Westinghouse Non-Proprietary Class 3

WCAP-16500-NP-A
Revision 0

August 2007

CE 16x16 Next Generation Fuel Core Reference Report

Westinghouse

WESTINGHOUSE NON-PROPRIETARY CLASS 3

WCAP-16500-NP-A

Revision 0

CE 16x16 Next Generation Fuel Core Reference Report

Original Version: February 2006

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Table of Contents

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Section A

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July 30, 2007

Mr. James A. Gresham, Manager
Regulatory Compliance and Plant Licensing
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: FINAL SAFETY EVALUATION FOR WESTINGHOUSE ELECTRIC COMPANY (WESTINGHOUSE) TOPICAL REPORT (TR) WCAP-16500-P, REVISION 0, "CE [COMBUSTION ENGINEERING] 16X16 NEXT GENERATION FUEL [(NGF)] CORE REFERENCE REPORT" (TAC NO. MD0560)

Dear Mr. Gresham:

By letter dated February 28, 2007, Westinghouse submitted TR WCAP-16500, "CE 16x16 Next Generation Fuel Core Reference Report," to the U.S. Nuclear Regulatory Commission (NRC) staff. By letter dated June 15, 2007, an NRC draft safety evaluation (SE) regarding our approval of TR WCAP-16500, Revision 0, was provided for your review and comments. By letter dated June 21, 2007, Westinghouse commented on the draft SE. The NRC staff's disposition of Westinghouse's comments on the draft SE are discussed in the attachment to the final SE enclosed with this letter.

The NRC staff has found that TR WCAP-16500, Revision 0, is acceptable for referencing in licensing applications for Combustion Engineering designed pressurized water reactors with a 16x16 fuel assembly lattice to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that Westinghouse publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include an "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, Westinghouse and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Ho K. Nieh, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 700

Enclosure: Final SE

cc w/encl:
Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
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Pittsburgh, PA 15230-0355

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ADAMS ACCESSION NO.:ML071920269 *No major changes to SE input. NRR-043

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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT (TR) WCAP-16500-P, REVISION 0,

"CE [COMBUSTION ENGINEERING] 16X16 NEXT GENERATION FUEL [(NGF)]

CORE REFERENCE REPORT"

WESTINGHOUSE

PROJECT NO. 700

1.0 INTRODUCTION AND BACKGROUND

By letter dated February 28, 2006 (Reference 1), as supplemented by letters dated November 29, 2006 (Reference 2), January 29, 2007 (Reference 3), February 15, 2007 (Reference 4), March 16, 2007 (Reference 5), and April 5, 2007 (Reference 6), Westinghouse requested review and approval of TR WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report." This TR describes the 16x16 lattice NGF assembly mechanical design for the CE nuclear steam supply system (NSSS). In addition to the reference product description, this TR describes the fuel mechanical and reload design methodology intended to support fuel design and licensing applications up to a rod average burnup of 62 Gigawatt Days per Metric Ton Uranium (GWd/MTU).

2.0 REGULATORY EVALUATION

Regulatory guidance for the review of fuel system designs and adherence to applicable General Design Criteria (GDC) is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System Design" (Reference 7). In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

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

In addition to licensed reload methodologies, an approved mechanical design methodology is utilized to demonstrate compliance to SRP Section 4.2 fuel design criteria. The U.S. Nuclear

Regulatory Commission (NRC) staff's objectives for review of TR WCAP-16500-P are to ensure that the approved reload and fuel mechanical design methodologies (1) remain applicable to the NGF design, and (2) adequately address SRP Section 4.2. criteria. In addition, based upon Lead Test Assemblies (LTAs), post-irradiation examinations (PIEs), mechanical testing, past operating experience of similar designs and materials, and fuel performance model predictions, the NRC staff reviewed expected performance of the CE 16x16 NGF assembly to ensure it satisfies these objectives. The NRC staff's review is similar in scope to past reviews on SVEA-96 Optima2 fuel assembly design (Reference 8).

3.0 TECHNICAL EVALUATION

The NRC staff's review of TR WCAP-16500-P is summarized below:

- Verify that the fuel assembly component and fuel rod design criteria are consistent with regulatory criteria identified in SRP Section 4.2 or otherwise acceptable and justified.
- Verify that the fuel mechanical design methodology is capable of accurately or conservatively evaluating each component with respect to its applicable design criteria.
- Verify that the CE 16x16 NGF assembly design satisfies regulatory requirements.
- Verify that the Westinghouse experience database (in-reactor residence, post-irradiation examinations, and out-of-pile testing) supports the operating limits being requested and provides reasonable assurance that no anomalous behavior will occur during batch implementation.
- Verify that the impact of the new fuel assembly design on the reload design methodology, safety analyses, and setpoints process has been properly addressed.

The layout of this SE closely follows that of TR WCAP-16500-P.

In addition to issuing requests for additional information (RAIs), the NRC staff conducted an audit of the supporting Westinghouse engineering calculations on January 30-31, 2007, at the Westinghouse Rockville office. Included in this audit was a presentation by Westinghouse on the implementation of multiple, axially-dependent critical heat flux correlations in the reload process. In addition, RAI responses (Reference 3) were discussed. Follow-on discussions with Westinghouse staff were required to assess the impact of the fuel assembly design on the Core Operation Limits Supervisory System (COLSS)/Core Protection Calculator System (CPCS) setpoints methodology. A subsequent audit was held on March 29, 2007, at the Westinghouse Rockville office. The material presented by Westinghouse is documented in Reference 6.

3.1 LTA Program and In-Reactor Experience

Section 2.4.7 of TR WCAP-16500-P describes the ongoing LTA programs and previous in-reactor experience with the features being implemented with use of the CE 16x16 NGF assembly. The Westinghouse fleet has extensive experience with ZIRLO™ grids and guide tubes as well as mixing vane grids and intermediate flow mixing (IFM) grids. In addition, full batch implementation of a CE 14x14 advanced fuel assembly design with many of the same

features as the 16x16 NGF has been in-service at Calvert Cliffs Nuclear Power Plant. Westinghouse states that any new data (as it becomes available from these programs) will be assessed for its impact on the approved models and methods.

In response to RAI 1a in Reference 2 regarding validation of the fuel performance models, Westinghouse described ongoing LTA programs for their Optimized ZIRLO™ cladding. Table 1-1 of the response to RAI 1a provides details of the ongoing LTA irradiations in several different reactors along with expected burnup and scheduled PIEs. This LTA data, along with the Vogtle Creep and Growth Test Program, will be used to validate the fuel performance models. As part of a continuing condition to validate its models ahead of burnup achieved by batch implementation, Westinghouse submitted the first SE compliance letter to the NRC staff (Reference 9). Based upon conversations with Westinghouse, the NRC staff anticipates a second letter validating its models against more recent, higher burnup data prior to the first full batch implementation of Optimized ZIRLO™.

In response to an RAI regarding in-reactor experience of the various components of the NGF design (clarification of response to RAI 1a, Reference 3), Westinghouse provided a summary of reactor operating experience. Examination of the table revealed that each of the assembly components and the fuel rod design has significant in-reactor experience. For example, the Inconel straight strip top grid has extensive service in Westinghouse fuel designs. Based upon the information provided in TR WCAP-16500-P, and in response to RAIs coupled with the commitment (i.e., Limitation and Condition 4) to validate fuel performance models, the NRC staff concludes that Westinghouse has provided sufficient evidence to demonstrate with reasonable assurance that the CE 16x16 NGF assembly design will not experience anomalous behavior during batch application.

3.2 CE 16x16 NGF Assembly - Fuel Assembly Design

Section 1 of TR WCAP-16500-P provides a description of the CE 16x16 NGF fuel assembly design along with a comparison to the current CE fuel designs. Figure 1-1 of TR WCAP-16500-P illustrates the distribution of grid-types for five CE reactor designs. The NRC staff identified that Westinghouse's Fuel Criteria Evaluation Process (FCEP) fuel design change process was not currently applicable to CE fuel assembly designs. In response to RAI No. 2, Westinghouse more clearly identified design variations needed to address plant differences. The description of the fuel assembly design, along with the variances defined in response to this RAI, specifies the extent of the regulatory evaluation of the NGF design. Therefore, changes in the CE 16x16 NGF fuel assembly design may require NRC review prior to implementation.

Section 2.3 of TR WCAP-16500-P describes the fuel assembly mechanical design basis and evaluations.

3.2.1 Fuel Assembly Growth

The Westinghouse criterion is that sufficient allowance for irradiation-induced axial growth exists to prevent solid interference between the assembly and the core internals. Interference loads could lead to fuel assembly bowing or guide tube distortion (i.e., challenge control rod insertion). This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The design evaluation employs the previously approved SIGREEP computer code (Reference 12). However, this code was approved for the growth evaluation of Zircaloy-4 guide tubes (NGF uses ZIRLO™ tubing). Westinghouse proposed to use an adjustment factor applied to the best-estimate SIGREEP calculated fuel assembly length change along with the upper/lower 95 percent values directly from the code. Justification for this approach is based upon Zircaloy-4 and ZIRLO™ cladding growth data along with two measured fuel assembly growth data points (displayed on Figure 2-15 of TR WCAP-16500-P). In response to RAI 1b of Reference 2 regarding NRC staff concerns with the use of this limited database to validate the adjusted SIGREEP calculations, Westinghouse identified further sources of in-reactor growth data on ZIRLO™ guide tubes and agreed to compare this data (as it becomes available) to predictions and modify the correlation if necessary (keeping the NRC informed via update meetings). The NRC staff does not accept this informal approach. An SE condition on the timely validation of these guide tube growth predictions is required to ensure that the design requirements are satisfied. Similar to the NRC staff's approval of Optimized ZIRLO™, Westinghouse shall demonstrate the accuracy of its growth predictions based upon measured data and this validation shall be ahead of the burnups achieved by batch implementation.

3.2.2 Fuel Assembly Hydraulic Stability

The Westinghouse criterion for fuel assembly hydraulic stability is that the fuel assembly will not experience significant flow-induced, resonant fuel assembly vibration under any operating conditions. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The design evaluation employs both single bundle and dual bundle flow testing. Both the NGF and standard CE 16x16 fuel designs were tested over a range of reactor operating flow rates to verify that the design basis is satisfied. In response to RAI 4 regarding the degree of flow testing on the different design variations in Figure 1-1 of TR WCAP-16500-P, Westinghouse stated that the limiting NGF design was tested based upon past experience and the characteristics of that particular design. The NRC staff had concerns with the difficulty predicting assembly vibration between the NGF design variations. In response to RAI 4a clarification of Reference 3, Westinghouse provided further discussion and evidence to support its finding that flow induced vibration (FIV) would not occur for any of the five fuel assembly designs shown in Figure 1-1 of TR WCAP-16500-P. Based upon the information and FIV test results provided in TR WCAP-16500-P and in response to RAIs, the NRC staff finds that the NGF assembly design satisfies its design criteria.

3.2.3 Fuel Assembly Structural Integrity

The Westinghouse criterion is that the assembly must maintain its structural integrity under all operating conditions, including seismic and loss-of-coolant accident (LOCA) loads. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The design evaluation is based upon results from full-scale testing of the skeleton and the fuel assembly which were used to determine input characteristics to predict bundle deflected shapes and grid impact loads. Dynamic grid crush testing was performed for comparison to predicted grid impact loads. Stress intensities in the remaining components were evaluated against applicable limits. In response to RAI 8a of Reference 2 regarding the structural integrity testing,

Westinghouse provided a description of all of the mechanical testing performed to validate the structural integrity of the fuel assembly design. The NRC staff finds this test matrix acceptable to demonstrate the design basis.

In response to RAI 8b of Reference 2 regarding irradiation induced spring relaxation on grid crush strength, Westinghouse cited an evaluation within TR WCAP-12488-P-A and stated that similarities between the CE NGF grid design and Westinghouse design were such that the conclusions were applicable. The NRC staff accepts this justification and finds the grid crush test program acceptable to demonstrate the design basis.

3.2.4 Fuel Assembly Shipping and Handling Loads

The Westinghouse criterion is that the fuel design must be able to accommodate shipping and handling loads without exceeding the specified limits in Table 2-2 of TR WCAP-16500-P. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

3.2.5 Fuel Assembly Components

The design evaluation for each of the assembly components is discussed below.

3.2.5.1 Fuel Assembly Guide Tube Wear

The Westinghouse criterion is that the fuel design must continue to satisfy all stress limits with the maximum predicted reduction in cross-sectional area of the guide tube due to friction wear caused by the control element assembly (CEA). This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The design evaluation cites past experience with chrome-plated wear sleeves which have essentially eliminated wear as an issue. The NGF wear sleeves cover the possible range of wear associated with the CEAs residing at the all-rods-out (ARO) elevation. Based upon operating experience, the NRC staff finds that the NGF guide tube sleeves will adequately protect against wear. As for non-sleeved NGF designs, the NRC staff finds the use of current CE 16x16 wear methods acceptable for application to NGF designs.

3.2.5.2 Fuel Assembly Bottom Nozzle

The Westinghouse criterion is that the stress level of the bottom nozzle must be less than the specified limits in Table 2-2 of TR WCAP-16500-P. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The design evaluation states that the NGF bottom nozzle is structurally identical to the standard design with one minor difference. Westinghouse analyses have demonstrated that the nozzle continues to satisfy the design basis.

3.2.5.3 Fuel Assembly Top Nozzle

The Westinghouse criterion is that the stress level of the top nozzle must be less than the specified limits in Table 2-2 of TR WCAP-16500-P. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The design evaluation states that the NGF top nozzle is almost identical to the standard design with only minor differences. Westinghouse analyses have demonstrated that the nozzle continues to satisfy the design basis.

3.2.5.4 Fuel Assembly Holddown Springs

The Westinghouse criterion is that the combination of fuel assembly wet weight and holddown spring force must maintain a net downward force during all Condition I and II events. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The design evaluation states that the NGF holddown spring provides more force than the standard design. Changes were made to the design to compensate for increased pressure drop across the assembly. Plant-specific analyses will demonstrate that the design basis is satisfied.

3.2.5.5 Fuel Assembly Guide Thimbles and Instrumentation Tube

The Westinghouse criterion is that the stress levels of the guide thimbles and instrumentation tube must be less than the specified limits in Table 2-2 of TR WCAP-16500-P. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The design evaluation stated that the "yield and ultimate strengths of the two materials are almost identical." In response to RAI 9 of Reference 2 regarding this statement, Westinghouse provided unirradiated properties for OPTIN and ZIRLO™. Unirradiated properties are used and no credit is given for irradiation-induced hardening. Based upon satisfying the stress limits for all operating conditions, the NRC staff finds the guide thimble and instrument tube performance acceptable.

3.2.5.6 Fuel Assembly Joints and Connections

The Westinghouse criterion for threaded joint components is that the stress levels must be less than the specified limits in Table 2-2 of TR WCAP-16500-P. The Westinghouse criterion for bulged connections between the guide thimble and the grid sleeves or guide thimble flange is that their strength must exceed the loads applied to the connection under all operating conditions. The Westinghouse criterion for welded connections is that they will not fail under all operating conditions. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The design evaluation demonstrates that these joints and connections satisfy the applicable stress limits. Based upon the described analyses and testing performed by Westinghouse, the NRC staff finds the various joints and connections of the NGF assembly design acceptable.

3.2.5.7 Fuel Assembly Spacer Grids

The Westinghouse criterion is that the lateral strength must be sufficient to withstand seismic and LOCA events with no channel closure greater than that which would significantly impair the coolability of the fuel rod array or insertability of the CEAs. The Westinghouse criterion for the grid springs is that the cumulative fatigue usage will not exceed 1.0 at end-of-life (EOL). In addition, the Westinghouse criterion on spacer grid design is that its width must be small enough to provide adequate clearances between the grid assemblies and the reactor internals to ensure functionality during the fuel assembly lifetime. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The design evaluation states that the grid strength exceeds the predicted impact forces associated with the seismic/LOCA events. This evaluation also concludes that the cumulative fatigue usage factor for the mid grid springs satisfies the 1.0 limit, consistent with Westinghouse methodology. The top and bottom grids have extensive operating experience with no signs of fatigue failure. Further, based upon in-reactor behavior, Westinghouse concludes that the low tin ZIRLO™ grids will maintain adequate clearance within the reactor cavity. Based upon the described analyses and testing performed by Westinghouse, the NRC staff finds the spacer grid design acceptable.

3.3 CE 16x16 NGF Assembly - Fuel Rod Design

Section 2.5 of TR WCAP-16500-P describes the fuel rod mechanical design basis and evaluations. Figure 2-14 of TR WCAP-16500-P illustrates changes in the fuel rod design relative to the standard CE 16x16 design.

During the review of a new fuel rod design, it is recommended that the NRC staff include an independent assessment. The fuel rod performance code FRAPCON has been developed and maintained to provide this support capability. At this time, FRAPCON-3 does not have either ZIRLO™ or Optimized ZIRLO™ properties nor the ability to simulate the helium production associated with ZrB2 integral fuel burnable absorber (IFBA) fuel pellets. As such, independent calculation would be of limited value. Instead, the NRC staff chose to conduct an audit of the Westinghouse fuel rod design analyses.

3.3.1 Fuel Rod Internal Pressure and Departure from Nucleate Boiling (DNB) Propagation

The Westinghouse criterion is that the fuel rod internal hot gas pressure shall not exceed the critical pressure determined to cause an outward creep rate that is in excess of the fuel pellet swelling rate. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design. Additional criteria for ZrB2 IFBA fuel, as listed in Section 2.5.1 of TR WCAP-16500-P, also exist.

During the Westinghouse audit, the NRC staff reviewed the supporting Westinghouse engineering calculations to ensure that the NGF design criteria were satisfied using approved models and methods. The Westinghouse engineering calculation included demonstration analyses for both UO2 and ZrB2 NGF fuel rods. Compared to the current CE fuel rod design, the UO2 NGF design has less void volume; whereas, the ZrB2 NGF fuel rod may be designed with higher void volume (i.e., with the use of annular axial blanket pellets) to accommodate the production of helium gas. A bounding pin power history was employed in the demonstration analysis. The calculated rod internal pressure, using the approved models and methods, remained below the critical no-clad liftoff (NCLO) pressure, thus ensuring that the design

criteria was satisfied. Plant and cycle-specific evaluations, using the currently approved models and methods, will be performed to ensure that this criteria is satisfied based upon future core loading patterns and fuel rod designs. Based upon the audit of the Westinghouse calculations, which included reasonable conservative inputs along with approved models and methods, the NRC staff finds the NGF fuel rod design acceptable with respect to rod internal pressure.

Using the currently approved methodology, Westinghouse will continue to satisfy the respective DNB propagation criteria including the SE conditions associated with ZrB2 IFBA fuel designs.

