ATTACHMENT 8

Westinghouse Application for Withholding, Affidavit, and Non-Proprietary Versions of Attachments 6 and 7

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Our ref: CAW-05-2005

June 15, 2005

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: NF-BEX-05-64 P-Attachment,
"Westinghouse Input to Request for Licensing Amendment Regarding Transition to Westinghouse Fuel"

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-05-2005 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-05-2005 and should be addressed to J. A. Gresham, Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

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

Enclosures

cc: D. G. Holland/NRR
L. W. Rossbach/NRR
B. J. Benney/NRR
L. M. Feizollahi /NRR (only affidavit)

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared J. S. Galembush, 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. S. Galembush, Supervisory Engineer
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 15th day
of June, 2005

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 the Supervisory Engineer, 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-05-64 P-Attachment Westinghouse Input to Request for Licensing Amendment Regarding Transition to Westinghouse Fuel" (Proprietary), dated June 15, 2005, for approval of the Optima2 License Amendment, being transmitted by the Exelon Nuclear letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted for use by Westinghouse for Exelon Nuclear is expected to be applicable for other licensee submittals in response to certain NRC requirements for justification of Revised Optima2 Amendment Requests.

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

(a) Support Exelon Nuclear in obtaining a license amendment for Optima2 fuel.

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to sell the use of similar information to its customers for purposes of obtaining license amendments for Optima2 fuel.

(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 license amendments 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).

NF-BEX-05-64 NP-Attachment

Westinghouse Input to Request for Licensing Amendment Regarding Transition to Westinghouse Fuel

June 15, 2005

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

Attachment 6
Applicability of Westinghouse Fuel and Analytical Methods

1.0 Introduction

Attachment 6 provides details justifying the first application of Westinghouse fuel and analytical methods to the extended power uprate (EPU) conditions existing at Dresden Nuclear Power Station (DNPS) and Quad Cities Nuclear Power Station (QCNPS). In particular, Attachment 6 supports EGC's request by demonstrating the following:

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2.0 Analytical Methods Applied Within NRC-Approved Applicability Ranges

This section identifies the NRC approved applicability ranges associated with the steady-state and transient neutronic and thermal hydraulic analytical methods and code systems used to perform the safety analyses. Plant specific information will be provided to demonstrate that the analytical methods and code systems are applicable to DNPS and QCNPS EPU conditions.

Prior to providing the list of safety analyses and associated applicability ranges, it is worth noting that the analytical methods and code systems have been previously used for applications with average assembly powers considerably higher than DNPS and QCNPS. Table 1 provides a comparison of the power levels among a sample of boiling water reactors (BWRs) for which Westinghouse has previously been or is currently the fuel vendor. As can be seen from the table, the core thermal power, assembly average power and rod average power conditions at DNPS and QCNPS after the extended power uprate are lower than those at other plants.

Table 1 – Comparison among BWRs with Fuel Supplied by Westinghouse* a,c

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Applicability of Westinghouse Fuel and Analytical Methods

The set of safety analyses performed during a reload licensing campaign is noted in Table 2. The table also includes the steady-state and transient neutronic and thermal hydraulic analytical methods and code systems used to perform each of the safety analyses, as well as the governing topical reports. As noted in the table, nuclear analyses are performed with the neutronic codes PHOENIX4 and POLCA7. Mechanical analyses are performed with the STAV, VIK, and COLLAPS codes. Transient analyses are performed with the PHOENIX4, POLCA7, BISON, GOBLIN, CHACHA and RAMONA3 codes.

Table 2 – Safety Analyses Analytical Methods and Code Systems

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Table 3 lists the applicability ranges for the code systems noted in Table 2. As indicated in the DNPS/QCNPS value column, the values fall within the applicability ranges, except as noted.

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Applicability of Westinghouse Fuel and Analytical Methods

Table 3 – Analytical Methods and Code Systems Applicability Ranges

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3.0 Uncertainties Applied to the Thermal Limits Analyses Are Valid

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Table 4 – Thermal Limits Analyses Uncertainties

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4.0 Assessment Database and Uncertainty of Models Remain Valid

This section identifies the assessment database and the assessed uncertainty of models used in all licensing codes that interface with and/or are used to simulate the plant's response under steady state, transient or accident conditions. Plant specific information is provided to demonstrate that the assessment database and the assessed uncertainty of models are applicable to DNPS and QCNPS EPU conditions.

Important parameters are identified to compare between the assessment database and DNPS/QCNPS conditions. The assessment database includes examples presented in the respective topical reports, as well as conditions in previous U. S. applications [
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A Westinghouse assessment database in the form of key parameters for plants in which Westinghouse fuel has been utilized is provided in WCAP-15942-P. The data shown in

Attachment 6
Applicability of Westinghouse Fuel and Analytical Methods

WCAP-15942-P demonstrates the DNPS/QCNPS application is within the Westinghouse experience base.

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Applicability of Westinghouse Fuel and Analytical Methods

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Table P-1: Pin Power and Assembly Power Uncertainty Measurements

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Figure P-1: Total Fission Rate and U^{238} Capture Rate RMS Errors from PROTEUS Phase 1 Measurements

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Figure P-2: TIP Measurement Results in Successive KKL Cycles as Part-Length Rod Fuel is Introduced

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Figure P-3: Peak Relative Pin Power Comparison Between HELIOS and PHOENIX as a Function of Lattice Burnup

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Figure P-4: Difference Between HELIOS and PHOENIX Relative Pin Power Predictions for a SVEA-96 Optima2 Lattice at BOL

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ATTACHMENT 7
Evaluation of Licensing Basis Events

1.0 Introduction

CENPD-300-P-A provides an overview of the Westinghouse boiling water reactor (BWR) reload fuel methodology. CENPD-300-P-A references other topical reports, which have been reviewed and approved by the NRC, that provide additional details on the methodology. In addition, there are other topical reports that have been reviewed and approved by NRC that were issued subsequent to the approval of CENPD-300-P-A. The entire set of topical reports and the limitations and conditions placed on them by the NRC comprise the Westinghouse BWR reload methodology.

For each application, Westinghouse reviews the plant's licensing basis. An engineering evaluation is performed for each licensing basis event to determine whether the event will be reanalyzed to support the introduction of Westinghouse fuel. While CENPD-300-P-A identifies certain events that will be evaluated for the introduction of Westinghouse fuel, others may also be identified as a result of this review. Table 1 provides a summary of the events that have been reviewed for the introduction of SVEA-96 Optima2 fuel at Quad Cities Nuclear Power Station (QCNPS). This table identifies each licensing basis event, cross-referenced to the QCNPS Updated Final Safety Analysis Report (UFSAR) section that describes the event. A similar evaluation will be performed to support the introduction of SVEA-96 Optima2 fuel at Dresden Nuclear Power Station (DNPS). Table 1 also indicates whether the event falls into one of the following categories:

- **Category 1** – Event affected by the introduction of the SVEA-96 Optima2 fuel design and is potentially limiting for each cycle. Requires that the event be reanalyzed for each reload.

- **Category 2** – Event affected by the introduction of the SVEA-96 Optima2 fuel design, but not for different cycles using that design. Requires that the event be reanalyzed for the introduction of the new fuel design, but not for subsequent reloads.

- **Category 3** – Event potentially affected by the introduction of the SVEA-96 Optima2 fuel design, but is bounded by a more limiting event of the same frequency category. No analyses will be performed for events in this category.

- **Category 4** – Event not affected by the introduction of the SVEA-96 Optima2 fuel design. No analyses will be performed for events in this category.

Table 1 identifies the licensing topical reports that provide detailed information regarding Westinghouse's methodology for analyzing each applicable event. Topical reports that are referenced by CENPD-300-P-A are incorporated by reference. The analysis for each event will be performed using NRC-approved methodologies, within the ranges upon which NRC approval was granted as described in Attachment 6.

Table 1 provides a cross reference to the limitations and conditions placed upon the Westinghouse methodology by the NRC including the limitations and conditions placed on all topical reports. The individual limitations and conditions for each topical report are listed in Table 3 through Table 21.

Exelon Generation Company, LLC (EGC) recognizes that the NRC's June 25, 2003, letter advises that licensees must submit plant-specific analysis results for the NRC to review. EGC is providing the detailed criteria described herein as an alternative to the results of plant-specific analysis. These criteria directly address the NRC's concerns regarding the approach EGC and Westinghouse will use to constrain both Westinghouse fuel and other co-resident fuel types to within NRC-approved limits on design, considering plant and fuel type specific uncertainties, methods, and the range of applicability of these methods. This alternative approach provides better clarity on constraints that must be applied to future fuel cycles following the initial introduction of SVEA-96 Optima2 fuel, and thus provides a solid regulatory envelope within which the SVEA-96 Optima2 fuel will be utilized by EGC such that the NRC can approve the proposed change. In addition, the task reports for analyses performed for the introduction of SVEA-96 Optima2 as well as the analyses performed each cycle are available for NRC inspection.

2.0 Evaluation of Licensing Basis Events

As indicated above, Westinghouse performs an evaluation of all licensing basis events to determine the events that require re-analysis due to the introduction of a new fuel design. In this case, analyses described in the UFSAR sections listed in Table 1 were evaluated. As a result of this evaluation, each analysis falls into one of the four categories defined above. This section provides example licensing basis events that are illustrative of these four category types and are selected to illustrate how mixed cores are treated and the rationale behind the analysis categorization. The ATWS example not only illustrates these points, but also outlines a proposed analysis methodology.

