ATTACHMENT 3
Retyped Technical Specifications Page for Proposed Change

<u>QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2</u>

<u>RENEWED FACILITY OPERATING LICENSE NOS. DPR-29 AND DPR-30</u>

<u>REVISED TECHNICAL SPECIFICATIONS PAGE</u>

5.6-4

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

10. Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors/Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors: Methodology for Analysis of Assembly Channel Bowing Effects/NRC Correspondence, ANF-524(P)(A).

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.

16. NEDC-3298IP, "GEXL96 Correction for ATRIUM 9B Fuel."

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

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

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

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

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

(continued)

ATTACHMENT 4

Westinghouse Application for Withholding, Affidavit, and Non-Proprietary Version of Attachment 2

Westinghouse

Westinghouse Electric Company
Nuclear Services
P.O. Box 355
Pittsburgh, Pennsylvania 15230-0355
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555-0001

Direct tel: (412) 374-4419
Direct fax: (412) 374-4011
e-mail: maurerbf@westinghouse.com

Our ref: CAW-06-2095

January 25, 2006

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Westinghouse Input to Dresden Nuclear Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2 - Request for Additional Information Regarding Transition to Westinghouse SVEA-96 Optima2 Fuel (Proprietary/Non-Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-06-2095 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying affidavit by Exelon Nuclear.

Correspondence with respect to the proprietary aspects of the application for withholding or the Westinghouse affidavit should reference this letter, CAW-06-2095 and should be addressed to B. F. Maurer, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

R. M. Span/for

B. F. Maurer, Acting Manager
Regulatory Compliance and Plant Licensing

Enclosures

cc: F. M. Akstulewicz/NRR
P. M. Clifford/NRR
M. Banerjee/NRR
G. S. Shukla/NRR
L. M. Feizollahi/NRR (affidavit only)

CAW-06-2095

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. A. Gresham, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J A Gresham

J. A. Gresham, Manager
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 25th day
of January, 2006

Sharon L. Fiori

Notary Public

Notarial Seal
Sharon L. Fiori, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires January 29, 2007
Member, Pennsylvania Association Of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse application for withholding accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in NF-BEX-06-15 P-Attachment, " Westinghouse Input to Dresden Nuclear Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2 - Request for Additional Information Regarding Transition to Westinghouse SVEA-96 Optima2 Fuel" (Proprietary), for response to request for additional information , being transmitted by Exelon Nuclear letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse for the Dresden Units 2 and 3 and Quad Cities Units 1 and 2 is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of SVEA-96 Optima2 License Amendment Request.

This information is part of that which will enable Westinghouse to:

(a) Provide technical information in support of License Amendment Request.

(b) Assist customer to respond to NRC RAIs.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this information to further enhance their licensing position with their competitors.

(b) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar analyses and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

Proprietary Information Notice

Transmitted herewith are proprietary and/or non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

Copyright Notice

Westinghouse Input to Dresden Nuclear Power Station, Units 2 and 3; Quad Cities Nuclear Power Station, Units 1 and 2 -Request for Additional Information Regarding Transition to Westinghouse SVEA-96 Optima2 Fuel

January 25, 2006

Westinghouse Electric Company LLC
P.O. Box 355
Pittsburgh, PA 15230-0355

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

Updated Tables 16 and 18 of Attachment 7 (i.e., Reference 1), which correspond to the conditions and limitations of WCAP-15836-P and WCAP-15942-P are attached. NRC formal approval of WCAP-15942-P has not been obtained. Therefore, conditions and limitations documented in Table 18 are based on the NRC's draft safety evaluation for WCAP-15942-P.

Since WCAP-15836-P-A has not been issued, the applicability tables in Attachment 6 of Reference 1 remain valid.

The geometrical compatibility of SVEA-96 Optima2 fuel with existing GNF (GE14) and FANP (ATRIUM-9B Offset) fuel, core internals and handling equipment in the Dresden Nuclear Power station (DNPS), Units 2 and 3 and Quad Cities Nuclear power Station (QCNPS), Units 1 and 2 plants has been evaluated according to References 2 and 3. The results from the geometrical study, based upon input data from Exelon Generating Company, LLC (EGC) and from Westinghouse experience show that the SVEA-96 Optima2 fuel is compatible with existing fuel, core internals, fuel storage facilities and handling equipment during the design life of the fuel. For detailed specifics refer to the resolution to Condition 2 in Table 18 attached.

The control rod insertability evaluation required by the draft SER for Reference 2 has been performed by Westinghouse for SVEA-96 Optima2 fuel in the DNPS and QCNPS plants by combining plant specific assembly pitch and control rod dimensional information with the measured channel bow and channel creep experience database. The conclusion is that both the calculated maximum channel-to-control rod interference and available control rod insertion force-time for SVEA-96 Optima2 in DNPS and QCNPS are bounded by proven Westinghouse successful operational experience and are demonstrated to be acceptable following the methodology defined by References 2 and 3 and Condition #4 of Reference 3. For detailed specifics refer to the resolution to Condition 4 in Table 18 attached.

Table 16 WCAP-15836-P Conditions and Limitations

WCAP-15836-P Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1		
No.	**Condition / Limitation**	**Resolution**
1	STAV7.2 is approved for modeling BWR fuel with the following limitations: a. Solid UO_2 fuel pellet with a maximum gadolinia content of [ ]^{a,c}. b. No substance beyond gadolinia and nominal trace elements shall be added to the fuel pellet for the purposes of altering its physical characteristics. c. Nominal fuel pellet density between [ ]^{a,c} percent theoretical. d. Fully RXA Zircaloy-2 fuel clad material. e. For fuel rods with clad liner (e.g. natural zirconium), the liner thickness shall be no greater than [ ]^{a,c} (nominal). f. Peak rod average burnup limit 62 GWd/MTU.	The conditions are met via: a. The pellet in SVEA-96 Optima2 fuel used in DNPS and QCNPS is solid UO_2 with the maximum gadolinia content of [ ]^{a,c} b. No substance beyond gadolinia and nominal trace elements is added to the fuel pellet for the purposes of altering its physical characteristics c. Nominal fuel pellet density of [ ]^{a,c} (between 92-97 percent theoretical) d. Fully RXA Zircaloy-2 fuel clad material e. Nominal liner thickness of [ ]^{a,c} mils f. Peak rod average burnup limit of 62 GWd/MTU
2	STAV7.2 shall not be used to model fuel above incipient fuel melting temperatures.	The highest fuel temperature will be encountered in the fuel temperature calculation. The maximum fuel temperature is shown to be below the fuel melting temperatures.
3	STAV7.2 shall not be used to model fuel rods with an average cladding temperature above [ ]^{a,c} at any axial node.	The maximum possible average cladding temperature is from cladding strain or fuel temperature anticipated operational occurrence (AOO) calculations, where the power is ramped [ ]^{a,c} above thermal-mechanical operation limits (TMOL). Even at the peak of the power ramps the average cladding temperature is below [ ]^{a,c}.
4	STAV7.2 shall be used only within the range for which fuel performance data were acceptable or for which verifications discussed in WCAP-15836-P and responses to RAIs were performed. For example, Section 3.8 describes a LHGR limit based upon the calibration and verification database of STAV7.2.	The SVEA-96 Optima2 fuel rod properties and assembly design are in the calibration and verification database. The TMOL linear heat generation rate (LGHR) which is the highest LHGR that can be experienced during normal operation is lower than the LHGR limit specified in Section 3.8 of the safety evaluation.
5	Due to the empirical nature of the STAV7.2 calibration and validation process, the specific values of the equation constants and tuning parameters derived in WCAP-15836-P (as updated by RAIs, e.g. Attachment 2 of Reference 3) become inherently part of the approved models. Thus these values may not be updated without further NRC review. Exceptions include the BWR cladding corrosion constants (Table 2.2.51), crud deposition constants (Table 2.2.5-2), and rod nodal power uncertainties for the BWR "Older" data (Uncontrolled and Controlled Cells in Table 3.3-1). These exceptions will be addressed as part of the implementation methodology in WCAP-15942-P.	The released STAV7.2 is based on the approved models. There is no update on the constants and tuning parameters.