3.3.2 Fuel Rod Cladding Stress and Strain

The Westinghouse criteria for primary tensile stress in the clad and the end cap welds are stress must not exceed 2/3 of the minimum unirradiated yield strength during Condition I and II. For Condition III, the primary tensile stress limit is the yield strength and for Condition IV seismic and LOCA conditions, the stress limit is the lesser of 0.7 times the ultimate yield strength (Su) or 2.4 times the allowable stress intensity (Sm). The design criteria for primary compressive stress in the clad and the end cap welds is that stress must not exceed the minimum unirradiated yield strength during Conditions I, II, and III. During Condition IV seismic and LOCA conditions, the stress limit is the lesser of 0.7 Su or 2.4 Sm. This design criteria is consistent with the current CE fuel design methodology.

The Westinghouse criterion for cladding strain is that the net unrecoverable circumferential tensile cladding strain shall not exceed 1 percent for fuel rods less than or equal to 52 Megawatt Days per kilogram Uranium (MWd/kgU). A total (elastic plus plastic) circumferential cladding strain of less than 1 percent is applied for fuel rods exceeding 52 MWd/kgU. This design criterion is consistent with past CE fuel designs; however, it differs from both the SRP and the strain criterion dictated as part of the approval of Optimized ZIRLO™ (Reference 8). In response to RAI 10 of Reference 2 regarding the cladding strain limit, Westinghouse stated that the plastic strain capability of Optimized ZIRLO™ was greater than 1 percent up to the transition point of 52 MWd/kgU. The NRC staff requested mechanical testing data on irradiated fuel specimens be provided to demonstrate allowable strains. In response to RAI 10 clarification of Reference 3, Westinghouse provided a table of measured plastic strain from axial tensile and ring tensile tests on high burnup ZIRLO™ fuel rods. As part of the January 30-31, 2007 audit, the NRC staff questioned Westinghouse staff on the applicability of this mechanical testing data and reviewed the basis document, a Studsvik Laboratory Report (Reference 13). Examination of the laboratory report revealed that the tabulated plastic strain was ultimate strain (at failure), not uniform plastic strain. Based upon the lack of irradiated mechanical testing data to support the proposed strain limit, the NRC staff determined that the SRP strain limit of 1 percent total (plastic plus elastic) applies to the CE 16x16 NGF assembly design. This position is reflected in Westinghouse's amended RAI response (RAI 10 clarification, Reference 4) which states that Westinghouse will apply a 1 percent total strain limit for all burnups.

During the Westinghouse audit, the NRC staff reviewed the supporting Westinghouse engineering calculations to ensure that the NGF design criteria were satisfied using approved models and methods. The Westinghouse engineering calculation included demonstration analyses of both fuel rod stress and strain. Calculations were based upon measured, unirradiated Optimized ZIRLO™ properties and demonstrated that the design criteria were satisfied. Note that calculated circumferential cladding strain remained below 1 percent total.

Based upon the audit of the Westinghouse calculations, which included measured Optimized ZIRLO™ properties and approved methods, the NRC staff finds the NGF fuel rod design acceptable with respect to fuel rod stress and strain.

3.3.3 Fuel Cladding Oxidation and Hydriding

The Westinghouse criterion is that fuel rod damage will not occur due to excessive clad oxidation and hydriding. The specific limits on fuel cladding corrosion were developed as part of the NRC staff's approval of Optimized ZIRLO™ (Reference 9) and are applicable to the CE 16x16 NGF assembly design.

3.3.4 Fuel Temperature

The Westinghouse criterion is that fuel rod damage will not occur due to excessive fuel temperature. For Condition I and II events, the fuel system and protection system are designed to assure that the calculated centerline fuel temperature does not exceed the fuel melting temperature. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

Using approved models and methods, Westinghouse will continue to limit peak local power experienced during Condition I and II events to ensure that fuel temperature remain below melting temperature at all burnups. This evaluation may be plant- and cycle-specific. Based upon this commitment, the NRC staff finds the NGF fuel rod design acceptable with respect to fuel temperature.

3.3.5 Fuel Cladding Fretting Wear

The Westinghouse criterion is that the fuel system will not be damaged due to fuel rod fretting. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design. A main objective of the NGF design is to add fretting margin relative to the current CE 16x16 Standard assembly. As such, the design requirement is that cladding wear due to contact with the grid rod supports must be less than the observed wear on the existing CE 16x16 Standard assembly.

Out-of-pile long-term wear flow testing was performed in the Westinghouse VIPER test loop. In addition to earlier flow testing and associated wear measurements on the CE Standard assembly design, the VIPER tests included both the NGF and the Standard designs in adjacent locations within the test loop. Results from the flow testing confirm that cladding wear margin is improved in the NGF design. In response to RAI 4b of Reference 2 regarding the long-term wear testing, Westinghouse provided further details of the flow testing. Based upon the information presented in TR WCAP-16500-P and in the RAI response related to the out-of-pile long-term wear testing, the NRC staff finds the NGF spacer and fuel rod design acceptable with respect to fuel cladding fretting wear.

3.3.6 Fuel Cladding Fatigue

The Westinghouse criterion is that for the number and type of transients which occur during Condition I reactor operation, EOL cumulative fatigue damage in the clad and in the end cap

welds must be less than 0.8. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

During the Westinghouse audit, the NRC staff reviewed the supporting Westinghouse engineering calculations to ensure that the NGF design criteria were satisfied using approved models and methods. The Westinghouse engineering calculation included demonstration analyses of fuel rod cladding fatigue. The calculated EOL cumulative fatigue damage, using methods consistent with the current CE 16x16 Standard design, was well below the 0.8 requirement. Based upon the audit of the Westinghouse calculations, which included measured Optimized ZIRLO™ properties and approved methods, the NRC staff finds the NGF fuel rod design acceptable with respect to fuel rod cladding fatigue.

3.3.7 Fuel Cladding Flattening

The Westinghouse criterion is that the time required for the radial buckling of the cladding in any fuel or integral burnable absorber rod must exceed the reactor operating time necessary for the appropriate fuel batch to accumulate its design average discharge burnup. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

In response to RAI 11b of Reference 2 regarding the plenum spring radial support capacity, Westinghouse stated that validation of the spring's radial support characteristics will consist of either an assessment of the spring design relative to previously justified designs, or performing autoclave testing with the particular plenum spring. Westinghouse noted that the characteristics of the Waterford Nuclear Power Plant LTA design were such that previous autoclave test results were conservative.

During the Westinghouse audit, the NRC staff reviewed the supporting Westinghouse engineering calculations to ensure that the NGF design criteria were satisfied using approved models and methods. The Westinghouse engineering calculation included demonstration analyses of fuel rod cladding flattening. The calculation demonstrated that instability did not occur during the maximum residence time. Based upon the audit of the Westinghouse calculations and the justification of the spring support in the plenum region, the NRC staff finds the NGF fuel rod design acceptable with respect to fuel rod cladding flattening.

3.3.8 Fuel Rod Axial Growth

The Westinghouse criterion is that the axial length between the end fitting must be sufficient to accommodate differential thermal expansion and irradiation-induced differential growth between fuel rods and guide tubes such that it can be shown with 95 percent confidence that no interference exists. Interference may lead to rod bow. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

Shoulder gap adjustments for tolerances, guide tube growth, differential thermal expansion, and rod growth are done statistically to determine the lower 95 percent shoulder gap. The evaluation is based upon growth data available to date, along with an ongoing commitment to collect PIE data and validate predictions (See Section 3.1). Based upon these calculations and the ongoing commitment to validate growth predictions, the NRC staff finds the NGF fuel rod design acceptable with respect to fuel rod axial growth and shoulder gap.

3.3.9 Fuel Materials

The treatment of the ZrB2 coating on the fuel pellet was previously addressed in Reference 11. The treatment of Optimized ZIRLO™ and its material and mechanical properties was previously addressed in Reference 9.

3.3.10 Burnable Absorbers

The utilization of ZrB2 coating, Gd2O3, and Er2O3 burnable absorbers have been previously addressed with respect to application in CE plants (See Section 2.5.10 of TR WCAP-16500-P). Implementation of these IFBAs within CE 16x16 NGF designs should be in compliance with any and all SE limitations and conditions imposed during their past approvals.

3.3.11 Pellet Cladding Interaction

The Westinghouse criterion is that the fuel system will not be damaged due to excessive pellet-cladding interaction. As indicated in Section 4.2 of the SRP, there are no generally applicable criteria for PCI failure. However, two acceptable criteria of limited application are presented in the SRP for PCI: 1) less than 1 percent transient-induced cladding strain, and 2) no centerline fuel melting. Both of these criteria were addressed above in Sections 3.3.2 and 3.3.4, respectively.

3.4 Fuel Rod Average Burnup Limit - 62 Mwd/kgU

Section 2.6 of TR WCAP-16500-P provides justification for a burnup limit of 62 MWd/kgU for the CE 16x16 NGF design. Current CE fuel assembly designs are limited to 60 MWd/kgU. In response to RAI 5 of Reference 2 regarding the validity of the methods up to the proposed burnup for the different fuel rod configurations, Westinghouse provided information on its physics and fuel performance models. Current physics and fuel performance models and methods have been previously reviewed and approved for the different fuel rod designs (e.g., Erbia, Gadolinium, ZrB2). Based upon the fuel assembly and fuel rod design evaluations provided in TR WCAP-16500-P and RAI responses, the NRC staff finds the extension in burnup from 60 MWd/kgU to 62 MWd/kgU acceptable. This burnup extension applies to the fuel assembly mechanical design and to the analytical methods used to evaluate this fuel design. A fuel burnup limit may exist, either explicitly or implicitly, in other portions of a plant's licensing basis. For example, a limit on fuel assembly burnup may be implicit in the reported fuel handling accident dose consequences, fuel burnup may be implicit in the spent fuel pool criticality analysis, or a burnup limit may be explicitly stated in the plant's environmental impact statement. Further, the NRC staff's SE for Optimized ZIRLO™ (Addendum 1 to TR WCAP-12610-P-A and TR CENPD-404-P-A) specified a 60 MWd/kgU burnup limit and this limitation must be revised prior to extending the peak rod average burnup for the NGF design.

The NRC staff's approval of this TR allows the CE 16x16 NGF assembly to reach a rod average burnup of 62 MWd/kgU. However, the licensee may need to address other portions of its license prior to extending burnup beyond current levels.

3.5 Nuclear Design

Section 3.0 of TR WCAP-16500-P describes the impact of the CE 16x16 NGF assembly design on the currently approved nuclear design models and methods. The major change affecting the nuclear design characteristics is the change in fuel pellet and fuel rod clad diameter. Other primary parameters such as fuel assembly pitch, fuel rod pitch, and burnable absorbers are unchanged. In Section 3.2 of TR WCAP-16500-P, Westinghouse concludes that no changes to currently approved models and methods are required to design and analyze cores containing CE 16x16 NGF assemblies.

In response to a NRC staff inquiry regarding the impact of the change in fuel pin diameter on current physics biases and uncertainties, Westinghouse stated that the PARAGON benchmarks show no significant difference in measured-to-predicted errors between plants with different rod diameters (Reference 7). Westinghouse committed to continue updating physics uncertainties as necessary to maintain accuracy with measurements. The NRC staff finds this acceptable.

The NRC staff agrees that the current models and methods are capable of analyzing the NGF assemblies.

3.6 Thermal and Hydraulic Design

Section 4.0 of TR WCAP-16500-P describes the thermal-hydraulic evaluation of the CE 16x16 NGF assembly design. The addition of side supported mixing vanes on the mid grids and the introduction of Intermediate Flow Mixer (IFM) grids have a significant impact on DNB thermal margin.

3.6.1 DNB Design Basis

The Westinghouse criterion is that there will be at least a 95 percent probability at a 95 percent confidence level that DNB will not occur on the limiting fuel rods during Condition I and II events. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The Critical Heat Flux (CHF) correlations associated with the non-vaned and vaned grids (e.g., ABB-NV and WSSV-T) have been previously reviewed and approved. These correlations will be implemented in the current core thermal-hydraulics codes (VIPRE-01, TORC, and CETOP-D) and used to calculate the departure from nucleate boiling ratio (DNBR). As part of the audit, Westinghouse presented the nodalization of the CETOP-D and TORC models and described the calculation of DNBR. The NRC staff agrees that the methods used for calculation of DNBR are acceptable.

3.6.2 Fuel Assembly Holddown Force

The Westinghouse criterion is that the fuel assembly will not lift and will remain in contact with the lower core plate under all Condition I and II events. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

The design evaluation calculated the net force on the assembly including the downward force of the holddown springs, the weight of the assembly, and the upward forced flow and buoyancy forces. The evaluation concluded that sufficient holddown force is available to maintain the assembly seated on the lower core plate. Each licensee will verify holddown force based on

plant-specific conditions. Based upon this design evaluation and plant-specific verification, the NRC staff finds the NGF fuel assembly design acceptable with respect to holddown force.

3.6.3 Thermohydrodynamic Stability

The Westinghouse criterion is that thermohydrodynamic instability will not occur under Condition I and II events. This design basis is consistent with SRP Section 4.2 and, therefore, is acceptable for application to the CE 16x16 NGF design.

Employing methods consistent with staff review of other Westinghouse fuel designs, thermohydrodynamic stability was evaluated and found unlikely to occur during Condition I and II events. Based upon this design evaluation, the NRC staff finds the NGF fuel assembly design acceptable with respect to thermohydrodynamic stability.

3.7 Accident Analysis

Section 5.0 of TR WCAP-16500-P describes the impact of the NGF design features on non-LOCA and LOCA analyses. The incorporation of mixing vanes, addition of IFM grids, and change in fuel rod dimensions will have a significant impact on performance and need to be explicitly addressed in the accident analyses. Impacts related to Optimized ZIRLO™ fuel rod cladding and IFBA fuel rod designs have been previously addressed.

During an audit, the NRC staff questioned the DNB degradation experienced during the loss-of-flow (LOF) event. The NRC staff was concerned that the reduction in forced flow (during the LOF transient) may result in a larger degradation in vaned assemblies (than non-vaned designs). The Westinghouse LOF analysis calculated identical required overpower margin for both designs. In response to the NRC staff's concerns, Westinghouse provided a plot of measured CHF as a function of local mass velocity from the CHF testing program (Reference 6). Examination of this figure reveals that the trend in CHF is parallel between the two correlations in the LOF range (2.6 - 2.1 Million pounds mass per hour per feet squared (Mlbm/hr-ft2)). The degradation (per unit mass flow) is larger at reduced flows. However, this occurs well after reactor trip (for the LOF event). This demonstrates that, all else equal, the DNB margin degradation will be approximately the same between the two correlations. Based on the information presented in Reference 6, the NRC staff finds the LOF transient acceptable.

In general, the non-LOCA system transient codes (e.g., CENTS, CESEC-III, RETRAN) are not sensitive to details of the fuel assembly design, but are capable of being calibrated to match fuel performance aspects. The details of the fuel design are captured by the core thermal-hydraulics models which explicitly model the fuel assembly subchannels. Section 5.1.3 provides a qualitative evaluation of the non-LOCA accidents; however, plant-specific non-LOCA accident analyses may be required by each applicant.

Appendix A of TR WCAP-16500-P, describing changes to the large-break LOCA (LBLOCA) model, was reviewed separately and is not documented in this SE. Aspects of the NGF assembly design (e.g., hydraulic pressure loss, fuel rod diameter) necessitate re-analysis of the emergency core cooling system performance analysis. Plant-specific calculations are necessary and will be performed to capture the impact of the NGF assembly design. Due to the requirement of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.46, licensees are required to submit their revised LOCA analyses for review. During transition

reload cycles, the effects of mixed fuel assembly design cores must be specifically addressed. Upon final approval, the full-core NGF LOCA analyses will constitute the analysis-of-record and become the baseline which future errors or model changes will be measured against, in accordance with 10 CFR 50.46(a)(3).

Section 5.3 of TR WCAP-16500-P briefly describes the impact of the NGF assembly on the reload setpoints methodology. It concludes that the previously approved analog setpoint methodology (TR WCAP-8745-P-A), using the VIPRE-01 thermal-hydraulics code with the approved NGF CHF correlations, may be applied to reload cores with CE 16x16 NGF assemblies. The NRC staff finds this approach acceptable.

With respect to the digital setpoints process, Westinghouse concluded that the standard reload uncertainty methodology (MSCU) will provide appropriate uncertainty factors for the on-line systems COLSS and CPCS such that the DNB design bases are maintained. During its review, the NRC staff identified concerns with the application of the MSCU methods to NGF reloads where the CHF correlation within COLSS and CPCS were inconsistent with the axial-dependent CHF correlations of the NGF design. Specifically, the two NGF CHF correlations each have the potential to introduce separate temperature-dependent, pressure-dependent, and flow-dependent biases as a function of axial power shape.

In response to NRC staff concerns, Westinghouse submitted Supplement 1-P (Reference 5) which documented, in more detail, the setpoints process with respect to the NGF design. Supplement 1 to TR WCAP-16500-P states:

> "The overall uncertainty factors determined using the MSCU methodology described in Reference 1 [TR CEN-356(V)-P-A Revision 01-P-A, May 1988] and the MSCU process as modified to reflect the NGF design and CHF correlations continue to ensure that the COLSS DNB POL calculations and the CPCS DNBR calculations will be conservative to at least a 95% probability and 95% confidence [95/95] level."

During a subsequent audit, Westinghouse presented additional material to address NRC staff concerns (Reference 6). Examination of Slides #10 through #15 of Reference 6 reveal the temperature-, pressure-, and flow-dependent biases in the core thermal-hydraulic calculation which are the result of differences between the ABB-NV and WSSV-T CHF correlations inherent in the NGF design and the CE-1 CHF correlation (which is fixed within COLSS and CPCS algorithms). This bias would be introduced into the MSCU setpoints process in the statistical comparison between the "truth" and the randomly-perturbed on-line algorithm calculations. As a result, the calculated penalty factor would be biased and may not preserve the 95/95 level of protection.

As part of the revised setpoints process proposed by Westinghouse, both the COLSS and CPCS analysis range will be subdivided into 64 portions (1/4 range of flow, pressure, and temperature) and analyzed to produce the most conservative uncertainty factors (e.g., BERR1 and EPOL2). The limiting 1/64 hypercube would then be analyzed over a range of axial power distributions to determine Axial Shape Index (ASI) dependent penalty factors. The NRC staff's concerns with the analytical process defined in Response No. 6 of Reference 6 include the following:

1. As illustrated in the comparison of Slide #6 (DNB Power Operating Limit (POL) uncertainty versus flow) and Slide #14 (CETOP-D POL versus flow), the absolute bias (resulting from the inconsistent CHF correlations) may be diminished when combined with the randomly applied variables in the overall uncertainty analysis.

2. Due to the proposed statistical treatment of the biases, a small portion of the analytical range may not be guaranteed 95/95 protection. The 1/64 hypercube approach limits both the operating space exposed and the magnitude of any potential bias. However, this approach still does not absolutely ensure the 95/95 protection provided for current reload cores.

