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10 CFR 50.90

RS-05-078

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Dresden Nuclear Power Station, Units 2 and 3
Renewed Facility Operating License Nos. DPR-19 and DPR-25
NRC Docket Nos. 50-237 and 50-249

Quad Cities Nuclear Power Station, Units 1 and 2
Renewed Facility Operating License Nos. DPR-29 and DPR-30
NRC Docket Nos. 50-254 and 50-265

Subject: Request for License Amendment Regarding Transition to Westinghouse Fuel

References:

1. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Request for License Amendment Regarding Transition to Westinghouse Fuel," dated January 20, 2005

2. Letter from J. A. Bauer (Exelon Generation Company, LLC) to U. S. NRC, "Withdrawal of Request for License Amendment Regarding Transition to Westinghouse Fuel," dated March 11, 2005

3. Letter from A. Mohseni (U. S. NRC) to C. M. Crane (Exelon Generation Company, LLC), "Dresden and Quad Cities Nuclear Power Stations Optima2 Fuel Amendment Request Acceptance Review (TAC Nos. MC5802, MC5803, MC5804, and MC5805)," dated March 17, 2005

4. Letter from P. R. Simpson (Exelon Generation Company, LLC) to U. S. NRC, "Outline of Revised License Amendment Request Regarding Transition to Westinghouse Fuel," dated April 28, 2005

In Reference 1, Exelon Generation Company, LLC (EGC) requested an amendment to Renewed Facility Operating License Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS) Units 2 and 3, and Renewed Facility Operating License Nos. DPR-29 and

DPR-30 for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2, to support the transition to Westinghouse SVEA-96 Optima2 fuel at DNPS and QCNPS. The NRC discussed several concerns with EGC's justification for the proposed amendment during a conference call on March 4, 2005. Since the NRC indicated during the March 4, 2005, conference call that the submittal lacked information that was needed to continue the NRC review, EGC withdrew the proposed change in Reference 2. The NRC provided specific examples of areas that required additional information in Reference 3. In Reference 4, EGC submitted an outline that identified the type of information to be submitted to the NRC to address the concerns identified in Reference 3.

On April 14 and May 5, 2005, meetings were held between Westinghouse, NRC, and EGC, to address NRC questions regarding the Westinghouse methodologies that will be used to support reload core designs containing SVEA-96 Optima2 fuel at DNPS and QCNPS. During these meetings, EGC stated its intent to continue efforts to transition to SVEA-96 Optima2 fuel beginning with the QCNPS Unit 2 refueling outage in Spring 2006, including submittal of a revised license amendment request in June 2005.

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," EGC requests an amendment to Renewed Facility Operating License Nos. DPR-19 and DPR-25 for DNPS Units 2 and 3, and Renewed Facility Operating License Nos. DPR-29 and DPR-30 for QCNPS Units 1 and 2. This license amendment request is consistent with the information submitted in Reference 4, and supersedes the Reference 1 amendment request in its entirety. This license amendment request addresses the NRC's concerns identified in Reference 3.

This request is subdivided as follows.

- Attachment 1 provides an evaluation supporting the proposed change.

- Attachments 2 and 3 contain the marked-up Technical Specifications (TS) pages for DNPS and QCNPS, respectively, with the proposed change indicated.

- Attachments 4 and 5 provide retyped TS pages for DNPS and QCNPS, respectively, with the proposed change incorporated.

- Attachment 6 provides information to justify the applicability of the Westinghouse fuel and analytical methods to extended power uprate conditions at DNPS and QCNPS.

- Attachment 7 provides an overview of the evaluations that are performed, as part of the Westinghouse reload methodology, to determine the impact to licensing basis events. Attachment 7 also addresses each of the NRC safety evaluation limitations and conditions related to the applicability of Westinghouse's topical reports.

The proposed change has been reviewed by the Plant Operations Review Committee and approved by the Nuclear Safety Review Board for the respective facilities in accordance with the requirements of the EGC Quality Assurance Program.