Table 18 WCAP-15942-P Conditions and Limitations

WCAP-15942-P Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors – Supplement 1 to CENP-287		
No.	**Condition / Limitation**	**Resolution**
1	Following the fuel assembly and fuel rod mechanical design methodology described in WCAP-15942, as amended by RAI responses, the licensee must ensure that all of the design criteria are satisfied on a cycle-specific basis.	The amended methodology is followed in the design analysis for DNPS and QCNPS. Cycle specific design changes and power histories will be checked to evaluate whether this reference design analysis is still valid. If this analysis does not bound the specific cycle, a new design analysis will be performed.
2	The reference fuel assembly design SVEA-96 Optima2 is approved up to a peak rod average burnup of 62 GWd/MTU. a. In addition to referencing this report in their LAR submittal for implementing SVEA-96 Optima2, licensees must include a description of the plant-specific changes which are being made to ensure mechanical compatibility with core components and co-resident fuel. Further, the licensee must demonstrate that these changes are within the envelope of approved plant-specific changes to the reference design description in Section 3.1. b. Modifications to the fuel assembly design, beyond the mechanical compatibility changes identified in Section 3.1, will invalidate the staff's approval of the SVEA-96 Optima2 reference fuel design. The provisions described in Section 3.1.4 of WCAP-15942-P, "New Design Features", are not approved.	The calculations and evaluations performed for DNPS and QCNPS are valid to a maximum fuel assembly burnup of [ ]^{a,c}, which supports a peak rod average burnup of 62 MWd/kgU. (a) In order to ensure compatibility with DNPS and QCNPS Legacy fuel and core internals, the SVEA-96 Optima2 fuel (i.e., fuel rods, active fuel length and fuel channel) have been shortened by [ ]^{a,c} and "Style 2" in Section 5 of WCAP-15942-P was used, compared with previously evaluated and delivered SVEA-96 Optima2 fuel to other reactors of the GE/KWU type. Other plant specific changes are partial symmetrisation (same level as Legacy fuel) of the originally asymmetric core lattice and minor adaptations of the fuel assembly handle in order to be compatible with Legacy fuel and core internals at all conditions. Also the inlet piece bypass flow holes are adapted so that the SVEA-96 Optima2 fuel is thermal hydraulically compatible with the Legacy fuel and current core conditions. These changes are consistent with RAI 7 of Reference 3 and are within the envelope of approved plant-specific changes to the reference design in Section 3.1 of the NRC safety evaluation. (b) There are no design feature changes to the Reference SVEA-96 Optima2 fuel design defined in Chapter 2 of WCAP-15942 for DNPS and QCNPS other than those identified in the response to RAI 7 of Reference 3.
3	The fuel mechanical design methodology and design criteria are approved up to a peak rod average burnup of 62 GWd/MTU. In addition: a. These methods are approved for application to currently approved Westinghouse SVEA fuel assembly designs. b. These methods are also approved for the calculation of gap heat transfer coefficients (as described in Section 4.4 and RAI#23) for mixed cores containing non-Westinghouse fuel designs.	The peak rod average burnup in this analysis is 62 GWd/MTU. Additionally, a. The assembly design for DNPS and QCNPS is the approved SVEA-96 Optima2 design. b. Gap heat transfer calculations are the only analyses performed for non-Westinghouse fuel.

WCAP-15942-P Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors – Supplement 1 to CENP-287		
No.	**Condition / Limitation**	**Resolution**
4	During initial implementation, licensees must submit to the NRC an evaluation of control blade interference taking into account manufacturing tolerance, channel bulge, and channel bow over the life of the fuel assemblies (similar to RAI#15 response). As part of this evaluation, the licensee must demonstrate the following: a. Calculated maximum channel-to-control rod interference (blade and roller/pad) must be less than that determined for [ ]^{a,c}. b. Westinghouse channel bow database remains valid. This demonstration must consider the materials and manufacturing process employed in the fabrication of the SVEA channels. c. Following the methodology outlined in RAI#15, the calculated control rod force-time [$((P_{accumulator} \times A_{annulus})\ t_{CR\text{-}73\%}) / M_{CR}$] for the target plant must be greater than or equal to the force-time parameter for [ ]^{a,c}. d. Confirm SVEA channel experience is applicable for the specific application and continues to be bounded by the database presented in RAI#15 by assessing the trend in control rod insertion time (e.g. the number of "slow" control rods) in US plants which have implemented SVEA fuel channels since the time of issuance of this SER. This demonstration should identify the number of "slow" control rods as well as the historical significance of these indications. Updates to the database reflecting new channel bow data measurements may be used to address increasing trends in the numbers of slow rods. The updated database will be used as the bases to evaluate control blade interference	The mechanical compatibility analyses to implement SVEA-96 Optima2 fuel in DNPS and QCNPS confirm that: a. The Maximum channel-to-control rod interference (blade and roller/pad) for DNPS and QCNPS is less than that determined for [ ]^{a,c}. b. The current [ ]^{a,c} data base used for the evaluation of control blade interference contains data from the 10x10 SVEA designs including the SVEA-96, SVEA-96+, SVEA-96 Optima and SVEA-96 Optima2 designs. The mechanical design of the channels for these designs has not been modified in a manner that would affect channel bow or bulge. The channel material has evolved from Zircaloy-4 in earlier designs to the current Zircaloy-2 channels. Furthermore, the annealing process has been improved to provide greater uniformity. Both of these changes tend to reduce channel bow. Therefore, the entire data base provides a conservative description of the current SVEA-96 Optima2 channels, with respect to channel bow, which will be installed in the DNPS and QCNPS units. c. The calculated control rod force-time for DNPS and QCNPS is greater than the force-time parameter for [ ]^{a,c}. [ ]^{a,c} d. The [ ]^{a,c} database in the response to RAI 15 of Reference 3 and used for the DNPS and QCNPS application is current. It will be updated as new data becomes available. Scram times in the DNPS and QCNPS units containing SVEA-96 Optima2 fuel will be evaluated to detect any systematic increase in scram times or the numbers of slow rods which would indicate that the [ ]^{a,c} data base is not representative.
5	The lined SVEA fuel PCI threshold on LHGR must be shown to exceed the TMOL LHGR over the entire burnup range, otherwise fuel PCI conditioning guidelines applicable to non-lined fuel must be applied beginning at LHGRs in excess of the lined fuel PCI threshold.	For lined SVEA PCI thresholds that do not exceed the TMOL, fuel PCI conditioning guidelines applicable to non-lined fuel will be applied beginning at LHGRs in excess of the lined fuel PCI threshold.

NRC Request 3

Describe the interaction between the General Electric (GE) emergency core cooling system (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

The response to Question 14 in WCAP-16078-P-A describes the interaction between the Westinghouse ECCS performance analysis and the ECCS performance analyses performed by other vendors of fuel present in the reactor during the transition to Westinghouse fuel.

For the EGC application, the system response to a loss of coolant accident (LOCA) for the limiting break / single failure combination is determined using three core models:

1. An equilibrium core comprised of 100% SVEA-96 Optima2 fuel.
2. An equilibrium core comprised of 100% GE14 fuel.
3. A representative transition core comprised of a mixture of SVEA-96 Optima2 and GE14 fuel.