3. It is unclear how the distinct axial regions (WSSV-T and ABB-NV) will be treated with respect to the temperature-, pressure-, and flow-dependent biases and the overall uncertainty analysis. These two populations may not be poolable nor may they be treated as a single normal distribution. At the end of the analytical steps defined in Reference 6, Westinghouse concludes that "these steps have not been tested in detail" and that the steps "may have to be adjusted in order to assure conservative results at 95/95". The NRC staff has determined that the modified analytical process needs to be thoroughly tested and documented by Westinghouse, and reviewed by the staff.

Based upon these concerns, the NRC staff is unable to conclude that the proposed digital setpoints methodology is (1) consistent with the currently approved methods and (2) will preserve the required 95/95 protection level when applied to the NGF assemblies.

The NGF assembly design offers many advanced features which will benefit fuel performance. In order to allow batch implementation while the setpoint issues are being resolved, the NRC staff concludes that an interim DNB margin penalty shall be imposed. Re-examining the material presented in Supplement 1 (Reference 5) and during the subsequent audit (Reference 6), the NRC staff estimated that a heat flux penalty of 6 percent, in combination with the 1/64 hypercube setpoints process (Response No. 6 of Reference 6), would be sufficient to compensate for the above issues. The 6 percent heat flux penalty should be applied to the final addressable constants (e.g., BERR1* 1.06, [(1+EPOL2)*1.06 - 1.0]). Removal of this interim margin penalty will be considered after the digital setpoints methods have been formalized, documented (e.g., revision to TR WCAP-16500-P), and approved by the NRC staff.

3.8 Reactor Vessel and Internals (RVI) Evaluation

Section 6.0 of TR WCAP-16500-P describes the impact of the NGF design on the reactor vessel and reactor internals design evaluations. Plant-specific analyses for RVI thermal-hydraulic performance, seismic and pipe break response, and structural evaluations will be performed, using currently approved methodology, to demonstrate that design criteria are met. The NRC staff finds this acceptable.

The ability to insert control rods and scram within the time requirements assumed within the safety analysis is crucial. While the NGF design maintains the same interface configuration with the control rods as the standard 16x16 CE assembly design, the NGF's increased pressure drop has the potential to lengthen scram times. Plant-specific CEA scram time analysis will confirm insertion time criteria. Further, Technical Specification surveillance requirements will be

performed, as usual, and will confirm control rod drop times. The NRC staff finds this acceptable.

3.9 Radiological Assessment

Section 7.0 of TR WCAP-16500-P documents the impact of the NGF assembly design on the accident radiological assessments. The NGF rod and assembly uranium loadings are not significantly different from the current fuel assembly design. In addition to uranium loading, burnup, and power history will continue to be evaluated against bounding assumptions in the plant-specific dose calculations. The transition to the NGF fuel design will not significantly impact the LOCA, non-LOCA, or fuel-handling accident source terms. However, any change in licensed burnup limits will need to be specifically addressed by the licensee.

3.10 Mixed Core Evaluation

Differences in assembly component design and hydraulic resistance between the NGF and the standard CE 16x16 fuel bundle may impact all aspects of the reload design and thus must be properly addressed. With respect to fuel mechanical design, Westinghouse has performed dual assembly tests in the VIPRE test loop to evaluate FIV and fuel rod wear during transition cycles. In response to RAI 4.b of Reference 2, Westinghouse provided further information on mixed-core FIV evaluations. While these tests confirmed that co-resident fuel satisfy fuel design criteria, it is not possible to dismiss fuel rod fretting and potential fuel rod failures during transition cycles. As the NGF design occupies a higher percentage of the core, core flow will be preferentially directed toward the remaining standard CE design (due to lower pressure drops). This increased flow will be amplified within peripheral assembly locations (due to higher core bypass flow). Hence, any plant currently experiencing fuel rod fretting damage may expect further problems during transition cores. Plant technical specifications limiting reactor coolant system (RCS) activity ensure that this transition core effect does not introduce a public safety concern.

With respect to core thermal-hydraulics analyses, both VIPRE-01 and TORC have detailed models to capture axial-dependent and radial-dependent differences in assembly component designs and hydraulic resistance. Details of the TORC code's modeling capability was presented during the January 30-31, 2007 audit (Reference 3). Westinghouse performed testing in the FACTS loop to measure pressure drop characteristics across the entire assembly and individual components. Further, dual assembly testing in the VIPRE loop confirmed computed flow splits between the co-resident assemblies. While these detailed core thermal-hydraulic models have good capabilities, and pressure drop characteristics have been measured under controlled test conditions, licensees must consider the uncertainty associated with predicting local flow characteristics in a mixed-core environment (Limitation and Condition No. 6).

The LOCA and non-LOCA simulation codes do not have the same level of detail with respect to capturing assembly-specific characteristics in a mixed core. The potential impact of a mixed core on these analyses would be plant-specific and cycle-specific. As a result, licensees must detail the analytical methods and results of their transition core LOCA and non-LOCA analyses.

4.0 LIMITATIONS AND CONDITIONS

Licensees referencing TR WCAP-16500-P (along with Supplement 1-P) must ensure compliance with the following conditions and limitations:

1. Using approved methods, the licensee must ensure that all of the design criteria specified in TR WCAP-16500-P are satisfied on a cycle-specific basis (SE Section 3.3.1).

2. Fuel assembly component design and configuration (e.g., type and distribution of spacer grids and IFM grids) are limited to the five designs described in TR WCAP-16500-P and in response to RAI 2 (SE Section 3.2).

3. The reference fuel assembly design, CE 16x16 NGF, its fuel mechanical design methodology and design criteria, are approved up to a peak rod average burnup of 62 GWd/MTU. A fuel burnup limit may exist, however, either explicitly or implicitly, in other portions of a plant's licensing basis. The NRC staff's approval of this topical report allows the CE 16x16 NGF assembly to reach a rod average burnup of 62 GWd/MTU. However, a license amendment request, specifically addressing each plant's licensing basis including radiological consequences, is required prior to extending burnup beyond current levels. Further, the NRC staff's SE for Optimized ZIRLO™ (Addendum 1 to TR WCAP-12610-P-A and TR CENPD-404-P-A) specified a 60 MWd/kgU burnup limit and this limitation must be revised prior to extending the peak rod average burnup for the NGF design (SE Section 3.4).

4. Licensees shall demonstrate the accuracy of their growth predictions based upon measured data and this validation shall be ahead of the burnups achieved by batch implementation. The growth model validation (e.g., measured versus predicted) should be documented in a letter(s) to the NRC (SE Section 3.2.1).

5. To compensate for NRC staff concerns related to the digital setpoints process, an interim margin penalty of 6 percent must be applied to the final addressable constants (e.g., BERR1* 1.06, [(1+EPOL2)*1.06 - 1.0]) calculated following the 1/64 hypercube setpoints process (Response No. 6 of Reference 6). Removal of this interim margin penalty will be considered after the digital setpoints methods have been formalized, documented (e.g., revision to TR WCAP-16500-P), and approved by the NRC (SE Section 3.7).

6. Licensees are required to demonstrate that during transition cores, DNB margin gains associated with the NGF design offset (1) any impacts of flow starvation due to increased pressure drop and (2) uncertainty associated with predicting local flow characteristics. Further, licensees must detail the analytical methods and results of their transition core LOCA and non-LOCA analyses (SE Sections 3.7 and 3.10).

7. Implementation of CE 16x16 NGF assemblies necessitate re-analysis of the plant-specific LOCA analyses. Licensees are required to submit a license amendment containing the revised LOCA analyses for NRC review. Upon approval, the revised LOCA analyses constitute the analysis-of-record and baseline for which future changes will be measured against in accordance with 10 CFR 50.46(a)(3) (SE Section 3.7).

8. Using approved models and methods, Westinghouse will continue to limit peak local power experienced during Condition I and II events to ensure that fuel temperature remains below melting temperature at all burnups. This evaluation may be both plant and cycle-specific (SE Section 3.3.4).

9. The NRC staff's approval of TR WCAP-16500-P establishes the licensing basis for batch implementation of the CE 16x16 NGF assembly design. Licensees wishing to implement this fuel design are required to submit a license amendment request, where applicable, updating their Core Operating Limits Report list of methodologies with the "A" version of this TR.

10. The NRC staff's review did not include the LOCA model changes described in Appendix A of TR WCAP-16500-P. Therefore, a licensee, will have to submit a license amendment, if they desire to use The Appendix A LOCA model changes.

5.0 CONCLUSIONS

The reference CE 16x16 NGF design reviewed by the NRC staff meets design and regulatory requirements. Plant-specific and cycle-specific evaluations are required to ensure that allowable variances of this assembly design continues to satisfy all criteria.

Based upon its review of TR WCAP-16500-P, Supplement 1-P, and RAI responses, the NRC staff finds the CE 16x16 NGF assembly design, fuel design criteria, and supporting fuel mechanical and reload design methodology acceptable subject to the conditions and limitations listed in Section 4.0.

6.0 REFERENCES

1. WCAP-16500, Revision 0, "Submittal of WCAP-16500-P/WCAP-16500-NP, 'CE 16x16 Next Generation Fuel, Core Reference Report,'" LTR-NRC-06-04, February 28, 2006, (ADAMS Package Accession No. ML060670508).

2. Letter, J. A. Gresham (W) to USNRC, "Response to NRC's Request for Additional Information By the Office Of Nuclear Reactor Regulation Topical Report WCAP-16500-P, 'CE 16x16 Next Generation Fuel Core Reference Report,'" LTR-NRC-06-66, November 29, 2006 (ADAMS Accession No. ML063400056).

3. Letter, J. A. Gresham (W) to USNRC, "Slide Presentation in Support of NRC Audit on WCAP-16500-P, 'CE 16x16 Next Generation Fuel Core Reference Report' and Clarification of RAI Responses to Questions 1a, 4a and 10," LTR-NRC-07-6, January 29, 2007 (ADAMS Accession No. ML070470485).

4. Letter, J. A. Gresham (W) to USNRC, "Further Clarification to RAI Response 7 and 10 for CE 16x16 Next Generation Fuel Core Reference Report WCAP-16500-P," LTR-NRC-07-8, February 15, 2007 (ADAMS Accession No. ML070530113).

5. Letter, B. F. Maurer (W) to U.S. Nuclear Regulatory Commission, "Supplement 1-P to WCAP-16500-P, 'Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)," LTR-NRC-07-13, March 16, 2007 (ADAMS Accession No. ML070860997).

6. Letter, B. F. Maurer (W) to USNRC, "Presentation Material on Audit Responses to Questions on Setpoints Supplement 1-P to WCAP-16500-P," LTR-NRC-07-20, April 5, 2007 (ADAMS Accession No. ML071030085).

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

8. WCAP-15942-P-A, "Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors Supplement 1 to TR CENP-287," March 31, 2006 (ADAMS Accession No. ML061110244).

9. Letter, J. A. Gresham (W) to USNRC, "SER Compliance with WCAP-12610-P-A & CENPD-404-P-A Addendum 1-A 'Optimized ZIRLO™," LTR-NRC-07-1, January 4, 2007 (ADAMS Accession No. ML070100385).

10. WCAP-12610-P-A and TR CENPD-404-P-A Addendum 1-A, "Optimized ZIRLO™," July 2006 (ADAMS Accession No. ML062080563).

11. WCAP-16072-P-A, "Implementation of Zirconium Diboride Burnable Absorber Coating in CE Nuclear Power Fuel Assembly Designs," August 2004 (ADAMS Accession No. ML042510056).

12. CEN-386-P-A, "Verification of the Acceptability of a 1-pin Burnup Limit of 60 Mwd/kgU for Combustion Engineering 16x16 PWR Fuel," August 1992.

13. Studsvik Laboratory, "PIE of North Anna PWR Rods: Results from hydrogen analyses, ring tensile tests and axial tensile tests," N(H)-03/014, July 30, 2003.

Attachment: Resolution of Comments

Principle Contributor: P. Clifford

Date: July 30, 2007

RESOLUTION OF WESTINGHOUSE ELECTRIC COMPANY (WESTINGHOUSE)

COMMENTS ON DRAFT SAFETY EVALUATION FOR TOPICAL REPORT (TR) WCAP-16500,

REVISION 0, "CE [COMBUSTION ENGINEERING] 16X16 NEXT GENERATION FUEL [(NGF)]

CORE REFERENCE REPORT"

By letter dated June 21, 2007, Westinghouse provided sixteen comments on the Draft Safety Evaluation for TR WCAP-16500, Revision 0, "CE 16x16 Next Generation Fuel Core Reference Report." The following are the NRC staff's resolution of these comments:

Westinghouse Comment 1 (Page 2, Line 1):

Missing Punctuation:
After "Section 4.2 criteria," add a period.

NRC Resolution for Comment 1:

The proposed change is adopted.

Westinghouse Comment 2 (Page 3, Lines 20-21):

Clarification:
It is assumed that the commitment referred to is Limitation and Condition #4. Suggest adding a parenthetical statement (i.e., Limitation and Condition #4)" after the word models.

NRC Resolution for Comment 2:

The proposed change is adopted.

Westinghouse Comment 3 (Page 3, Lines 34-35):

Clarification:
Suggest adding to the statement as follows: "Changes in the fuel assembly design beyond those permitted by 10 CFR 50.59 would require NRC review."

NRC Resolution for Comment 3:

The sentence is amended to read: "Therefore, changes in the CE 16x16 NGF fuel assembly design may require NRC review prior to implementation."

Westinghouse Comment 4 (Page 4, Lines 3-4):

Proprietary Statement:
Suggest revising the wording to "an adjustment factor applied to the best-estimate."

ATTACHMENT

NRC Resolution for Comment 4:

The proposed change is adopted.

Westinghouse Comment 5 (Page 4, Line 30):

Proprietary Statement:
Suggest deleting the parenthetical statement.

NRC Resolution for Comment 5:

The proposed change is adopted.

Westinghouse Comment 6 (Page 8, Lines 3-5):

Incomplete sentence:
Suggest rewording and merging the second and third sentences as follows: "Compared to the current CE fuel rod design, the UO_2 NGF design has less void volume; whereas, the ZrB_2 NGF fuel rod may be designed with higher void volume (i.e., with the use of annular axial blanket pellets) to accommodate the production of helium gas."

NRC Resolution for Comment 6:

The proposed change is adopted.

Westinghouse Comment 7 (Page 9, Lines 44-45):

Proprietary Statement:
Suggest deleting the sentence.

NRC Resolution for Comment 7:

The proposed change is adopted.

Westinghouse Comment 8 (Page 12, Lines 47-48):

Missing concluding statement:
All of the evaluation sub-sections have a concluding statement, except for this section. Suggest the following sentence be added: "The NRC staff agrees that the methods used for calculation of DNBR are acceptable."

NRC Resolution for Comment 8:

The following sentence was added. "The NRC staff agrees that the current models and methods are capable of analyzing the NGF assemblies."

Westinghouse Comment 9 (Page 13, Lines 33-34):

Incorrect transient:
LOF refers to Loss-of-Flow transient, not Feedwater. This discussion is associated with the Loss-of-Flow event. Suggest correcting the transient naming.

NRC Resolution for Comment 9:

The proposed change is adopted.

Westinghouse Comment 10 (Page 14, Lines 1-3):

Clarification:
Non-LOCA analyses are not impacted directly by the fuel design, but can be impacted indirectly by peaking factor changes or system parameter changes. Suggest modifying the last part of this sentence to "however, additional plant specific evaluation/analyses may be necessary if other changes are made to the plant."

NRC Resolution for Comment 10:

The last part of this sentence is amended to read: "however, the evaluation/analyses conducted in Section 5.1.3 may not be sufficient and plant-specific non-LOCA accident analyses may be required by each applicant."

Westinghouse Comment 11 (Page 14, Lines 13-15):

Fragmented sentence:
Suggest the following wording: "Upon final approval, the full-core NGF LOCA analyses will constitute the analysis-of-record and become the baseline which future errors or model changes will be measured against, in accordance with 10 CFR 50.46(a)(3)."

NRC Resolution of Comment 11:

The proposed change is adopted.

Westinghouse Comment 12 (Page 15, Lines 37-38):

Incomplete sentence:
Suggest merging the second and third sentence as follows: "In order to allow batch implementation while the setpoint issues are being resolved, the NRC staff concludes that an interim DNB margin penalty shall be imposed."

NRC Resolution of Comment 12:

The proposed change is adopted.

Westinghouse Comment 13 (Page 17, Lines 4-5):

Clarification:
On Page 16, Lines 47-48 and on Page 17, Lines 1-2, the testing of the fuel assemblies is documented. The statement on Page 17, Lines 4-5, "licensees must consider a potential increase in the uncertainty associated with predicting local flow characteristics," does not appear to be an appropriate or needed statement for this SE. Recommend deleting the last sentence in this paragraph, Lines 2-6.

NRC Resolution of Comment 13:

The text on Page 17, Lines 4-5 has been amended to clarify the NRC staff's meaning to read: "licensees must consider the uncertainty associated with predicting local flow characteristics in a mixed core environment (Limitation and Condition No. 6)."

Westinghouse Comment 14 (Page 17, Lines 10-11):

Clarification:
Delete "and non-LOCA." Non-LOCA analyses are not impacted directly by the fuel. A transition core analysis would not be required.

NRC Resolution for Comment 14:

The proposed change is not adopted. This text remains unchanged. The NRC staff has a concern regarding the potential impact of fuel assembly design changes on non-LOCA accident analyses.

Westinghouse Comment 15 (Page 18, Lines 6-7):

Clarification:
Same comment as above. Recommend deleting "and (2) any increase in uncertainty associated with predicting local flow characteristics."

NRC Resolution for Comment 15:

The referenced text will be changed to: "and (2) uncertainty associated with predicting local flow characteristics." The text regarding non-LOCA accident analyses remains as discussed above.

Westinghouse Comment 16 (Page 18, Lines 17-20):

Clarification:
Recommend deleting the Limitation and Condition since this design criterion must be met to comply with TR WCAP-16500-P. If this criterion is to be spelled out in the Limitation and Conditions, then all the other design criterion should be considered and listed.

NRC Resolution for Comment 16:

The proposed change is not adopted. The NRC staff considers this criterion to be of sufficient importance to list as a limitation and condition. This text remains unchanged.

Note: The resolution of Westinghouse comments 4, 5, & 7 involved the removal of proprietary information from language contained in the draft Safety Evaluation for TR WCAP-16500-P, Revision 0. As such, the draft Safety Evaluation for this TR contains proprietary information and will remain non-public. The final Safety Evaluation for WCAP-16500-P, Revision 0 contains no proprietary information and is therefore publicly available.