As described above, EGC plans to transition fuel vendors for DNPS and QCNPS beginning with the QCNPS Unit 2 refueling outage in March 2006. Therefore, EGC requests approval of the proposed change by March 15, 2006, since the core operating limits using the new analytical methods added to TS Section 5.6.5, "Core Operating Limits Report (COLR)," will become effective upon startup following the QCNPS Unit 2 refueling outage. Once approved, the amendments shall be implemented within 60 days. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

The core reload analyses for the affected units may result in the need for additional TS changes to support the transition to SVEA-96 Optima2 fuel, such as a change to the safety limit minimum critical power ratio. These changes, if any, will be submitted to the NRC in a separate license amendment request.

In accordance with 10 CFR 50.91(b), EGC is notifying the State of Illinois of this application for changes to the TS by transmitting a copy of this letter and its attachments to the designated State Official.

Attachments 6 and 7 contain information proprietary to Westinghouse Electric Company LLC; it is supported by an affidavit signed by Westinghouse, the owner of the information. The affidavit, provided in Attachment 8, sets forth the basis on which the information may be withheld from public disclosure by the NRC and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, it is respectfully requested that the information be withheld from public disclosure in accordance with 10 CFR 2.390. Non-proprietary versions of the information contained in Attachments 6 and 7 are also provided in Attachment 8.

There are no regulatory commitments contained in this letter. Any actions discussed in this letter represent intended or planned actions by EGC. They are described for the NRC's information and are not regulatory commitments. Should you have any questions related to this letter, please contact Mr. Kenneth M. Nicely at (630) 657-2803.

I declare under penalty of perjury that the foregoing is true and correct. Executed on the 15th day of June 2005.

Respectfully,

Patrick R. Simpson

Patrick R. Simpson
Manager – Licensing

Attachments:

Attachment 1: Evaluation of Proposed Change
Attachment 2: Markup of Proposed Technical Specifications Pages for DNPS
Attachment 3: Markup of Proposed Technical Specifications Pages for QCNPS
Attachment 4: Retyped Technical Specifications Pages for Proposed Change for DNPS
Attachment 5: Retyped Technical Specifications Pages for Proposed Change for QCNPS
Attachment 6: Applicability of Westinghouse Fuel and Analytical Methods (PROPRIETARY)
Attachment 7: Evaluation of Licensing Basis Events (PROPRIETARY)
Attachment 8: Westinghouse Application for Withholding, Affidavit, and Non-Proprietary Versions of Attachments 6 and 7

cc: Illinois Emergency Management Agency – Division of Nuclear Safety

ATTACHMENT 1
Evaluation of Proposed Change

1.0 DESCRIPTION

In accordance with 10 CFR 50.90, "Application for amendment of license or construction permit," Exelon Generation Company, LLC (EGC) requests an amendment to Renewed Facility Operating License Nos. DPR-19 and DPR-25 for Dresden Nuclear Power Station (DNPS) Units 2 and 3, and Renewed Facility Operating License Nos. DPR-29 and DPR-30 for Quad Cities Nuclear Power Station (QCNPS) Units 1 and 2. EGC will be transitioning to Westinghouse SVEA-96 Optima2 fuel at each DNPS and QCNPS unit beginning with QCNPS Unit 2 Cycle 19, which is currently scheduled to begin in late March 2006. Technical Specifications (TS) 3.1.4, "Control Rod Scram Times," TS Section 4.2.1, "Fuel Assemblies," and TS Section 5.6.5, "Core Operating Limits Report (COLR)," require revision to support this transition. The proposed change is described below.

2.0 PROPOSED CHANGE

TS 3.1.4 requires that each control rod scram time be within the limits specified in Table 3.1.4-1 and that no more than 12 control rods or two adjacent rods be "slow" in accordance with the Table. An editorial change is proposed to remove the phrase "for GE analyzed cores" from Table 3.1.4-1.

TS Section 4.2.1 provides a description of fuel assemblies. The current description states, in part, "The assemblies may contain water rods or a water box." The proposed change revises this sentence to read, "The assemblies may contain water rods or other assembly bypass channels."

TS Section 5.6.5.b specifies the analytical methods used to determine the core operating limits. Specifically, TS Section 5.6.5.b states:

> "b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:"

The proposed change revises the list of analytical methods to add references to the Westinghouse analytical methods that will be used to determine the core operating limits to support cores containing SVEA-96 Optima2 fuel. Specifically, the following references are added.

1. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel."

2. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2."

3. WCAP-15682-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application."

4. WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel."

5. WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1."

6. WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, Supplement 1 to CENPD-287-P-A."

7. CENPD-390-P-A, "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors."

Attachments 2 and 3 provide TS marked-up pages indicating the proposed change for DNPS and QCNPS, respectively. Attachments 4 and 5 provide the retyped TS pages incorporating the proposed change for DNPS and QCNPS, respectively.

3.0 BACKGROUND

EGC will be transitioning to SVEA-96 Optima2 fuel at each DNPS and QCNPS unit beginning with QCNPS Unit 2 Cycle 19, which is scheduled to begin in late March 2006. Westinghouse, in conjunction with EGC, will be designing future core reloads for DNPS and QCNPS beginning with QCNPS Unit 2 Cycle 19. TS 3.1.4, TS Section 4.2.1, and TS Section 5.6.5 require revision to support this transition.

4.0 TECHNICAL ANALYSIS

4.1 Proposed Change to TS 3.1.4

TS 3.1.4 requires that each control rod scram time be within the limits specified in Table 3.1.4-1 and that no more than 12 control rods or two adjacent rods be "slow" in accordance with the Table. Currently, Table 3.1.4-1 lists scram time acceptance criteria, with an annotation that the times listed are for General Electric (GE) analyzed cores. Since Westinghouse will be designing future core reloads for DNPS and QCNPS, an editorial change is proposed to remove the phrase "for GE analyzed cores" from Table 3.1.4-1.

The need for including the phrase "for GE analyzed cores" in the existing TS is documented in Reference 1, which requested NRC approval of TS changes to support the change in fuel vendors from Siemens Power Corporation (SPC) to GE. Specifically, at the time of the change in fuel vendors, the methodology of modeling control rod insertion during a scram was different for SPC and GE. Therefore, Table 3.1.4-1 was modified to include scram times applicable to both the SPC and GE methodology. The NRC approved these changes in References 2 and 3 for DNPS and QCNPS, respectively. The associated NRC safety evaluation stated that this change is acceptable, based on consistency with the NRC-approved GE methodology.

The proposed change does not alter the acceptance criteria for control rod scram times. Future core reloads will be analyzed using the NRC-approved methodology for modeling control rod insertion during a scram as described in Reference 1.

4.2 Proposed Change to TS Section 4.2.1

A complete description of the SVEA-96 Optima2 fuel design is contained in Westinghouse topical report WCAP-15942-P. The SVEA-96 Optima2 design is an improvement from the basic SVEA-96 10x10 design irradiated in the Columbia Generating Station and the SVEA-96+ design irradiated in the Hope Creek Generating Station. SVEA-96 Optima2 is the second generation Westinghouse 10x10 fuel product line to use part-length rods. The SVEA-96 Optima2 fuel channel consists of an inlet piece and a channel. This channel consists of an outer channel with a square cross section and an internal double-walled, cruciform structure, or "watercross," which forms channels for non-boiling water. The watercross structure is composed of a square central water channel and smaller water channels in each of the four wings. The watercross structure, along with the outer channel walls, form four sub-channels in which sub-bundles are positioned.

TS Section 4.2.1 provides a description of fuel assemblies contained within the reactor. The description currently states, in part, that "The assemblies may contain water rods or a water box." Since the SVEA-96 Optima2 fuel design contains a watercross structure, the description of fuel assemblies contained in TS Section 4.2.1 must be revised. Therefore, the proposed change revises the current description to state "The assemblies may contain water rods or other assembly bypass channels."

4.3 Proposed Change to TS Section 5.6.5

TS Section 5.6.5 requires, in part, the establishment of core operating limits prior to each reload cycle and that these limits be documented in the COLR. As stated in TS Section 5.6.5.b, the analytical methods used to determine the core operating limits shall be previously reviewed and approved by the NRC and documented in this section of the TS. The COLR contains a complete identification for each of the referenced topical reports used in the preparation of the COLR.