The thermal hydraulic compatibility analysis, which is performed using the 3D simulator, is used as a benchmark to ensure that the LOCA core model provides an accurate representation of each core configuration (e.g., flow splits between the active core and the intra-assembly/ inter-assembly bypass channels; core pressure drop distribution, etc.) at nominal conditions. The thermal hydraulic compatibility analysis, which is established based on extensive geometrical and hydraulic information provided to Westinghouse by EGC, provides an accurate thermal hydraulic prediction of flow and pressure distributions for a variety of core configurations. The following table summarizes the important thermal hydraulic features of the GE14 and SVEA-96 Optima2 fuel designs for a full core of the designated fuel assembly in the DNPS and QCNPS units.

a,c

a,c

These characteristics are included in the LOCA model, which is tuned to match the corresponding pressure drops, flow splits, etc. After ensuring that the three LOCA core models are accurate at nominal conditions relative to the thermal hydraulic compatibility analysis, a LOCA system response analysis is performed to determine the system response for each configuration.

The system response model also includes boundary conditions to the hot channel model, which is used to determine the response of the hot assembly. The boundary conditions from the hot assembly are used to determine the thermal response of the fuel rods (i.e., peak cladding temperature, maximum cladding oxidation) and ultimately to develop the (maximum average planar linear heat generation rate MAPLHGR) limits. The impact of different core configurations on these boundary conditions is determined by the timing of three key events, which impact the MAPLHGR analysis. These are [ ][a,c].

Westinghouse will consider the system response from the three configurations to determine the limiting one to evaluate the Optima2 MAPHGR limit. For determining the limiting system response, Westinghouse will evaluate the time of uncovery, the time Core Spray pumps achieve rated flow, and time two-phase conditions are re-established. Should the system response for the mixed core be more limiting than that for GE14, Westinghouse will inform EGC to have GNF perform an evaluation of the impact of the mixed core on the GE14 MAPLHGR limits for the transition to SVEA-96 Optima2 fuel.

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

CENPD-288-P-A, Reference 4, describes the general Westinghouse methodology which demonstrates that the Westinghouse reload fuel assembly satisfies the following design bases under a postulated seismic/LOCA event:

a. Fuel fragmentation will not occur as a result of combined normal operation, seismic, and LOCA loads.

b. Control rod insertability will not be impaired.

c. Spacer grid distortion will not be sufficiently great that fuel rod coolability would be prevented.

Per Section 2 of Reference 4, the seismic/LOCA evaluation is performed for each plant application of Westinghouse BWR fuel. The methodology is defined in a clear and generalized format that can be applied:

- To both Westinghouse and non-Westinghouse designed BWR fuel.
- In all BWR reactors (e.g. BWR/2 through BWR/6).
- Accommodating a variety of plant licensing bases and available seismic and LOCA data.

Reference 4, has been approved by the NRC with the conclusion that it presents an adequate and acceptable methodology to evaluate all Westinghouse BWR fuel assemblies subjected to postulated seismic/LOCA events with no restrictions imposed. Therefore, it can be concluded that CENPD-288-P-A is applicable to the Westinghouse SVEA-96 Optima2 fuel assembly.

Reference 4, documents mechanical tests of spacer grids that have been performed to verify the performance under seismic-type loads. The primary tests performed to address potential seismic loads are the lateral load cycling tests. [

]^{a,c}

The SVEA-96 Optima2 fuel is a further development of the SVEA-96 design. The SVEA-96 Optima2 spacer grid design is based on the same Westinghouse SVEA-96 grids with the same

principal design of the grid cell and of the same material. However, there are differences between the two fuel types that may lead to different dynamic responses under a seismic load. [ ][a,c]

[ ][a,c]

For example, for a typical application these design changes were determined to change the natural frequency of the SVEA-96 Optima2 assembly by about [ ][a,c] percent relative to SVEA-96. This leads to less than 1 percent change in deflection.

As discussed in Section 8.3 of Reference 2, Westinghouse has also performed lateral load cycling tests with low-cycle fatigue for the SVEA-96 Optima2 fuel to qualify spacer and channel welds for seismic loads. The test conditions were [ ][a,c]. These tests were performed at room temperature, and scaling factors were used to translate test results to operating conditions in accordance with ASME Section III, Appendix II-1520. The scaling factors include the effects of the temperature and irradiation as well as experimental uncertainty.

The tests have verified that the spacer grids and welds will withstand the following lateral seismic type acceleration at operating conditions without failure and with negligible deformation:

- Spacer grid: [ ][a,b,c]
- Channel welds: [ ][a,b,c]

For more detail refer to Section 8.3 of Reference 2.

For DNPS and QCNPS, the mechanical behavior of the SVEA-96 Optima2 fuel during a postulated combined Safe Shutdown Earthquake (SSE) and Loss of Coolant Accident (LOCA) event is currently in progress and will be completed prior to plant start-up. The methodology for the calculation of stress intensities and component deflections documented in Reference 4 will be followed. The structural analysis of the fuel assembly is based on fuel support and core grid response spectra for SSE and channel pressure load from the most limiting LOCA event. The acceptability of the results will be evaluated against a set of material and component acceptance criteria or experimentally based acceptable external forces, Reference 4 and 5, consistent with the USNRC Standard Review Plan, Section 4.2, Reference 8, and ASME Section III, Appendix F, Reference 9. All tests necessary to support the methodology have been performed or are judged to be unnecessary.

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 10CFR40.56, EGC 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

A report will be provided upon completion to justify the acceptability of the application of the Westinghouse ECCS evaluation methodology for the transition to SVEA-96 Optima2 fuel at DNPS and QCNPS. The report will describe a single 'Unit 5' model that bounds, from a LOCA perspective, all four DNPS and QCNPS units. The report will describe the application of Westinghouse methodology in sufficient detail to demonstrate that the ECCS models are applied properly and in conformance with all limitations / conditions placed on approved topical reports.

This report will provide the basis for future 10 CFR 50.46 evaluations of plant changes; errors discovered in the approved evaluation model; or errors in the application of the approved evaluation model.

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

[

]a,c

1. [

]a,c

Example: USAG14 is the Westinghouse-developed CPR correlation for GE14 legacy fuel. The renormalization consists of [

]a,c

a,c

Where

USAG14 = CPR correlation for GE14

D4.1.1 = CPR correlation for SVEA – 96 Optima2

$f = \text{massflux}\left(Kg/m^2 - s\right)$

$p = \text{assembly exit pressure (bar)}$

$h = \text{assembly inlet enthalpy } (J/gm)$

Correction coefficients:

a,c

[

]a,c

[

]a,c

Table 8-1 Cosine Axial Power Shape (node 1 = bottom)

1	0.349
2	0.496
3	0.636
4	0.769
5	0.893
6	1.006
7	1.107
8	1.194
9	1.267
10	1.325
11	1.366
12	1.392
13	1.4
14	1.392
15	1.366
16	1.325
17	1.267
18	1.194
19	1.107
20	1.006
21	0.893
22	0.769
23	0.636
24	0.496
25	0.349

2. [

]a,c

Example: Comparison of CPR_Exelon/CPR_Westinghouse for GE14 legacy fuel

[a,c

]

3. Apply the conservative multiplier C (note that this is labeled as an adder to the OLMCPR since it will in effect increase the OLMCPR) to the OLMCPR calculation

[

]a,c

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

In the Westinghouse BWR methodology, the Lattice/Depletion code system is used to generate cross sections and other cell data for the core simulator. Uncertainties, biases or even reactivity coefficients are neither generated by nor computed directly from the Lattice/Depletion code system. The cross sections and cell data are generated at the particular plant's conditions, yet as shown in Table 9.1, the DNPS and QCNPS extended power uprate (EPU) conditions fall within Westinghouse's experience base. The plants in which Westinghouse BWR fuel has been used are referred to in bold type in Table 9.1. The application to the DNPS and QCNPS units is indicated by the use of italics for these plants.

