation Company, LLC, et al. (Exelon, the licensee) requested changes to the technical specifications (TSs) and surveillance requirements (SRs) for Dresden Nuclear Power Station, Units 2 and 3 (DNPS), and Quad Cities Nuclear Power Station, Units 1 and 2 (QCNPS). The supplements dated January 26, January 31, February 22, 2006, March 3, 2006, and March 23, 2006, contained clarifying information and did not change the NRC staff's initial proposed finding of no significant hazards consideration.

The proposed changes would revise the following TS sections:

1. An editorial change to TS Section 3.1.4, "Control Rod Scram Times," to remove specific reference to "GE analyzed cores."

2. A revision of the TS Section 4.2.1, "Fuel Assemblies," fuel assembly description to encompass the Westinghouse SVEA-96 Optima2 fuel design.

3. Westinghouse analytical methods added to Core Operating Limits Report (COLR) references in TS 5.6.5, "Core Operating Limits Report (COLR)."

4. Added TS SR 3.1.7.10, to TS SR 3.1.7, "SLC [Standby Liquid Control] System," to verify sodium pentaborate enrichment.

Specifically, the proposed changes would permit the licensee to transition from General Electric (GE) 14 fuel to Westinghouse SVEA-96 Optima2 fuel and use the supporting Westinghouse analytical methods.

Exelon originally requested this license amendment by letter dated January 20, 2005 (Reference 5). As part of the acceptance review, the NRC staff determined that the request did not provide technical information in sufficient detail to enable the NRC staff to complete its safety review. Therefore, by letter dated March 11, 2005 (Reference 10), the licensee withdrew the original license amendment requested by Reference 5. By letter dated March 17, 2005 (Reference 6), the NRC staff provided the results of its acceptance review, including specific areas in the original license amendment that required additional information. Being the first implementation of the Westinghouse SVEA-96 Optima2 fuel assembly design in the United States and the first application of Westinghouse analytical methods to extended power uprate (EPU) operating conditions, the NRC staff requested further information and conducted several audits of the supporting Westinghouse engineering calculations.

The proposed changes are required to support the transition to Westinghouse SVEA-96 Optima2 fuel and reload analysis methods beginning with QCNPS, Unit 2 Cycle 19 (spring 2006).

2.0 REGULATORY EVALUATION

The Commission's requirements for fuel system design are set forth in the Title 10 of the *Code of Federal Regulations* (10 CFR), including Section 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors," Appendix K to Part 50, "ECCS Evaluation Models," and certain General Design Criteria (GDC) in Appendix A to Part 50, "General Design Criteria for Nuclear Power Plants." The applicable GDC are as follows: GDC 10, "Reactor Design;" GDC 27, "Combined Reactivity Control System Capability;" and GDC 35, "Emergency Core Cooling." The regulatory requirements limit the fuel cladding temperature, specify maximum cladding oxidation and hydrogen generation, and require reactivity control and a coolable geometry such that long term cooling could be maintained. Regulatory guidance for the review of fuel system design and adherence to applicable GDC is provided in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 4.2, "Fuel System Design" (Reference 7). In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

a. the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),
b. fuel system damage is never so severe as to prevent control rod insertion when it is required,
c. number of fuel rod failures is not underestimated for postulated accidents, and
d. coolability is always maintained.

In addition to licensed reload methodologies, an approved mechanical design methodology is utilized to demonstrate compliance to SRP Section 4.2 fuel design criteria. The NRC has previously reviewed and approved the Westinghouse SVEA-96 Optima2 fuel assembly design (WCAP-15942-P-A[1], "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, Supplement 1 to CENPD-287-P-A," and WCAP-15836-P, "Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1"). 10 CFR 50.36 specifies the content of the TSs. TS 5.6.5 lists the NRC-approved methods that are used to determine core operating limits. This list is extended to include additional methodology documents.

3.0 TECHNICAL EVALUATION

The license amendment request includes changes to the DNPS and QCNPS TSs to support the transition to Westinghouse SVEA-96 Optima2 fuel and reload analysis methods beginning with QCNPS, Unit 2 Cycle 19 (spring 2006). These changes are discussed in Attachment 1 of Reference 1 and addressed below.