Future core reloads utilizing SVEA-96 Optima2 fuel will use analytical methods described in topical reports that are currently not listed in TS Section 5.6.5.b. Since these methodologies will be used for the design and analysis of the DNPS and QCNPS core reloads, they must be added to the references included in TS Section 5.6.5.b.

The references being added have all been submitted to the NRC, and have either been approved or are currently under NRC review. Those methodologies that are currently under NRC review are scheduled to receive NRC approval prior to the first use of SVEA-96 Optima2 fuel in a reload core at either DNPS or QCNPS. The following table lists the topical reports being added to TS Section 5.6.5.b, along with the current status of the NRC's review and approval.

Topical Report	Title	Submitted for NRC Review	NRC Approval
CENPD-300-P-A	Reference Safety Report for Boiling Water Reactor Reload Fuel	12/8/94	5/24/96
WCAP-16081-P-A	10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2	5/12/03	12/9/04

Topical Report	Title	Submitted for NRC Review	NRC Approval
WCAP-15682-P-A	Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application	2/8/02	3/10/03
WCAP-16078-P-A	Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel	4/30/03	10/15/04
WCAP-15836-P-A*	Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1	6/25/02	Currently under NRC review
WCAP-15942-P-A*	Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, Supplement 1 to CENPD-287-P-A	10/31/04	Currently under NRC review
CENPD-390-P-A	The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors	4/15/99	7/24/00

* The proposed change includes the "-A" (i.e., designating accepted) following the topical report identification symbol due to the expectation that Westinghouse will reissue the topical report prior to NRC approval of the proposed change.

A brief description of each of these topical reports is provided below. EGC and Westinghouse have reviewed the associated NRC safety evaluations for those topical reports that have been previously approved by the NRC. The use of the topical reports for DNPS and QCNPS reload core designs will be limited to the extent specified in the topical reports and under the conditions and limitations delineated in the NRC safety evaluations. The safety evaluation limitations/conditions and the measures taken to ensure compliance for each approved topical report are listed in Attachment 7.

4.3.1 CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel"

Topical report CENPD-300-P-A for BWR reload fuel describes the reload fuel design and safety analysis process used to evaluate reload applications. This methodology has been applied previously for U. S. BWR reload applications. A comparison of the operating parameters of these applications to those anticipated for the DNPS and QCNPS applications is presented in Attachment 6.

Specific aspects of the Westinghouse BWR reload fuel design and safety analysis methodology (e.g. fuel mechanical design, thermal-hydraulic stability, control rod drop accident, etc.) and supporting codes (e.g. the steady state nuclear codes PHOENIX/POLCA, the fast transient analysis code BISON, the loss-of-coolant accident (LOCA) codes GOBLIN and CHACHA, etc.) are contained in numerous licensing topical reports incorporated by reference in CENPD-300-P-A. Topical report CENPD-300-P-A integrates the separate reports into a single comprehensive reload fuel design and safety analysis methodology.

The six additional topical reports updating calculational methods and supporting new fuel designs, listed in the Table above, are described below. These topical reports collectively provide a comprehensive description of the code methods, code qualifications, design bases, methodologies, and sample applications for reload licensing performed in support of plant modifications. The NRC approved topical report CENPD-300-P-A in Reference 4. The NRC safety evaluation limitations/conditions, and the measures taken to ensure compliance, are listed in Attachment 7. Attachment 7 also addresses the NRC safety evaluation limitations/conditions for those topical reports incorporated by reference in CENPD-300-P-A.

The scope of CENPD-300-P-A includes:

- Descriptions of the Westinghouse BWR assembly mechanical designs (i.e., by reference to other topical reports). These descriptions do not include the SVEA-96 Optima2 fuel design, which was designed after the issuance of CENPD-300-P-A. A description of the SVEA-96 Optima2 fuel design is provided in WCAP-15942-P,
- A description of the Westinghouse BWR assembly design and reload analysis methodology (i.e., by reference to other topical reports),
- A summary of the reload nuclear design methodology,
- A summary of the thermal-hydraulic design process,
- The treatment of anticipated operational occurrences,
- The treatment of fuel loading accidents,
- The treatment of the LOCA and control rod drop accidents (by reference),
- The treatment of special events (e.g., stability analysis and over-pressure protection), and
- An overview of the flow of information between the analysis disciplines.