Section B

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Attention: F. M. Akstulewicz, Chief
Nuclear Performance & Code Review Branch
Division of Safety Systems

Our ref: LTR-NRC-06-4

February 28, 2006

Subject: Submittal of WCAP-16500-P/WCAP-16500-NP, "CE 16x16 Next Generation Fuel, Core Reference Report," (Proprietary/Non-proprietary)

Dear Mr. Akstulewicz:

Enclosed are 5 Proprietary and 3 Non-Proprietary copies of WCAP-16500-P/WCAP-16500-NP, "CE 16x16 Next Generation Fuel, Core Reference Report," submitted to the NRC for review and approval. It is requested that the above topical be approved by February 2007. It is also requested that the NRC provide an estimate on the man-power resources required for the review and a tentative date for the acceptance meeting.

WCAP-16500-P/WCAP-16500-NP describes and presents the generic fuel assembly design for CE 16x16 Next Generation Fuel (NGF) and the methods and models used for evaluating its acceptability to CE NSSS plants.

Also enclosed are:

1. One (1) copy of the Application for Withholding, AW-06-2107 with Proprietary Information Notice and Copyright Notice.
2. One (1) copy of Affidavit, AW-06-2107.

This submittal contains Westinghouse proprietary information of trade secrets, commercial or financial information which we consider privileged or confidential pursuant to 10 CFR Section 2.390. Therefore, it is requested that the Westinghouse proprietary information attached hereto be handled on a confidential basis and be withheld from public disclosure.

Correspondence with respect to the affidavit or Application for Withholding should reference AW-06-2107 and should be addressed to B. F. Maurer, Acting Manager, Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P.O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

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

Enclosures

cc: G. S. Shukla, NRR
L. M. Feizollahi, NRR

Westinghouse

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Attention: F. M. Akstulewicz, Chief
Nuclear Performance & Code Review Branch
Division of Safety Systems

Our ref: AW-06-2107

February 28, 2006

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: Submittal of WCAP-16500-P, "CE 16x16 Next Generation Fuel, Core Reference Report," (Proprietary)

Reference: Letter from B. F. Maurer to F. M. Akstulewicz, LTR-NRC-06-4, dated February 28, 2006

Dear Mr. Akstulewicz:

The application for withholding is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-06-2107 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-06-2107 and should be addressed to B. F. Maurer, Acting Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

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

AW-06-2107

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared B. F. Maurer, 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:

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

Sworn to and subscribed
before me this 28th day
of February, 2006

Sharon L. Fiori
Notary Public

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

AW-06-2107

(1) I am Acting Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse) and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking 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.

AW-06-2107

(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 which 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.

AW-06-2107

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in Submittal of WCAP-16500-P, "CE 16x16 Next Generation Fuel, Core Reference Report," (Proprietary)," February 2006, for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-06-4) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company is for NRC review and approval.

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

(a) Obtain generic NRC licensed approval for the Westinghouse CE 16x16 Next Generation Fuel assembly design and applicable methodology and models.

(b) Assist customers in improving their fuel performance (zero defects).

Further this information has substantial commercial value as follows:

(a) Westinghouse plans to continue to implement corrective actions to ensure the highest quality of fuel in order to meet the customer needs.

(b) Assist customers to obtain license changes.

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 technical evaluation justifications 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 for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

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Table of Contents
CE 16x16 Next Generation Fuel
Core Reference Report

Table of Contents (cont.)
CE 16x16 Next Generation Fuel
Core Reference Report

Table of Contents (cont.)
CE 16x16 Next Generation Fuel
Core Reference Report

Table of Contents (cont.)
CE 16x16 Next Generation Fuel
Core Reference Report

List of Tables
CE 16x16 Next Generation Fuel
Core Reference Report

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List of Figures
CE 16x16 Next Generation Fuel
Core Reference Report

List of Figures (cont.)
CE 16x16 Next Generation Fuel
Core Reference Report

1.0 Introduction and Summary

1.1 Introduction

This topical report describes the CE 16x16 Next Generation Fuel (CE 16x16 NGF) assembly design and the methods and models used for evaluating its acceptability.

The driving forces and goals of the CE 16x16 NGF design include improving fuel reliability to resolve grid to rod fretting failures, improving fuel performance for high duty operation, and providing enhanced margin. The significant design features for the CE 16x16 NGF design include:

- A top Inconel grid to improve fretting margin at that axial location relative to a top Zircaloy grid
- Advanced Mid grids with "I" spring rod supports and side supported mixing vanes at selected elevations to improve fretting and thermal margin
- Intermediate Flow-Mixing (IFM) grids to improve fuel thermal performance
- Optimized ZIRLO™ material for cladding and ZIRLO™ material (including low tin ZIRLO™) for guide tubes and grid straps to improve corrosion resistance and dimensional stability
- Advanced 0.374" OD rod to accommodate the higher pressure drop of the Mid and IFM grids
- Axial blankets (solid or annular pellet fuel) and ZrB_2 Integrated Fuel Burnable Absorbers (also referred to as IFBA) fuel rods to improve fuel cycle economics
- Guardian™ grid with solid lower fuel rod end plug to provide enhanced performance with respect to both debris and non-debris related fretting at the bottom grid elevation

Some of the features described above: top Inconel grid, advanced Mid grids, axial blankets, ZrB_2 IFBA fuel rods, and Guardian™ grids have already been implemented and are operating successfully in CE NSSS units.

The CE 16x16 NGF fuel rod may incorporate burnable absorber variations to meet specific rod internal pressure requirements based on burnup and power level conditions. The primary burnable absorber will be ZrB_2; however, NGF may also include other burnable absorbers such as Erbia or Gadolinia in the fuel assembly. These design aspects will be addressed as needed in plant specific evaluations.

This topical report provides a licensing basis for evaluating the CE 16x16 NGF fuel assembly design and, once approved, will serve as the basis for applications incorporating CE 16x16 NGF design features into any of the CE 16x16 plants. Plant specific analyses/evaluations will need to be done for each initial application of CE 16x16 NGF. These analyses/evaluations will address the transition core effects from the co-resident fuel (referred to as CE 16x16 Standard Fuel) to a full core of CE 16x16 NGF. The licensing basis for the CE 16x16 Standard Fuel design includes References 7, 8, 18, and 19. Any changes to this licensing basis for implementing NGF in CE 16x16 plants will be defined in this report.

To facilitate regulatory review, this topical report contains a cross-reference of the CE 16x16 NGF design evaluation with the Standard Review Plan 4.2 – Fuel System Design given in the NRC Standard Review Plan (NUREG 0800)[2], refer to Table 1-1. In addition, where appropriate, reference is made to prior NRC approvals or where an NGF feature has been previously applied in an operating reactor.

The report is organized along functional lines, consistent with the sub-chapters of a typical FSAR (i.e., Section 2.0 - Mechanical Design, Section 3.0 - Nuclear Design, Section 4.0 - Thermal and Hydraulic Design, Section 5.0 - Accident Analyses – Non-LOCA and LOCA, Section 6.0 - Reactor Vessel and Internals Evaluation, and Section 7.0 - Radiological Assessment) which support the CE 16x16 NGF design.

The CE 16x16 NGF design, licensing bases, and criteria as described in this report have been reviewed with respect to the individual NSSS plant conditions where the CE 16x16 design may be utilized and the licensing bases and criteria have been found to be generically applicable. Plant specific analyses will be performed to confirm the acceptability of the NGF design prior to implementation as a part of the standard reload process.

A brief summary of the CE 16x16 NGF design follows. A comparison of the Standard and NGF features is given in Table 1-2 to help identify the improvements made to NGF relative to the current Standard fuel. The NGF features and figures, illustrating the design details, are presented in Section 2.0.

A top Inconel grid was introduced in NGF fuel compared to a Zircaloy-4 one in Standard fuel to improve fretting margin, since grid to rod fretting failures have occurred in CE plants at that axial location (Note: some CE 16x16 designs have already implemented a similar Inconel top grid to address this issue). This grid is equivalent to the design used in Westinghouse plants to date where extensive experience has been obtained with no fretting failures at the top grid location. The Inconel grid rod supports firmly hold the fuel rod throughout life but still allow the fuel rods to grow vertically, thus reducing fuel rod bowing.

The upper nozzle of the NGF assembly is similar to the Standard assembly except [

] a, c The thimbles tubes are attached to the top guide tube flange by making a double bulge joint.

The Mid grids design features an "I-spring" rod support system and side supported mixing vanes. This grid was specifically developed to improve the grid-to-rod fretting margin over current Standard fuel at Mid grid locations and to improve thermal margin and heat transfer performance. This design, adopted from the CE 14x14 Turbo grid design has been demonstrated in reactor to improve fretting margin[3]. The grid material is low tin ZIRLO™ (to improve corrosion resistance) and the straps are stamped in the [ ] a, c The rod supports also alternate at each grid elevation like the current Standard grid design to maintain fuel rod stability. Extensive CHF testing of the new Mid grid was done at the Columbia University Heat Transfer Research Facility (HTRF). The CHF data and a corresponding new DNB correlation for the NGF design is being submitted to the NRC in a separate topical report[4].

Intermediate Flow Mixing (IFM) Grids have been added in the NGF assembly to improve thermal performance for selected grid span locations. The IFM grids use the same side supported mixing vanes as the Mid grids. The IFM grids contain [ ] ^{a,c} in each grid cell and the grid straps are [ ] ^{a,c} like the Mid grids. The grid strap material is low tin ZIRLO™.

The outer straps on the Mid and IFM grids are designed to assure strength in the grid corner region like the Standard grid design. The straps are low tin ZIRLO™ and are [ ] ^{a,c}.

ZIRLO™ is used for guide thimbles to improve corrosion resistance and dimensional stability relative to the Zircaloy-4 guide thimbles used in Standard fuel. The top, Mid, and IFM grid joints are made by bulging guide thimbles into grid sleeves that are attached to grids.

The Guardian™ grid with solid fuel rod lower end plug will continue to be used at the bottom grid elevation since no debris or fretting related failures have been detected with this feature. A new bottom nozzle to Guardian™ grid joint has been designed, where Stainless Steel sleeves that are welded to the Guardian™ grid thimble openings are captured between the lower thimble end plug and the bottom nozzle screws.

The rod diameter is reduced from 0.382" to 0.374" to accommodate the higher pressure drop of the Mid and IFM grids. The Westinghouse advanced 0.374" OD rod with low volume plenum spring, solid lower end plug, and Optimized ZIRLO™ cladding is used for the NGF assemblies. The use of Optimized ZIRLO™ cladding will improve corrosion resistance to support future higher fuel duty and burnup increases. Optimized ZIRLO™ is a new feature that has recently been approved by the NRC(5)(6). The applicable SER requirements, specified by the NRC, will be met in plant specific applications.

Axial blanket pellets (including annular pellets) and ZrB_2 integrated fuel burnable absorbers may also be used in the fuel rods. The use of ZrB_2 has been previously approved by the NRC for use in CE plants by the approval of WCAP-16072-P-A(7). NGF may also include other burnable absorbers such as Gadolinia(18) or Erbia(19) in the NGF assembly. These features have already been implemented in CE plants to improve fuel cycle economics.

In this topical report, the NRC reviewed and approved fuel performance models and methods(5)(6)(7)(8) were used to evaluate the CE 16x16 NGF fuel assembly up to a peak rod average burnup of [ ] ^{a,c}. However, at this time Westinghouse is requesting licensing approval of this design to 62 MWd/kgU peak rod average burnup for use in CE NSSS units using the current CE Reload methodology. Thus, inherent margin has been built into the design.

This topical covers the application of NGF fuel for the CE 16x16 plants. Figure 1-1 demonstrates the expected distribution of the vaned, non-vaned, and IFM grids for NGF fuel in the CE 16x16 plants. Minor variations in assembly configurations will be required to fit plant specific applications. These variations will be assessed using the standard CE Reload methodology and the licensing basis presented

in this topical. As a result, all of the design bases will continue to be satisfied. For example, one of the CE plants will require a stronger Mid grid at selected locations to satisfy high seismic requirements.

Plant specific analyses/evaluations will be done as needed for each first-time (initial) application of CE 16x16 NGF. The licensing for full region implementation of NGF fuel will require that each plant reference this topical in the COLR reference section as an administrative Technical Specification change and then will meet the requirements of a 10 CFR 50.59 evaluation.

1.2 Summary

a. The results of the Mechanical and Fuel Performance Design evaluations performed on the CE 16x16 NGF fuel assembly design confirmed that:
   - The CE 16x16 NGF fuel assembly design is mechanically compatible with the CE 16x16 Standard fuel design, the reactor core components and internals, in-core detector system, and the fuel handling equipment.
   - The design bases and limits for the CE 16x16 NGF fuel assembly and fuel rod performance are satisfied.
   - The grid impact force for seismic and LOCA events were determined to be within the allowable limits as determined by grid crush tests.
   - Hydraulic flow testing of the CE 16x16 NGF fuel assembly with the CE 16x16 Standard fuel design confirmed that the design provides additional fretting margin relative to current designs.

b. The results of the Nuclear Design evaluation performed on the CE 16x16 NGF fuel assembly design confirmed that:
   - Standard nuclear design analytical models and methods accurately describe the neutronic behavior of the CE 16x16 NGF design.
   - The CE 16x16 NGF nuclear design bases are satisfied.

c. The results of the Thermal and Hydraulic Design evaluation on the CE 16x16 NGF fuel assembly design confirmed that:
   - With the implementation of mixing vanes and IFM grids, thermal margins are increased. This margin can be made available for use in improved fuel management, increased plant availability, uprates, and transition core effects.
   - The transition core DNBR penalty is more than offset by the available margin from the mixing vane grids.
   - The ABB-TV correlation gives conservative predictions relative to the NGF DNB test data. As a result of this NGF DNB test data, a new DNB correlation for the NGF fuel assembly is being submitted to the NRC in a separate report[4].

- Hydraulic flow tests with the addition of mixing vanes and IFM grids indicated an increase in a CE 16x16 NGF core pressure drop compared to a CE 16x16 Standard fuel core. The value is dependent on the features included in the Standard fuel. This increase in pressure drop can be accommodated and thermal hydraulic design bases are satisfied.

d. The results of the Safety and Setpoints evaluation performed on the CE 16x16 NGF fuel assembly design confirmed that:
- For the non-LOCA accidents, the CE 16x16 NGF design met the acceptable safety criteria. All established methods/procedures and computer codes used in previous analyses for the CE 16x16 Standard fueled cores were found applicable for CE 16x16 NGF safety evaluations.
- For the LOCA accidents, using the Westinghouse ECCS Performance Evaluation Models for CE plants (either Best-Estimate or Appendix K), the NRC-accepted component models and their range of applicability are adequate. For LBLOCA and SBLOCA, plant-specific calculations will be performed to determine the effect of the CE 16x16 NGF design on ECCS Performance. The Appendix K steam cooling heat transfer component model in the Westinghouse LBLOCA Evaluation Model for CE plants has been modified to include spacer grid heat transfer effects.
- For setpoints, current methods will be applied such that DNB design bases are maintained.

e. The results of the Structural evaluation performed on the CE 16x16 NGF fuel assembly design confirmed that:
- The methodology used in thermal hydraulic analysis of the reactor vessel internals (RVI) remains valid for implementing the NGF design.
- The methodology used in seismic and pipe break analysis of the reactor vessel, RVI, and fuel is valid.
- The analyses performed demonstrate that the stresses and deflections in the RVI meet design basis criteria.

Table 1-1
Standard Review Plan Section 4.2
Subsection II - Acceptance Criteria

		SRP Subsection	Topical Report Section
Design Bases	Fuel System Damage	II.A.1.(a) - Stress, Strain or Loading Limits on grids, GT, fuel rods, control rods & other fuel system structural members	2.3.1.3, 2.3.1.4, 2.4.1, 2.4.2, 2.4.4, 2.4.5, 2.4.6, 2.5.2
		II.A.1.(b) - Strain Fatigue	2.5.6
		II.A.1.(c) - Fretting Wear	2.3.1.2, 2.3.1.5, 2.5.5
		II.A.1.(d) - Oxidation, Hydriding and Crud	2.5.3
		II.A.1.(e) - Dimensional Growth, Rod Bow, Irradiation Growth	2.3.1.1, 2.5.8, 4.2
		II.A.1.(f) - Rod/BA Internal Gas Pressure	2.5.1, 2.5.10
		II.A.1.(g) - Holddown Forces	2.4.3, 4.1.2
		II.A.1.(h) - Control Rod Reactivity	3.3, 6.4
	Fuel Rod Failure	II.A.2.(a) - Hydriding	2.5.3
		II.A.2.(b) - Cladding Collapse	2.5.7
		II.A.2.(c) - Fretting	2.3.1.2, 2.5.5
		II.A.2.(d) - Clad Overheating	2.5.4, 5.1.3
		II.A.2.(e) - Pellet Overheating	2.5.4, 5.1.3
		II.A.2.(f) - Excessive Fuel Enthalpy	5.1.3.4
		II.A.2.(g) - PCI	2.5.11
		II.A.2.(h) - Burst	2.5.1, 5.2
		II.A.2.(i) - Mechanical Fracturing	2.5.3, 5.1.3
	Fuel Coolability	II.A.3.(a) - Cladding Embrittlement	2.5.9, 5.2
		II.A.3.(b) - Violent Expulsion of Fuel	5.1.3.4
		II.A.3.(c) - Clad Melting	2.5.4, 5.1, 5.2
		II.A.3.(d) - Fuel Rod Ballooning	2.5.9, 5.2
		II.A.3.(e) - Structural Deformation (Seismic/LOCA)	2.3.1.3, 2.4.6, 5.2.7, 6.2
Description & Design		II.B	1.1, 2.2
Design Evaluation		II.C.1 - Operating Experience	2, 3, 4
		II.C.2 - Prototype (LTA) Experience	2.4.7
		II.C.3 - Analytical Predictions	2.0 thru 7.0
Testing, Inspection and Surveillance Plans		II.D - Test, Inspections, Surveillance	2.4.7

To meet the requirements of General Design Criterion 10 as it relates to Specified Acceptable Fuel Design Limits for normal operation, including anticipated operational occurrences, fuel system damage criteria should be given for all known damage mechanisms. Fuel system damage includes fuel rod failure, which is discussed in subsection II.A.2. In addition to precluding fuel rod failure, fuel damage criteria should assure that fuel system dimensions remain within operational tolerances and that functional capabilities are not reduced below those assumed in the safety analysis.