3.1 Change to Technical Specification Section 3.1.4

The first proposed change involves an editorial change to TS Section 3.1.4, "Control Rod Scram Times" to remove specific reference to "GE analyzed cores." The change does not alter the control rod scram time requirements nor related surveillance. Based upon its editorial nature, the NRC staff finds the proposed change to TS Section 3.1.4 acceptable.

3.2 Change to Technical Specification Section 4.2.1

The second proposed change involves a revision of the TS Section 4.2.1 fuel assembly description to encompass the Westinghouse SVEA-96 Optima2 fuel design. Instead of water rods or a single water box, the Westinghouse SVEA-96 Optima2 fuel assembly design contains a watercross structure. This unique design feature necessitates a revision to the current TS Section 4.2.1 assembly description. The NRC staff finds the revised assembly description acceptable.

3.3 Change to Technical Specification Section 5.6.5

The third proposed change involves updating the list of COLR approved methodologies in TS Section 5.6.5 to include the Westinghouse analytical methods. In response to the NRC staff's acceptance review comments, the licensee provided a description of the Westinghouse analytical methods along with a list of topical reports being added to TS Section 5.6.5 (Reference 5). All of the Westinghouse methods have been previously reviewed and approved by the NRC staff. In Attachment 7 of Reference 1, the licensee specifically addressed each of the conditions and limitations delineated in the NRC safety evaluation for each topical report.

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[1] At the time this safety evaluation was prepared, the NRC-approved ("A") version of WCAP-15836-P, "Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1," and WCAP-15942-P were not yet issued. The respective safety evaluations, request for additional information (RAI) responses, and Proprietary ("P") versions for these two topical reports were used to complete the NRC staff's review of the Exelon license amendment request.

Further, the licensee stated that future core reloads utilizing Westinghouse SVEA-96 Optima2 fuel along with the approved methods will continue to satisfy these limitations and conditions (Attachment 1 of Reference 1).

3.3.1 Applicability of Westinghouse Analytical Methods

In response to the NRC staff's acceptance review comments (Reference 5), the licensee provided information to demonstrate that the DNPS and QCNPS operating parameters are within the applicability ranges of the NRC staff's original review of the Westinghouse analytical methods and computer models. Table 3 of Attachment 6 to Reference 1 provides the requested information.

In Attachment 6 of Reference 1, Exelon provided details justifying the first application of the Westinghouse analytical methods to Westinghouse SVEA-96 Optima2 fuel at DNPS and QCNPS. The application of these methods was performed at the EPU conditions. Specifically, the analyses were conducted by Westinghouse to purposely demonstrate the following:

5. That the steady-state and transient neutronic and thermal hydraulic-analytical methods and code systems used to perform the safety analyses supporting the EPU conditions are being applied within the NRC-approved applicability ranges;
6. That for the EPU conditions, the calculation and measurement uncertainties applied to the thermal limits analyses are valid for the predicted neutronic and thermal-hydraulic core and fuel conditions; and
7. That the assessment database and the assessed uncertainty of models used to simulate the plant's response during steady state, transient or accident conditions remain valid and applicable for the EPU conditions (RAI responses 9 through 13 of Reference 2).

Exelon also examined code systems to demonstrate that the analytical methods are applicable to DNPS and QCNPS EPU conditions. These analytical methods and code systems have been previously used for applications with average assembly powers considerably higher than the EPU conditions intended for use at DNPS and QCNPS as noted in Table 1 of Attachment 6 to Reference 1. Table 1 of Attachment 6 to Reference 1 provided a comparison of the power levels among a sample of boiling water reactors (BWRs) for which Westinghouse has previously been or is currently the fuel vendor. The data provided in Table 1 clearly demonstrates that the core thermal power, assembly average power and rod average power conditions at DNPS and QCNPS at the EPU conditions are lower than those at other plants. Table 2 of Attachment 6 to Reference 1 provided the safety analyses conducted for both steady-state and transient conditions. The NRC staff finds the Westinghouse analytical methods acceptable based on DNPS and QCNPS conditions being within the BWR fleet's operating conditions for which the topical reports were approved.