CENPD-300-P-A focuses on the treatment of mixed cores. Either by reference to supporting topical reports, or by specific description, CENPD-300-P-A describes the process by which mixed cores are treated in the cycle nuclear design and the methodology for assuring mechanical and thermal-hydraulic compatibility. Since fuel rod nodal linear heat generation rate (LHGR) limits are independent of core design, the LHGR limits established by the legacy fuel vendor are provided by EGC and used in mixed cores. Similarly, the maximum average planar linear heat generation rate (MAPLHGR) limits established by the legacy fuel vendor are also used for the mixed core. A LOCA system analysis is performed to confirm the continued applicability of these MAPLHGR limits.

Critical power ratio (CPR) limits, however, can be cycle-dependent. Therefore, EGC will provide detailed information to Westinghouse to establish a

Westinghouse CPR correlation for the legacy fuel following the methodology described in CENPD-300-P-A. Specifically, EGC will provide CPR values calculated using the legacy fuel vendor's CPR correlation to Westinghouse. Since the raw CPR data that was used to develop the legacy fuel vendor's CPR correlation will not be provided, a conservative adder will be applied to the legacy fuel operating limit minimum CPR (OLMCPR) which satisfies the 95/95 statistical criterion.

EGC and Westinghouse apply a systematic process of obtaining all the data required for reload analysis, including those data required for a mixed core application. Subsequent to the data transfer, EGC will confirm that all data used for the reload evaluation have been appropriately applied. A summary description of the methodology used in the development of the Westinghouse CPR correlation for legacy fuel and its application to the calculation of the associated OLMCPR is described in Attachment 7.

4.3.2 WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2"

Topical report WCAP-16081-P-A describes the CPR correlation for SVEA-96 Optima2 fuel design and the experimental data supporting the correlation. The correlation for the SVEA-96 Optima2 assembly and the bases for its acceptance are presented in this report. The NRC approved topical report WCAP-16081-P-A in Reference 5. The NRC safety evaluation limitations/conditions, and the measures taken to ensure compliance, are listed in Attachment 7.

4.3.3 WCAP-15682-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application"

Topical report WCAP-15682-P-A describes changes to the Westinghouse emergency core cooling system evaluation model for BWRs. This version of the evaluation model is identified as USA4. The only difference between this version of the evaluation model and the previously approved evaluation model (i.e., USA2) is the methodology used to determine when the fuel rod cladding will rupture. This document provides the basis for improving the cladding rupture criteria such that rupture occurs when either there is contact between adjacent rods or the burst stress criterion has been exceeded. The MAPLHGR is limited in the application of the USA4 evaluation model to ensure that the 10 CFR 50.46 criteria are met. The NRC approved topical report WCAP-15682-P-A in Reference 6. The NRC safety evaluation limitations/conditions, and the measures taken to ensure compliance, are listed in Attachment 7.

4.3.4 WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel"

Topical report WCAP-16078-P-A describes changes to the Westinghouse emergency core cooling system evaluation model for BWRs. This version of the evaluation model is identified as USA5. The differences between this version and the previously approved version (i.e., USA4) are (1) a change to the counter-current flow limit correlation, (2) the addition of a fuel rod plenum model that is applicable to part-length fuel rods, and (3) incorporation of the applicable features of the STAV7.2 fuel performance model. In addition, this topical report also provides the basis for applying the USA5 evaluation model to the SVEA-96 Optima2 fuel design. The NRC approved topical report WCAP-16078-P-A in Reference 7. The NRC safety evaluation limitations/conditions, and the measures taken to ensure compliance, are listed in Attachment 7.

4.3.5 WCAP-15836-P, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1"

Topical Report WCAP-15836-P describes improvements to the Westinghouse BWR fuel rod performance codes in support of the program to extend the accepted application of these codes to a fuel rod average burnup of 62 MWd/kgU. This report describes the latest versions of the STAV, VIK, and COLLAPS codes. This supplement provides a description of the revised models implemented in the latest code versions along with the qualification actions that demonstrate that these codes are qualified for fuel rod design and safety analyses to a rod average burnup of 62 MWd/kgU.