Table 1-2

Comparison of Standard and NGF Designs

Feature	Standard Fuel Design Feature Description	NGF Fuel Design Feature Description
Top Grid	Zr-4 Wavy Strip grid or Inconel Straight Strip grid, both with L-shaped outer strap	Inconel, Straight Strip grid with corner weld outer strap
Upper Nozzle	CE Std Nozzle	Same but tabs added to guide tube flange & keyways in flow plate
Top GT flange joint	Zr-4	Zr-4
Mid grids	Zr-4 Wavy grid, no mixing vanes, alternating rod supports	Low tin ZIRLO™ "I" spring grid with Side Supported Vanes on selected grids, alternating rod supports & ZIRLO™ Sleeves
IFM Grids	None	Low tin ZIRLO™ grid with Side Supported Vanes, non-contacting arches with ZIRLO™ sleeves
Mid & IFM Grid Outer Strap Design	Zr-4 strap	Low tin ZIRLO™ strap
Top, Mid & IFM Grid to GT Joints	Welded	Sleeves Bulged to GT above and below Mid grids and below IFM grid
GTs and Wear Sleeves	Zr-4 GTs with dashpot and SS Inner Wear Sleeve	Same but use ZIRLO™ GTs and Short SS Inner wear sleeves
Guardian™ Grid and joint with lower nozzle	Inconel grid with skirt	Same, but uses perimeter strap modified for no welding to lower nozzle and added SS sleeves in GT openings
Bottom Nozzle	CE Std Nozzle	Same except features for welding Guardian™ grid not required
Fuel Rod	0.382" OPTIN™ Zr-4 rod with Std Plenum Spring and Guardian™ solid end cap	Westinghouse 0.374" Optimized ZIRLO™ rod with low volume plenum spring and Guardian™ solid end plug

Figure 1-1

Distribution of Vaned, Non-Vaned, and IFM Grids for NGF Fuel

a, c

2.0 Next Generation Fuel (NGF) Mechanical Design

2.1 Introduction

The Standard Review Plan Section 4.2[2] provides the guidance for demonstrating the acceptability of a fuel design for use in-reactor. Table 1-1 provides an overview of those parameters that should be addressed with a new fuel design and indicates the sections in this report where these parameters are addressed.

Note: to build in inherent margin, the CE 16x16 NGF design was designed to achieve a lead rod average burnup of [ ] ^{a,c} consistent with standard methodology described in sections below; however, at this time Westinghouse is requesting licensing approval of this design to 62 MWd/kgU peak rod average burnup for use in CE NSSS units.

2.2 Fuel System Design Description

The CE 16x16 NGF fuel assembly is designed to be mechanically compatible with the Standard CE 16x16 design for reactor operation with mixed fuel cores. Typical 16x16 NGF fuel assembly design data are given in Table 2-1 and in Figure 2-1. Both the table and the figure show the corresponding information for the Standard 16x16 fuel assembly so that the two designs can be easily compared.

The CE 16x16 NGF fuel assembly incorporates many of the same features and geometry as the Standard 16x16 fuel assembly. Both designs have the same overall length at beginning of life. The basic structure consists of 5 large guide thimble tubes connected to spacer grids at intermediate locations and to nozzles at the ends. In both designs the guide thimbles have the same diameters and spacing, the structural spacer grids are at essentially the same elevations, the top and bottom nozzles are very similar, and the guide thimbles are connected to the top and bottom nozzles using the same type of connections. Both designs have 236 fuel rods with the same pitch, and the two designs have a very similar Guardian™ (bottom Inconel debris-filtering/retention) grid. In addition, structural testing has demonstrated that the response to external loads is similar and meets the design criteria for both designs.

The major differences between the two designs are the following:

- The guide thimbles (Figure 2-5) are made of Zircaloy-4 in the standard design and ZIRLO™ in the NGF design. This change was made because of ZIRLO™'s improved corrosion resistance and dimensional stability under irradiation.
- Mid Spacer Grids
    - o The standard design Mid grids (Figure 2-8) are made using wavy strap OPTIN™, while the NGF grids use straight strap low tin ZIRLO™. The material change was made because of ZIRLO™'s (including low tin ZIRLO™'s) improved corrosion resistance and dimensional stability under irradiation. The change to straight straps was made to improve fabrication and to facilitate the incorporation of mixing vanes.
    - o The standard design Mid grids have cantilever springs, while the NGF grids have vertical "I-springs" (Figure 2-9), which are designed [

]^{a,c}

- o The NGF Mid grids have side-supported mixing vanes (Figure 2-9) to improve thermal performance.
- The welded top grid in the standard design is made of Inconel or Zircaloy-4, and has cantilever springs. The NGF top grid is made of Inconel, has vertical springs (Figure 2-10), and has an extensive history of successful operation in Westinghouse NSSS operating nuclear power plants.
- The NGF design incorporates Intermediate Flow Mixer (IFM) grids (Figure 2-11) to improve thermal performance in critical grid spans. The IFMs are short, non-structural grids with side-supported mixing vanes and opposing dimples [

  ]^{a,c} (Figure 2-12).
- The top, Mid, and IFM grids are attached to the guide thimbles by bulging the thimbles into sleeves that are connected to the grids (Figure 2-6). On the standard design the Zircaloy-4 grids are attached to the thimbles by welding, while the Inconel top grid, if applicable, is retained by rings that are welded to the guide tubes above and below the grid. The bulged design was selected for NGF to improve fabrication while preserving the rigidity of the fuel assembly structure.
- The NGF guide tube flange is connected to the guide thimble by bulging the guide thimble into the flange (Figure 2-6), in lieu of by welding as in the standard design. The bulged design was selected for NGF to improve fabrication while retaining adequate strength.
- Changes were made to the Guardian™ grid (Figure 2-13) [

  ]^{a,c}:
  - o In the standard design, the bottom edges of the outer straps on the Guardian™ grid are welded to the bottom nozzle. In NGF, the Guardian™ grid is retained by insert tubes that are welded to the Guardian™ grid guide thimble openings and are clamped between the bottom of the thimble and the bottom nozzle.
  - o The bottom edge of the Guardian™ grid outer strap was modified to reduce potential for hangup with adjacent fuel assemblies during fuel handling.
- [

  ]^{a,c}.
- A minor change has been made to the top nozzle flow plate and the portion of the guide tube posts within the flow plate to accommodate the guide thimble flange (Figure 2-4).
- The holddown spring has been modified slightly to provide additional holddown force to compensate for the increased pressure drop across the assembly.
- Fuel rod design changes (Figure 2-14):
  - o The NGF fuel rod cladding OD and thickness have been reduced slightly and the pellet design has been made consistent with the standard design that has operated successfully for many years in Westinghouse fuel.
  - o Cladding material has been changed from OPTIN™ to Optimized ZIRLO™ to take advantage of ZIRLO™'s improved corrosion resistance and dimensional stability under irradiation.
  - o Fuel rod length has been increased to provide more fuel rod internal void volume while

still accommodating irradiation growth.

2.3 NGF Fuel Assembly

The design bases for the CE 16x16 NGF fuel assembly and each of the assembly components are similar to the design bases for the Standard 16x16 fuel assemblies except where new design features (e.g., bulged connections in lieu of welded connections) have required the bases to be modified or supplemented.

2.3.1 Fuel Assembly Design Bases and Evaluations

2.3.1.1 Fuel Assembly Growth

Design Basis: The fuel assembly design must include sufficient allowance for irradiation-induced axial growth such that there is no solid axial interference between the assembly and the core internals at any time during the fuel lifetime. The clearances provided to accommodate fuel assembly growth shall be demonstrated to be adequate at the 95% confidence level or greater.

Evaluation: This criterion assures that excessive forces on a fuel assembly will not be generated by the hard contact between the fuel assembly and the reactor internals. Such forces could lead to fuel assembly bowing or guide thimble distortion.

The CE licensed model for predicting axial length changes of a fuel assembly is the NRC reviewed and approved SIGREEP computer code (Section 4.2.2.a of Reference 10). Section 4.2.2.a of Reference 10 discusses the SIGREEP computer code in detail and presents the specific models used for growth and creep of guide thimbles made with Zircaloy-4 tubing.

The tubing material used for the CE 16x16 NGF guide thimbles is ZIRLO™ instead of Zircaloy-4. [

] a, c. Therefore, the best-estimate fuel assembly length change predictions for the CE 16x16 NGF design are taken as [

] a, c The variations between the best-estimate value and the upper/lower 95% values for the CE 16x16 NGF are taken directly from the SIGREEP results with no reduction.

This approach has been benchmarked against available post-irradiation data for fuel assemblies with ZIRLO™ guide thimbles. Adjusted SIGREEP results are presented in Figure 2-15, along with the length change data for the two fuel assemblies that were measured. Figure 2-15 shows that the measured data agree very well with the adjusted best estimate curve from the SIGREEP model. It is, therefore, concluded that the SIGREEP model with a [ ] a, c can be used to predict the axial

dimensional change of the ZIRLO™ guide thimbles used in the CE 16x16 NGF design fuel.

Application of the approach described above to the CE 16x16 NGF design demonstrates that the design includes sufficient axial clearances to operate to a peak rod axial average burnup of [ ] a,c.

2.3.1.2 Fuel Assembly Hydraulic Stability

Design Basis: Flow through the assembly should not cause wear that exceeds the Westinghouse guideline that the fuel system will not be damaged due to fuel clad fretting wear. Specifically, the CE 16x16 fuel assembly shall have [ ] a, c resulting from coolant flow through the fuel assembly over a continuous range of flow rates that cover all CE 16x16 PWR plants.

Evaluation: The fuel assembly hydraulic stability is evaluated using vibration [ ] a, c. Both CE 16x16 NGF and CE 16x16 Standard prototypical fuel assemblies have been flow tested in the Westinghouse Fuel Assembly Compatibility Test System (FACTS) and the Vibration Investigation and Pressure-drop Experimental Research (VIPER) test loops. The testing in the FACTS loop was used to confirm the pressure drop characteristics across the entire assembly and individual components as well as verifying that [ ] a, c is observed over a range of reactor operating flow rates. The tests were performed with simulated core internal support components. A dual test was performed in the VIPER loop to evaluate rod wear as well as confirm [ ] a,c the CE 16x16 NGF and CE 16x16 Standard assemblies and to verify that [ ] a,c. In addition, testing for [ ] a,c.

2.3.1.3 Fuel Assembly Structural Integrity

Design Basis: The fuel assembly must maintain its structural integrity under all operating conditions.

Evaluation: For other than seismic and LOCA loads, the fuel assembly's structural integrity is assured by each component complying with its appropriate design criteria through testing and/or analyses (see Section 2.4). Since the applicable design criteria are based on stress values compared to unirradiated material properties, this criterion is not affected by burnup.

For seismic and LOCA loads, a combination of testing and analysis was performed on the CE 16x16 NGF design to verify that structural integrity would be maintained, i.e. component strength or stress criteria of Table 2-2 were satisfied. Results of full-scale testing of the skeleton and the fuel assembly were used to determine the appropriate input characteristics necessary to predict bundle deflected shapes and grid impact forces. Dynamic crush testing of the CE 16x16 NGF Mid and IFM grids was performed to determine grid crush strengths for comparison to predicted grid impact loads. Stress intensities in the remaining components were evaluated against applicable limits. The evaluation of the CE 16x16 NGF fuel subjected to the seismic and LOCA events of a typical 16x16 plant demonstrated that the criteria of Table 2-2 were satisfied. Due to differences in the seismic/LOCA inputs to the analyses, the implementation of CE 16x16 NGF in CE NSSS plants will include a plant-specific 50.59 evaluation done as part of the standard reload process that confirms compliance with this design criterion.

2.3.1.4 Fuel Assembly Shipping and Handling Loads

Design Basis: The fuel design must be able to accommodate shipping and handling loads without exceeding the limits specified in Table 2-2.

Evaluation: A combination of testing and analysis were performed on the fuel assembly to verify that shipping and handling load requirements were met. Section 2.4 gives more detail on what was done for the different components. Since the applicable design criteria are based on stress values compared to unirradiated material properties, this criterion is not affected by burnup.

2.3.1.5 Fuel Assembly Guide Tube Wear

Design Basis: The fuel assembly must continue to satisfy all stress limits with the maximum predicted reduction in the cross-sectional area of the guide thimble due to wear caused by the CEA.

Evaluation:

The use of chrome-plated wear sleeves has been demonstrated to eliminate guide thimble wear as an issue[13]. Wear sleeves employed in CE 16x16 NGF designs are functionally equivalent to the sleeves in Standard 16x16 designs since the sleeve thickness, chrome-plate requirements, and installed diameters are the same. Although the NGF wear sleeves are slightly shorter at both ends for compatibility with the bulged connections at the top two grids, the wear sleeve still protects the guide thimble through the possible range of wear associated with the CEAs residing at the all-rods-out elevation (including any planned programmed insertions). Therefore, the CE 16x16 NGF wear sleeve design continues to eliminate guide thimble wear as an issue.

Any unsleeved CE 16x16 NGF designs would be evaluated against, and shown to comply with, this design criterion using the same guide thimble wear extrapolation technique employed for the Standard CE 16x16 designs.

2.4 Structural Components Design Bases and Evaluations

2.4.1 Bottom Nozzle

Design Basis: The stress levels of the bottom nozzle must be less than the limits specified in Table 2-2.

Evaluation: The CE 16x16 NGF bottom nozzle (Figure 2-2) is structurally identical to the Standard 16x16 bottom nozzle with the only difference being a machined recess around the upper edge of the standard nozzle has been replaced with a lead-in chamfer. The recess had accommodated the outer strap of the Guardian™ grid assembly (which was welded to the nozzle to secure the grid axially), but is no longer needed since the CE 16x16 NGF Guardian™ grid assembly is secured by four inserts that are captured by the bottom nozzle to guide thimble joints. Analyses of the CE 16x16 NGF bottom nozzle have demonstrated that the nozzle continues to satisfy the stress limits defined in Table 2-2 for all applicable operating conditions.

2.4.2 Top Nozzle

Design Basis: The stress levels of the top nozzle components must be less than the limits specified in Table 2-2.

Evaluation: The CE 16x16 NGF top nozzle (Figure 2-3) is virtually the same as the Standard CE 16x16 design except for a minor change to the holddown springs and the addition of [ ] a,c The holddown spring change increases the holddown spring force to accommodate higher uplift forces associated with the CE 16x16 NGF design (see Section 2.4.3). Analyses of the CE 16x16 NGF top nozzle have demonstrated that the nozzle continues to satisfy the stress limits defined in Table 2-2 for all applicable operating conditions.

2.4.3 Fuel Assembly Holddown Springs

Design Basis: The combination of the fuel assembly wet weight and holddown spring force must maintain a net downward force on the fuel assembly during all Condition I and II events.

Evaluation: The CE 16x16 NGF holddown springs provide more force than the Standard CE 16x16 design to compensate for increased pressure drop across the assembly. Full-scale flow testing was performed on the CE 16x16 NGF design to quantify the hydraulic characteristics of the bundle. Analyses for the application of the CE 16x16 NGF design in a typical 16x16 plant demonstrate that the revised holddown spring design provides sufficient force to satisfy the holddown design criterion (discussed in more detail in Section 4.1.2). Due to differences in system designs and operation, the implementation of the CE 16x16 NGF design in other plants will include a plant-specific analysis done as part of the standard reload process to confirm compliance with this design criterion.

2.4.4 Guide Thimbles and Instrumentation Tube

Design Basis: The stress levels of the guide thimbles and instrumentation tube must be less than the limits specified in Table 2-2.

Evaluation: There are two differences between the CE 16x16 NGF guide thimbles (Figures 2-4 and 2-5) and the Standard CE 16x16 guide thimbles: the tubing material and the attachment of the flange to the guide thimble tube at the top end of the guide thimble assembly (addressed in Section 2.4.5). The yield and ultimate strengths of the two materials are almost identical; the slight difference can be explicitly accounted for in the determination of the allowables for the NGF tubing (ZIRLO™) versus the standard tubing (Zircaloy-4). Analyses of the CE 16x16 NGF guide thimbles have demonstrated that the thimbles continue to satisfy the stress limits defined in Table 2-2 for all applicable operating conditions.

2.4.5 Joints and Connections

Design Basis: The stress levels in threaded joint components must be less than the limits specified in Table 2-2.

Evaluations: The CE 16x16 NGF design includes the same three threaded joints as the Standard CE 16x16 design. These include the outer guide post to guide thimble flange joint, the center guide post to flow plate joint, and the bottom nozzle to guide thimble end plug joint. For each joint configuration, the thread sizes and length of engagements are the same for both the NGF and standard designs. An analysis of the CE 16x16 NGF joints demonstrates that the joints continue to satisfy the applicable stress limits.

Design Basis: The strength of the bulged connections between the guide thimble and the grid sleeves or the guide thimble flange must exceed the loads applied to the connection under all operating conditions.

Evaluations: The bulged connections (Figure 2-6) are similar to those used for the Westinghouse designs that have operated successfully in a variety of plants. Confirmatory testing was completed to verify the strength of the bulged connections exceeded the loads applied to the connection under all operating conditions.

Design Basis: Welded connections between the grids and their respective sleeves/inserts must not fail under all operating conditions.

Evaluations: The sleeves/inserts that are used to secure the spacer grid assemblies to the guide thimbles are welded to the spacer grid (Figure 2-7). Testing performed on each of the weld types confirmed that the welds can sustain the applied loads under all operating conditions without failure.

2.4.6 Grid Assemblies

Design Basis: The lateral strength of the spacer grids must be sufficient to withstand seismic and LOCA events with no channel closure greater than that which would significantly impair the coolability of the fuel rod array or insertability of the CEAs.

Evaluation: The evaluation of the CE 16x16 NGF grid impact strengths was performed in accordance with the licensed CE methodology, as defined in Reference 11. One-sided and through-grid impact forces associated with the seismic/LOCA events of a typical 16x16 plant were generated for the CE 16x16 NGF grids, including IFMs. The impact forces were based on characteristics developed from full-scale testing of grids, skeleton, and fuel assembly. One-sided and through-grid impact strengths of the grids were also determined from testing. The grid strengths of the CE 16x16 NGF design exceed the predicted impact forces associated with the seismic/LOCA events. Due to differences in the seismic/LOCA inputs to the analyses, the implementation of CE 16x16 NGF in CE NSSS plants will include a plant-specific analysis done as part of the standard reload process to confirm compliance with this design criterion.

Design Basis: The cumulative fatigue usage in the grid springs must not exceed 1.0 at EOL.

Evaluation: The Inconel top grid design and the Guardian™ bottom grid design have extensive operating experience that demonstrates the acceptability of the fatigue capability of their grid springs. The IFM grids do not have [

] ^{a,c} An analysis was performed for the Mid grid springs that verifies that the cumulative fatigue usage factor satisfies the 1.0 limit, consistent with Westinghouse methodology.

Design Basis: The spacer grid width must be small enough to provide adequate clearances between the spacer grid assemblies and the reactor internals to ensure functionality during the fuel assembly lifetime.

Evaluation: CE fuel designs have successfully used [ ] ^{a,c} Zircaloy-4 grids for many years without clearance issues between the grids and the reactor internals after irradiation. The CE 16x16 NGF fuel design uses [ ] ^{a,c} low-tin ZIRLO™ strips for the Mid and IFM grids. Since the [

] ^{a,c}, it is concluded that the use of low-tin ZIRLO™ grids will provide adequate clearance within the reactor cavity. Therefore, satisfactory performance is expected for the low-tin ZIRLO™ grids used in the 16x16 NGF design.