2.1 Inadvertent Closure of the Main Steam Line Isolation Valves (UFSAR Section 15.2.4)

The inadvertent closure of the main steam isolation valves (MSIVs) is an example of an event that is bounded by another event of the same frequency category (i.e., Category 3).

Inadvertent MSIV closure may occur by a number of nuclear system malfunctions or operator actions. These include spurious signals resulting in a low steam line pressure, high steam line flow, low reactor water level, or low condenser vacuum. Intended or inadvertent manual action can also cause one or more MSIVs to close.

Closure of all MSIVs is the most limiting MSIV closure event as it causes the largest increase in reactor vessel pressure and the largest resulting increase in core power due to void collapse. The reactor protection system initiates a scram on MSIV position when the valves reach 10% closed. Alternately, depending on the timing of the power increase, the scram can be triggered by high flux or high reactor pressure. When the pressure increases to the setpoint of the first group of safety/relief valves (SRVs), the SRVs will open and close to control reactor vessel pressure.

The Westinghouse BWR reload methodology categorizes this event as an anticipated operational occurrence (AOO), which is an event of moderate frequency.

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2.2 Steam System Line Break Outside Containment (UFSAR Section 15.6.4)

The steam system line break outside containment event is an example of an event considered unaffected by the introduction of a new fuel design (i.e., Category 4). This event may occur if the main steam line breaks downstream of the outboard isolation valve. The event is analyzed as a postulated event without cause. In addition, since the pipe break is located outside of containment, it can result in the release of radioactivity directly to the environment.

The general plant response to a main steam line break is the rapid depressurization of the reactor and the closure of the MSIVs due to high steam flow. The loss of inventory to the environment is terminated by the closure of the MSIVs and the reactor is shutdown by the reactor trip on MSIV position. Following the closure of the MSIVs, the reactor vessel pressure increases until it is controlled by the SRVs opening and closing as they discharge steam to the suppression pool.

Generic sensitivity studies have demonstrated that there are no significant changes to the core thermal hydraulic conditions due to introduction of a new fuel or core design that would affect the overall system response to this accident. Furthermore, the total inventory discharged to the environment before the break is isolated is independent of reload core or fuel design and the core coolant activity is limited by the plant's Technical Specifications (TS), which are not changed as a result of the fuel transition. Therefore, this event is unaffected by the introduction of the new fuel design.

2.3 Emergency Core Cooling System Performance Evaluation (UFSAR Section 6.3.3)

The Emergency Core Cooling System (ECCS) performance evaluation is an example of a potentially limiting event that is affected by the introduction of a new fuel design. Parts of the evaluation are performed only on the introduction of a new fuel design (i.e., Category 2) and parts of the evaluation are performed each time new lattice designs are introduced (i.e., Category 1). The loss-of-coolant accident (LOCA) is postulated as an event without cause. The LOCA analysis considers the full spectrum of line breaks, up to and including the double-ended guillotine break of the largest recirculation line. The ECCS performance evaluation demonstrates compliance with the 10 CFR 50.46 acceptance criteria by calculating peak cladding temperature, maximum cladding oxidation, and maximum hydrogen generation. The analysis also demonstrates that the geometry remains coolable and that the decay heat can be removed over an extended period of time. In addition to demonstrating compliance with the regulation, this analysis generates the burnup-dependent maximum average planar linear heat generation rate

(MAPLHGR) limits, which ensure that the regulatory criteria continue to be met for each new lattice design. Since the analysis results are dependent on fuel design, the Westinghouse BWR reload methodology requires an evaluation of the MAPLHGR limits for each new lattice design being introduced in the core. This usually results in new exposure dependent MAPLHGR limits being provided for each reload cycle.

In order to determine the limiting conditions for the calculation of MAPLHGR limits, the analyses provided in the UFSAR are reviewed. A break spectrum / single failure study, which is similar to the studies shown in the UFSAR by the other fuel suppliers, is performed using the Westinghouse ECCS evaluation model to determine the limiting break size and limiting single failure. Consistent with the current licensing basis, TS pump flows will be used as inputs to the LOCA analysis, without adjustment for uncertainty. A plant-specific system model for the limiting break size/single failure and core configuration (i.e., transition core or equilibrium core), is used to determine the limiting peak cladding temperature and local oxidation for the Westinghouse fuel design. The confirmed limiting break case is used to determine the peak cladding temperature, maximum local oxidation and bounding core-wide oxidation for the unit. A summary of this analysis of record for the SVEA-96 Optima2 fuel design will be added to the UFSAR to provide the benchmark for the reporting of future changes or discovered errors in the evaluation model as required by 10 CFR 50.46. This result is also used in the calculation of the burnup-dependent MAPLHGR limits to support operation with SVEA-96 Optima2 fuel. The MAPLHGR limits are selected to ensure that these bounding values are not exceeded.

The scope of the ECCS performance analysis is summarized in Table 1 along with a cross reference to the applicable topical reports, NRC-imposed limitations and conditions stated in the applicable safety evaluations, and Westinghouse's method of resolution.

The analysis of the ECCS performance provides an example of how mixed cores will be treated for this event. The GOBLIN system models will be developed for a transition core using SVEA-96 Optima2 and GE14 fuel designs and a SVEA-96 Optima2 equilibrium core to establish the limiting core configuration that must be used for establishing bounding SVEA-96 Optima2 MAPLHGR limits. A GOBLIN system model will also be developed for an equilibrium GE14 core to assess the potential impact of the transition on the MAPLHGR limits for the GE14 fuel. Since the GNF-established MAPLHGR limits will be used for the legacy GE14 fuel, EGC will contact GNF regarding any potential impact resulting from the transition.

The same LOCA model that is used in the ECCS performance analysis is used to support the seismic / LOCA loads analysis. A description of the Westinghouse methodology to evaluate the Westinghouse fuel assembly when subjected to postulated seismic and LOCA events is provided in CENPD-288-P-A. As indicated in Table 19, there are no SER limitations or conditions placed on this methodology.

2.4 Thermal-Hydraulic Stability (UFSAR Section 4.3.4)

General Design Criterion 12 requires that the reactor core be designed to assure that power oscillations that can result in conditions exceeding specified acceptable fuel design limits are not possible, or can be reliably detected and suppressed. As a result of generic concerns following several stability events at operating reactors, DNPS and QCNPS installed the Oscillation Power Range Monitor (OPRM) that will automatically detect and suppress power instabilities. This design modification is in accordance with Option III of the BWR Owners' Group (BWROG) recommendations provided in NEDO-31960-A [BWR Owners' Group Long-Term Stability Solutions Licensing Methodology]. The OPRM uses the LPRM signals to detect core instabilities using a period-based algorithm. The OPRM also uses amplitude and growth rate algorithms, which are implemented for defense-in-depth, but are not relied upon for detecting instabilities. If an unacceptable oscillation is detected by any of these algorithms, a trip signal will be generated by the OPRM. The methodology used to establish the bounding DIVOM curve, which is used to establish the OPRM setpoint, is similar to that described in NEDO-32465-A [Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications], except that a plant-specific DIVOM curve will be used instead of a generic DIVOM curve. The plant-specific DIVOM curve will be derived in accordance with the generic guidance established by the BWROG [Plant-Specific Regional Mode DIVOM Procedure Guideline, GE-NE-0000-0028-9714-R0].

The Westinghouse methodology for developing the plant-specific DIVOM curve makes use of the RAMONA3 computer code to generate the channel power and flow oscillations during a regional instability. These transient nuclear and thermal-hydraulic conditions are then used as boundary conditions for a hot-channel calculation that utilizes the BISON/SLAVE code to determine the ratio of delta-CPR to initial CPR. The combined results, RAMONA3 for power oscillation magnitude and BISON/SLAVE for delta-CPR over initial CPR, are used to generate plant-specific DIVOM curves for a range of cycle exposures and initial conditions. The limiting DIVOM curve is determined from these results.

Since Westinghouse uses a full-core RAMONA3 model for this analysis, the hydraulic and neutronic designs of each fuel design in the core are accounted for explicitly each reload (i.e., Category 1). [

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In the event the OPRM is out of service, analyses will be performed each reload to establish or confirm existing exclusion boundaries on the power/flow map. These analyses are performed consistent with procedures described in CENPD-294-P-A and CENPD-295-P-A.

2.5 ATWS (UFSAR Section 15.8)

This section provides a summary of the ATWS evaluation methodology for the SVEA-96 Optima2 fuel transition at DNPS and QCNPS. ATWS will be evaluated each reload to ensure compliance with the peak reactor vessel pressure acceptance criterion. For the initial transition, a SVEA-96 Optima2 equilibrium core will be evaluated for the ATWS long-term plant response.

2.5.1 Acceptance Criteria

The acceptance criteria for the ATWS analysis are summarized in Table 2. Stability criteria are not part of the DNPS and QCNPS ATWS licensing basis, and are therefore not evaluated.

