NRC Request 1

The license amendment request was completed prior to the staff's approval of WCAP-15836-P-A and WCAP-15942-P-A. Now that these two topical reports have been completed, please update the applicability tables in Attachment 6 and the conditions and limitations tables in Attachment 7 to reflect the approved documents. Include the following:

a. Detailed descriptions of the plant-specific changes to the SVEA-96 Optima2 fuel design and the evaluation to ensure mechanical compatibility with core components and co-resident fuel (WCAP-15942-P, Condition #2a).

b. Detailed description of the control blade interference evaluation in accordance with WCAP-15942-P, Condition #4.

Response

Response is provided in Attachment 2.

NRC Request 2

Identify all fuel design operational limits (e.g. thermal mechanical operating limit, burnup, pellet/cladding interaction maneuvering restrictions, etc.) for each fuel type and describe how each is monitored by plant operations.

Response

Key fuel design operational limits are monitored by EGC consistent with Technical Specification (TS) Surveillance Requirements and procedural requirements.

The key fuel operational limits are monitored in accordance with TS Surveillance Requirements via the online core monitoring system. These limits have been established based on the appropriate generic fuel licensing methodology or cycle-specific reload licensing analyses. These limits are:

- Minimum Critical Power Ratio (MCPR)
- Linear Heat Generation Rate (LHGR)
- Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

MCPR limits are based on cycle-specific analyses. For operation of DNPS and QCNPS with Westinghouse SVEA-96 Optima2 fuel, the operating limit MCPR (OLMCPR) values for all fuel types are established by Westinghouse based on the NRC-approved methodology described in CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel." Westinghouse performs plant and cycle-specific analyses to determine the OLMCPR based on an evaluation of anticipated operational occurrences. Westinghouse also determines appropriate power and flow-dependent multipliers to be applied to the OLMCPR values. In order to determine the margin to the OLMCPR, the online monitoring system will calculate the assembly critical power ratio (CPR) using an approved correlation for the SVEA-96 Optima2 fuel and for the co-resident GE14 fuel.

LHGR limits have been established by each fuel vendor for application to their fuel product lines. Westinghouse has generated thermal-mechanical operating limits (TMOL) that are applicable to the Westinghouse SVEA-96 Optima2 fuel that will be loaded in DNPS and QCNPS. TMOLs are established to ensure that the fuel will comply with all fuel design bases during operation for any credible fuel rod power history during the life of the assembly. These design bases are related to the fuel rod internal pressure, cladding stresses, cladding strain, hydriding, corrosion, cladding collapse (i.e., elastic and plastic instability), cladding fatigue, cladding temperature, fuel temperature, and fuel rod bow. Westinghouse provides the TMOLs for the SVEA-96 Optima2 fuel in terms of an LHGR operating limit that should not be exceeded during plant operation. The co-resident GE14 fuel in DNPS and QCNPS will continue to be monitored with the LHGR limits previously provided by Global Nuclear Fuel (GNF) for the GE14 fuel. These were generated by GNF to ensure that the GE14 fuel will comply with all fuel design bases during operation for any credible fuel rod power history during the life of the assembly.

MAPLHGR limits have been established by each vendor based on an evaluation of loss-of-coolant accidents. The MAPLHGR limits for SVEA-96 Optima2 fuel have been established using the methodology described in Westinghouse topical report CENPD-300-P-A, and the MAPLHGR limits for GE14 fuel have been established using the methodology described in GNF topical report NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel." MAPLHGR limits provide protection related to the Emergency Core Cooling System (ECCS). At DNPS and QCNPS, MAPLHGR limits are usually not as restrictive to reactor operation as LHGR limits. That is, the core usually maintains more operating margin to MAPLHGR limits than to LHGR limits.

Key fuel design operational limits that are monitored in accordance with procedural requirements via the online core monitoring system include:

- Fuel exposure limits, and
- Fuel preconditioning restrictions.

Westinghouse and GNF have established fuel exposure limits for their respective fuel product lines.