2.4.7 LTA Program

Westinghouse has acquired extensive in-plant experience with the features being implemented in the CE 16x16 NGF fuel assembly design. The Westinghouse fleet has many years of successful operation with Mid grid and IFM grid mixing vaned fuel, including experience with Turbo fuel in CE NSSS plants. The Westinghouse fleet experience includes extensive use of ZIRLO™ grids and guide tubes. This operating experience base provides adequate justification for implementation of the CE 16x16 NGF fuel design. Ongoing confirmatory irradiation programs are also being performed as described below. It is expected that data from these programs will continue to confirm the models and methods described herein. As always, any new data will be assessed for its impact on the approved models and methods to assure the conclusions remain valid. Since these data are only confirmatory, it is intended that NRC approval of this design report will be referenced in plant specific 50.59 evaluations for implementation of the CE 16x16 NGF fuel assembly design.

In addition to the [ ][a,c] LTA program, selected NGF features have been implemented in Turbo fuel as full regions at [ ][a,c]; 17x17 NGF LTA programs at [ ][a,c]; and Westinghouse 16x16 NGF LTA programs at [ ][a,c].

CE 16x16 NGF Lead Test Assemblies (LTAs) are currently in operation at [ ][a,c], as part of an irradiation demonstration program that will provide confirmatory performance data for the CE 16x16 NGF design. Four LTAs were inserted in Spring 2005 and will be irradiated for three 18 month cycles. It is expected that the LTAs will reach a peak rod burnup near [ ][a,c]. The LTAs contain all the NGF features except axial blankets and ZrB_2 burnable absorber. These features have already been implemented in CE type plants and will be implemented in full regions of NGF fuel. Post-Irradiation Examinations (PIE) will be performed on selected LTAs at the end of each cycle. The details of these examinations are described in a letter to the NRC[9].

2.5 Fuel Rod Design Bases and Evaluations

Evaluations have been done to verify that the current licensed fuel rod bases and design criteria can be met for the CE 16x16 NGF design. The CE 16x16 NGF fuel rod design (Figure 2-14) has been evaluated using the NRC-approved Westinghouse fuel rod performance code [14][15][16]. The fuel rod design bases and criteria are described below.

The design bases and limits for the CE 16x16 NGF fuel are the same as the CE 16x16 standard and the CE 16x16 value added fuel [14][15][16][17][7][18][19][8][5][6].

2.5.1 Fuel Rod Internal Pressure and DNB Propagation

Design Basis: The fuel system will not be damaged due to excessive fuel rod internal pressure.

The fuel rod internal hot gas pressure shall not exceed the critical maximum pressure determined to cause an outward clad creep rate that is in excess of the fuel radial growth rate anywhere locally along the entire active fuel length of the fuel rod.

The criterion precludes the outward clad creep rate from exceeding the fuel swelling rate, and therefore ensures that the fuel-to-clad diametrical gap will not reopen during normal (Condition I) operation and Condition II moderate frequency events. Restricting the fuel-to-clad gap from reopening will prevent potential accelerated fission gas release at high burnup, with commensurate increases in fuel rod internal pressure and possible eventual failure of the fuel rod. This NRC-approved fuel rod internal pressure limit is currently justified in References 17 and 8.

Mechanistic high temperature strain correlations are used to determine total accumulated strain during a DNB transient. An NRC approved mechanistic DNB propagation methodology is described in References 8, 17, and 49.

For CE Westinghouse PWRs the following additional conditions for clad burst must be met for ZrB_2 IFBA fuel[7]:

a. For Condition I (normal), Condition II (moderate frequency), and Condition III (infrequent) events, fuel cladding burst must be precluded for ZrB_2 fuel rods. Using models and methods approved for CE designs, licensees must demonstrate that the total calculated stress remains below cladding burst stress at the cladding temperatures experienced during any potential Condition II or Condition III event. Within the confines of the plant's licensing basis, licensees must evaluate all Condition II events in combination with any credible, single active failure to ensure that fuel rod burst is precluded.

b. For Condition IV non-LOCA events which predict clad burst, the potential impacts of fuel rod ballooning and bursting need to be specifically addressed with regard to the coolable geometry, RCS pressure, and radiological source term.

Evaluation: The CE 16x16 NGF fuel rod internal pressures are evaluated in the same manner as is used for other Westinghouse CE PWR fuel types. Gas inventories, gas temperature, and rod internal volumes are modeled and the resulting rod internal pressure is compared to the design limit. The design evaluations verify that the fuel rod internal pressure as calculated will meet the design basis.

DNB propagation is evaluated using approved methodology. The currently approved methods are those in References 8, 17, and 49. Incremental cladding high temperature creep strain is calculated. The time-dependent DNB transient local properties are obtained from the appropriate licensed transient analysis methodology for any given plant. These inputs include time, heat flux, quality, mass flow, system pressure, rod internal pressure, and fuel rod initial geometry. To evaluate the potential for DNB propagation against design criteria, the plant's limiting DNB transients are used. For ZrB_2 IFBA fuel the additional conditions of Reference 7 are required to be met. This is accomplished with NRC approved methods and models.

2.5.2 Fuel Rod Clad Stress and Strain

Design Basis: During Conditions I and II, primary tensile stress in the clad and the end cap welds must not exceed 2/3 of the minimum unirradiated yield strength of the material at the applicable temperature. The primary tensile stress limit is yield strength under Condition III. During Condition IV seismic and LOCA conditions (mechanical excitation only), the stress limit is the lesser of 0.7 S_u or $2.4S_m$.

During Conditions I, II and III, primary compressive stress in the clad and the end cap welds must not exceed the minimum unirradiated yield strength of the material at the applicable temperature. During Condition IV seismic and LOCA conditions (mechanical excitation only), the stress limit is the lesser of 0.7 S_u or $2.4S_m$.

Evaluation: The method used to evaluate the cladding stress accounts for power dependent and time dependent changes (e.g., fuel rod void volume, fission gas release and gas temperature, differential cladding pressure, cladding creep and thermal expansion) that can affect stresses in the fuel rod cladding. The same analytical techniques are used for the evaluation of the CE 16x16 NGF design as for other CE fuel designs. All calculated primary tensile and compressive stresses are less than their allowable limits.

Design Basis: At any time during the fuel rod lifetime, the net unrecoverable circumferential tensile cladding strain shall not exceed 1%, based on the Beginning-Of-Life (BOL) cladding dimensions. This criterion is applicable to normal operating conditions and following a single Condition II or III event.

For rod average fuel burnups greater than 52 MWd/kgU, the total (elastic plus plastic) circumferential cladding strain increment produced as a result of a single Condition II or III event shall not exceed 1.0%.

Evaluation: The method used to evaluate the strain accounts for power dependent and time dependent changes (e.g., fuel rod void volume, fission gas release and gas temperature, differential cladding pressure, cladding creep, and thermal expansion) that can produce strain in the fuel rod cladding. In addition, the strain analysis accounts for both long term, normal operation, and short term, transient conditions. The same methods are used for the

evaluation of the CE 16x16 NGF design as for other CE fuel designs since Reference 5 documents that the strain capability of Optimized ZIRLO™ cladding is consistent with the 1% criterion. All calculated cladding strains are less than their allowable limits.

2.5.3 Fuel Clad Oxidation and Hydriding

Design Basis: Fuel rod damage will not occur due to excessive clad oxidation and hydriding.

For Optimized ZIRLO™, the best estimate clad oxide thickness is limited to a licensed peak value of [ ]^{a,c}. The clad hydrogen pickup is also limited to [ ]^{a,c} at end of life to preclude loss of ductility due to hydrogen embrittlement by formation of zirconium hydride platelets.

Evaluation: The cladding oxide thickness and hydriding of the CE 16x16 NGF fuel rod is evaluated by the same methods as are used for Westinghouse fuel designs. The best estimate oxide thickness for ZIRLO™ cladding has been representatively shown to be less than [ ]^{a,c}. Based on References 5 and 6, Optimized ZIRLO™ has been shown to have less oxidation than standard ZIRLO™. Therefore, the limit is met. The calculations show that the clad hydriding of the CE 16x16 NGF fuel rod meet the design limit.

2.5.4 Fuel Temperature

Design Basis: Fuel rod damage will not occur due to excessive fuel temperatures.

For Condition I and II events, the fuel system and protection system are designed to assure that a calculated centerline fuel temperature does not exceed the fuel melting temperature. The melting temperature of UO_2 is taken to be 5080 °F (unirradiated) and to decrease by 58°F per 10 MWd/kgU of fuel burnup. [

]^{a,c}

Evaluation: The temperature of the CE 16x16 NGF fuel pellets is evaluated by the same methods as are used for all Westinghouse CE PWR fuel designs. Rod geometries, thermal properties, heat fluxes, and temperature differences are modeled to calculate the temperature at the surface and centerline of the fuel pellets. Fuel centerline temperatures are calculated as a function of local power and rod burnup. To preclude fuel melting, the peak local power experienced in Condition I and II events can be limited to a maximum value which is sufficient to ensure that the fuel centerline temperatures remain below the melting temperature at all burnups. Design evaluations for Condition I and II events have shown that fuel melting will not occur for achievable local powers and licensed fuel rod burnup.

2.5.5 Fuel Clad Fretting Wear

Design Basis: The fuel system will not be damaged due to fuel rod clad fretting. Consistent with the objective for the CE 16x16 NGF design to add margin relative to the current designs, it is a requirement that the fuel rod cladding wear due to contact with the grid rod supports must be less than the observed wear on the existing CE 16x16 Standard assembly.

Evaluation: The baseline for fretting wear for the CE 16x16 Standard fuel design was based on extensive out-of-pile tests, including full scale flow tests with flow test velocities that exceeded the calculated maximum velocity at operating conditions. The fretting wear evaluation for the CE 16x16 NGF is performed using [

]^{a,c}. The CE 16x16 NGF and CE 16x16 Standard fuel assemblies have been [ ]^{a,c} flow tested in the Westinghouse VIPER test loop. The test in the VIPER Loop had a CE 16x16 NGF fuel assembly adjacent to a CE 16x16 Standard fuel assembly. The test was conservatively performed with [

]^{a,c}. The flow was set to conservatively cover [

]^{a,c} with the CE 16x16 NGF assembly and was run for a duration of 500 hours. Results of these tests confirmed that the fuel rod wear due to contact with the spacer and IFM grids for the CE 16x16 NGF assembly is [ ]^{a,c} than the fuel rod wear on the CE 16x16 Standard assembly. The measured wear on the CE 16x16 NGF assembly was also [ ]^{a,c} than the wear measured on the RFA/RFA-2 test assemblies.

2.5.6 Fuel Clad Fatigue

Design Basis: For the number and type of transients which occur during Condition I reactor operation, End-Of-Life (EOL) cumulative fatigue damage in the clad and in the end cap welds must be less than 0.8.

Evaluation: The fatigue damage associated with [ ]^{a,c} was calculated. In addition, the clad fatigue damage due to startups/shutdowns and reactor trips was also calculated. The same methods are used for the evaluation of the CE 16x16 NGF design as for other CE fuel designs since Reference 5 documents the applicability of the fatigue damage criterion to Optimized ZIRLO™ cladding. The calculated cumulative fatigue damage factors for the CE 16x16 NGF design are all less than the 0.8 criterion.

2.5.7 Fuel Clad Flattening

Design Basis: The time required for the radial buckling of the clad in any fuel or integral burnable absorber rod must exceed the reactor operating time necessary for the appropriate fuel batch to accumulate its design average discharge burnup. This criterion must be satisfied for continuous reactor operation at any reasonable power level and during any Condition I, II, or III situation. It will be considered satisfied if it can be demonstrated that axial gaps longer than 0.125 inch will not occur between fuel pellets and that the plenum spring radial support capacity is sufficient to prevent clad collapse under all design conditions.

Evaluation: The method used to evaluate cladding collapse accounts for power dependent and time dependent changes (e.g., differential cladding pressures, cladding temperature, cladding flux, and oxide buildup) that can affect the ovalization of the cladding during operation. The same methods are used for the evaluation of the CE 16x16 NGF design as for other CE fuel designs since Reference 5 documents that the application of these methods is conservative for rod designs with Optimized ZIRLO™ cladding. The calculated cladding collapse times for the CE 16x16 NGF design in the active fuel region exceed the operating time of the fuel. The evaluation of cladding collapse in the plenum region demonstrated that the CE 16x16 NGF plenum spring design provides sufficient radial support to the cladding to preclude collapse.

2.5.8 Fuel Rod Axial Growth

Design Basis: The axial length between end fittings must be sufficient to accommodate differential thermal expansion and irradiation-induced differential growth between fuel rods and guide thimbles such that it can be shown with 95% confidence that no interference exists.

Evaluation: This requirement provides assurance that the fuel rods are not fully constrained axially between the top and bottom end fittings. If a fuel rod were to be constrained in this manner, any additional length change of the fuel rod due to irradiation-induced growth or thermal expansion could result in additional fuel rod bowing.

The nominal BOL hot shoulder gap (i.e. the available axial clearance between the fuel rods and the top/bottom end fittings) is calculated using the cold dimensions for the appropriate fuel and internal component, adjusted for differential thermal expansion between fuel rods and guide thimbles. This initial hot shoulder gap is further adjusted for component tolerances, guide thimble growth, and fuel rod growth to determine the hot shoulder gap at other points in life. The adjustment for tolerances, guide thimble growth, and fuel rod growth is done statistically to determine the lower 95% shoulder gap prediction for comparison to the criterion. The CE 16x16 NGF shoulder gap evaluation used the fuel assembly growth model discussed in Section 2.3.1.1 and the previously approved fuel rod growth model for Westinghouse fuel designs with Optimized ZIRLO™ cladding[5]. [

]$^{a, c}$ The shoulder gap calculation for the CE 16x16 NGF design demonstrated compliance with the criterion at an axially averaged fuel rod burnup of [ ]^{a,c}.

2.5.9 Fuel Materials

The fuel rod design will use design values for properties of materials as given in References 14, 15, 16, 7, 18, 19, 8, 5, and 6, for UO_2, Gadolinia, Erbia, ZIRLO™, and Optimized ZIRLO™ material.

The material properties of the UO_2 fuel are not affected by the presence of a thin [ ]$^{a, c}$ ZrB_2 coating on the fuel pellet surface, therefore, the properties described in Reference 14 for UO_2 are also applicable, with due consideration to temperature and irradiation effects. The irradiation behavior of the thin IFBA coating material has been evaluated and is presented in Reference 20 and in Reference 7.

ZIRLO™ is a modification of the Zircaloy-4 alloy. The comparative properties of the ZIRLO™ and Zircaloy-4 alloy are described in detail in Reference 1 and in Reference 8. Some of these properties, including density, thermal expansion, thermal conductivity, and specific heat, have been verified in testing programs described therein. Appropriate ZIRLO™ materials properties models are used in fuel rod evaluations.

Optimized ZIRLO™ is a modification of the ZIRLO™ alloy. The comparative properties of the ZIRLO™ and Optimized ZIRLO™ alloy are described in detail in References 5 and 6. Some of these properties, including density, thermal expansion, thermal conductivity, and specific heat, have been verified in testing programs described therein.

2.5.10 Burnable Absorbers

The CE 16x16 NGF fuel design is expected to use the ZrB_2 IFBA burnable absorber. In the ZrB_2 IFBA fuel rod, the fuel pellets in the center portion of the rod are coated with a thin layer of ZrB_2. The B^{10} in the thin layer acts as a burnable absorber. The B^{10} may be enriched. The ZrB_2 IFBA fuel rod design has been reviewed and approved for used in Westinghouse CE PWR's in Reference 7.

However, other NRC approved burnable absorbers may also be used in the CE 16x16 NGF applications. Gadolinia and erbia burnable absorber fuel rod designs are currently approved for use in Westinghouse CE PWR's and could be used in the CE 16x16 NGF application. In the gadolinia burnable absorber fuel rod, a small amount of Gd_2O_3 is mixed with the UO_2 and sintered together to act as a burnable absorber. The use of the gadolinia burnable absorber fuel rod design has been reviewed and approved for used in Westinghouse CE PWR's in Reference 18. Similarly, in the erbia burnable absorber fuel rod, a small amount of Er_2O_3 is mixed with the UO_2 and sintered together to act as a burnable absorber. The use of the erbia burnable absorber fuel rod design has been reviewed and approved for used in Westinghouse CE PWR's in Reference 19.

2.5.11 Pellet Cladding Interaction

Design Basis: The fuel system will not be damaged due to excessive pellet-cladding interaction (PCI).

The fuel rod cladding is protected against damage from PCI by limiting the clad deformation due to pellet thermal expansion. While there is no current criterion for fuel failure resulting from PCI, two related design criterion are applied. For Condition I and II events, the fuel rod cladding is protected against damage from PCI 1) by limiting the fuel cladding strain to 1% and 2) by precluding fuel melting.

Evaluation: The CE 16x16 NGF fuel strain criterion and evaluation of Section 2.5.2 limits the fuel cladding strain to 1%. Fuel melting is controlled through the criterion and evaluation discussed in Section 2.5.4.

2.6 Rod Average Burnup to 62 MWd/kgU

The CE 16x16 NGF fuel assembly has been designed for burnups beyond a peak rod average burnup of 62 MWd/kgU. Justification for a burnup limit of 62 MWd/kgU is provided by Westinghouse experience and by approved fuel performance model predictions as discussed below.

Westinghouse Optimized ZIRLO™ clad fuel rod performance has been demonstrated to be satisfactory in Westinghouse NSSS's and is approved by the NRC to a burnup of 62 MWd/kgU[5][6] for the Westinghouse fuel design. Successful performance to date in CE NSSS's is similar. Design analyses of the NGF assembly structural components demonstrate burnup capability well beyond 62 MWd/kgU. The justification for evaluations of CE 16x16 NGF structures (skeleton) up to a peak rod average burnup of 62 MWd/kgU and beyond is provided in this topical report. Consequently, the CE 16x16 NGF assembly hardware is capable of performing satisfactorily to rod average burnups well beyond 62 MWd/kgU. Further, CE 16x16 NGF LTA's are in place to confirm this acceptability.

The approved fuel rod performance model[16] has been demonstrated to provide conservative over-predictions of fission gas release in fuel rods to [ ] [a,c] rod average burnup as shown in Figure 3-2 of Reference 16. The peak rod average burnup was [ ] [a,c]. Additional fission gas release and temperature data well above burnups of 62 MWd/kgU have also been analyzed and the results indicated that both fuel parameters (fission gas release and fuel centerline temperatures) are satisfactorily predicted for conditions consistent with design and licensing.

[

] [a,c] Thus, fission gas release predictions for design and licensing are acceptable to 62 MWd/kgU.

Additional temperature data available from [

] a,c Thus, it is concluded that temperature predictions for design and licensing at high burnup are satisfactory.