2.5.2 Methodology

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2.5.3 ATWS Operator Actions

Operator actions will be credited for the long-term portion (i.e., after Standby Liquid Control (SLC) injection) of the limiting ATWS scenario. The long-term portion of the ATWS event addresses primarily the challenges to the containment limits. The following operator actions credited for the SVEA-96 Optima2 fuel transition are consistent with the current licensing basis ATWS analysis. The time delay from the symptom to initiating operator action credited for the SVEA-96 Optima2 fuel transition will be equal to or conservatively longer than the current licensing basis long-term ATWS analysis. The specific steps credited for the SVEA-96 Optima2 fuel transition are contained within current station procedures for addressing ATWS scenarios.

- After the ATWS signal trips the recirculation pumps, operators will manually terminate all reactor pressure vessel injection, except boron injection (if initiated) and Control Rod Drive (CRD) flow to reduce water level. These actions result in a higher void fraction in the core, which directly reduces reactor power. The operator will control the water level in the core between the top of active fuel and the minimum steam cooling water level using the preferred ATWS systems (i.e., Feedwater, CRD, High Pressure Coolant Injection (HPCI)).
- After the suppression pool temperature has reached the criteria in the emergency procedures, operators will manually initiate SLC.

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2.5.4 ATWS Containment Model

The BWR containment is modeled using the GOTHIC code. Lumped parameter control volumes are used to represent the containment drywell, vent pipes, ring header, downcomer pipes, and torus. Thermal conductors are used to model the metal and concrete heat sinks.

The ATWS mass and energy releases are calculated externally with BISON and supplied as forcing function input tables to the containment model.

2.5.5 SLC System Modeling

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2.5.6 Boron Mixing Model in BISON

The transition to SVEA-96 Optima2 fuel does not affect the actual mixing of boron in the reactor and recirculation system. The current licensing basis assumes a "perfect" mixing of boron in the reactor and recirculation system, and

75% effectiveness factor. The SVEA-96 Optima2 model retains these mixing assumptions, and incorporates them into BISON.

The "perfect" mixing assumption is a conservative mixing model. In the real reactor, the boron enters the lower plenum, and passes directly into the active fuel zone. From the active fuel zone, the boron flows into the upper plenum and then into other volumes of the reactor and recirculation system. Boron injection occurs after the recirculation pumps have tripped. The "perfect" mixing model predicts a conservative (i.e., lower) active core boron concentration because it will "divert" boron to all connected volumes of water in the reactor and recirculation system, reducing the boron that goes to the active core region.

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2.5.7 Reactivity Calculation for Boron in BISON

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2.5.8 Sensitivity Cases

Westinghouse will evaluate several base cases and sensitivity cases to ensure the reasonableness and conservatism of the containment response model.

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2.5.9 Justification for Slip and Void Correlation If Pressure Exceeds 1450 psia

NRC SER Condition 4 in topical report RPA 90-90-P-A (Table 9) requires justification if the slip and void correlation models will be used above 1450 psia (i.e., 10 MPa).

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To demonstrate the applicability of the AA78 slip correlation in combination with the EPRI boiling model at pressures above 1450 psia, an extension of the comparison of these models is made below. In this new verification, the comparison is extended to higher pressures and higher steam qualities than originally used in the topical report.

The EPRI slip/boiling correlation has been verified for a wide range of pressures. It was developed to fit not only the rod bundle data which forms the basis of the AA78 correlation, but also other data including measurements in rectangular tubes to above [

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Figure 1 shows the total change of void fraction with increasing pressure starting at 7 MPa (i.e., 1015 psia) as calculated using the EPRI slip and boiling model, at constant steam quality, for typical BWR channel conditions. [

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Comparison of Figure 1 with the corresponding curves calculated with the AA78 void correlation (i.e., Figure 2) indicates that the change of void fraction with pressure in the range [

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Thus the application of the AA78 void correlation is considered verified for pressures up to [ ]^{a,c}. As can be seen from the figures below, the AA78 correlation does not have any discontinuity or threshold effect above [ ]^{a,c}.

2.5.10 References

1. S. Helmersson, Void Correlation AA78, AA PM RCA 79-87 (unpublished). AA PM RCA 81-90, 1981

2. G. S. Lellouche, B. A. Zolotar, A Mechanistic Model for Predicting Two-Phase Void Fraction for Water in Vertical Tubes, Channels, and Rod Bundles, Electric Power Research Institute, EPRI NP-2246-SR, 1982

3. CENPD-292-P-A BISON – One Dimensional Dynamic Analysis Code for boiling Water Reactors: Supplement 1 to Code Description and Qualification

4. Martin, R., “Mesure de Tauxde Vide a Haute Pression dans un Canal Chauffant,” University of Grenoble, 1967

Figure 1: EPRI Void Correlation, Differential Void vs. Steam Quality

a,c

Figure 2: AA78 Void Correlation, Differential Void vs. Steam Quality

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Table 1 Licensing Basis Events Evaluated for Introduction of a New Fuel Design

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Table 2 Acceptance Criteria

Event or Event Classification	Westinghouse Reload Methodology Acceptance Criteria (these criteria may be supplemented by additional plant-specific limitations)
Moderate Frequency Events (M)	Radioactive Effluents ≤ 10 CFR 20 Limits Specified Acceptable Fuel Design Limits Satisfied: ✓MCPR ≥ SLMCPR ✓LHGR ≤ Overpower Limit (< 1 % plastic strain, no fuel melting) ✓Average Fuel Pellet Enthalpy ≤ 170 cal/gm Peak Reactor Vessel Pressure ≤ 110 % Design Suppression Pool ≤ Heat Capacity Temperature Limit
Infrequent Events (I)	Moderate frequency event acceptance criteria are applied
Limiting Faults (L)	Radiological Consequences (LOCA, Pipe Breaks Outside of Primary Containment, Fuel Handling Accident, Recirculation Pump Shaft Break / Seizure Accidents): Offsite Dose ≤ 10 CFR 100 Limit Control Room Dose ≤ GDC-19 Limit Barrier Performance Control Rod Drop Accident ✓Failure Threshold Enthalpy ≤ 170 cal/gm ✓Peak Fuel Enthalpy ≤ 280 cal/gm Misplaced Bundle Accident[1] ✓MCPR ≥ SLMCPR
Core Thermal Hydraulic Stability (STAB)	Detect and Suppress (Option III) ✓MCPR ≥ SLMCPR Interim Corrective Action (Backup to Option III) ✓Exit Region (1.0 < decay ratio ≤0.8) ✓Manually Scram Reactor (decay ratio ≥1.0)
Over Pressure Protection (OP)	Reactor Vessel Pressure ≤ 110% Design
Containment Functional Design and Heat Removal (CONT)	Drywell Pressure ≤ 62 psig Drywell Shell Temperature ≤ 281 °F (sustained) Suppression Chamber Pressure ≤ 62 psig Suppression Pool Temperature ≤ 202 °F
Combustible Gas Control (CGC)	Using the hydrogen generated as a result of the metal water reaction equal to the maximum of (a) five times the total amount calculated demonstrating compliance with §50.46(b)(3), or (b) the amount resulting from oxidation of the cladding surrounding the fuel to a depth of 0.00023 inch, show that flammable conditions do not occur [O_2 concentration ≤ 5% and H_2 concentration ≤ 6%] within the first 30 days after an accident without venting or exceeding one half of the containment design pressure.

[1] The misplaced bundle accident, which was classified as an event with a lower frequency of occurrence than moderate frequency in the UFSAR, is categorized as an accident in Westinghouse's methodology. However, the acceptance criterion that is applied protects the SLMCPR, which is consistent with the criteria applied to Moderate Frequency Incidents. The term "Misplaced" includes both an assembly loaded in an inappropriate core location and an assembly in the correct location, which is rotated by 90° or 180°.

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Event or Event Classification	**Westinghouse Reload Methodology Acceptance Criteria** (these criteria may be supplemented by additional plant-specific limitations)
ECCS Performance	- Peak cladding temperature ≤ 2200 °F - Maximum cladding oxidation ≤ 0.17 times the total cladding thickness before oxidation - Maximum hydrogen generation ≤ 0.1 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume were to react - Calculated changes in core geometry shall be such that the core remains amenable to cooling - After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core
ATWS	- Peak cladding temperature ≤ 2200 °F - Peak containment pressure < containment design pressure - Peak reactor vessel pressure ≤ 120% reactor vessel design pressure (ASME Limit C) - Offsite dose ≤ 10 CFR 100 acceptance limits - Demonstrated equipment availability
Reactor Shutdown Without Control Rods (C_1)	$K_{eff} < 1.0$
New Fuel Vault Criticality (C_2)	$K_{eff} < 0.90$ dry and $K_{eff} < 0.95$ flooded when the vault is fully loaded
Spent Fuel Pool Criticality (C_3)	$K_{eff} < 0.95$ with racks fully loaded
Shutdown Margin	≥ 0.38% $\Delta k/k$, with the highest worth control rod analytically determined; or ≥ 0.28% $\Delta k/k$, with the highest worth control rod determined by test