The Westinghouse BWR methodology uses uncertainties associated with the power calculations performed by both the Lattice/Depletion code system and the 3D core simulator. The nodal, assembly and pin nodal relative power uncertainties currently used by Westinghouse and included in Westinghouse Topical Report CENPD-390-P-A were noted in Attachment 6, page 9, first row of Table P-1. Those uncertainties were generated from comparisons against

measurements for four plants (see page 90 of CENPD-390-P-A). Since approval of CENPD-390-P-A, Westinghouse has performed additional gamma scans, to support the introduction of new fuel types (including SVEA-96 Optima2), as well as reactor thermal power increases (including EPUs), well beyond the power level and bundle average power level at DNPS and QCNPS. Those additional gamma scans were presented in Attachment 6, page 9, rows two through four of Table P-1. As can be seen in the table, neither the introduction of new fuel types, nor higher power levels have degraded the accuracy initially documented in CENPD-390-P-A.

Westinghouse Topical Report CENPD-390-P-A Chapter 3 presents the qualification of the Lattice/Depletion code system and its associated library. In that chapter, several critical experiments are modeled with the Lattice/Depletion code system. Nevertheless, the chapter starts with the following statement – "The primary application of PHOENIX is to generate the few-group nodal cross sections and other physics constants for POLCA. Therefore, the benchmarking of POLCA to plant data described in Chapter 5 provides the best overall qualification of PHOENIX." Thus, isotopic validation is not performed directly with the Lattice/Depletion code system, but more as part of an integral method, including the core simulator. Nevertheless, Westinghouse continually evaluates its Lattice/Depletion code system by comparing calculated global parameters (reactivity, power distributions, fission/capture rates) against higher order methods.

Westinghouse Topical Report CENPD-390-P-A Chapter 5 presents the qualification of the 3D core simulator. In that chapter, multiple comparisons are presented, including gamma scans and traversing in-core probe (TIP) instrumentation comparisons for four different plants. The gamma scans and TIP comparisons included in CENPD-390-P-A include SVEA-96 fuel assemblies. As previously mentioned, additional gamma scans have been performed to address new fuel types and more demanding operating conditions. Regarding new TIP comparisons including SVEA-96 Optima2 at more demanding operating conditions, Figure P-2 in page 10 of Attachment 6 notes very consistent results for multiple cycles at [ ]^{a,c}. The figure notes the nodal and radial root mean square (RMS) differences, as well as the fraction of loaded fuel containing part-length rods. This figure notes that for more challenging conditions than those at DNPS and QCNPS, the introduction of SVEA-96 Optima2 did not cause a degradation in the TIP comparison results with the Westinghouse neutronic methods.

Westinghouse has previously applied the Lattice/Depletion code system as well as the core simulator for neutronic and thermal-hydraulic conditions that cover the EPU conditions at DNPS and QCNPS. In addition, Westinghouse has continued to validate its power uncertainties with additional testing and measurements at more demanding conditions than those at DNPS and QCNPS. Westinghouse also continually evaluates the Lattice/Depletion code system and core simulator performance, and how they are used within the BWR methodology. It is Westinghouse's conclusion that its neutronic methods are capable of accurately modeling the EPU conditions at DNPS and QCNPS.

a,c

Table 9.1 – Experience Data Base

* Plants in which Westinghouse BWR fuel has been used.

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

The comparisons presented in Westinghouse Topical Report CENPD-390-P-A Chapter 5, "POLCA Qualification," predate the introduction of the SVEA-96 Optima2 fuel. Nevertheless, the information provided in Section 4.0 of CENPD-390-P-A, "POLCA," does include model descriptions for part-length rods assemblies. One of the main objectives of this new version of POLCA was the treatment of part-length rods assemblies.

Westinghouse Topical Report CENPD-390-P-A includes TIP analyses for four different plants (see Table 5.6 on page 90). The information in the Topical Report includes both normal and EPU conditions for the reactors identified as A, B, and C. The information for Reactor D is for normal operating conditions only. The topical report includes pin power comparisons for two different plants (see Section 5.3.2 on page 72). The assemblies analyzed were for reactors prior to undergoing their EPUs. However, those reactors at pre-uprate conditions were at a higher power density than DNPS and QCNPS.

Although Westinghouse Topical Report CENPD-390-P-A does not include results with SVEA-96 Optima2 fuel nor with the challenging EPU conditions present today, Westinghouse has performed additional gamma-scan and TIP comparisons to address new fuel types as well as increased power levels and more challenging operating conditions. The nodal, assembly and pin nodal relative power uncertainties currently used by Westinghouse and included in Westinghouse Topical Report CENPD-390-P-A were noted in Attachment 6, page 9, first row of Table P-1. Rows two through four present the results for the additional gamma scans. As can be seen in the table, neither the introduction of new fuel types, nor higher power levels have degraded the uncertainties initially documented in CENPD-390-P-A. Regarding new TIP comparisons including SVEA-96 Optima2 at more demanding operating conditions, Figure P-2 in page 10 of Attachment 6 notes very consistent results for multiple cycles at [ ][a,c]. The figure notes the nodal and radial RMS differences, as well as the fraction of loaded fuel containing part-length rods. This figure notes that for more challenging conditions than those at DNPS and QCNPS, the introduction of SVEA-96 Optima2 did not cause a degradation in the TIP comparison results with the Westinghouse neutronic methods.

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

The second and third paragraphs on page 6 of 11, Attachment 6 of Reference 1, provide background information on Westinghouse's methods for establishing the power uncertainties, which includes plant TIP comparisons and pool-side gamma scan measurements. The fourth paragraph refers to experiments performed at the KRITZ facility and at the LWR-PROTEUS facility. Attachment 6 already contains results for the latest set of TIP comparisons, gamma-scan measurements, and the PROTEUS experiments.

Westinghouse Topical Report CENPD-390-P-A presents results for the KRITZ facility experiments, as well as several plant TIP comparisons and pool-side gamma scan results. TIP comparison results are presented in Chapter 5 for four different plants. Gamma scan results are also presented in Chapter 5, for two sets of measurements. Since the approval of the Topical Report, Westinghouse has performed additional TIP comparisons and gamma-scan measurements to address new fuel types as well as increased power levels and more challenging operating conditions. The nodal, assembly and pin nodal relative power uncertainties currently used by Westinghouse and included in Westinghouse Topical Report CENPD-390-P-A were noted in Attachment 6, page 9, first row of Table P-1. Rows two through four present the results for three additional sets of gamma scan measurements. Regarding new TIP comparison results, Figure P-2 in page 10 of Attachment 6 presents results for multiple cycles at [ ][a,c]. The figure notes the nodal and radial RMS differences, as well as the fraction of loaded fuel containing part-length rods.

The PROTEUS experiment results are also included in Attachment 6. Rows five and six of Table P-1 and Figure P-1 present those results. The second paragraph on Attachment 6, page 7 of 11 provides some discussion on the PROTEUS experiment results.

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

The description and equations used for the statistical calculations are included on pages 73-74 of Westinghouse Topical Report CENPD-390-P-A. In the Topical Report, the differences between measurements and POLCA calculated values are noted as $RMS_{overall}$, RMS_{radial}, and RMS_{axial}. The "overall" label represents the nodal differences, whereas the "radial" label represents the assembly differences. Note also that in the Topical Report, the differences are left in percentages.