Thus, Westinghouse is requesting NRC approval for the use of the CE 16x16 NGF assembly in CE NSSS's to a peak rod average burnup limit of 62 MWd/kgU using the existing approved fuel rod performance models and methodology.

Table 2-1
Typical Standard CE 16x16 and CE 16x16 NGF Fuel Design Comparison

a, c

Table 2-1 (continued)
Typical Standard CE 16x16 and CE 16x16 NGF Fuel Design Comparison

a, c

Table 2-2
Stress Limits of Structural Components

Loading Condition	Components (except Spacer Grids[1])	Stress Limits[2]
Condition I and II	All components except Holddown Springs	$P_m \leq S_m$ $P_m + P_b \leq F_s S_m$
	Holddown Springs	Shear stress $\leq$ Minimum Yield Stress in Shear
Condition III	All components except Holddown Springs	$P_m \leq 1.5\ S_m$ $P_m + P_b \leq 1.5\ F_s S_m$
	Holddown Springs	Shear stress $\leq$ Minimum Yield Stress in Shear
Condition IV	All components	See Reference 11

Notes:

1. Spacer grid strength requirements per Reference 11.
2. Nomenclature
   a. P_m = Calculated general primary membrane stress, defined by Section III, ASME Boiler and Pressure Vessel Code.
   b. P_b = Calculated general bending stress, defined by Section III, ASME Boiler and Pressure Vessel Code.
   c. S_m = Design stress intensity value, equal to one of the following (adjusted for the appropriate temperature):
      - For zirconium alloys, 2/3 of the specified minimum unirradiated yield strength, or 2/3 of the lower 95% value of yield strength derived from a distribution of test results from representative specimens.
      - For other materials, the value from the ASME Boiler and Pressure Vessel Code, Section III Stress Intensity for Class 1 Components, or a value based on the formulas used to establish the Section III values, with the yield and tensile strengths used in the formulas equal to the lower 95% values derived from a distribution of test results from representative samples.
   d. Fs = Shape factor, defined as the ratio of the bending moment required to produce a fully plastic cross section to the bending moment required to first produce yielding at the extreme fiber of the cross section.

Figure 2-1
Typical Comparison of 16x16 NGF Design with a Standard 16x16 Design

a, c

Standard 16x16 (Plant B)

CE 16x16 NGF (Plant B)

Figure 2-2

16 CE NGF Bottom Nozzle Design

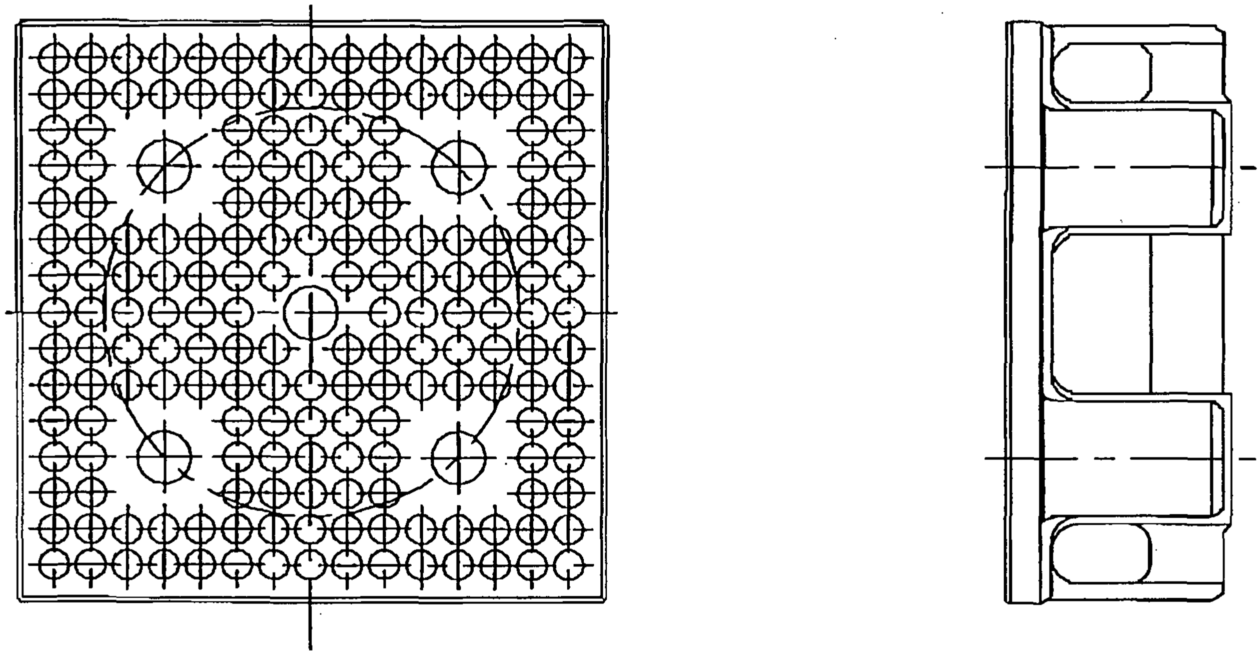
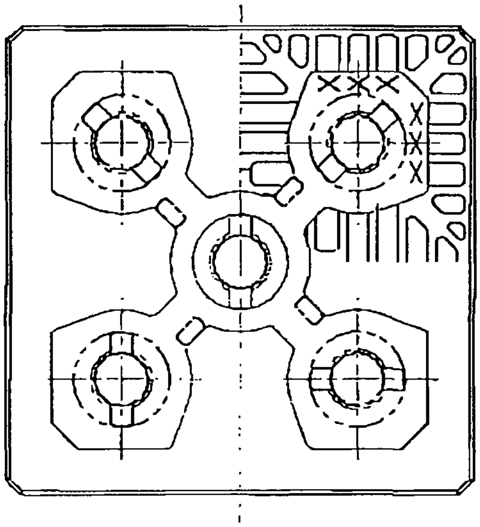


Figure 2-3
16 CE NGF Top Nozzle Design



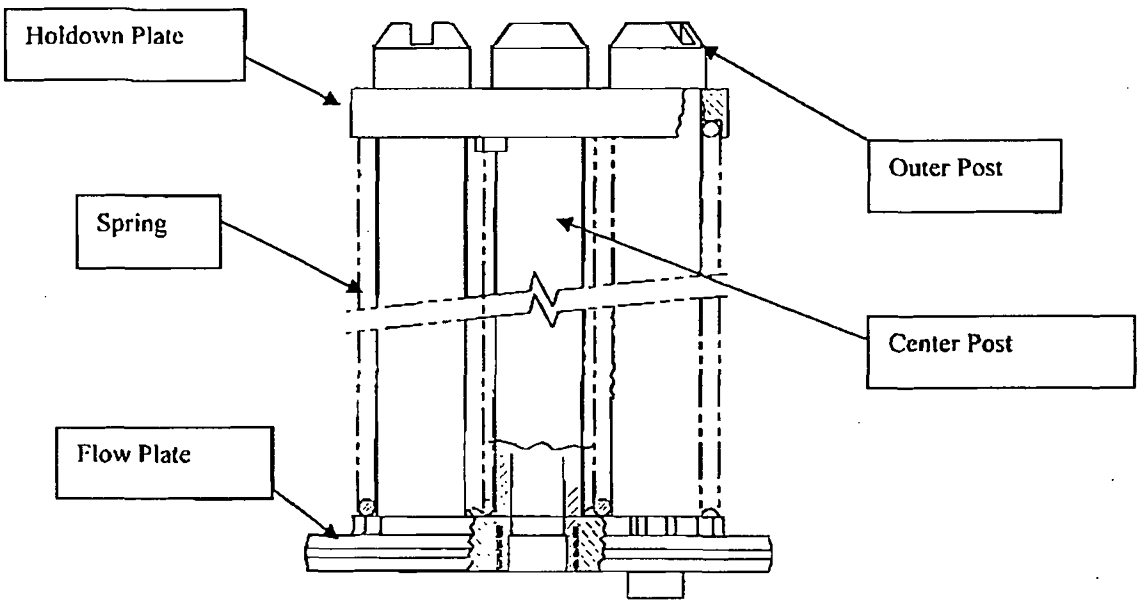


Figure 2-5
16 CE NGF Guide Thimble Assembly Design

a, c

Figure 2-6
16 CE NGF Grid to Guide Thimble / Instrument Joints

a, c

Figure 2-7

16 CE NGF IFM / Mid Grid to Sleeve Joint

a, c

Figure 2-8

16 CE NGF Vaned Mid grid 3-D Configuration with Sleeves

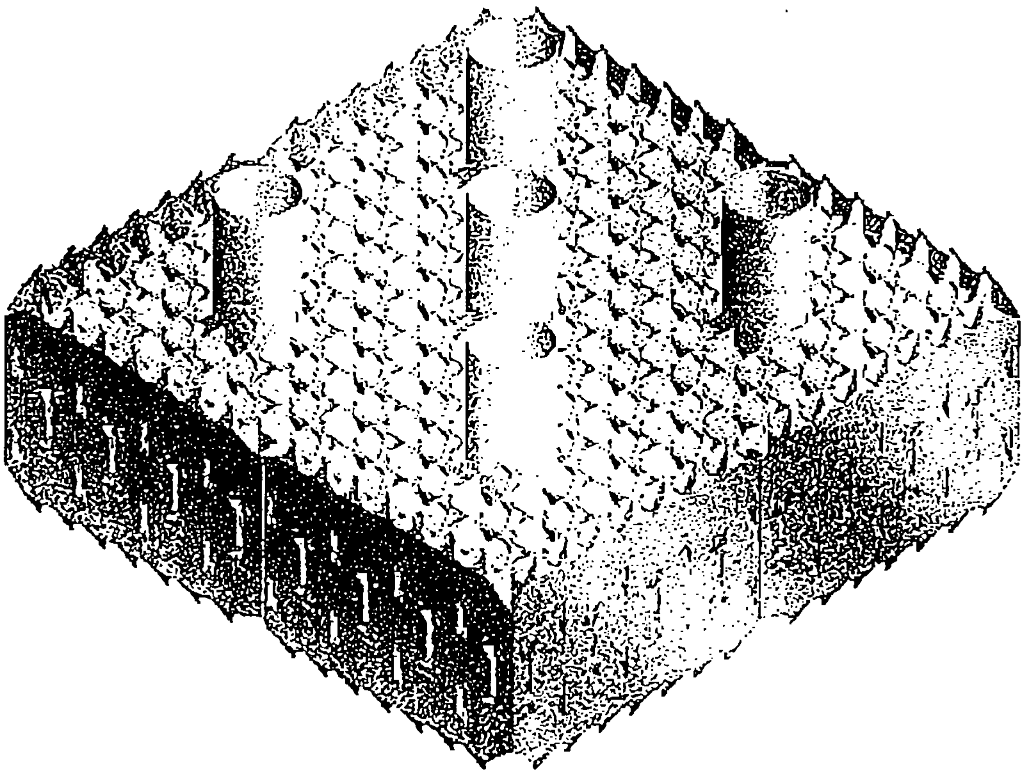


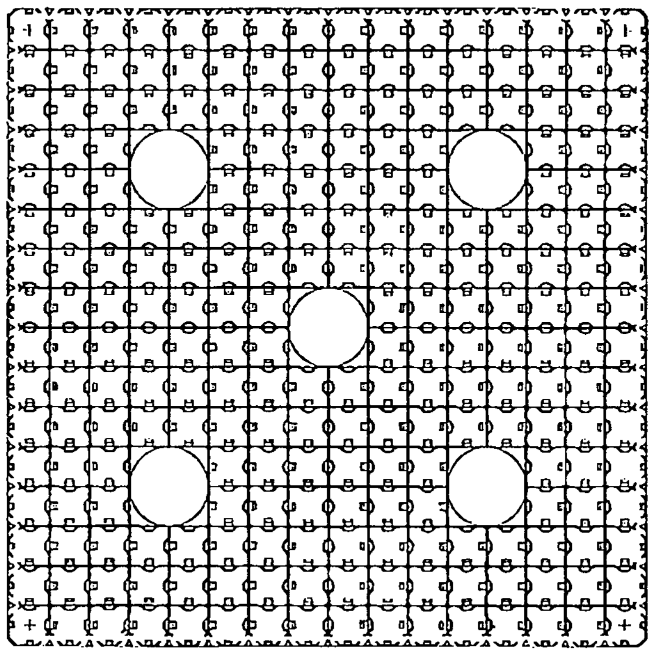
Figure 2-9
16 CE NGF Mid Grid "I'Spring" Design

[ ] a, c

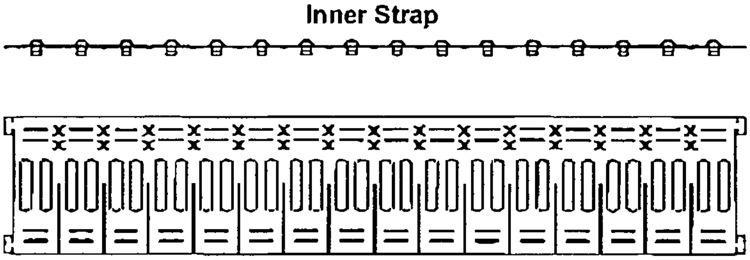
Figure 2-10

CE NGF Top Grid Assembly, Inner Strap, Outer Strap and Sleeve

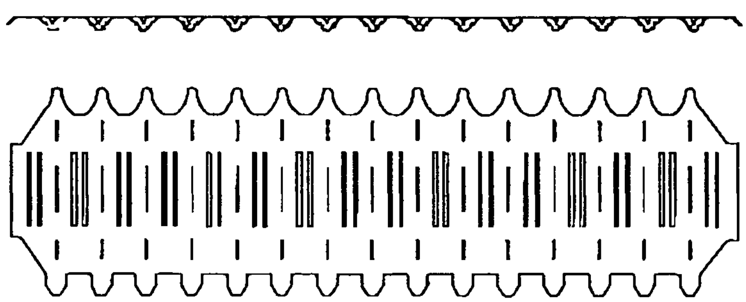
Grid Assembly

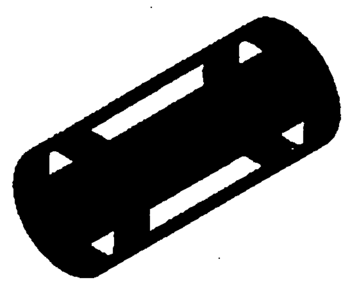


Inner Strap



Outer Strap





Sleeve

Figure 2-11

16 CE NGF Vaned IFM Grid 3-D Configuration with Sleeves

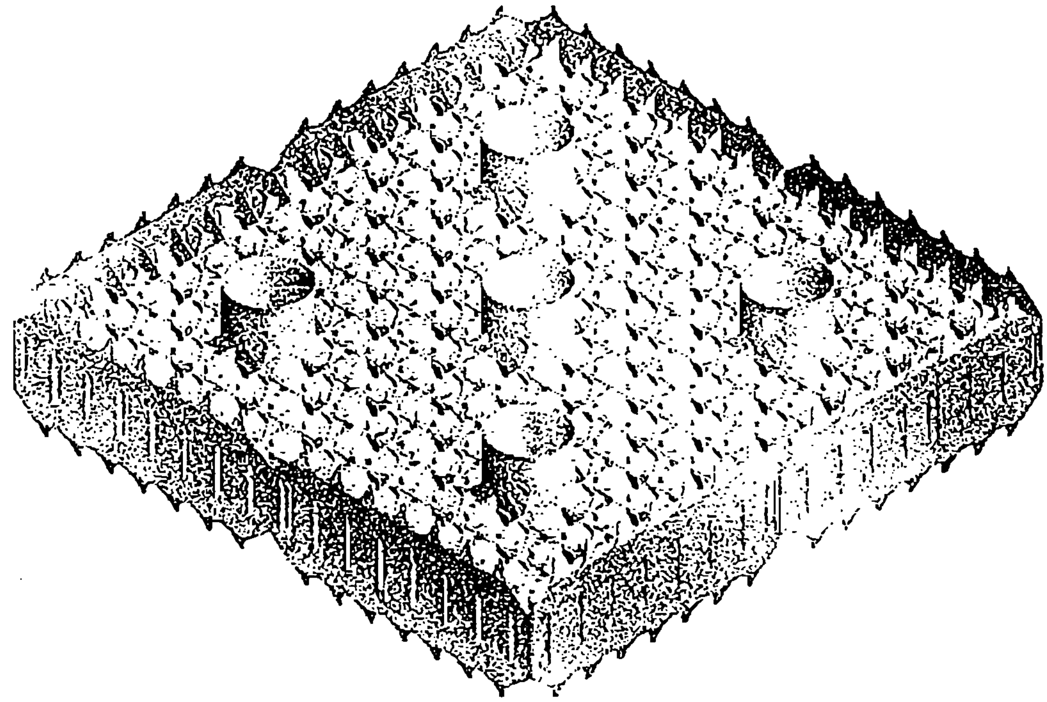


Figure 2-12
16 CE NGF Intermediate Flow Mixing (IFM) Grid Design

a, c

Figure 2-13

16 CE NGF Guardian™ Grid Assembly with Inserts

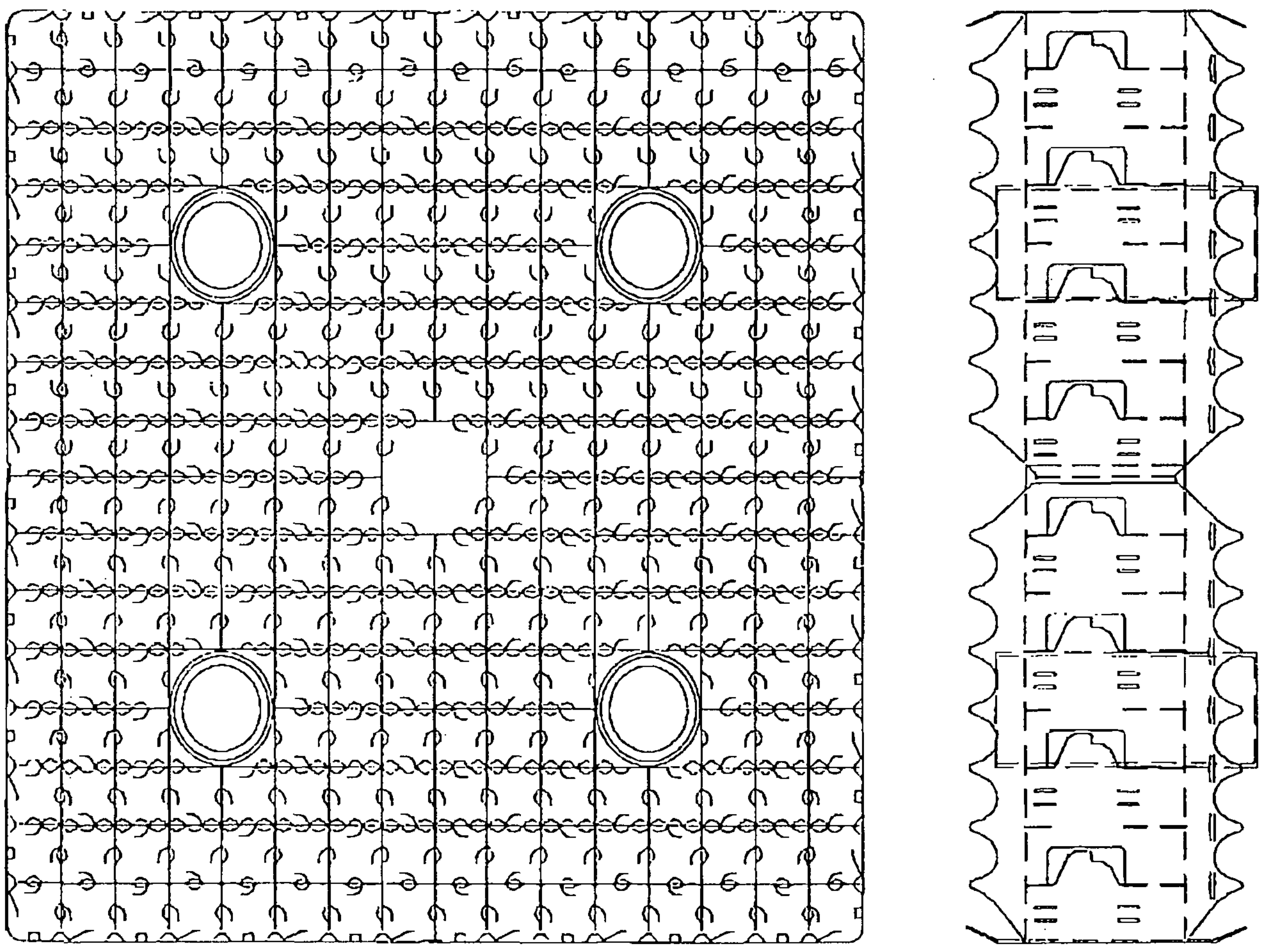


Figure 2-14
Typical 16 CE NGF Fuel Rod Design (Plant B)

a, c

Figure 2-15
Comparison of Model Predictions to Measured Data

a, b, c

[ ]