Table 3 CENPD-300-P-A Conditions and Limitations

CENPD-300-P-A – Reference Safety Report for Boiling Water Reactor Reload Fuel		
No.	**Condition / Limitation**	**Resolution**
1	Acceptability of this topical report is subject to review finding of the other relevant topical reports cited in the topical report, and all conditions set forth therein are applicable to this topical report. Furthermore, acceptability of reload analysis is subject to conditions cited in methodology topical reports.	Responses for each SER condition and limitation are provided below.
2	ABB/CE's [Westinghouse] uncertainty analysis approach is not generically acceptable since the acceptability is highly application dependent. The Operating Limit MCPR must be calculated with Method A.	The uncertainty analysis supporting the OLMCPR is performed using Method A.
3	The use of ANS79 decay heat curve is not acceptable for LOCA analysis. For compliance with Appendix K, ABB/CE [Westinghouse] must use 1.2 times the ANS71 as stated in the current 10 CFR 50, Appendix K.	Westinghouse continues to use 1.2 times the ANS71 decay heat standard in its Appendix K evaluation model.
4	No evaluation of validity of sample analyses was performed. Furthermore, the approval recommended in this report does not imply any endorsement of analyses nor of the quantified uncertainties set forth in Appendix D. Therefore, no reference should be made to Appendix D as demonstration in support of any future reload.	Appendix D was included only to illustrate the methodology by presenting sample applications and not as a reference for later reload designs. Specific results from Appendix D are not used to support plant-specific applications.
5	At the minimum, each reload safety evaluation report should contain all the items referred to in Appendix B of the topical report.	The items identified in Appendix B of CENPD-300-P-A are addressed in the reload safety evaluation report.
6	ABB/CE [Westinghouse] must use 110% of vessel design pressure for the peak reactor vessel pressure limit unless otherwise governed by a plant specific licensing basis.	110% of reactor vessel design pressure for the peak reactor pressure limit is used unless specified otherwise in the plant-specific licensing basis.
7	The ABB/CE [Westinghouse] methodology for determining the operating limit maximum (sic) critical power ratio (OLMCPR) for non-ABB/CE fuel as described in CENPD-300-P and additional submittals (References 1, 3, 4, 5 and 6) is acceptable only when each licensee application of the methodology identifies the value of the conservative adder to the OLMCPR. The correlation applied to the experimental data to determine the value of the adder must be shown to meet the 95/95 statistical criteria. In addition, the licensee's submittal must include the justification for the adder and reference the appropriate supporting documentation.	Each licensee application identifies the value of the conservative adder to the OLMCPR. The value of the adder meets the 95/95 statistical criterion and is based on the comparisons of predicted CPR (using a Westinghouse CPR correlation renormalized to match the CPR performance for non-Westinghouse fuel) to the reference data provided by the utility (using the vendor CPR correlation for the vendor fuel). The comparisons cover the applicable range of the CPR correlation and are documented in a Westinghouse calculation note. The EGC specific analysis for determining the value of the conservative adder to the OLMCPR is documented in task report, NF-BEX-05-10 Rev. 0, "Task Report for TSD DQW04-020, CPR Correlation for Design." A summary of the analysis and process is provided as footnote (a) to this Table.

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CENPD-300-P-A – Reference Safety Report for Boiling Water Reactor Reload Fuel		
No.	**Condition / Limitation**	**Resolution**
8	For the rotated fuel assembly analysis ABB/CE [Westinghouse] stated its intent to vary gap sizes to reduce conservatism in the analysis accompanied by uncertainty analyses to establish the impact. Since the acceptability of this approach depends upon the validity of the uncertainty analysis, which has not been validated this approach is not acceptable.	Constant gap sizes are used in the rotated fuel assembly analysis.

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

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Where

$$USAG14 = CPR\ correlation\ for\ GE14$$
$$D4.1.1 = CPR\ correlation\ for\ SVEA-96\ Optima2\ fuel$$
$$f = mass\ flux\left(Kg/m^2-s\right)$$
$$p = assembly\ exit\ pressure\ (bar)$$
$$h = assembly\ inlet\ enthalpy\ (J/gm)$$

Correction coefficients:

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2. [

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Example: Comparison of CPR_Exelon/CPR_Westinghouse for GE14 legacy fuel

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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

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Table 4 RPB 90-93-P-A and RPB 90-94-P-A Conditions and Limitations

RPB 90-93-P-A [Referenced by CENPD-300-P-A] – Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Description and Qualification RPB 90-94-P-A [Referenced by CENPD-300-P-A] – Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity		
No.	Condition / Limitation	Resolution
1	The staff concludes that the Westinghouse BWR ECCS EM provides an acceptable evaluation model of loss-of-coolant accidents for use in calculations of peak clad temperature (PCT) and hydrogen generation made in accordance with Appendix K licensing calculations for large-break and small-break LOCAs in boiling water reactor BWR/2 through BWR/6 plants. The basis for this position is the staff review of the licensing topical reports WCAP-11284 (Ref. 1) [RPB 90-93-P-A] and WCAP-11427 (Ref. 2) [RPB 90-94-P-A] and the evaluation summarized in this safety evaluation. This conclusion is subject to the conditions described in paragraphs 2 and 3 below.	The DNPS and QCNPS units are BWR/3 designs and therefore fall within the approved classes of BWRs. Conditions 2 and 3 are dispositioned below.
2	The staff concludes that the Westinghouse BWR ECCS EM has provisions and options to conform with the required modeling features of Appendix K. Conformance to plant-specific requirements of Appendix K (e.g., I.C.6, Pump Modeling) for use in licensing calculations must be specified in the license application reload safety analysis report. This report should include or reference a sensitivity study for the BWR type identified in the license application.	A LOCA report will be submitted to EGC, which can be included in the license application reload safety analysis report as appropriate. This report will include the results of plant-specific sensitivity studies, benchmarking (e.g., pump modeling), the conclusions of a break-spectrum analysis, and limiting peak cladding temperature.
3	Certain specific model areas of the Westinghouse BWR ECCS EM discussed in WCAP-11284 [RPB 90-94] are specific to a fuel design (QUAD+). These areas are the critical heat flux (CHF) and fuel design characteristics for the QUAD+ fuel assemblies. A staff-approved CHF correlation must be used when the subject ECCS methodology is used in a licensing analysis (Section 3.1.8). The experimental data used to verify the convective spray heat transfer coefficients should be justified as applicable to the particular fuel design for which the overall methodology is to be applied (Section 3.2.2). The use of a fuel design other than QUAD+ fuel in a transition core should also be addressed.	SVEA-96 Optima2 fuel will be inserted in the DNPS and QCNPS cores. The NRC-approved CPR correlation for this fuel design, which is described in WCAP-16081-P-A, is used in the analysis. Convective spray heat transfer coefficients approved by the NRC in WCAP-16078-P-A for application to cores containing SVEA-96 Optima2 fuel will be used for the DNPS and QCNPS analyses.

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Table 5 CENPD-293-P-A Conditions and Limitations

CENPD-293-P-A [Referenced by CENPD-300-P-A] – BWR ECCS Evaluation Model: Supplement 1 to Code Description and Qualification		
No.	**Condition / Limitation**	**Resolution**
1	A large number of models from the fuel performance code STAV6.2 are incorporated into the GOBLIN/DRAGON/CHACHA code series, especially CAHCHA.USA2. Their acceptability and compliance to Appendix K requirements were beyond the scope of this review and are subject to the review findings of CENPD-285-P.	When the STAV6.2 fuel performance features are used in CHACHA, their use is in accordance with the review findings of CENPD-285-P-A. See the resolution of SER #1 for WCAP-16078-P-A (Table 8).
2	The new strain and rupture model produced acceptable results against test data only when a bias was introduced. Therefore the use of this model is approved provided that the bias is always incorporated.	Consistent with the discussion in RAI #12, a bias of [ ]^{a,c} is included in the CHACHA input to comply with this condition.
3	The remainder of the ABB/CE's [Westinghouse] ECCS methodology remains subject to the SER conditions on the RPB 90-93-P-A and RPB 90-94-P-A.	The RPB 90-93-P-A and RPB 90-94-P-A SER conditions are dispositioned above (Table 4).
4	The use of the XL-S96 CPR correlation is subject to the SER conditions in UR 89-210-P-A and Reference 1 [RPB 90-93-P-A and RPB 90-94-P-A].	The XL-S96 CPR correlation is not applicable to the SVEA-96 Optima2 fuel design that is being inserted in the DNPS and QCNPS cores. Therefore, this condition is not applicable to the insertion of this fuel design. However, a similar condition/limitation (#4) is prescribed in the SER for WCAP-16081-P-A (Table 14), requiring the use of an approved CPR correlation for SVEA-96 Optima2 fuel.