On page 122 of Westinghouse Topical Report CENPD-390-P-A, the relative fuel rod power uncertainty of [ ]^{a,c}, relative nodal power uncertainty of [ ]^{a,c}, and relative assembly power uncertainty of [ ]^{a,c} are noted. Those same values are noted as "fractional standard deviations" in Attachment 6, page 9, first row of Table P-1. Thus, the term fractional standard deviation implies the uncertainty, in fractional (not percentage) form.

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

The pin power testing performed in the PROTEUS facility is performed at the mid-plane. The reason is that the facility is a small critical core, with significant axial leakage. In order to facilitate the validation of the lattice codes, the measurements are performed at the mid-plane, where the spectral conditions are least sensitive to leakage. The experimental conditions in the axial direction are constant; for measurements performed at [ ]^{a,c}, the entire experiment's axial distribution has constant (non-voided) density. Similarly, for measurements performed at [ ]^{a,c}, the entire experiment's axial distribution is set to [ ]^{a,c}.

As noted in Attachment 6 of Reference 1, page 10 of 11, Figure P-1, the PROTEUS experiments were performed at four different conditions. The first set, [

]^{a,c}

In connection with the introduction of 10x10 fuel designs with part-length rods, Westinghouse performed new void measurements at its FRIGG loop to confirm the validity of the void correlation. The new measurements were performed on a SVEA-96 Optima model, with [

]^{a,c}. Figure 13.1 shows the void correlation prediction – measured void results, as a function of measured void. Two things to note in the figure are the lack of a [

]^{a,c}.

In order to conclude that the void measurements taken cover the range observed at DNPS and QCNPS under EPU conditions, Figure 13.2 was generated. This figure presents the axial void distribution for three sample hot channels at [

][a,c]. Based on the values shown in Figure 13.2, the void measurements performed at FRIGG clearly cover the range observed at DNPS and QCNPS under EPU conditions.

Although there is uncertainty in the measurements and data gathering [

][a,c]

Figure 13.1 Void Measurements Results

a,c

Figure 13.2 DNPS and QCNPS Hot Channel Void Contents

a,c

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

Description of the AA78 void correlation data base

The AA78 slip correlation is described in the BISON Topical Report RPA 90-90-P-A. This correlation is basically a bubble flow correlation modified to cover annular flow for BWR fuel bundle. [

]a,c

The correlation is a best fit to void measurements performed with full-scale (36 and 64 rods) test sections in Westinghouse's FRIGG test loop. The original recommended range of applicability was:

Pressure: 3.0 to 9.0 MPa (435 to 1305 psia)
Max flux: 500 to 2900 kg/m^2 s (0.30 to 2.1 Mlb/h-ft^2)
Quality: 0 to 1.0

Covered ranges in these early FRIGG void measurements were:

Table 14-1 Covered Ranges in the AA78 Database

Test Section	Pressure (bar)	Mass Flux (kg/m^2s)	Steam Quality (%, max)	Void Fraction (%, max)
0F-36	30-90	550-2900	40	90
0F-64A	48, 68	500-2500	40	90
0F-64B	68	500-2000	55	95

Additional void measurements were later performed for SVEA-96 geometries (sub-bundle test sections) which extended the validity of AA78 correlation to 400 kg/m^2s. The void predicted by the AA78 correlation was compared to these new measurements and extrapolation below the data range for mass flux is considered acceptable at least down to 400 kg/ m^2s.

This new data covered the following ranges:

Table 14-2 Additional void measurement data utilized for V&V of the AA78 void correlation

Test Section	Pressure (bar)	Mass Flux (kg/m^2s)	Steam Quality (%, max)	Void Fraction (%, max)
SF24VA	55, 70	400-2000	35	90
SF24VB	55, 70, 80	400-1625	40	87

The error distribution and standard deviation for the AA78 void correlation as a function of the void is showed in Table 14-3 and the comparison against each measurement series in Table 14-4.

Table 14-3 Error distribution as a function of the AA78 predicted void

a,c

Table 14-4 Mean error and standard deviation of the AA78 predicted void compared to the measured void for the different series

a,c

EPRI void correlation

The EPRI void correlation is based on a larger data base which includes not only rod bundle measurement but also measurements from heated rectangular channels and round tubes. The description of the correlation is given in EPRI Report NP-2246-SR, "A Mechanistic Model for Predicting Two-Phase Void Fraction for Water in Vertical Tubes, Channels, and Rod Bundles," G.S. Lellouche and B.A. Zolotar, 1982.

The statistical Analysis of the Model versus Data for the different type of measurements is provided in Table 3, 8, and 11 of the EPRI report and summarized in the following table. In addition EPRI NP-2246-SR Table 13 gives the Model versus Data - Pressure and Flow Range Comparison.

Table 14-5 Mean error and RMS error of the EPRI predicted void compared to the measured void for the different type of experimental data

Experimental Data	Mean error	RMS Error σ	Sample Size
Rod Bundles	-0.0002 ± 0.0010	0.028	784
Rectangular Channels	-0.0021 ± 0.0018	0.051	776
CISE Tube Data	-0.0007 ± 0.0010	0.022	440

Information provided during the NRC revision of the AA78 void correlation (Topical Reports RPA-90-90-P-A and CENPD-292-P-A)

During the NRC review of the Topical Report RPA 90-90-P-A, questions regarding the void models were discussed further. Some of the information provided in responses is relevant to the discussion of the applicability of the correlation to pressures higher than [ ]^{a,c}.

Question 5 regarding the limitations of several correlations, including AA78, are answered on RPA-90-90-P-A pages Q5-1 to Q5-6 and included comparisons with FRIGG loop data. The following text has been extracted from the response regarding the AA78 void correlation.

"The verified data range covers most BWR applications. However, in some extreme cases, such as design basis pressurization transients (MSIV closure without position scram) or trip of all recirculation pumps, the limits of the above data range may be exceeded. However, the dependencies in pressure and mass flux are smooth and continuous, and the correlation prediction outside the above range follows the expected trend."

To justify that extrapolation beyond the test conditions is acceptable, two figures, Q5.1 and Q5.2, were provided. Figure Q5.1 plots measured void against steam quality for two pressures, 7 and 9 MPa (1015 and 1305 psia), at the same inlet subcooling. Also shown are the BISON calculated curves for various pressures. These calculated curves show that there is a smooth trend in void as a function of pressure. Figure Q5.2 shows measured versus calculated void at different pressures, and demonstrate that there is no significant trend in the error as a function of pressure.

Question Q24 requested further justification of the use of the void and boiling correlations in BISON at pressures higher that [ ]^{a,c}.

Comparison with other correlations with somewhat larger ranges of applicability has verified that the correlation behavior is also correct outside the above ranges. Further discussion and justification is provided through the response to NRC Question 24 on RPA-90-90-P-A pages Q24-1 to Q24-6. Comparative graphs, of the same type as now provided in Reference 1, of pressure trends up to [ ]^{a,c}. The graphs compare void change trends predicted with AA78 combined with the Solberg boiling/ condensation model and with the Lellouche-Zolotar EPRI slip correlations described above.

The EPRI correlation has been verified for a wide range of pressures. It was developed to fit not only the rod data which forms the basis of the AA78 correlation, but also other data including measurement in rectangular channel experiments at 10.3 and 11.0 MPa (1493 and 1598 psia). Thus, it serves as a reference for the variation of void fraction with pressure for a range of geometries.

The following text has been extracted from the response to NRC Question 24:

[

]^{a,c}

Comparison of this figure with the corresponding curves calculated with AA78 and the Solberg models using parameters derived for a single channel application (AA, Figure Q24.2), and using parameters for application to core average conditions (W, Figure Q24.3), and also with curves calculated using the modified Bryce-Holmes correlation (Figure Q24.4), indicates that the change of void fraction with pressure over the range [ ]^{a,c} is the same for all methods. "

The application of the AA78 void correlation to pressures up to [ ]^{a,c} was justified through the response to Questions Q5 and Q24.