Westinghouse submitted topical report WCAP-15836-P to the NRC for review on June 25, 2002, and this topical report is currently under NRC review. Limitations and conditions placed on the application of this methodology will be incorporated in the design and safety analysis process when available. It should be noted that both DNPS and QCNPS have license conditions that limit the maximum rod average burnup for any rod to 60 GWD/MTU until the completion of an NRC environmental assessment supporting an increased limit.

4.3.6 WCAP-15942-P, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, Supplement 1 to CENPD-287-P-A"

WCAP-15942-P describes improvements to the methodology contained in CENPD-287-P-A associated with adoption of the latest versions of the STAV, VIK, and COLLAPS codes. Utilization of these improved code versions, in conjunction with the required revisions to the methodology associated with the use of these codes, justifies an extension of the burnup range for which the methodology can be applied to a rod-average burnup of 62 MWd/kgU.

WCAP-15942-P also describes a generic SVEA-96 Optima2 design for which NRC approval is being requested. The report identifies plant specific mechanical compatibility parameters that may need to be changed to assure compatibility with the plant specific core internals and the co-resident fuel, as discussed below. The topical report provides examples of the application of the methodology to the SVEA-96 Optima2 fuel design. In conjunction with an expanded fuel rod and assembly inspection database and test basis, this sample application demonstrates that the SVEA-96 Optima2 assembly satisfies the Westinghouse design criteria to a rod-average burnup of 62 MWd/kgU for the sample plant application. However, the sample analysis is not used to justify plant specific application. The methodology described and illustrated in WCAP-15942-P is applied on a plant specific basis.

The report identifies the following plant specific mechanical compatibility parameters that may need to be changed to assure compatibility with the core internals and the co-resident fuel: (1) channel length; (2) fuel rod/bundle length; (3) channel bypass flow hole size; (4) channel offset and dimple sizes. For example, for DNPS and QCNPS the active fuel length and assembly length will be adjusted to assure compatibility with the resident fuel and the plant. The bottom nozzle will be designed to assure optimum performance in a D-lattice plant. The by-pass flow hole will be adjusted to preserve the current inter-assembly by-pass flow rate.

Westinghouse submitted topical report WCAP-15942-P to the NRC for review on October 31, 2004, and this topical report is currently under NRC review. Limitations and conditions placed on the application of this methodology will be incorporated in the design and safety analysis process when available.

4.3.7 CENPD-390-P-A, "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors"

The two principal computer programs for BWR steady-state nuclear design and analysis used by Westinghouse are PHOENIX and POLCA. 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 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.

The NRC approved topical report CENPD-390-P-A in Reference 8. The NRC safety evaluation limitations/conditions, and the measures taken to ensure compliance, are listed in Attachment 7.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

Exelon Generation Company, LLC (EGC) will be transitioning to Westinghouse SVEA-96 Optima2 fuel at Dresden Nuclear Power Station (DNPS), Units 2 and 3, and Quad Cities Nuclear Power Station (QCNPS), Units 1 and 2. Westinghouse, in conjunction with EGC, will be designing future core reloads for DNPS and QCNPS beginning with

QCNPS Unit 2 Cycle 19, which is scheduled to begin in late March 2006. Technical Specifications (TS) 3.1.4, "Control Rod Scram Times," TS Section 4.2.1, "Fuel Assemblies," and TS Section 5.6.5, "Core Operating Limits Report (COLR)," require revision to support this transition. Specifically, the proposed change: (1) incorporates an editorial change to TS 3.1.4 to clarify that the control rod scram times specified in Table 3.1.4-1 will continue to apply independent of whether the core is analyzed by General Electric; (2) revises TS Section 4.2.1 to modify the description of fuel assemblies to be more generic yet envelope key fuel assembly characteristics; and (3) revises TS Section 5.6.5 to add analytical methods that support design of core reloads utilizing SVEA-96 Optima2 fuel.