3.3.2 Physics Biases and Uncertainties

In response to the NRC staff's acceptance review comments (Reference 5), the licensee provided information to demonstrate that the current core physics model uncertainties remain applicable to the DNPS and QCNPS operating conditions (Attachment 6 of Reference 1). The two principal computer programs for BWR steady-state nuclear design and analysis used by Westinghouse are the PHOENIX and POLCA codes. The PHOENIX code is a two-dimensional multi-group transport theory code used to calculate the lattice physics constants of BWR fuel assemblies. The POLCA code is a two-group nodal code used for the three-dimensional simulation of the nuclear and thermal-hydraulic conditions in BWR cores. In addition, several auxiliary codes are used to facilitate calculations and to transfer data between the aforementioned codes.

Topical Report CENPD-390-P-A, "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors," describes an improved cross section library used in the PHOENIX code and changes to the POLCA code relative to the PHOENIX/POLCA topical report previously reviewed and approved by the NRC. It also provides an assessment against operational data and measurements to demonstrate that the codes are capable of predicting power distributions, thermal limits, and critical conditions necessary for BWR nuclear design and analyses.

The report provides a detailed description of the verification that has been performed to qualify the computer codes and analysis methods that are used for the nuclear design and analysis of BWRs. This version of the code package is described and qualified in CENPD-390-P-A, Reference 8 of the June 15, 2005, submittal. Relative pin power and assembly power uncertainties as well as the overall uncertainty analyses were addressed in support of the determination of linear heat generation rate and critical power ratio (CPR) limits. The result of the analysis demonstrated to the NRC staff the reliability of the methods to predict any up-coming condition in the core. Indeed, Westinghouse engages in an on-going qualification program, which includes experimental and numerical comparisons, to confirm the continued applicability of the pin power and assembly power uncertainties in particular, and to confirm on a continuing basis the acceptability of power distribution predictions with the PHOENIX/POLCA code system in general. This is particularly important as assembly designs are improved. The NRC staff finds it acceptable for licensees to use this code combination.

Evaluations of relative rod power uncertainties were conducted by Westinghouse and reviewed by the NRC staff. These evaluations were based on comparisons with critical experiments, higher order code predictions (e.g., HELIOS), and pool-side gamma scan measurements. Evaluations were also conducted of relative assembly power uncertainties based on comparisons with pool-side gamma scan measurements and plant traversing incore probe (TIP) measurements. The plant TIP measurements are an excellent on-going plant-specific data source with which to confirm the continued adequacy of node and assembly average power uncertainties. Westinghouse and other vendor's experience base indicates that the most useful methods of establishing uncertainties are pool-side gamma scan measurements for both relative rod power and relative assembly power uncertainties and critical experiments for relative rod power uncertainties.

During the NRC staff's review of the Westinghouse neutronic and thermal-hydraulic methodology application to the QCNPS and DNPS, the NRC staff requested Westinghouse to provide additional clarification information and supportive calculations pertaining to the determination of physics biases, uncertainties and thermal-hydraulic limits and predictions. The requested information and the respective detailed responses are provided in Reference 2.

Specifically, the NRC staff was concerned with how computer codes' biases and uncertainties, and core monitoring codes' uncertainties were calculated. In particular, the NRC staff requested additional information in support of the basis and technical justification provided in the submittal for the uncertainties and thermal-hydraulic predictions used in their calculations (RAI responses 4, 8, and 19 of Reference 2).

In addition, the NRC staff requested additional information associated with thermal- hydraulic issues, such as the minimum critical power ratio (RAI responses 4, 8, and 19 of Reference 2), high void fractions (RAI responses 14 through 18 of Reference 2) that are typically associated with cores operating in ranges of high power and low flow, and gamma scanning-versus-computer code predictions comparisons. Westinghouse demonstrated to the NRC staff that QCNPS and DNPS can operate at all the state points, within the allowable operating conditions (as specified in the TSs) including minimum core flow condition, without violating their safety limit minimum critical power ratio (SLMCPR) during an anticipated transient.