The matter was further discussed in the supplement to the Topical Report RPA-90-90-P-A, CENPD-292-P-A "BISON – One Dimensional Dynamic Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification," July 1996. This supplement to the BISON topical report was submitted, among other improvements, to change the boiling and condensation model (core void profile) [

]^{a,c}

The qualification was provided in Section 6.5.3.2 (comparison against the Peach Bottom Turbine Trip data) and in Appendix A in the response to NRC Question A1 to CENPD-292-P-A. The same qualification as the one performed in response to Question Q5 to RPA-90-90-P-A, was repeated for the EPRI boiling/condensation model in combination with the AA78 slip (void) correlation. The results of the prediction against the FRIGG loop data are presented in Figures A1-1 and A1-2. These figures show that the correlation gives comparable results with no systematic deviations over the entire range of void fractions up to [ ]a,c.

Justification for Extending the Validity Range of AA78 Correlation

To calculate the pressure response during an anticipated transient without scram (ATWS) up to the acceptance criterion of 1500 psia, [

]a,c This range increase is supported by extended comparative graphs of the same type as the ones presented in the response to NRC Question 24 to RPA 90-90-P-A shown below.

The two differential voids versus steam quality figures for AA78 and EPRI respectively, show that both correlations have the same trends. [

]a,c

The AA78 correlation is as shown above verified against measured data for pressures up to [ ]a,c. In Figure 6.4 of Topical Report CENPD-292-P-A, "BISON – One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification," the RMS error of the AA78 correlation as implemented in BISON is given to be [ ]a,c by direct comparisons to measurement data. The mean error is [ ]a,c. When extrapolating further a comparison with the EPRI correlation is used.

The EPRI void correlation (equivalent to the Chexal-Lellouche drift flux correlation) is described in the Paul Coddington and Rafael Macian paper "A study of the performance of void fraction correlations used in the context of drift-flux two-phase flow models," Nuclear Science Engineering and Design, 215 (2002) 199-216. In this paper, void fraction results were compared to a wide-range of experimental data with various geometry, inlet subcooling, power distribution, and pressure values (up to 15 MPa = 2176 psia).

Comparing the differential void changes versus 7.0 MPa and calculating the standard deviation and the bias between AA78 and EPRI for pressures between [ ]a,c generates the following graphs.

Figure 14-1 [
]a,c

a,c

[

]a,c

a,c

[

]a,c

The major explanation for [ ]a,c when extrapolating and the good agreement between EPRI and AA78 at pressure higher than the AA78 data base maximum pressure (FRIGG loop measurements) is given by the fact that [

]a,c

The conclusion of the comparison against experimental data provided in the response to Questions Q5 and Q24 to RPA-90-90-P-A demonstrate that there is a smooth trend in void as a function of the pressure and that there is no significant trend in the error as a function of pressure. [

]a,c

This is even confirmed by comparing the AA78 void to other void correlations based on experimental data for a wider range of pressures, similar to the EPRI void correlation which includes measurements up to 11 MPa. Also Table 13 of the EPRI report NP-2246-SR shows the lack of trend in the Model versus Data bias with pressure. The comparison between AA78 void correlation to other methods as shown in Figures Q24.1 to Q24.4 and the figures in Attachment 7 of Reference 1 (also presented below) indicate that the change in void fraction with pressure predicted over the range [ ][a,c] is the same for all methods.

At EPU conditions with increased flow window, the core average void is expected to increase since the core average power is higher even though the sub-cooling also increases due to increased Feedwater flow. However, the highest void fractions occur in the hot channels. The hot channels at EPU conditions still have approximately the same exit void fraction, since they still are limited by the thermal limits (e.g. CPR, that limits the bundle power). At EPU conditions the highest power channels have practically unchanged exit void fractions. The main difference at EPU conditions is that more channels have higher powers.

For this reason, all correlations valid at high voids (e.g. AA78 which is based on rod bundles void measurements up to [ ][a,c]) are still within range at EPU conditions. Further justification of the applicability of the void correlation to EPU conditions and the comparison of the POLCA predicted void to the more recent FRIGG measurement for SVEA-96 Optima2 is provided in the response to NRC Request 13 above.

Figure 14-2 [ ][a,c]

a,c

Figure 14-3 [ ]a,c

a,c

NRC Request 15

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

Response

Per Reference 5, CENPD-300-P-A, the Westinghouse BWR fuel assembly is designed to maintain the inter-assembly bypass flow within the same range as the original plant design or within the same range provided by the current resident fuel. In addition, the BWR fuel assembly is also designed to assure sufficient flow to the water cross, in order to prevent significant boiling in the water cross at full power. An axially-averaged void content of [

]^{a,c}. The inlets to fuel assembly bypass flow regions are sized to provide the required flow rate to meet the design criteria. Therefore, it can be concluded that the implementation of the Westinghouse Optima2 fuel in the DNPS and QCNPS cores has negligible impact on the existing voiding criteria or specification or the performance and accuracy of the TIP and the local power range monitor (LPRM) readings.

The effect of boiling in the bypass regions on the accuracy of the simulated in-core detector signal is addressed here. An important result of the POLCA calculation for use in comparing POLCA predictions to measurements is the simulation of the signals from the neutron and gamma sensitive detectors in the core. The simulated detector signals are determined for both the TIP and the LPRM. The neutron sensitive response calculation is based on computing the reaction rate induced in the detector by the fast and thermal flux at the detector region. A combined model (explicit detector modeling during lattice calculation, together with a detector model in the 3D core simulator) is used to simulate the response of neutron-sensitive instrumentation. The detector reaction rate is calculated via:

[ ]^{a,c} (1)

Where [

]^{a,c}.

In the core simulator, [

]^{a,c} The detector formula used in the simulator relies on the known homogeneous flux.

[ ]^{a,c} (2)

Where, to be consistent with Equation (1),

$$\left[\qquad \qquad \right]^{a,c} \qquad (3)$$

$$\left[\qquad \qquad \right]^{a,c} \qquad (4)$$

The [

]^{a,c}

The [

]^{a,c}

The best evidence of the reliability of the model is the excellent agreement obtained between simultaneous measurement of TIP distributions and bundle gamma scan. For details refer to response to NRC Request 10.

NRC Request 16

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

Response

As discussed in the response to NRC Request 15, the Westinghouse BWR fuel assembly is designed to maintain the inter-assembly bypass flow within the same range as the original plant design or within the same range provided by the current resident fuel. In addition, the BWR fuel assembly is also designed to assure sufficient flow to the water cross, in order to prevent significant boiling in the water cross at full power. An axial averaged void content of [

]^{a,c}. The inlets to fuel assembly bypass flow regions are sized to provide the required flow rate to meet the design criteria.

The system simulation code, POLCA, is used to calculate that bypass flow rate in the bypass region. The size of the holes in the bypass region is used by POLCA to determine the bypass flow rates utilizing the same thermal hydraulic governing equations of conservation of mass, momentum and energy, which are used in the active region of the fuel assembly. Similarly, the

constitutive relations that close the solution of the mentioned governing equations in the active region are also used in the bypass region. Since the active region and the bypass region communicate with each other through several paths, the equations are solved iteratively until the calculated pressure drop through the active and bypass regions are reasonably close. [

]^{a,c}

The Westinghouse safety analysis and design codes, BISON and GOBLIN, model the transport of momentum, mass and energy of single phase and two phase coolant in the core and bypass channels and the external coolant loops. No distinction is made between the active and the bypass regions of the core. The same conservation equations and constitutive relations are used in the core and bypass regions. The conservation equations are solved iteratively until the pressure drops in the active and the bypass regions are reasonably close.