According to 10 CFR 50.92, "Issuance of amendment," paragraph (c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EGC has evaluated the proposed change to the TS for DNPS, Units 2 and 3, and QCNPS, Units 1 and 2, using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No

   The proposed change has no effect on any accident initiator or precursor previously evaluated and does not change the manner in which the core is operated. The type of fuel is not a precursor to any accident. The new methodologies for determining core operating limits have been validated to ensure that the output accurately models predicted core behavior, and use of the methodologies will be within the ranges previously approved. The new methodologies being referenced have all been submitted to the NRC, and have either been approved or are currently under NRC review. Those methodologies that are currently under NRC review are scheduled to receive NRC approval prior to the first use of SVEA-96 Optima2 fuel in a reload core at either DNPS or QCNPS.

   There is no change in the consequences of an accident previously evaluated. The proposed change in the administratively controlled analytical methods does not affect the ability to successfully respond to previously evaluated accidents and does not affect radiological assumptions used in the evaluations. Source term from SVEA-96 Optima2 fuel will be bounded by the source term assumed in the accident analyses. There is no effect on the type or amount of radiation released, and there is no effect on predicted offsite doses in the event of an accident.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change does not affect the performance of any DNPS or QCNPS structure, system, or component credited with mitigating any accident previously evaluated. The use of new analytical methods, which have either been reviewed and approved by the NRC or are currently being reviewed by the NRC, for the design of a core reload will not affect the control parameters governing unit operation or the response of plant equipment to transient conditions. The proposed change does not introduce any new modes of system operation or failure mechanisms.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No

The proposed change to TS 3.1.4 clarifies that analyses for design basis accidents and transients will continue to support the scram times listed in TS Table 3.1.4-1, independent of whether General Electric analyzes the core. The proposed change does not alter the acceptance criteria for control rod scram times. Future core reloads will be analyzed using the NRC-approved methodology for modeling control rod insertion during a scram. The proposed change to TS Section 4.2.1 revises the description of fuel assemblies to envelope the SVEA-96 Optima2 fuel characteristics. The proposed change to TS Section 5.6.5 adds new analytical methods for design and analysis of core reloads to the list of methods currently used to determine the core operating limits. The NRC has either previously approved the analytical methods being added, or is currently reviewing the methods.

The proposed change does not modify the safety limits or setpoints at which protective actions are initiated, and does not change the requirements governing operation or availability of safety equipment assumed to operate to preserve the margin of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based upon the above, EGC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements/Criteria

10 CFR 50.36, "Technical specifications," provides the regulatory requirements for the content required in a licensee's TS. 10 CFR 50.36, paragraph (c)(5) states that TS will include administrative controls that address the provisions relating to organization and management, procedures, recordkeeping, review and audit, and reporting necessary to assure operation of the facility in a safe manner. The COLR is required as part of the reporting requirements specified in the DNPS and QCNPS TS administrative controls. In addition, it is required that the analytical methods used to determine the core operating limits be approved and described in the administrative controls section of the TS. The proposed change ensures that these requirements are met.

10 CFR 50.36, paragraph (c)(4) states that design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety. The proposed change revises TS Section 4.2.1 to modify the description of fuel assemblies to be more generic yet envelope key fuel assembly characteristics. The revised description meets the requirements of 10 CFR 50.36, paragraph (c)(4).

In conclusion, based on the considerations discussed above, (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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 ENVIRONMENTAL CONSIDERATION

EGC has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, "Standards for Protection Against Radiation." However, the proposed amendment does not involve: (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22, "Criterion for categorical exclusion; identification of licensing and regulatory actions eligible for categorical exclusion or otherwise not requiring environmental review," Paragraph (c)(9). Therefore, pursuant to 10 CFR 51.22, Paragraph (b), no environmental impact statement or environmental assessment needs be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Letter from R. M. Krich (Commonwealth Edison Company) to U. S. NRC, "Request for Technical Specifications Change, Transition to General Electric Fuel," dated September 29, 2000

2. Letter from S. N. Bailey (NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Issuance of Amendments (TAC Nos. MB0170, MB0171, MB1337, MB1338, MB2715, and MB2716)," dated November 2, 2001

3. Letter from S. N. Bailey (NRC) to O. D. Kingsley (Exelon Generation Company, LLC), "Issuance of Amendments (TAC Nos. MB0168, MB0169, MB1327, and MB1328)," dated December 20, 2001