On November 7 - 10 and again on December 14, 2005, the NRC staff conducted an audit of the Westinghouse engineering calculations supporting the Exelon license amendment request. As part of this audit, the NRC staff reviewed several calculation workbooks, computer code outputs, and were briefed collectively and in one-on-one sessions on all of the above stated subjects. This effort resulted in providing the NRC staff with in-depth insight into the basis and justifications for the calculation results provided in the original and subsequent submittals. Westinghouse provided RAI responses (References 2 and 3) that provided the NRC staff with qualitatively and quantitatively analytical foundations and derivations of the respective biases and uncertainties discussed in the RAIs. Based upon the information provided in the license amendment request, RAI responses, as well as the NRC staff's audit of the supporting Westinghouse engineering calculations, the NRC staff finds the Westinghouse neutronic and thermal-hydraulic analytical methods, and the associated uncertainties and biases, are acceptable for application to DNPS and QCNPS.

3.3.3 Licensing Basis Events

Attachment 7 of Reference 1 provides a description of the licensing basis events including the acceptance criteria and applicable Westinghouse topical reports addressing the approved methodology for each event. With the exception of the Jet Pump Malfunction (Updated Final Safety Analysis Report (UFSAR) 15.3.5), the event classifications on Table 1 of Attachment 7 to Reference 1 are consistent with the current licensing basis (as documented in the DNPS and QCNPS UFSARs). In discussions with the licensee, it was determined that the Jet Pump Malfunction event had been mistakenly classified as a limiting fault, whereas the current licensing basis treats this event as an AOO (i.e., precluding fuel failure). The Jet Pump Malfunction event will remain classified and treated as an AOO.

The design basis accidents (DBAs) listed in Table 1 of Attachment 7 to Reference 1 are classified as limiting faults with the acceptance criteria denoted in Table 2 of Attachment 7 to Reference 1. The acceptance criteria on radiological consequences for the DBAs are listed as "Offsite Dose $\leq$ 10 CFR 100 Limit." Examination of the current licensing basis reveals that not all of the DBAs have been allotted the full 10 CFR 100 guidelines (i.e. within 10 CFR 100). For example, QCNPS UFSAR Table 15.7-4a identified the offsite regulatory dose limit for the fuel-handling accident at 75 REM thyroid and 6.25 REM whole body (i.e. well within 10 CFR 100). Table 2 of Attachment 7 to Reference 1 identifies that the acceptance criteria "may be supplemented by additional plant-specific limitations." The license amendment request did not include any revised dose calculations supporting changes to the DBA radiological consequences. As such, the currently docketed regulatory dose limits for each DBA (e.g., within, well within, or small fraction of 10 CFR 100), as well as the currently reported doses, remain the licensing basis for DNPS and QCNPS.

3.3.3.1 BWR Loss of Coolant Accident (LOCA) Methodology

The transition to the Westinghouse SVEA-96 Optima2 fuel was demonstrated through application of the Westinghouse emergency core cooling system (ECCS) Evaluation Model identified in Reference 4 as supplemented by Reference 9, which consisted of the GOBLIN thermal-hydraulic code and CHACHA-3D fuel rod mechanical and heat-up methodologies.

Westinghouse did not make any changes to the NRC staff-approved methodology (GOBLIN and CHACHA-3D codes) that was previously approved for the DNPS and QCNPS nuclear steam supply systems containing the transition core and Westinghouse SVEA-96 Optima2 fuel design.

In Tables 4 through 8 of Attachment 7 to Reference 1, the licensee provided a resolution for each of the conditions and limitations as delineated in the NRC safety evaluation for each applicable topical report. The NRC staff verified that all of the conditions and limitations of the NRC approved BWR LOCA methods were satisfied for this application with the following clarifications:

Table 4 of Reference 1, Attachment 7

> Condition 2: The NRC staff reviewed the reported application results and sensitivity studies applicable to DNPS and QCNPS. Analysis of the most limiting break, the double-ended recirculation line break with the single failure of the low-pressure coolant injection (LPCI) valve, was identified with a PCT of 2150 °F.