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Table 6 CENPD-283-P-A Conditions and Limitations

CENPD-283-P-A [Referenced by CENPD-300-P-A] – Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel		
No.	**Condition / Limitation**	**Resolution**
1	The review did not include evaluation of the adequacy of the implementation of the XL-S96 CPR correlation, since the vendor chose to address this issue in report CENPD-293.	This limitation states that the SER for this report does not address the implementation of the XL-S96 CPR correlation. Since this correlation is not applicable to the fuel being inserted in the DNPS and QCNPS cores, the condition does not require a resolution. The adequacy of the applicable CPR correlation is addressed in the SER for WCAP-16078-P-A (Table 8).
2	This methodology documented in CENPD-283-P is approved for extension to the SVEA-96 fuel only. Previous approvals remain unchanged and this methodology cannot be extended to other fuel and plant designs without NRC approval.	The application of the SVEA-96 Optima2 fuel design, which is being inserted in the DNPS and QCNPS cores, is presented in the NRC-approved report WCAP-16078-P-A (Table 8). The SER for RPB 90-94-P-A approved the Westinghouse ECCS methodology for BWR/2 through BWR/6 plant designs.
3	Since the convective spray heat transfer coefficients selected for the SVEA-96 fuel design were selected by procedure to show conservatism, but not supported by experimental data, this procedure should not be extended to other fuels without experimental verification.	A design different than the SVEA-96 fuel design is being inserted in the DNPS and QCNPS cores. A justification for applying the SVEA-96 spray heat transfer coefficients to the SVEA-96 Optima2 fuel design, which is being inserted in the DNPS and QCNPS cores, is presented in the NRC-approved report WCAP-16078-P-A (Table 8).
4	Similarly, the coefficients in the CCFL option that were shown to be insensitive to these coefficients for the SVEA-96 fuel should not be extended to other fuels without being validated by experimental data.	A design different than the SVEA-96 fuel design is being inserted in the DNPS and QCNPS cores. However, the CCFL correlation was shown to be applicable also to the SVEA-96 Optima2 fuel design, which is being inserted in the DNPS and QCNPS cores. This justification is provided in the NRC-approved report WCAP-16078-P-A (Table 8).

CENPD-283-P-A [Referenced by CENPD-300-P-A] – Boiling Water Reactor Emergency Core Cooling System Evaluation Model: Code Sensitivity for SVEA-96 Fuel		
No.	**Condition / Limitation**	**Resolution**
5	It is expected that the insensitivity demonstrated by the selected CCFL and convective heat transfer coefficients to the predicted system parameters would be plant design independent; however, the vendor will be required to demonstrate the acceptability of both of these in any instance when the calculated PCT approaches the Appendix K limit. The vendor submitted PCT data by facsimile dated September 19, 1995, and September 27, 1995, that showed the CCFL correlation has a 100 °F sensitivity in the temperature range of the Appendix K limit (1800 °F to 2200 °F). The data showed that as the peak linear heat generation rate approaches 36 kw/m the PCT goes above the Appendix K limit. Therefore, the vendor must include a conservative bias. By a facsimile dated October 16, 1995, the vendor stated that when the PCT is greater than 2100°F, the CCFL correlation shall include a conservative bias that bounds the scatter in the data base. The bias introduced to the base CCFL correlation will be such that conservative bounding predictions are obtained from the data base of all fuel assembly components that were used to derive the basic CCFL correlation.	WCAP-16078-P-A (Table 8) presents a method for applying a conservative bias such that the scatter in the database is bounded. This method will be applied to the DNPS and QCNPS analyses. The SER for WCAP16078-P-A states in part "...The staff also found, given that the modified CCFL correlation acts in the same manner as the imposed restriction, that there is no justification for the continued use of a separate conservative bias when the calculated PCT approaches the 10 CFR 50.46 limit."
6	The overall acceptability of the vendor's ECCS methodology for BWR remains subject to the restrictions and limitations of all other governing SEs of relevant computer codes, models, and fuel designs and their previous approvals.	Other applicable ECCS methodology SER conditions are dispositioned in Table 4 and Table 5.

Table 7 WCAP-15682-P-A Conditions and Limitations

WCAP-15682-P-A – Westinghouse BWR ECCS Evaluation Model: Supplement 2 to Code Description, Qualification and Application		
No.	**Condition / Limitation**	**Resolution**
	No conditions or limitations were identified in the SER	NA

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Table 8 WCAP-16078-P-A Conditions and Limitations

WCAP-16078-P-A – Westinghouse BWR ECCS Evaluation Model: Supplement 3 to Code Description, Qualification and Application to SVEA-96 Optima2 Fuel		
No.	**Condition / Limitation**	**Resolution**
1	All of the STAV7.2 features cannot be used pending completion of the staff review and approval of the STAV7.2 code. The previously approved STAV 6.2 model in the CHACHA-3D code can continue to be used for LOCA analysis. Upon receipt of staff approval of STAV7.2, Westinghouse shall provide written notification that STAV7.2 models are now being used in their ECCS methods and shall submit a revision to WCAP-10678-P, if it is determined necessary by the NRC staff, to document any changes in the STAV7.2 models, methods, or implementation currently described in this TR.	The application of the STAV7.2 models will be applied only if the STAV7.2 topical report (WCAP-15836-P) is approved. When WCAP-15836-P is approved, Westinghouse will provide NRC written notification that the applicable STAV7.2 models have been incorporated in the CHACHA-3D code and will be used for licensing applications. Otherwise, the previously approved STAV6.2 models will be used. If it is deemed necessary, a revision will be made to WCAP-16078-P-A to incorporate changes to the description of the STAV7.2 models being implemented in the CHACHA-3D code.
2	A mixed core GOBLIN model shall be developed during the first reload analysis of a transition core to verify the validity of the full core Westinghouse fuel approach. If it is confirmed that the analysis with a full core of Westinghouse fuel is bounding, then the LOCA ECCS evaluation can be performed using the full core Westinghouse fuel approach. Otherwise, the mixed core model shall be used.	A mixed core GOBLIN model will be developed to be representative of the first transition core. This model will be used to confirm that the model simulating a full core of Westinghouse fuel is bounding. Otherwise, the mixed core model will be used.
3	The USA5 EM cannot be used to calculate the MAPLHGR limits for non-Westinghouse fuel for a mixed core analysis. If the transition core analysis indicates that the system performance of the mixed core is more limiting than the full core analysis of legacy fuel, Westinghouse will request the utility to contact the legacy fuel vendor for an evaluation of the impact of the mixed core on the MAPLHGR limits for their fuel.	The USA5 EM will not be used to calculate MAPLHGR limits for non-Westinghouse fuel. If the transition core analysis indicates that the system performance of the mixed core is more limiting than the full core of legacy fuel, Westinghouse will request EGC to contact the legacy fuel vendor for an evaluation of the impact of the mixed core on the MAPLHGR limits for their fuel.
4	The methodology cannot be used until the SVEA-96 Optima2 fuel CPR correlation is approved by the NRC.	The SVEA-96 Optima2 fuel CPR correlation was approved [WCAP-16081-P-A] (Table 14). The correlation has been installed in GOBLIN code [Version 3.12.4].
5	The overall acceptability of the Westinghouse BWR ECCS methodology remains subject to the restrictions and limitations of all other governing SEs of relevant computer codes, models, and fuel designs, as previously approved.	See responses to restrictions and limitations to RPB 90-93-P-A, RPB 90-94-P-A, CENPD-293-P-A, CENPD-283-P-A and WCAP-15682-P-A above. Note that the use of the new CCFL model described in WCAP-16078-P-A implements SER condition #5 for CENPD-283-P-A as described in Table 6.

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Table 9 RPA 90-90-P-A Conditions and Limitations

RPA 90-90-P-A [Referenced by CENPD-300-P-A] – BISON – A One Dimensional Dynamic Analysis Code for Boiling Water Reactors		
No.	**Condition / Limitation**	**Resolution**
1	We require justification of the core flow when an unbalance from either recirculation loop can cause a thermal-hydraulic gradient across the core.	This condition was resolved in CENPD-292-P-A by the introduction of a two loop model in BISON. The model was qualified by comparison with plant data and was considered acceptable by the NRC technical reviewer.
2	We require justification for use of the recirculation pump model when transients are in other than the first quadrants of the Karman-Knapp diagram.	The BISON code will not be used for any calculations in which the recirculation pump model is used outside the first quadrant of the Karman-Knapp diagram.
3	We require justification for use of the recirculation pump model when two-phase flow conditions are calculated.	The BISON code will not be used for any calculations in which two-phase flow conditions occur in the recirculation loop.
4	We require justification of the slip and void correlation used if the core pressure exceeds 1305 psia while the quality exceeds 40 percent or if the pressure exceeds 1450 psia.	The BISON code will not be used for any calculations in which the slip and void correlation are used when core pressure exceeds 1305 psia while the quality exceeds [ ]a,c
5	We require justification of the isentropic coefficient used during a depressurization event.	Only the BISON code standard equilibrium model is used for all reactor vessel depressurization events.
6	We require a staff-approved model when modeling is necessary to simulate control systems.	The control systems will not be explicitly modeled. The control systems will be assumed to operate ideally or simulated by supplying boundary conditions.

Table 10 CENPD-292-P-A Conditions and Limitations

CENPD-292-P-A [Referenced by CENPD-300-P-A] – BISON – One Dimensional Dynamic Analysis Code for Boiling Water Reactors: Supplement 1 to Code Description and Qualification		
No.	**Condition or Limitation**	**Resolution**
1	With use of the PARA steamline model, the user has flexibility of modeling valves and control system functions through the use of user supplied table and control systems. Modeling of these systems greatly affects the amount of conservatism in the transient outcome in certain event analysis. Therefore as required in the original SER for BISON, ABB/CE [Westinghouse] is required to provide justification for these user controlled items, which include valve performance, to assure conservatism in licensing applications.	Conservative values (e.g., from Technical Specifications) will be used for user-controlled items, such as valve performance, to assure conservatism in the application.
2	The modeling changes or upgrades did not affect the basic modeling of the recirculation pump model. Therefore, the SER condition regarding the pump model remains unchanged. Similarly, the other SER conditions with respect to the simulation of the control system and the use of the slip and void correlation in the range of applicability are not affected by this submittal, and therefore remain unchanged.	Conditions 2, 3, 4, 5, and 6 from RPA 90-90-P-A remain (Table 9).
3	The use of the EPRI void model for licensing analysis must be demonstrated to be used within its range of applicability for each such application.	Should the EPRI void model be used for licensing analysis, it will be demonstrated to be used within its range of applicability.
4	ABB [Westinghouse] stated that the turbine assembly model will not be used, thus this model was not qualified in this review. The use of the turbine assembly model is restricted until qualified.	The turbine assembly model will not be used.