EGC currently monitors the fuel during operation with respect to fuel preconditioning restrictions that have been established to reduce the potential for pellet-clad interaction (PCI) fuel failures. These restrictions consist of a set of fuel preconditioning criteria that establish the nodal power threshold, envelope, deconditioning rate, ramp rate, etc.

EGC currently monitors DNPS and QCNPS using the Framatome-ANP (FANP) POWERPLEX-III (PPLX) online core monitoring system. PPLX provides for monitoring of all fuel operational limits. PPLX allows the fuel operational limits to be input and resulting values to be calculated and edited as a function of fuel product line, assembly nuclear type, exposure (i.e., of an assembly, node, or individual fuel rod), core thermal power, or core flow, as appropriate to the specific limit. This includes the thermal limits (i.e., OLMCPR limits, TMOL LHGR limits, and MAPLHGR limits) and the fuel exposure limits. PPLX also tracks the fuel nodal powers and the changes in nodal powers to allow for monitoring to the fuel preconditioning envelopes and to the ramp rates associated with the fuel preconditioning criteria.

At approximately two-hour intervals during reactor operation or on demand, PPLX determines the power distribution and exposure distribution for the reactor core and calculates the

associated reactor operating margins to thermal limits, margins to fuel exposure limits, and information related to fuel preconditioning criteria. PPLX can also be used to predict how changes to control rod positions and/or core flow would impact the core thermal power, the power distribution, and associated margins.

NRC Request 3

Describe the interaction between the General Electric (GE) emergency core cooling (ECCS) performance analyses of the GE14 fuel design and the Westinghouse ECCS performance analyses of the Optima2 fuel design with respect to developing the bounding maximum average planar linear heat generation rate limits. Include within this description an explanation of the flow characteristics of each bundle design and how this information is addressed in each respective ECCS analysis.

Response

Response is provided in Attachment 2.

NRC Request 4

Provide the basis for the 0.05 relative assembly power uncertainty used in the safety limit minimum critical power ratio calculation.

Response

The relative assembly power uncertainty used in the Safety Limit MCPR calculation is dependent on the core monitoring system being used. A bounding core radial bundle power distribution uncertainty value of 0.05 was determined by Framatome ANP (formerly Siemens Power Corporation) for use with the POWERPLEX-III core monitoring system. A Framatome ANP Proprietary letter (DEG:00:174 dated July 20, 2000) containing this information was shared with the NRC during the audit that took place during the week of November 7, 2005. Although the Framatome letter provided to the NRC during the audit was specific to the POWERPLEX-II radial bundle power distribution uncertainty, the value of 5% quoted in the letter bounds the actual FANP proprietary radial power distribution uncertainty value for the POWERPLEX-III core monitoring system.

NRC Request 5

Discuss the applicability of seismic/loss-of-coolant accident methodology in CENPD-288-P-A to the SVEA-96 Optima2 fuel design. Include a discussion of the mechanical testing done on the Optima2 grids.

Response

Response is provided in Attachment 2.

NRC Request 6

Section 2.5.5 of the license amendment request identifies a physical change to the standby liquid control (SLC) system being credited in the anticipated transient without scram (ATWS)

analysis. Specifically, the sodium pentaborate in the SLC tank has been upgraded from natural boron (19.8 a/o B10) to enriched boron (30 a/o B10). This enhancement is directly responsible for the mitigation of the accident analysis and therefore must be surveilled in accordance with the ATWS rule, Title 10 of the *Code of Federal Regulations* (10 CFR) Part 62. The Exelon license amendment request does not include a change to technical specification (TS) 3.1.7 and thus, is deficient in that it does not capture this important physical change to the plants nor any surveillance requirements (SRs). Standard TS SR 3.1.7.10 defines the SR for verifying enriched sodium pentaborate. The licensee should address this deficiency to their amendment request.

Response

EGC's response to NRC Request 6 has not yet been finalized. The response to NRC Request 6 will be submitted to the NRC in a separate letter.