The process described above is consistent with other thermal hydraulic codes used by the industry, such as VIPRE, RETRAN, RELAP etc, with no restriction on the amount of bypass flow through the bypass channels.

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

The PHOENIX two-dimensional physics lattice code is used to generate cross sections used by the core simulator code POLCA, including the detector relative signal, as a function of
[

]^{a,c}

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

The relevant parameter in evaluating core is the [

]a,c regardless of the existence of EPU. DNPS and QCNPS, as discussed in the response to NRC Request 9 and shown in Table 9-1, even after EPU are [

]a,c.

Higher power densities, primarily, lead to [

]a,c

As discussed in response to NRC Requests 15 through 17, there is the possibility of a small amount of boiling in the bypass channels during normal operation and anticipated operational transients. It was also discussed that small amounts of boiling in the bypass region have [

]a,c.

Figure 18-1 Average Void Content Before/After EPU a,c

[

]

As can be seen, following a 10 percent power uprate results in a negligible change in the core average void content.

Figure 18-2 Void Content Before/After EPU

a,c

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 to Part a

The generic SLMCPR methodology is described in Section 5.3.2.1 and the Response to RAI's F11 and F13 of Reference 5. The methodology was further clarified in the Response to RAI D-13 of Reference 6.

The requirement to calculate the SLMCPR at the worst case conditions for minimum core flow conditions at rated power is covered by the general requirement that "the SLMCPR is established based on a single conservative radial power distribution used to represent that cycle" in Section 5.3.2.1 of Reference 5. [

]a,c While CPR increases at reduced core power relative to rated conditions, it is necessary to specifically evaluate single-loop conditions since [ ]a,c. An example of application of the methodology is provided in the response to Part c which is an outline of the QCNPS Unit 2 Cycle 19 SLMCPR analysis.

Response to Part b

Since the SLMCPR is based on the number of fuel rods expected to be in boiling transition, the SLMCPR increases as the number of assemblies with CPRs close to the limiting CPR assembly and the number of fuel rods with CPRs close to the limiting fuel rod CPR increase. Consequently, the SLMCPR increases as the relative assembly and fuel rod power (and, therefore, CPR) distributions become more uniform. Only an assembly at the OLMCPR has the potential to challenge the SLMCPR during an AOO. [

]a,c

Response to Part c

The SLMCPR analysis for QC2, Cycle 19, is currently in progress, and the scope of that analysis provides an illustration of the process described above.

Two-Loop

Since the SLMCPR will be the interplay of various factors (e.g. assembly power and fuel rod power distributions), it is calculated throughout the Reference Core cycle to determine a conservative SLMCPR as follows.

a. [

]a,c

b. Based on the results of Step a, additional state points are evaluated to find the most limiting point(s) in the cycle.

c. At the most limiting cycle burnup(s), [

]^{a,c}

Single Loop

[

]^{a,c} Additional points may be evaluated as required to establish the limiting credible single-loop SLMCPR for the cycle.

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

a. The process followed by Westinghouse to implement the Long-Term Stability Solution III is the process developed by the BWROG as described in References 10 and 20. In the following table the Westinghouse methodology is compared to the cycle-specific DIVOM procedure guideline:

Table 20-1 Plant Specific DIVOM Procedure

No.	Element	Plant Specific Regional Mode DIVOM Procedure Guideline (Ref. 10)	Westinghouse Methodology
1	Plant-specific model	Generate base deck for plant to be evaluated.	Consistent with the guideline, Westinghouse sets up a RAMONA3 input deck following the procedures established in References 11 and 13.
2	Cycle-specific model	Incorporate cycle-specific characteristics into base deck (bundle types, CPR correlation, etc.).	Consistent with the guideline, Westinghouse sets up a steady-state operating conditions following the procedures established in References 11 and 13.
3	3D simulation data	Generate best-estimate steady-state neutronic and thermal-hydraulic data with a 3D methodology at the desired power/flow state point.	Consistent with the guideline, Westinghouse generates best-estimate steady-state neutronic and thermal-hydraulic data using the 3D POLCA code (Ref. 14) and procedures established in References 10 and 12.

No.	Element	Plant Specific Regional Mode DIVOM Procedure Guideline (Ref. 10)	Westinghouse Methodology
4	Channel grouping	Optionally, group channels based on established criteria (e.g., channel power). The least stable channel should be considered. The first harmonic flux distribution needs to be computed for regional mode oscillations.	Westinghouse models the entire core such that each channel is represented. Different stability modes can therefore be evaluated directly. Therefore, there is no need to artificially group channels.
5	Cycle exposure[1]	Generate 3D simulation data for a minimum of three exposures, e.g., BOC, PHE, and EOC exposures, analyzed at NC at the highest licensed rod line. Nominal rod patterns are used for each exposure.	Consistent with the guideline, Westinghouse generates 3D simulation data for at least three exposure conditions, at the analytical NC conditions along the highest licensed rod line.
6	Power/flow conditions	Power/flow state points along the highest rod line (limited to MELLLA) beginning with NC, then NC+5%, NC+10%, etc., until oscillations fail to develop or the slope of DIVOM data decreases with increased flow.	Consistent with the guideline, Westinghouse simulates conditions along the highest licensed rod line beginning at the calculated NC condition and in 5% increasing flow increments until oscillations fail to develop or the slope of the DIVOM data decreases with increased flow.
7	Xenon condition	Use rated core power equilibrium xenon.	Consistent with the guideline, the Westinghouse methodology uses rated core power equilibrium xenon
8	Feedwater temp	Use off-rated equilibrium temperature (nominal feedwater heating).	Consistent with the guideline, the Westinghouse methodology uses off-rated equilibrium feedwater temperature.
9	Radial peaking factor of limiting channel.	Include consideration for changes in radial peaking from the design calculations. The goal is to reasonably represent expected variations in radial peaking factor as the result of normal operation.	Consistent with the guideline, the Westinghouse methodology increases the hot bundle power in order to cover expected cycle variations.
10	Transient simulation	Run the transient until the MCPR equals 1.00, until the oscillations are no longer increasing, or until sufficient information is obtained to generate a DIVOM.	Consistent with the guideline, the Westinghouse methodology runs the transient until the MCPR equals 1.00, until the oscillations are not longer increasing, or until sufficient information is obtained to generate a DIVOM.
11	DIVOM calculation	Compute (initial-minimum)/initial CPR as a function of (peak-minimum)/average oscillation magnitude. Connect data to generate piecewise linear curve.	Consistent with the guideline, the Westinghouse methodology computes points of ΔCPR/initial CPR as a function of oscillation magnitude for a representative group of hot channels. A point on the DIVOM curve is established by the channel producing the highest ΔCPR/initial CPR and the channel producing the highest oscillation magnitude. The points are connected to form a piecewise linear curve.

[1] Beginning of Cycle (BOC), Peak Hot Excess (PHE), End of Cycle (EOC) and Natural Circulation (NC).

No.	Element	Plant Specific Regional Mode DIVOM Procedure Guideline (Ref. 10)	Westinghouse Methodology
12	DIVOM uncertainties	Not evaluated because best-estimate DIVOM of reasonably limiting conditions is used in conservative methodology (95/95 HCOM) to compute OPRM amplitude setpoint.	The Westinghouse methodology is consistent with guideline.

In the event the OPRM is out of service, a Backup Stability Protection (BSP) analysis is performed. The results of this analysis are the locations of the Exit and Scram region boundaries on the power-flow map. The BSP analysis is performed according to Westinghouse References 11, and 13, which incorporate the following steps:

[

]^{a,c}

The process is used in order to verify that the SCRAM REGION and the EXIT REGION boundaries are well chosen. The Westinghouse methodology is consistent with the BWROG guideline (Ref. 19).