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Table 11 CENPD-294-P-A Conditions and Limitations

CENPD-294-P-A [Referenced by CENPD-300-P-A] – Thermal Hydraulic Stability Methods for Boiling Water Reactors		
No.	**Condition / Limitation**	**Resolution**
1	The RAMONA-based stability methodology proposed by ABB/CE [Westinghouse] provides a reasonably accurate estimation of the stability of (1) the channel thermal-hydraulic mode, (2) the fundamental or core-wide coupled neutronics thermal-hydraulic mode, and (3) the out-of-phase or regional coupled neutronics thermal-hydraulics mode.	No response necessary.
2	Using the ABB/CE [Westinghouse] stability methodology, RAMONA decay ratio calculations are accurate to within ±0.2 in a decay ratio range from 0 to 1.1 for all three modes.	An acceptance criterion decay ratio of [ ]^{a,c} is used to establish exclusion region boundaries in the Interim Corrective Action (ICA) analyses.
3	The RAMONA-based option described in CENPD-294-P and CENPD-295-P is an acceptable methodology for best-estimate stability prediction of operating boiling water reactors.	No response necessary.
4	As with all stability codes, input preparation is the major source of error; therefore, to maintain the ±0.2 accuracy, any new calculations must use procedures similar to those used in the qualification report. To insure that input errors do not compromise the accuracy of the calculations, best estimate RAMONA calculations must follow the input-generating procedures described in CENPD-294-P and CENPD-295-P. The RAMONA input must satisfy the following minimum requirements:	The stability calculations follow an input generating procedure that is consistent with the procedures described in CENPD-294-P-A and CENPD-295-P-A as well as the following.
4(1)	Each thermal-hydraulic region in the core (i.e., channel) model must be divided in a minimum of 24 axial nodes.	The RAMONA model for the DNPS and QCNPS units contains 25 axial nodes to represent the active channel.
4(2)	The core model must be divided into a series of radial nodes (i.e., thermal-hydraulic regions or channels) in such a manner that (a) No single region can be associated with more than 20% of the total core power generation. This requirement guarantees a good description of the radial power shape, especially for the high power channels. (b) The core model must include a minimum of three regions for every bundle type that accounts for significant power generation. (c) The model must include a hot-channel for each significant bundle type with the actual conditions of the hot channel.	The RAMONA model for the DNPS and QCNPS units represents the full core (i.e., every assembly).
4(3)	Each of the thermal-hydraulic regions must have its own axial power shape to account for 3-D power distributions. For example, high power channels are likely to have bottom peaked shapes.	The RAMONA model treats each channel explicitly such that each channel has its own power and axial power distribution.

CENPD-294-P-A [Referenced by CENPD-300-P-A] – Thermal Hydraulic Stability Methods for Boiling Water Reactors		
No.	**Condition / Limitation**	**Resolution**
4(4)	For out-of-phase calculations, a full-core representation is recommended. The minimum configuration, however, is two basic "symmetry units" (e.g., in a core with quarter core symmetry, RAMONA must model at least half the core).	The RAMONA model for the regional (out-of-phase) calculation uses a full core representation.
4(5)	Care must be taken in the selection of the perturbation used to excite each instability mode. A review must be performed to confirm that the perturbation actually excites each mode of oscillation (e.g., a perturbation along a symmetry line will not excite an out-of-phase oscillation).	A single control rod or groups of control rods are used to excite the perturbation. The results are reviewed to confirm the resulting mode of oscillation and to ensure that the perturbation is not along a symmetry line.
5	In addition to best-estimate calculations, the RAMONA-based ABB/CE [Westinghouse] methodology represents an adequate methodology to estimate *Exclusion Region* boundaries to be used with the so-called *BWR Stability Long Term Solutions*. Note that *Exclusion Region* calculations are not best-estimate and they require a well-defined input preparation procedure that has been specified by the Boiling Water Reactor Owners' Group (BWROG) and reviewed by the Nuclear Regulatory Commission. The so-called BWROG procedures are defined in NEDO-31960 (Refs. 3 & 4) "BWR Owner's Group Long Term Stability Solutions Licensing Methodology." In *Exclusion Region* applications using the RAMONA code, care must be taken to ensure that the axial and radial power shapes resulting from RAMONA's 3-D calculation represent as accurately as possible the power shapes prescribed in References 3 & 4. Any departure from the established BWROG procedures to calculate *Exclusion Regions* must be justified.	The long-term stability solution for the DNPS and QCNPS units is the detect-and-suppress approach (Option III) as described in NEDO-31960-A. This approach makes use of the LPRM-based OPRM. Westinghouse follows the procedure described in NEDO-32465-A [Reactor Stability Detect and Suppress Solutions Licensing Basis Methodology for Reload Applications] except that a plant-specific DIVOM curve is generated instead of the Generic DIVOM curve described in NEDO-32465-A, which was found to be non-conservative for some applications. Westinghouse follows the detailed procedure developed by the BWROG for this plant-specific analysis [GE-NE-0000-0028-9174-R0] As a backup to Option III (i.e., in the event the system became inoperable), Westinghouse also performs decay ratio calculations to establish conservative exclusion regions on the power-flow map. The process used to perform these calculations is described in CENPD-294-P-A and CENPD-295-P-A.

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Table 12 CENPD-295-P-A Limitations and Conditions

CENPD-295-P-A [Referenced by CENPD-300-P-A] – Thermal-Hydraulic Stability Methodology for Boiling Water Reactors		
No.	**Condition / Limitation**	**Resolution**
1	The conclusion stated in the staff's Safety Evaluation Report (SER) for the ABB/CE [Westinghouse] topical report CENPD-294-P are applicable to this review.	See Table 11.
2	Any departure from the established Boiling Water Reactor Owners Group (BWROG) procedures [NEDO-31960 and Supplement 1] to calculate Exclusion Regions must be justified.	See response to Condition #5 in Table 11.
3	The stability methodology described in CENPD-295-P is based primarily on RAMONA stability calculations, but it also includes options to use NUFREQ-NPW (Ref 6), experimental-loop instability limit measurements, in-plant channel flow measurements, and in-plant core stability measurements. Since CENPD-295-P only develops and documents in detail the RAMONA option, the use of any other option in CENPD-295-P (with the exception of NUFREQ-NPW, which is already licensed for limited purposes) will require a separate review.	Only the RAMONA option is used.
4	The acceptance criteria for RAMONA-3 code stated in [CENPD-295-P] are acceptable. They are: (1) Core-wide decay ratio calculations are set to a calculated decay ratio of 0.8 (i.e., expected error including input preparation is ±0.2). (2) Channel thermal-hydraulic decay ratio calculations are set to a calculated decay ratio of 0.8 (i.e., expected error including input preparation is ±0.2). (3) Out-of-phase instability-threshold power calculations are set to either: (a) The actual threshold power for out-of-phase instabilities calculated by RAMONA minus an uncertainty margin that is calculated as the power required to reduce by 0.2 the core-wide decay ratio under those operating conditions, or (b) the power at which the core-wide decay ratio is 1.0 (i.e., 20% higher than the core-wide acceptance criteria) if out-of-phase instabilities are not observed following an appropriate out-of-phase perturbation.	For the ICA analysis (i.e., backup to Option III), the core-wide and channel thermal-hydraulic acceptance criterion for establishing exclusion boundaries is that the predicted decay ratio must be [ ][a,c]. For the out-of-phase instability threshold, the threshold power is set to either the actual threshold power minus the power differential necessary to reduce the core-wide decay ratio by 0.2 at those conditions, or the power at which the core-wide decay ratio is 1.0, if out-of-phase instabilities are not observed following an appropriate out-of-phase perturbation.