NRC Request 7

Section 2.3 of the license amendment request identified a change to the Westinghouse ECCS evaluation methodology for the transition to SVEA-96 Optima2.

a. Per 10CFR50.46, Exelon needs to submit for staff review:

   i. Justification that the Westinghouse ECCS Models are acceptable for and properly applied to Dresden and Quad Cities.

   ii. Results of the plant-specific ECCS evaluation (detail sufficient for staff review).

Response

Response is provided in Attachment 2.

NRC Request 8

Section 4.3.1 states, "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 which satisfies the 95/95 statistical criterion." Demonstrate that the adder meets the 95/95 criterion.

Response

Response is provided in Attachment 2.

NRC Request 9

In Attachment 6, page 5 of 11, the last paragraph alludes to the Westinghouse Topical Report WCAP-15942-P as containing the Westinghouse experience base. Please provide this experience data base in Tabulated form, including as much detail as possible regarding Extended Power Uprates (EPU) and operation with high exit void fractions. That is specifically:

a. Demonstrates quantitatively and qualitatively, that the Lattice/Depletion code systems, and that the current uncertainties and biases established in the Lattice/Depletion code systems remain valid for the neutronic and thermal-hydraulic conditions predicted for the EPU operation. Specifically, demonstrate the uncertainties and biases that are used in the licensee's reactivity coefficients (e.g. void coefficient) are applicable or remain valid for the neutronic and thermal-hydraulic conditions expected for EPU operation.

b. Demonstrate quantitatively and qualitatively, that the fuel isotopic validations and testing performed in the Lattice/Depletion code systems remain applicable for prolonged operation under high void conditions for the fuel lattice designs that would be used for the expected EPU core designs.

c. Demonstrate qualitatively and quantitatively that the Westinghouse neutronic methodology experience base and demonstrate that the Westinghouse methodology is applicable to EPU conditions, specifically to EPU conditions at Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS).

d. Provide any validation data in support of the Westinghouse neutronic methodology prediction capability by comparison to gamma scans and Transverse Incore Probe (TIP) core follow benchmarking based on the current fuel designs operated under the current operating strategies and core conditions. This request pertains to any recent fuel, such as the SVEA-96+ and OPTIMA-2, in particular for first cycle and second cycle fuel.

Response

Response is provided in Attachment 2.

NRC Request 10

In Attachment 6, page 6 of 11, the first paragraph discusses briefly the contents of CENPD-390-P-A.

a. Does this topical include OPTIMA-2 data/analyses?

b. Does this topical contain TIP pin power comparisons for normal and extended power operations?

Response

Response is provided in Attachment 2.

NRC Request 11

Provide the TIP and Gamma comparisons and PROTEUS results, discussed in the 2nd, 3 and 4th Paragraphs on page 6 of 11, Attachment 6.

Response

Response is provided in Attachment 2.

NRC Request 12

In Attachment 6, page 7 of 11, the first four paragraphs on this page, and the Tables that go with them, require further clarification.

Response

Response is provided in Attachment 2.

NRC Request 13

In Attachment 6, page 8 of 11, the first paragraph alludes to pin power testing with results obtained for the mid-planes.

a. Does Westinghouse have any exit plane pin power behavior, particularly at very high exit void fractions?

b. Provide qualitative description of the void data base and the associated correlation. Specifically describe the uncertainty associated with the data gathering, specifying the uncertainties currently applied to the void fraction correlation and justify its applicability for EPU conditions.

Response

Response is provided in Attachment 2.

NRC Request 14

In Attachment 7, page 9 of 43, the justification provided on the next three pages to extend the AA78 slip correlation to pressures beyond those reviewed and approved in the topical report, will require additional quantitative technical justification. For example, nothing was stated regarding the possible effects on the uncertainties introduced due to extrapolation of the Westinghouse void correlation beyond its current data base. Please provide qualitative description of the void data base and the associated correlation. Specifically describe the uncertainty associated with the data gathering, specifying the uncertainties currently applied to the void fraction correlation and justify its applicability for EPU conditions.