Table 5 of Reference 1, Attachment 7

> Condition 1: The STAV7.2 fuel models have been implemented in CHACHA-3D and employed in the application analyses.

Table 8 of Reference 1, Attachment 7

> Condition 3: If the mixed core is found to be more limiting then the legacy fuel vendor will be notified to evaluate the impact on the maximum average planar linear heat generation rate (MAPLHGR) limits. The Westinghouse full core model was found to be more limiting in the Reference 1 analysis results.

Because of the similarity between the DNPS and QCNPS power plants, a single bounding analysis was performed that is considered conservative with respect to all four units. The break spectrum analysis consisted of an analysis of double-ended and slot breaks in the recirculation line covering break areas in the range of double-ended guillotine down to and including the 0.1 ft^2. Analyses of recirculation line, steam line, low pressure core spray (LPCS) lines, and feedwater line breaks were performed with the following worst single failures:

- LPCI injection line valve failure
- Emergency diesel generator failure
- High pressure coolant injection failure
- Loop select logic failure
- Automatic depressurization system failure

These analyses showed that the double-ended recirculation line break with a failure of the LPCI injection valve produced the highest PCT with a temperature of 2150 °F, which is below the 10 CFR 50.46 limit of 2200 °F. Additional sensitivity studies were performed investigating the effect of power shape on PCT. The limiting shape consisted of a slightly top skewed power shape. Axial shapes are always evaluated as part of the application analyses. Also, a time in life study identified, in determining the MAPLHGR limits, the maximum PCT of 2150 °F occurred at a burnup of 12500 MWD/MTU.

Review of the results of the limiting break showed that the evaluation model utilizes conservative assumptions. Once the peak power position uncovers following the LOCA, no convective heat transfer is assumed to cool the hot rod. This produces an early adiabatic heat-up period from 20 to 50 seconds for the limiting large break LOCA. This assumption is applied to all other LOCA evaluations as well. A convective heat transfer coefficient of 1.0 Btu/hr- ft^2 - °F is then assumed only after the spray flow reaches rated conditions. For the limiting break, this results in a very low heat transfer coefficient for the remainder of the event or until the core two-phase level recovers the peak power position, at which time the heat transfer coefficient increases to values between 20 and 30 Btu/hr- ft^2 - °F. During the application of the 1.0 heat transfer coefficient, the convective heat transfer coefficient has been shown to be higher, or in the range between 5 and 10 Btu/hr- ft^2 - °F. Thermal rod-to-rod radiation is also modeled during this period. It is also noted that top down quench does not terminate the clad temperature rise during the analyses. This approach is considered conservative for calculating PCT following a LOCA.

Additional analyses of transition cores and full cores of Westinghouse SVEA-96 Optima 2 fuel demonstrated that the full core Westinghouse fuel was most limiting. Furthermore, introduction of the Westinghouse SVEA-96 Optima2 fuel does not adversely affect an equilibrium core comprised mostly of GE14 fuel. These studies were performed for the limiting break size.

When questioned by the NRC staff, the licensee stated that residual heat removal, LPCI and LPCS delivery flow rates obtained through the surveillance program demonstrated that the flow deliveries assumed in the LOCA analyses conservatively bounded the data taken during the surveillance testing. The ECC pumped flows used in the analyses and verified by the surveillance testing are summarized in the RAI response of Reference 2.

The limiting peak local oxidation was found to be 7.1 percent and also included any initial oxidation present for the limiting hot rod calculation. Core wide oxidation was found to be considerably lower than the 0.01 limit in 10 CFR 50.46 for maximum hydrogen generation. Post-LOCA long-term cooling is also demonstrated as provided by the 4500 gallons per minute continued core spray or flooding to the top of the active fuel following reflood of the core.