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Table 13 CENPD-284-P-A (RPA 89-112-P-A) (RPA 89-053-P-A) Conditions and Limitations

CENPD-284-P-A (RPA 89-112-P-A) (RPA 89-053-P-A)[Referenced by CENPD-300-P-A] – Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification		
No.	**Condition / Limitation**	**Resolution**
CENPD-284-P-A		
1	Because of the present uncertainty in the rate of void production during the initial power transient, RDA licensing analyses should be conservatively calculated without moderator voids. While this submittal does not provide a sufficient basis for applying RAMONA-3B-SCP2 in RDA licensing calculations, this does not preclude a future NRC approval of these models if the necessary justification is provided by ABB-CE [Westinghouse].	The CRDA is performed conservatively without void reactivity feedback.
2	In RDA licensing analyses involving very high rod-worths and large inlet subcoolings, the nonconservatism introduced by the assumption of a linear scram should be evaluated and, if necessary, accounted for in the determination of the peak fuel rod enthalpy.	The scram worth is not modeled linearly. The dynamic scram worth is calculated by the code. The scram time is modeled linearly based on the Technical Specification scram time data.
3	ABB-Atom [Westinghouse] determines the fuel rod gap conductance using the STAV fuel performance code. STAV must receive NRC approval prior to its use in RAMONA-3B-SCP2 RDA licensing analyses.	The fuel rod gap conductance data used in the CRDA analysis is generated using an NRC-approved STAV code. The STAV7.2 code will be used for DNPS and QCNPS. NRC review of STAV7.2 is scheduled for completion by the end of June 2005.
4	The accuracy of the PHOENIX/POLCA code system in applications involving non-ABB [Westinghouse] fuel must be demonstrated by comparisons to previous cycle measurements of the core reactivity and power distribution.	The accuracy of the PHOENIX/POLCA code system in applications involving non-Westinghouse fuel is demonstrated by the qualification of these codes using core follow data from previous cycles.
5	In RDA licensing calculations, each fuel bundle should be represented by a unique thermal-hydraulic channel, or the predictions made by combining channels should be shown to be conservative or insensitive to this approximation.	The CRDA model describes each fuel assembly explicitly.
6	The RDA analyses described in CENPD-284-P assume a rod drop velocity of 3 ft/sec. The use of a lower (less conservative) rod drop speed in RDA licensing analyses will require additional justification.	The rod drop velocity used in CRDA licensing analyses is at least 3 ft/sec, unless a lower value is specifically justified.
7	In order to account for RAMONA-3B-SCP2 modeling and input uncertainties, RDA licensing evaluations should include a detailed uncertainty analysis.	Uncertainties and biases are included in the determination of the maximum peak fuel enthalpy as discussed in the response to RAI 19 and in Appendix A of CENPD-284-P-A.

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CENPD-284-P-A (RPA 89-112-P-A) (RPA 89-053-P-A)[Referenced by CENPD-300-P-A] – Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification		
No.	**Condition / Limitation**	**Resolution**
8	In RDA licensing analyses, the selection of the highest-worth control rod must account for the worst-case single equipment malfunction and operator error allowed by the plant Technical Specifications and licensing basis.	The selection of the highest worth control rod accounts for the worst-case single equipment malfunction and operator error allowed by the plant Technical Specifications and licensing bases.
9	Since the rod-worth and nodal peaking comparisons do not always ensure that the bounding RDA is limiting, it should be verified that changes in other parameters having a significant effect on the RDA have not made the cycle-specific RDA more limiting than the precalculated bounding RDA.	For the first transition cycle, dynamic analyses are performed. Therefore, a previous analysis is not being used as bounding. For subsequent cycles, changes in rod worth and nodal peaking along with other parameters having significant effect on the CRDA will be verified that they do not make the cycle-specific CRDA more limiting than the pre-calculated bounding CRDA.
10	When determining the limiting cycle-specific RDA, in order to ensure equal rod-worths and reactivity insertion rates, the RDA comparisons should be made for cases in which the rod drops at the same speed and over the same axial span.	CRDA analyses are performed using a bounding drop velocity over an axial span equal to or greater than the maximum span the rod could physically drop.

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CENPD-284-P-A (RPA 89-112-P-A) (RPA 89-053-P-A)[Referenced by CENPD-300-P-A] – Control Rod Drop Accident Analysis Methodology for Boiling Water Reactors: Summary and Qualification		
No.	**Condition / Limitation**	**Resolution**
RPA 89-112-P-A and RPA 89-053-P-A		
1	The accuracy of the PHOENIX/POLCA code system in applications involving non-ABB [Westinghouse] fuel must be demonstrated by comparison to previous cycle measurements of the core reactivity and power distribution.	See response to CENPD-284-P-A Condition / Limitation 4 (Table 13).
2	ABB Atom [Westinghouse] determines the fuel rod gap conductance using the STAV fuel performance code. STAV is presently being reviewed by the NRC and the approval of the ABB Atom [Westinghouse] RDA methodology is contingent on the approval of STAV.	See response to CENPD-284-P-A Condition / Limitation 3 (Table 13).
3	Because of the present uncertainty in the rate of void production during the initial power transient, RDA licensing analyses should be conservatively calculated without moderator voids. While this submittal does not provide a sufficient basis for applying RAMONA-3B-SCP2 in RDA licensing calculations, this does not preclude a future NRC approval of these models if the necessary justification is provided by ABB-CE [Westinghouse].	See response to CENPD-284-P-A Condition / Limitation 1 (Table 13).
4	In RDA licensing calculations, each fuel bundle should be represented by a unique thermal-hydraulic channel, or the predictions made by combining channels should be shown to be conservative or insensitive to this approximation.	See response to CENPD-284-P-A Condition / Limitation 5 (Table 13).
5	In RDA licensing analyses involving very high rod-worths and large inlet subcoolings, the nonconservatism introduced by the assumption of a linear scram should be evaluated and, if necessary, accounted for in the determination of the peak fuel rod enthalpy.	See response to CENPD-284-P-A Condition / Limitation 2 (Table 13).
6	In RDA licensing analyses involving high-worth control rods, the core-specific control rod-worth must be determined.	In CRDA licensing analyses involving high-worth control rods, the core-specific control rod worth will be determined.
7	In RDA licensing analyses, the selection of the highest-worth control rod must account for the worst-case single equipment malfunction and operator error allowed by the plant Technical Specifications and licensing basis.	See response to CENPD-284-P-A Condition / Limitation 8 (Table 13).

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Table 14 WCAP-16081-P-A Conditions and Limitations

WCAP-16081-P-A – 10x10 SVEA Critical Power Experiments and CPR Correlation: SVEA-96 Optima2		
No.	**Condition / Limitation**	**Resolution**
1	The range of application claimed for the D4.1.1 CPR correlation is acceptable insofar as it corresponds to the range of parameters in the database for the correlation. The correlation will be applied to the SVEA-96 Optima2 fuel assembly over the applicable range for mass flux, system pressure, sub-bundle R-factor, boiling length, and [ ]^{a,c} as specified in Section 8 of the subject topical report.	These parameters are monitored during licensing applications to assure that CPR values used to establish safety limits are within the approved range of the correlation.
2	The CPR correlation uncertainty used to determine the thermal margin limit at all mass flux values should be [ ]^{a,c} for system pressure below 45 bar, and [ ]^{a,c} for system pressure above 45 bar, based on the values given in tables 6.3 through 6.6 of the subject TR.	The uncertainties of [ ]^{a,c} for system pressure below 45 bar, and [ ]^{a,c} for system pressure above 45 bar, will be applied in the reload licensing analyses.
3	It is acceptable either to apply the power mismatch factor in the analysis of the full assembly or to evaluate the CPR on a sub-bundle basis, because the sub-bundle approach represents an improvement in the methodology and yields a better characterization of the local sub-bundle flow, and making the application of the correlation more consistent with its derivation.	Either the power mismatch factor or the sub-bundle model will be used in the licensing analyses of the fuel assembly.
4	It is acceptable for the multiplicative correction factor, Fcorr of [ ]^{a,c} to be based on the average uncertainty in dryout power in the normal operating range to account for the effect of applying the D4.1.1 CPR correlation to a full assembly.	An [ ]^{a,c}, will be used in accordance with Section 5.4 of WCAP-16081-P-A.
5	The [ ]^{a,c} term in Equation 5.3-2 of the TR is limited to a values of [ ]^{a,c} at flow rated below [ ]^{a,c} at all pressures and inlet subcooling values because of insufficient validation of the basis for extending the double-peak axial power profile correction factor to low flow rates.	[ ]^{a,c} is set equal to [ ]^{a,c}
6	It is acceptable to apply the same uncertainty and limitations to D4.1.1 CPR correlation for both transient calculation and safety steady-state analysis.	The uncertainty of the D4.1.1 CPR correlation is accounted for in the SLMCPR calculations for the licensing application.

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Table 15 CENPD-285-P-A Conditions and Limitations

CENPD-285-P-A [Referenced by CENPD-300-P-A] Fuel Rod Design Methods for Boiling Water Reactors		
No.	**Condition / Limitation**	**Resolution**
1	Based on the validity of fission gas release (FGR) model and other models, the application of CENPD-285-P is approved to a rod average burnup of 50 GWd/MTU.	This limitation is specific to STAV6.2. STAV7.2 will be used for DNPS and QCNPS.
2	The cladding creep model has too strong a dependence on cladding stress that results in a non-conservative estimate of the rod pressure. The staff concludes that the stress exponent of the ABB/CE [Westinghouse] creep equation model should be limited to 1.5. The uncertainty in the creep model estimated by ABB/CE [Westinghouse] is too small and should be increased by a factor of 2 for both the creep equation and the creep relationship in STAV6.2.	This limitation is specific to STAV6.2. STAV7.2 will be used for DNPS and QCNPS.
3	The ABB/CE [Westinghouse] BWR FGR model is approved to 40 GWd/MTU rod average, and the PWR FGR model is approved to 50 GWd/MTU rod average in STAV6.2. Thus, the PWR FGR model is the only acceptable model for fission gas release calculation for burnups between 40 and 50 GWd/MTU.	This limitation is specific to STAV6.2. STAV7.2 will be used for DNPS and QCNPS.
4	The STAV6.2 code is acceptable for application to urania-gadolinia fuel with gadolinia content up to 8 wt%. The ABB/CE [Westinghouse] urania-only fission gas diffusion constants should be used for both urania-only and urania-gadolinia fuel rod applications.	This limitation is specific to STAV6.2. STAV7.2 will be used for DNPS and QCNPS.