The DNPS and QCNPS stability analysis process is described below for more clarification.

The DNPS and QCNPS reactors are using Long Term Stability Solution Option III. The generic approach is currently changed to a cycle specific approach (defined by BWROG Guideline of Reference 10). The previous fuel vendor has performed the first cycle-specific implementation of Option III e.g. for QCNPS Unit2 (QC2). Starting with QC2 Cycle 19 reload, Westinghouse will perform the Option III stability analysis to establish the stability based operating limit MCPR as a function of oscillation power range monitor (OPRM) amplitude setpoint. Also, Westinghouse will perform the Backup Stability Protection (BSP) analysis starting with the QC2 Cycle 19. Westinghouse has performed a BSP evaluation for a representative first transition reload of SVEA-96 Optima2 fuel in QC2 (called c19mock). The result verifies the existing exclusion zones with a reasonable margin. Also, Westinghouse has performed DIVOM calculation in support of Option III evaluation for C19mock [

]^{a,c}

There are no interactions between Westinghouse and the previous fuel vendor on evaluation of stability for QC2.

b. Westinghouse provides cycle-specific information to support the OPRM setpoint and power-flow map exclusion boundaries to support continued operation should the OPRM be out of service.

The process followed to calculate the cycle-specific DIVOM curve is described in the response to item 'a' above. The OPRM setpoint is established or confirmed to ensure that oscillations initiated following a two-pump trip or steady-state operation at [ ]^{a,c}. In the first scenario, the initial MCPR is the MCPR that exists after the coast down to natural circulation and after the feedwater temperature reaches equilibrium. It is assumed that the reactor was operating at the MCPR operating limit prior to the two recirculation pump trip. In the second scenario, the plant is assumed to be in steady-state operation at [ ]^{a,c}. It is assumed that the reactor is operating at the MCPR operating limit corresponding to the specified power and flow conditions.

The process followed to determine the power-flow map exclusion boundaries, in the event the OPRM is out of service, is described in the response to item 'a' above.

For DNPS and QCNPS, Westinghouse will produce constant decay ratio lines and OPRM trip setpoint versus confirmation count setpoint in support of Option III stability analysis. This analysis is being verified and is scheduled to be completed and available for the NRC review by February 15, 2006.

c. The major codes used by Westinghouse and their uses for licensing applications involving stability are shown in the following table:

Table 20-2 Major Westinghouse BWR Stability Analysis Codes

Code	Description	Topical Report and NRC SER
PHOENIX4	This 2D lattice code is used to produce nuclear cross section dependencies that are used by RAMONA3 during transient conditions.	Refs 15 and 16
POLCA7	This is the steady-state 3D simulator code used to produce 3D burnup and Xenon distributions that are used by RAMONA3.	Refs 15 and 16
RAMONA3	This code is a time-domain 3D transient code. For backup stability protection calculations, the code is used to determine the limiting exposure point during the cycle with regard to decay ratio. At the limiting exposure point, the code is then used to determine exclusion boundaries on the power-flow map. For DIVOM calculation, this code is used to generate regional power oscillations, determine the transient oscillation magnitude of these oscillations and to provide boundary conditions for BISON hot channel calculation.	Refs. 11, 12, 13 and 14
BISON	For the DIVOM calculation, this code is used to calculate the CPR variations during the regional power-flow oscillations in selected assemblies.	Refs. 17 and 18

The configuration control of the codes is made according to Westinghouse standard where code release notes are the main tool.

The Westinghouse stability methods have originally been licensed as described by CENPD-294-P-A and CENPD-295-P-A as well as CENPD-300-P-A Topical Reports. The CPR application (BISON) has been licensed as described in CENPD-292-P-A. The new application codes (PHOENIX4 and POLCA) initiated a re-evaluation of the stability validation. This re-evaluation contains the previously used jet pump specific stability measurements [ ]^{a,c} as well as new measurements in [ ]^{a,c} (cycles 13 and 19). Here, QC2 Cycle 19 is a first reload of SVEA-96 Optima2 fuel for a plant licensed under EPU/MELLLA. This new validation confirms the ability of the RAMONA code to predict stability for part-length fuel at increased operating domains.

d. The licensing bases for the methodologies used by Westinghouse are presented in the table shown in the response to question c above. As shown in the table, all of the methodologies that are used in the stability calculations have been reviewed and approved by NRC.

e. QCNPS Unit 2 (QC2) has armed the plant Oscillation Power Range Monitor (OPRM) and is currently using long term stability solution Option 3 as the primary protection against damaging oscillations. The plant-specific DIVOM calculation for the Quad Cities and Dresden units will be done as part of the cycle-specific reload analysis. EGC and Westinghouse are performing the regional mode DIVOM analysis required to confirm that the OPRM setpoints provide protection of the plant MCPR safety limit for anticipated oscillations using the approved methodology established in NEDO-32465-A, Reference 20. The cycle-specific confirmation is being performed in accordance with the procedure guideline documented in OG04-01530-260, Reference 10, which was developed jointly by the BWROG Detect and Suppress Methodology Committee, GNF, AREVA, and Westinghouse. For each reload Westinghouse will determine the OPRM trip setpoint versus maximum confirmation count setpoint in support of Option III stability analysis.

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 June 15, 2005.

2. WCAP-15942-P, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors Supplement 1 to CENP-287," October 2004.

3. LTR-NRC-05-35, "Transmittal Letter to NRC of Responses to NRC Request for Additional Information on WCAP-15942-P Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors Supplement 1 to CENP-287."

4. CENPD-288-P-A, "ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel," July 1996.

5. CENPD-300-P-A, "Reference Safety Report for Boiling Water Reactor Reload Fuel," July, 1996.

6. WCAP-16081-P-A, "10x10 SVEA Fuel Critical Power Experiments and CPR Correlations: SVEA-96 OPTIMA2," March, 2005.

7. NEDO-32961, "Revision 1, Safety Analysis Report for Quad Cities 1 & 2 Extended Power Uprate," August 2001.

8. NUREG-0800, U.S. NRC Standard Review Plan, Section 4.2 Appendix A, June, 1987.

9. ASME Boiler and Pressure vessel Code, Section II, Part D, Appendix 2, 1992 Edition.

10. OG04-01530-260, "Plant-Specific Regional Mode DIVOM Procedure Guideline," June 15, 2004.

11. CENPD-294-P-A, "Thermal-Hydraulic Stability Methods for Boiling Water Reactors," July 1996.

12. "Acceptance for Referencing of ABB/CE Topical Report CENPD-294-P: Thermal Hydraulic Stability Methods for Boiling Water Reactors (TAC No. M92883)," February 22, 1996.

13. CENPD-295-P-A, "Thermal-Hydraulic Stability Methodology for Boiling Water Reactors," July 1996.

14. "Acceptance for Referencing of ABB/CE Topical Report CENPD-295-P: Thermal Hydraulic Stability Methodology for Boiling Water Reactors (TAC No. M93648)," February 22, 1996.

15. CENPD-390-P-A, "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors," December 2000.

16. “Acceptance for Referencing of CENPD-390-P, The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors' (TAC No. MA5659),” July 24, 2000.

17. ENPD-292-P-A, “BISON - One Dimensional Dynamic Analysis Code for Boiling Water Reactors: supplement 1 to Code Description and Qualification,” July 1996.

18. CENPD-292-P: “BISON – One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification,” (TAC No. M90165),” October 16, 1995.

19. OG 02-0119-260, “BWR Owner's Group Guidelines for Stability Interim Corrective Action.”

20. NEDO-32465-A, “Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications,” August 1996.