Based on the review of the application of the Westinghouse BWR LOCA methodology to DNPS and QCNPS, the NRC staff finds that Westinghouse properly and conservatively applied its approved methodologies to the evaluation of the Westinghouse SVEA-96 Optima2 fuel design, as well as the mixed core configurations anticipated during the transition fuel cycles. The NRC staff also verified that Westinghouse addressed the SER conditions and limitations, identified in Reference 1 in Tables 4 through 8 of Attachment 7, from the previous NRC staff review and approval of the Westinghouse BWR LOCA methods. Any changes to the results in Reference 4 should be addressed in accordance with the reporting requirements in 10 CFR 50.46(a)(3).

3.3.3.2 Thermal-Hydraulic Stability

The NRC staff and its consultant from Oak Ridge National Laboratory performed an audit of the thermal-hydraulic stability analysis at the offices of Westinghouse, in Monroeville, Pennsylvania on November 9, 2005. This review includes the results of that audit. It also includes the results of a February 15, 2006, audit of the calculation methodology for setpoints used for Stability Long Term Solution III in DNPS and QCNPS. The main purpose of these audits was to review Westinghouse methodology to calculate plant-specific DIVOM correlation for DNPS and QCNPS. The NRC staff has reviewed the relevant documents supporting DIVOM calculations. Both DNPS and QCNPS plan to use Westinghouse methods to arm Solution III in the near future.

Two scram setpoints are required for implementation of Long Term Solution III: the number of confirmation counts and the minimum oscillation amplitude. The oscillation amplitude setpoint requires a calculation to relate power oscillation amplitude to loss of CPR. The result of this calculation takes the form of the DIVOM correlation.

The DIVOM correlation methodology used to be generic in form, but as a result of a recent Part 21 communication is now calculated on plant- and cycle-specific basis (BWROG-03047). The Westinghouse methodology for calculating the Solution III setpoints involves three steps:

1. Determine the initial minimum critical power ratio (IMCPR) following a flow reduction event to natural circulation from the highest rod line.

2. A generic statistical treatment of oscillation contours determines the hot-channel peak-over-average oscillation amplitude. This procedure is known as HCOM, and provides a 95/95 confidence that the hot-channel oscillation amplitude is smaller than this quantity given that the OPRM cell is oscillating at a particular setpoint limit.

3. Use the 95/95 hot channel oscillation amplitude and the DIVOM correlation to determine the setpoint setting by iterating among different setpoint values. Currently, all Westinghouse plants are expected to use plant-specific DIVOM correlations, as recommended in BWROG-03047.

The Solution III setpoint methodology relies heavily on the plant-specific DIVOM correlation. The Westinghouse methodology for calculating the DIVOM correlation is based on the use of RAMONA-3, which is coupled to the BISON/SLAVE code. RAMONA-3 is used to define the hot channel operating conditions. BISON/SLAVE performs a single hot-channel calculation using the RAMONA-3 boundary conditions. BISON/SLAVE models the CPR correlation for this hot channel.

In addition, Westinghouse performs RAMONA-3 stability calculations runs to define the interim corrective action (ICA) exclusion regions. There are two ICA exclusion regions. The immediate-scram region is defined by a RAMONA-3 decay ratio > 0.8. The exclusion region is defined by the largest region that satisfies both: (1) DR > (0.8 – cu), and, (2) the immediate-scram line boundary minus 5 percent power.

RAMONA-3 is the main Westinghouse code used for stability calculations. It was reviewed and approved by the NRC staff for decay ratio calculations. In the DIVOM methodology, RAMONA-3 is used to define the hot channel boundary conditions; then the BISON/SLAVE code uses those boundary conditions to calculate the CPR.

The ability to predict both oscillation frequency and dry out conditions is the key ability required for predicting the DIVOM correlation. A complete review of the RAMONA-3 BISON/SLAVE methodology was not performed during the audit. However, the NRC staff concluded that this methodology is capable of performing DIVOM calculations with sufficient accuracy for DNPS and QCNPS because the validation of dry out calculations during coolant flow oscillation is performed by the DIVOM methodology.