Table 16 WCAP-15836-P Conditions and Limitations

WCAP-15836-P Fuel Rod Design Methods for Boiling Water Reactors – Supplement 1		
No.	**Condition / Limitation [expected]**	**Resolution**
	Currently under review by NRC staff. Any limitations resulting from the review process will be implemented.	

Table 17 CENPD-287-P-A Conditions and Limitations

CENPD-287-P-A [Referenced by CENPD-300-P-A] Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors		
No.	**Condition / Limitation**	**Resolution**
1	Based on the validity of fission gas release (FGR) and corrosion models, and other models, the application of CENPD-285-P is approved to a rod average burnup of 50 GWd/MTU.	This limitation is specific to STAV6.2. STAV7.2 will be used for DNPS and QCNPS.
2	The cladding creep model has too strong a dependence on cladding stress that results in a non-conservative estimate of the rod pressure. The staff concludes that the stress exponent of the ABB/CE [Westinghouse] creep equation mode should be limited to 1.5. The uncertainty in the creep model estimated by ABB/CE [Westinghouse] is too small and should be increased by a factor of 2 for both the creep equation and the creep relationship in STAV6.2.	This limitation is specific to STAV6.2. STAV7.2 will be used for DNPS and QCNPS.
3	The ABB/CE [Westinghouse] BWR FGR model is approved to 40 GWd/MTU rod average, and the PWR FGR model is approved to 50 GWd/MTU rod average in STAV6.2. Thus the PWR FGR model is the only acceptable model for fission gas release calculation for burnups between 40 and 50 GWd/MTU.	This limitation is specific to STAV6.2. STAV7.2 will be used for DNPS and QCNPS.
4	The STAV6.2 code is acceptable for application to urania-gadolinia fuel with gadolinia content up to 8 wt%. The ABB/CE [Westinghouse] urania-only fission gas diffusion constants should be used for both urania-only and urania-gadolinia fuel rod applications.	This limitation is specific to STAV6.2. STAV7.2 will be used for DNPS and QCNPS.
5	The calculation of uniform cladding strain should be the elastic plus inelastic strains due to power increases in normal operations and AOOs.	A uniform cladding strain defined as the elastic plus inelastic strains due to power increases in normal operations and AOOs will be used.

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**
	Currently under review by NRC staff. Any limitations resulting from the review process will be implemented in addition to any limitations on STAV7.2	

Table 19 CENPD-288-P-A [Referenced by CENPD-300-P-A] Conditions and Limitations

CENPD-288-P-A [Referenced by CENPD-300-P-A] ABB Seismic/LOCA Evaluation Methodology for Boiling Water Fuel		
No.	**Condition / Limitation**	**Resolution**
	No conditions or limitations were identified in the SER	

Table 20 CENPD-390-P-A Conditions and Limitations

CENPD-390-P-A The Advanced PHOENIX and POLCA Code for Nuclear Design of Boiling Water Reactors		
No.	**Condition / Limitation**	**Resolution**
a	When applying PHOENIX/POLCA to transition cores, CENP [Westinghouse] should use fuel specific data to model the thermal and hydraulic behavior of the non-ABB/CE [Westinghouse] fuel and confirm that the uncertainties derived for ABB [Westinghouse] fuel are applicable to the non-ABB/CE [Westinghouse] fuel.	For the DNPS/QCNPS application, fuel specific data has been received and is being used to model the thermal and hydraulic behavior of the non-Westinghouse fuel. As part of the benchmark and transition analyses in support of the DNPS/QCNPS application, Westinghouse will confirm that the uncertainties derived for Westinghouse fuel are applicable to the non-Westinghouse fuel.
b	PHOENIX/POLCA are approved for analysis of ABB/CE [Westinghouse] fuel types up to and including 10x10 lattices with a maximum enrichment of 5 w/o UO_2. Non ABB/CE [Westinghouse] fuel types may be analyzed assuming that analyses are performed consistent with (a) above. The code is approved for application to fuel with burnable absorbers composed of a mixture of UO_2 and Gd_2O_3 with concentrations up to 9 w/o Gd_2O_3. Application of the code to non-UO2 fuel or the fuel using burnable poisons other than Gadolinia will need to be justified.	For the DNPS/QCNPS application, the PHOENIX/POLCA codes are being used and will continue to be used for fuel types that meet all the conditions noted. DNPS/QCNPS legacy fuel, as well as the Westinghouse SVEA-96 Optima2 fuel are up to 10x10 lattices, with maximum enrichments lower than 5 w/o UO_2, use Gadolinia as the only burnable absorber, and have burnable absorber concentrations lower than 9 w/o Gd_2O_3.
c	When applying the PHOENIX/POLCA code to fuel other than what is approved in this SE (see (b) above), the NRC should be informed by letter of this application and be provided an opportunity for review.	For the DNPS/QCNPS application, the PHOENIX/POLCA codes are being used for fuel types that meet all the conditions noted in (b). In the event that this changes, the NRC will be informed and given an opportunity for review.
d	PHOENIX/POLCA contains several models for BWR analysis not used to generate the information contained in the topical report. If CENP [Westinghouse] determines that one of these models is needed for a licensing analysis, the staff should be informed of the application and be given an opportunity for review.	For the DNPS/QCNPS application, models for BWR analysis not used to generate the information contained in the topical report have not been used. If Westinghouse decides to use one of those models, the NRC will be informed and given an opportunity for review.

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Table 21 BR 91-255-P-A [Referenced by CENPD-300-P-A] Conditions and Limitations

BR 91-255-P-A – CONDOR: A Thermal-Hydraulic Performance Code for Boiling Water Reactors[2]		
No.	**Condition / Limitation**	**Resolution**
1	The CONDOR code is claimed, in the topical report, to be able to perform BWR loop calculations. However, no description of loop modeling is given. Therefore, the current version of CONDOR is restricted to the calculation of core flow and enthalpy distribution.	Only results between the inlet to the lower plenum and the steam dome are used in licensing calculations. Results applicable to other components in the BWR loop (e.g., steam line, turbine, etc.) are not utilized in licensing calculations. As discussed in conjunction with SER Limitation 3, plant-specific data are used to assure flow splits and pressure drops across the core region are adequately treated.
2	CONDOR does not have a verified CHF correlation in the code at this time. Any correlation to be incorporated in the code for MCPR licensing analysis has to be reviewed and approved by the NRC separately.	As described in CENPD-300-P-A, CPR correlations for Westinghouse fuel are submitted to the NRC for approval. In accordance with CENPD-300-P-A, in the absence of direct availability to Westinghouse of the NRC-approved correlation for a non-Westinghouse fuel design, information generated by an NRC-approved correlation is obtained from the utility to establish the CPR performance of the fuel.
3	Since the core bypass flow calculation is based on a simplified correlation with ΔP as the independent parameter, the correlation coefficients should be determined by comparing with the test data on a plant-specific basis. Factors affecting the coefficients, such as mixed core with fuels from different vendors, the crud buildup, and irradiation effects have to be considered.	As described in Reference CENPD-300-P-A, loss coefficients and flow splits for Westinghouse fuel components are based on test data. Loss coefficients and flow splits for non-Westinghouse fuel components are based on pressure drop and flow split data obtained from the utility for each reload plant application. In addition to assembly-specific information for non-Westinghouse fuel, these data include plant-specific information required to model the thermal-hydraulic characteristics of the core region between the lower plenum and the steam dome such as flow splits for paths external to the fuel assemblies (e.g., flow paths to inter-assembly bypass), core plate pressure drops, inlet orifice form losses, plant heat balances, etc. The impact of crud buildup is based on best available current information and is treated conservatively in design and licensing analyses.
4	The CONDOR code uses 24 nodes to represent a flow channel for code verification, this number of nodes should be used for licensing calculations. If any reduced number of nodes is used, additional calculations should be performed to identify uncertainties associated with the reduced number of nodes.	If less than 24 nodes are used to represent a flow channel, additional calculations are performed to identify uncertainties associated with the reduced number of nodes.
5	Selection of the loss coefficients with the effects of the crud build-up, geometry change due to irradiation, different fuel designs by different vendors should be considered on a plant-specific basis.	See Response to SER Limitation 3.
6	During the course of our review, we raised questions regarding the proposed models and data. Westinghouse responded to these questions in Reference 7. These questions and answers should be included (Ref. 2) in the final topical report on CONDOR submitted by Westinghouse.	Resolution of the NRC questions have been incorporated in WCAP-10107(P) and BR 91-255-P-A.

[2] Thermal hydraulic analyses will be performed with the POLCA7 code described in CENPD-390-P-A. There are no specific thermal-hydraulic conditions in the SER for CENPD-390-P-A. However, the thermal-hydraulic models in CENPD-390-P-A are updated versions of the models in the CONDOR topical report (BR 91-255-P-A) as described in CENPD-390-P-A. Therefore, it is appropriate to address the SER conditions in BR 91-255-P-A and how they are dispositioned in POLCA7.