4. Letter from R. C. Jones (NRC) to D. Ebeling-Koning (ABB CENO Fuel Operations), "CENPD-300-P, 'Reference Safety Report for Boiling Water Reactor Reload Fuel,' (TAC No. M91197)," dated May 24, 1996

5. Letter from H. N. Berkow (NRC) to J. A. Gresham (Westinghouse Electric Company), "Final Safety Evaluation for Topical Report (TR) WCAP-16081-P, '10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2' (TAC No. MB9011)," dated December 9, 2004

6. Letter from H. N. Berkow (NRC) to P. W. Richardson (Westinghouse Electric Company), "Acceptance for Referencing Topical Report WCAP-15682-P, 'Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application' (TAC No. MB4276)," dated March 10, 2003

7. Letter from H. N. Berkow (NRC) to J. A. Gresham (Westinghouse Electric Company), "Final Safety Evaluation for Topical Report WCAP-16078-P, 'Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel' (TAC No. MB8908)," dated October 15, 2004

8. Letter from S. A. Richards (NRC) to I. C. Rickard (Combustion Engineering Nuclear Power), "Acceptance for Referencing of CENPD-390-P, 'The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors' (TAC No. MA5659)," dated July 24, 2000

ATTACHMENT 2
Markup of Proposed Technical Specifications Pages

DRESDEN NUCLEAR POWER STATION, UNITS 2 AND 3

RENEWED FACILITY OPERATING LICENSE NOS. DPR-19 AND DPR-25

<u>REVISED TECHNICAL SPECIFICATIONS PAGES</u>

3.1.4-3
4.0-1
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Table 3.1.4-1 (page 1 of 1)
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(b) Scram times as a function of reactor steam dome pressure when < 800 psig are within established limits.

4.0 DESIGN FEATURES

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The low population zone shall be a five mile radius from the centerline of the reactor vessels.

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4.2.1 Fuel Assemblies

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4.2.2 Control Rod Assemblies

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5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

9. ANF-89-98(P)(A), Generic Mechanical Design Criteria for BWR Fuel Designs.

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11. Commonwealth Edison Company Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods."

12. EMF-85-74(P), RODEX2A (BWR) Fuel Rod Thermal Mechanical Evaluation Model.

13. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel (GESTAR)."

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14. NEDC-32981P, "GEXL96 Correlation for ATRIUM 9B Fuel," September 2000.

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d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

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When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

Insert 5.6.5

15. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel."

16. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlation: SVEA-96 Optima2."

17. WCAP-15682-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application."

18. WCAP-16078-P-A, "Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel."

19. WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1."

20. WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, Supplement 1 to CENPD-287-P-A."

21. CENPD-390-P-A, "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors."

ATTACHMENT 3
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QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

RENEWED FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30

<u>REVISED TECHNICAL SPECIFICATIONS PAGES</u>

3.1.4-3
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4.0 DESIGN FEATURES

4.1 Site Location

4.1.1 Site and Exclusion Area

The site consists of approximately 784 acres on the east bank of the Mississippi River opposite the mouth of the Wapsipinicon River, approximately three miles north of the village of Cordova, Rock Island County, Illinois. The exclusion area shall not be less than 380 meters from the centerline of the chimney.

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4.2.2 Control Rod Assemblies

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11. COTRANSA 2: A Computer Program for Boiling Water Reactor Transient Analyses, ANF-913(P)(A).

12. Advanced Nuclear Fuels Corporation Methodology for Boiling Water Reactors EXEM BWR Evaluation Model, ANF-91-048(P)(A).

13. Commonwealth Edison Topical Report NFSR-0091, "Benchmark of CASMO/MICROBURN BWR Nuclear Design Methods."

14. ANFB Critical Power Correlation Application for Coresident Fuel, EMF-1125(P)(A).

15. EMF-85-74(P), RODEX2A(BWR) Fuel Rod Thermal Mechanical Evaluation Model, Supplement 1(P)(A) and Supplement 2 (P)(A), Siemens Power Corporation, February 1998.

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22. WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors, Supplement 1 to CENPD-287-P-A."

23. CENPD-390-P-A, "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors."

ATTACHMENT 4
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21. WCAP-15836-P-A, "Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1."

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