During the February 15, 2006, audit, the NRC staff reviewed the results of the plant-specific oscillation power range monitor (OPRM) setpoint calculation in Reference 8 for QCNPS, Unit 2, Cylce 19, and found the generic values in Attachment 1, "Westinghouse DIVOM Evaluation," of Reference 8 for the OPRM Amplitude, hot channel oscillation magnitude (HCOM) and Confirmation Count Setpoint acceptable. However, the plant-specific setpoints should be included in the COLR report when the values are available.

The NRC staff has reviewed the license amendment request regarding the transition to Westinghouse fuel for DNPS and QCNPS. Based on the review of References 1, 2 and 8, the NRC staff concludes that the stability analysis incorporating the Westinghouse SVEA-96 Optima2 fuel design is acceptable. The final OPRM setpoints should be reported in the COLR.

Based upon the review of the license amendment request and RAI responses, the NRC staff finds the licensing basis events, including the application of Westinghouse analytical methods, event classifications, and acceptance criteria, acceptable. The NRC staff requests that the final DIVOM curve be provided to the NRC for information.

3.3.4 Westinghouse SVEA-96 Optima2 Fuel Mechanical Design and Fuel Performance

During the November 7 - 10 and December 14, 2005 audit at Westinghouse, the NRC staff reviewed the fuel rod design and fuel assembly design calculations. The NRC staff verified that all of the safety evaluation conditions and limitations within the applicable topical reports, WCAP-15836-P-A and WCAP-15942-P-A, were satisfied. Compliance to the safety evaluation conditions and limitations is also documented in response to RAI No. 1 in Reference 2. As part of the audit, the NRC staff also verified that all of the design requirements were satisfied for the Westinghouse SVEA-96 fuel being implemented at DNPS and QCNPS.

In response to an RAI regarding the plant-specific mechanical compatibility changes to the reference Westinghouse Optima2 fuel design, the licensee provided in RAI response 1a of

Reference 2 a brief description of the assembly changes and noted that these changes were within the envelope approved within WCAP-15942-P-A. The NRC staff has reviewed the design features and agrees that the DNPS and QCNPS plant-specific assembly design is within the regulatory envelope approved in WCAP-15942-P-A.

In response to an RAI regarding control blade interference, the licensee stated in RAI response 1b of Reference 2 that the Westinghouse experience database was applicable to their plant design and that the maximum channel-to-control rod interference for DNPS and QCNPS was less than that of the reference plant described in WCAP-15942-P-A. Further, the calculated rod force-time for DNPS and QCNPS was conservatively greater than that for the reference plant. During the audit, the NRC staff identified that the rod force-time calculation assumed that the control blade weight for DNPS, QCNPS, and the reference plant was equal without any reference to blade specifications. The licensee stated that this approximation was appropriate since any mass difference would be small relative to the control rod force-time difference (Table 18 of RAI No. 1). Based upon the significant difference in calculated rod force-time in the conservative direction, the NRC staff finds this assumption as well as the entire control blade interference assessment acceptable. In response to an RAI regarding the different fuel design limits for each of the different fuel designs, the licensee in RAI response 2 of Reference 2 described the use of the POWERPLEX-III online core monitoring system, which is used to monitor the various fuel design limits. In response to an RAI regarding the applicability of the Seismic/LOCA methodology in CENPD-288-P-A to the Westinghouse SVEA-96 Optima2 fuel design, the licensee in RAI response 5 of Reference 2 described load cycling tests done on the Westinghouse SVEA-96 Optima2 fuel to qualify spacer and channel welds for seismic loads. The licensee stated that all of the Seismic/LOCA fuel design requirements would be verified following the approved methodology in CENPD-288-P-A. In accordance with the approved methodology, the structural analysis of the fuel assembly is based on the fuel support and core grid response spectra for safe shutdown earthquake and channel pressure load from the most limiting LOCA event. Based upon the RAI response 5 of Reference 2, which commits to the use of an approved Seismic/LOCA methodology in combination with mechanical testing done on the Westinghouse SVEA-96 Optima2 fuel to demonstrate compliance with established design requirements, the NRC staff finds the Seismic/LOCA evaluation acceptable for Westinghouse SVEA-96 Optima fuel implementation at DNPS and QCNPS.