Response

Response is provided in Attachment 2.

NRC Request 15

State the bypass voiding criteria or specification that applies to the TIP and the local power range monitor.

Response

Response is provided in Attachment 2.

NRC Request 16

Evaluate the capability of the licensing code systems, including the core simulator, in determining the potential for bypass voiding.

Response

Response is provided in Attachment 2.

NRC Request 17

Provide evaluation and discussion of the lattice/depletion code capability to generate the cross-section with voiding in the in-channel water rods and bypass.

Response

Response is provided in Attachment 2.

NRC Request 18

Evaluate EPU core neutronic and thermal-hydraulic conditions and state for EPU core designs and operating conditions, if bypass voiding can occur during steady state or transient events. Consider operation at all limiting statepoints in the MELLLA domain.

Response

Response is provided in Attachment 2.

NRC Request 19

In August 30, 2004, General Electric Nuclear Energy (GENE) issued a Part 21 report (ML042720293), stating that using limiting control rod blade patterns developed for less than rated flow at rated power conditions could sometimes yield more limiting bundle-by-bundle MCPR distributions and/or more limiting bundle axial power shapes than using limiting control rod patterns developed for rated flow/rated power in the SLMCPR calculation. GNF-A evaluated the plants operating at the MELLLA operating domain and concluded that the potential exists for more limiting SLMCPR at the nonrated flow conditions for plants currently operating at the MELLLA domain as well. GNF-A also evaluated the plants operating at the MELLLA operating domain and identified four plants that may have more limiting SLMCPR calculated at the minimum core flow statepoint. The affected plants submitted amendment requests increasing their SLMCPR value. The staff understand that Framatome did not issue a Part 21 reporting on the SLMCPR methodology that addresses the calculation of the SLMCPR at minimum core flow and offrated conditions similar to GENE's Part 21 report (ML042720293). The following topics pertain to Framatome's methodology for calculating the SLMCPR at minimum core flow at rated power statepoint.

a. Provide reference(s) to the applicable sections of the SLMCPR Westinghouse methodology that specifies the requirement to calculate the SLMCPR at the worst-case conditions for minimum core flow conditions for rated power. Please demonstrate to the staff that the SLMCPR is calculated at different statepoints of the licensed operating

domain, including the minimum core flow statepoint and that the calculation is performed for different exposure points.

b. Discuss or reference the applicable Sections/Chapters that addresses what rod patterns are assumed in performing the nonrated flow SLMCPR calculations. State how it is established that the rod patterns assumed in the SLMCPR calculations for rated power, flow, and minimum core flow conditions, would reasonably bound the planned rod pattern that DNPS and QCNPS would operate under EPU conditions.

c. For implementation of ARTS/MELLLA using Westinghouse methods, show that the DNPS and QCNPS can operate at all statepoints, including the minimum core flow statepoint, without violating their SLMCPR in the event of an anticipated operational occurrence. The minimum core flow statepoint SLMCPR calculations should demonstrate that DNPS and QCNPS can operate at the minimum flow statepoint with some margin.

Response

Response is provided in Attachment 2.

NRC Request 20

Section 2.4 of Attachment 7 does not provide sufficient information regarding the Stability Analysis for the staff to reach a safety determination. The staff expects the following documentation to be submitted in a supplemental submittal to the TS Amendment that was previously reviewed by the staff:

a. Provide a summary of the process followed by Westinghouse and plants with Westinghouse fuel to implement Long-Term Stability Solution III.

b. Provide a summary of the process followed by Westinghouse to calculate plant-specific setpoints and core operating limits report items.

c. Provide a list and short description of the major codes used by Westinghouse and their uses for licensing applications.

d. Describe the status of the licensing basis for these methodologies and identify any topical reports that are NRC-approved or under review to support the methodologies.

e. Document the plant-specific DIVOM calculation for each plant.

Response

Response is provided in Attachment 2.