Based upon the use of an approved fuel assembly design (Westinghouse SVEA-96 Optima2) and continued compliance with conditions and limitations of the approved Westinghouse analytical methods, the NRC staff finds the proposed change to TS Section 5.6.5 acceptable.

3.4 Change to Technical Specification Surveillance Requirement

During review of the anticipated transient without scram analysis supporting the license amendment, the NRC staff identified that the change from natural boron to enriched boron in the SLC tank was not captured in TS 3.1.7 nor its SRs. In response to an RAI regarding this change to enriched sodium pentaborate in the SLC tank, the licensee in RAI response 6 of Reference 3 provided a proposed TS change for DNPS and QCNPS. This change follows the Standard TS SR Section 3.1.7.10 and requires that the sodium pentaborate enrichment be verified prior to addition into the SLC tank. The NRC staff has reviewed this proposed TS change and finds it acceptable.

Based on the NRC staff's review and evaluation of the proposed changes to transition from GE14 fuel to Westinghouse SVEA-96 Optima2 fuel and the use of supporting Westinghouse analytical methods, the NRC staff has determined that these TS changes to TS Sections 3.1.4, 4.2.1, and 5.6.5, as well as the change to TS SR Section 3.1.7, are in accordance with the guidance of SRP Section 4.2 and 10 CFR 50.36. Further, the NRC staff has reviewed the application of Westinghouse analytical methods supporting the implementation of the Westinghouse SVEA-96 Optima2 fuel design and finds it acceptable. Future core reloads utilizing Westinghouse SVEA-96 Optima2 fuel, along with the approved Westinghouse methods added to TS Section 5.6.5, will continue to satisfy the applicable safety evaluation limitations and conditions. Therefore, on the basis of the above review and justification, the NRC staff concludes that the proposed changes to TSs are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Illinois State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change the requirements with respect to the installation or use of a facility's components located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (70 FR 41445; July 19, 2005). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

8. Letter from P. R. Simpson (Exelon) to U.S. NRC, "Request for License Amendment Regarding Transition to Westinghouse Fuel," dated June 15, 2005, Agencywide Documents Access and Management System (ADAMS) Accession Number ML060620362.

9. Letter from P. R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting Request for License Amendment Regarding Transition to Westinghouse Fuel," dated January 26, 2006, ADAMS Accession No. ML060620375.

10. Letter from P. R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting Request for License Amendment Regarding Transition to Westinghouse Fuel," dated January 31, 2006, ADAMS Accession No. ML060480220.

11. Letter from P. R. Simpson (Exelon) to U.S. NRC, "Additional Information Supporting Request for License Amendment Regarding Transition to Westinghouse Fuel," dated February 22, 2006, ADAMS Accession No. ML060620386.

12. Letter from P. R. Simpson (Exelon) to U.S. NRC, "Request for License Amendment Regarding Transition to Westinghouse Fuel," dated January 20, 2005, ADAMS Accession No. ML050800371.

13. Letter from U.S. NRC to C. M. Crane (Exelon), "Dresden and Quad Cities Nuclear Power Stations Optima2 Fuel Amendment Request Acceptance Review," dated March 17, 2005, ADAMS Accession No. ML050660435.

14. NUREG-0800, Standard Review Plan, Section 4.2, "Fuel System Design," Draft Revision 3, April 1996.

15. Letter from K. R. Jury (Exelon) to U.S. NRC, "Additional Information Supporting Request for License Amendment Regarding Transition to Westinghouse Fuel and Request for Technical Specifications Change for Minimum Critical Power Ratio Safety Limit," dated March 3, 2006, ADAMS Accession No. ML060660556.

16. Letter from P. R. Simpson (Exelon) to U.S. NRC, "Request for License Amendment Regarding Transition to Westinghouse Fuel," dated March 23, 2006, ADAMS Accession No. ML060820353.

17. Letter from P. R. Simpson (Exelon) to U.S. NRC, "Withdrawal of Request for License Amendment Regarding Transition to Westinghouse Fuel," dated March 11, 2005, ADAMS Accession No. ML050730451.

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