3.0 Nuclear Design

The CE 16x16 NGF design results in small differences in nuclear design characteristics compared to prior 16x16 fuel designs. The major change affecting the nuclear design characteristics is the change in fuel pellet and fuel rod clad diameter. The other primary nuclear design parameters such as fuel assembly pitch, fuel rod pitch, and burnable absorber design are unchanged. (See Table 3-1)

3.1 Design Bases

The design bases and functional requirements used in the nuclear design of the 16x16 Next Generation Fuel (NGF) cores are the same as those employed in previous CE 16x16 fuel designs. The nuclear design requirements are based on plant specific documents. These documents are used to develop the reload core design and compliance to them assures that all applicable design bases will be satisfied. These documents will be revised as necessary to remain consistent with the plant safety analysis.

3.2 Design Methods

No changes to currently approved neutronics codes and methods are required to design and analyze cores containing 16x16 NGF fuel assemblies. The current neutronics design methods are given in References 22 through 26 and Reference 7.

3.3 Design Evaluation

The neutronic characteristics of the CE 16x16 NGF design results are very similar to previous 16x16 fuel designs (See Table 3-1). The change in fuel pellet diameter and rod diameter produce a slight increase in the core reactivity for low and intermediate burnups (See Figure 3-1). In addition there is also a slight increase in the power peaking and in the moderator temperature coefficient. All of these effects are easily compensated for by a decrease in the feed enrichment and/or increase in the number of burnable absorber rods loaded into the core. There is no significant impact on the control rod worth or core shutdown margin or any of the other reactivity related parameters.

The slight increase in assembly reactivity associated with the NGF assembly will require that the plant specific Tech Spec limits on maximum enrichment in the spent fuel pool be confirmed. Although it is anticipated that in most cases the actual assembly enrichment will be reduced to compensate for the reactivity increase, the assessment of spent fuel pool criticality is necessary for those cases where the increased reactivity of the NGF assembly will be used to support a decrease in the feed batch size.

The structural grid design has also been changed from prior designs to use an I-spring rod support and a mixing vane geometry. Intermediate Flow Mixing (IFM) grids have been added to the assembly slightly increasing the amount of structural material in the core region. The small increase in the amount of grid material in the core has a very small effect on core reactivity and power distribution. Appropriate allowances will be included in the cycle specific reload safety analysis to address these effects.

Table 3-1

Comparison of Typical CE 16x16 Design Parameters

a, c

Figure 3-1

Typical Difference in Assembly Reactivities

a, c

4.0 Thermal and Hydraulic Design

This section describes thermal-hydraulic evaluation of the CE 16x16 NGF design for general reload applications. The CE 16x16 NGF design improves heat transfer performance of the fuel design through the following design changes: (1) the addition of side-supported mixing vanes on both the Mid grids and Intermediate Flow Mixer (IFM) grids, and (2) the addition of IFM grids in the fuel assembly.

Similar to current Westinghouse fuel designs containing IFM grids, the IFM grids of the CE 16x16 NGF are placed [ ]^{a,c} to improve thermal performance. The IFM grids use the same side-supported mixing vanes as the Mid grids.

The new design features of the CE 16x16 NGF for thermal improvement have been verified with respect to applicable T/H design criteria through testing and analysis. Included are discussions of the T/H design bases, effect of the design changes on rod bow evaluation, the design methods, and effect of mixed core on Departure from Nucleate Boiling Ratio (DNBR).

4.1 Thermal and Hydraulic Design Bases and Evaluation

The thermal and hydraulic design bases for the CE 16x16 NGF design are described in this section. Each basis is followed by a discussion of the evaluation performed to verify that the basis is met.

4.1.1 DNB Design Basis

Design Basis: The fundamental criterion that must be met for core T/H design is the DNB design basis. SRP Sections 4.2(2) and 4.4(27) state that the DNB acceptance criterion provides assurance that there be at least a 95% probability at a 95% confidence level that the hot fuel rod in the core does not experience a DNB during Condition I or II events. Similar to all other Westinghouse fuel designs, the DNB design basis for the CE 16x16 NGF is that there will be at least a 95 percent probability at a 95 percent confidence level (95/95) that DNB will not occur on the limiting fuel rods during Condition I and II events. The DNB acceptance limit is the 95/95 DNBR limit defined by a DNB correlation applicable to the CE 16x16 NGF and approved by the NRC.

Evaluation: DNB tests (also referred to as Critical Heat Flux (CHF) tests) were performed with the CE 16x16 NGF side-supported vane grids with different grid spacing at the Columbia University Heat Transfer Research Facility (HTRF). The ABB-TV correlation developed in Reference 40 for Turbo fuel has been demonstrated to be conservative for the CE 16x16 NGF CHF test data. In order to more accurately reflect its thermal performance, a new DNB correlation has been developed for the CE 16x16 NGF design based on the test results. The correlation will be used only in the mixing vane region of the core with a

computer code that has been either used for the correlation development or qualified with its 95/95 DNBR limit. The Westinghouse version of the VIPRE-01 code[28], and the TORC and CETOP-D codes[29][30][31], can be used for thermal-hydraulic analysis of the core. The DNB correlation 95/95 DNBR limit and its applicable range are described in a separate topical report[4]. The application of the new DNB correlations in reload design is discussed in Section 6 of Reference 4. For the non-mixing vane region, the ABB-NV[32][40] correlation is used to calculate DNBR values in the hot channels.

The application of the correlation with VIPRE-01 will be in full compliance with the conditions of the Safety Evaluation Report (SER) on the VIPRE-01 code and modeling for CE-PWR[32]. The correlation will be used only with the currently USNRC-approved methodology for PWR safety analysis. The current methodology includes the Revised Thermal Design Procedure (RTDP)[33], the transition core evaluation method[34], and the reload evaluation method[35], as well as the current methods of Extended Statistical Combination of Uncertainties (ESCU)[36], and Modified SCU (MSCU)[37] for CE-PWR. The plant analysis will account for uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters in addition to uncertainty in the DNB correlation.

In the TORC code, the application of the correlation will be in full compliance with the conditions of the Safety Evaluation Reports (SER) for the TORC code and CETOP-D code [38]. The TORC code is used in reloads to perform detailed modeling of the core and the hot assembly and to determine minimum DNBR in the hot assembly. The CETOP-D code is a fast running tool, which is used in reload analysis to calculate the minimum DNBR in the hot subchannel. While the TORC code can be applied directly in the reload analyses[39], typically the TORC code is used to benchmark the CETOP-D DNBR results such that the CETOP-D results are conservative relative to TORC results. The correlation will be used with the currently USNRC-approved methodology for PWR safety analysis. The current methodology includes the setpoints topical[39], the methods of Extended Statistical Combination of Uncertainties (ESCU)[36] and Modified SCU (MSCU)[37]. The plant analysis will account for uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters in addition to uncertainty in the DNB correlation.

4.1.2 Fuel Assembly Holddown Force

Design Basis: The fuel assembly will not be allowed to lift due to flow during all Condition I and II events. The Westinghouse design limit is that the fuel assembly is designed to remain in contact with the lower core plate under all Condition I and II events.

Evaluation: The net force exerted on the fuel assembly consists of the downward force of the fuel assembly holddown springs, the downward force of the weight of the fuel assembly, the upward buoyancy force of the water and the upward force from axial flow interacting with resistances along the flow path within a control volume. The upward hydraulic force of the CE 16x16 NGF design was calculated using the same method as for other CE PWR fuel designs. The pressure loss coefficients used in the evaluation were determined from hydraulic tests of the CE 16x16 NGF design. The net holddown force evaluations are performed for a range of operating conditions from beginning of life (BOL) to end of life (EOL). The evaluation includes factors accounting for [

] $^{a, c}$. The evaluation concludes that the CE 16x16 NGF design has sufficient holddown force margin to meet the acceptance limit. The fuel assembly holddown force margin will be verified for each plant application with plant specific core operating conditions.

4.1.3 Thermohydrodynamic Stability

Design Basis: Operation under Condition I and II events will not lead to thermohydrodynamic instability in the reactor core. The types of instability considered are Ledinegg or flow excursion static instability and density wave dynamic instability. The Westinghouse design limits are that Ledinegg instability will not occur and that a large margin will exist to density wave instability[41].

Evaluation: For Westinghouse CE PWR designs, the Ledinegg instability is prevented because the slope of the reactor coolant system (RCS) pressure drop-flow rate curve is positive and the slope of the pump head curve is negative.

The margin to the density wave instability is evaluated using the method of Ishii[42] for the CE 16x16 NGF design, same as for other Westinghouse fuel designs. An inception of this type of instability will require typically increases on the order of 100% or greater of rated reactor power.

4.2 Effect on Fuel Rod Bowing

Effect of CE 16NGF rod bowing on DNB analysis are evaluated using the same NRC-approved methodology[43][44] for other fuel designs[45]. The methodology defined in References 43 and 44 remain applicable. The rod bow DNBR penalty in the non-IFM grid span will be offset by the same amount of DNBR margin retained in the DNBR Safety Analysis Limit for each plant analysis.

4.3 Thermal and Hydraulic Design Methods

No change in the T/H design methods currently used for other fuel designs is necessary for the incorporation of the CE 16x16 NGF design except for use of either the ABB-TV DNB correlation or the new DNB correlation described in Reference 4 for more accurate predictions of thermal margin. The ABB-TV DNB correlation yields conservative results relative to the new CE 16x16 NGF DNB data.

4.4 Transition Core DNBR Effect

Due to its relatively higher pressure drop, there will be a DNBR penalty on CE 16x16 NGF in a mixed core with the CE 16x16 Standard Fuel, as compared to the DNB analysis for a full core of CE 16x16 NGF.

Both VIPRE-01 and TORC are capable of accurately predicting fluid conditions in a transition core composed of different fuel designs. Consequently, the VIPRE-01 or TORC thermal hydraulic reload analysis methods as described in Section 4.1.1 will be used with the CE 16x16 NGF and ABB-NV CHF correlations for the CE 16x16 NGF and CE 16x16 Standard fuel assemblies. For CE 16x16 NGF fuel assemblies, some grid spans have the mixing vanes and some do not, so the ABB-NV correlation will be used for grid spans without the mixing vanes.

The application of the CE 16x16 NGF and ABB-NV correlations and codes, setpoints, and uncertainty analyses, as described in Section 4.1.1, will be the same for transition cores containing CE 16x16 NGF and CE 16x16 Standard fuel assemblies.

5.0 Accident Analysis

5.1 Non-LOCA Safety Evaluation

5.1.1 Introduction and Overview

This section addresses the effect of the CE 16x16 NGF design on the non-LOCA accident analyses. This evaluation addresses the following NGF features:

- Westinghouse standard 0.374 inch O.D. fuel rod,
- Intermediate Flow Mixing (IFM) grids (addition of IFM grids to the assembly),
- Side supported mixing vanes (for grids in the upper 2/3 of the core and IFMs),
- Use of NGF critical heat flux correlation.

The revised Mid grid design and the addition of IFM grids will improve the DNB performance of the fuel. This is beneficial for the non-LOCA analyses. Benefit for this will be taken into account in the implementation of the fuel in a plant application using the NGF critical heat flux correlation. Use of the assembly in a particular plant application may increase the core pressure drop, possibly resulting in increased bypass flow. Additionally, the decrease in fuel rod OD will increase the core average heat flux at the fuel rod surface. While this will not have a significant effect on the non-LOCA transients, this will also be addressed in the implementation.

An evaluation of the effect of the use of Optimized ZIRLO™ cladding has been addressed in References 5 and 6. The use of the IFBA burnable absorber has been addressed in Reference 7. Other minor design features of CE 16x16 NGF have a negligible effect on non-LOCA analysis results.

Note that this assessment summarizes the expected impacts on the non-LOCA analyses. When NGF is implemented at a given plant, the normal reload process will be followed to address the impact (if any) of the fuel changes.

An assessment of the impact of the CE 16x16 NGF fuel design features on the various non-LOCA events is provided below.

5.1.2 Evaluation of Effects on Non-LOCA Computer Codes and Methods

The evaluation of effects on non-LOCA will use codes and methods that have been NRC approved for CE NSSS applications. Currently the computer codes used in non-LOCA safety analysis for CE 16x16 plants are NRC approved and consist of the CENTS[46] or RETRAN[47] codes for calculating the NSSS transient response to accident events, the FACTRAN[48] and VIPRE[28] codes for hot rod fuel and clad temperature or heat flux evaluations, and the VIPRE, TORC[29] or CETOP-D[50] codes for the hot channel DNBR evaluation. In addition, the TWINKLE[51] or STRIKIN[52] code is used to calculate the core response for fast reactor transients where the RCS loop response is not important.

The system transient codes CENTS and RETRAN use a detailed nodalization of the RCS primary side components (RCS hot and cold loops, reactor vessel, steam generator, pressurizer, and reactor coolant pumps). In addition, they contain models of the reactor control and protection system, and engineered safeguards features. A simplified fuel rod radial heat transfer model is used in each node, which is calibrated to match a conservative set of fuel rod temperatures versus power. The core transient behavior is calculated with a point reactor kinetics model using pre-calculated kinetics coefficients (i.e., MTC, Doppler feedback, delayed neutron fraction, etc.). The core dynamic behavior is not sensitive to details of the fuel assembly design, and would be only very slightly affected by changes in the core pressure drop, flow rate, or core bypass caused by the implementation of the CE 16x16 NGF fuel assembly design.

The FACTRAN code uses a radial fuel pellet heat transfer model for calculating the transient temperature distribution in a cross-section of a fuel rod for a single axial node in the fuel channel. FACTRAN does not contain a detailed coolant thermal-hydraulics model. The FACTRAN code is used to calculate the hot channel average heat flux versus time for an external DNBR evaluation model such as VIPRE, or for calculating the hot spot fuel and clad temperature versus time with or without assuming DNB. FACTRAN includes the ability to input fuel or clad properties models to take into account changes in materials properties. The FACTRAN calculation is not sensitive to the details of the fuel assembly design changes addressed here, and the results would only be slightly affected by the small changes in the core pressure drop, flow rate, or core bypass expected with the implementation of the CE 16x16 NGF fuel design.

The VIPRE code includes both a radial fuel pellet heat transfer model and a detailed multi-dimensional core thermal-hydraulics model. The VIPRE code may be used in place of FACTRAN to calculate the hot spot fuel and clad temperature versus time for certain transients with or without DNB. Changes in fuel or clad properties models can be taken into account using the code input. In addition, the VIPRE code is used with a subchannel model to perform a DNBR analysis for selected transients. The effect of the changes in the fuel assembly design addressed here are either insignificant or are taken into account as described in Section 4.4 of this report.

The TORC code is used to determine the thermal margin of the hot rod in the core. TORC solves the conservation equations for a 3-dimensional representation of the open-lattice core to determine the local coolant conditions at all points within the core. These coolant conditions are then used with a critical heat flux (CHF) correlation supplied as a code subroutine to determine the minimum value of DNBR for the reactor core. A simpler, faster running code, CETOP-D is also used for thermal hydraulic evaluations and for plant monitoring in the online systems. During the reload process, the CETOP model is tuned to provide results conservative with respect to the more detailed TORC model.

The STRIKIN-II code is used to calculate core and fuel response to the CEA Ejection event. A point kinetics model predicts the core wide power response to the insertion of positive reactivity. In the fuel rod, a one-dimensional cylindrical heat conduction equation is solved for each axial region along the fuel rod. The conduction model explicitly represents the gas gap region and dynamically calculates the gap conductance in each axial region. A volume average temperature for each radial node is defined by

assuming spatially constant material properties within each radial node for one time step. The STRIKIN-II code uniquely determines a heat transfer regime at the clad\coolant boundary for the updated temperature distribution.

The TWINKLE code uses a finite-difference solution of the transient neutron diffusion equations with a relatively simple transient fuel and thermal-hydraulics model. It is used to calculate the core response for rapid reactivity insertion events (i.e., Bank CEA Withdrawal from Subcritical and Rod Ejection) where the RCS loop response is not important. The code is used in a one-dimensional model with multiple axial nodes representing the average core. The TWINKLE code models are not affected by the details of the fuel assembly design or the design changes which are addressed here.

In summary, the computer codes and methods used in the non-LOCA safety analysis are essentially unaffected by the fuel assembly design changes addressed here, and remain valid for use in the safety evaluation of a plant implementing the CE 16x16 NGF fuel design.

5.1.3 Non-LOCA Accident Evaluation

This section provides a qualitative assessment of the expected effect of the CE 16x16 NGF fuel design changes on the non-LOCA analyses. The assessment will rely on previous experience with similar changes for Westinghouse plants. The discussion that follows is divided into sections based on the following classifications of non-LOCA events:

- Increase in Heat Removal by the Secondary System,
- Decrease in Heat Removal by the Secondary System,
- Decrease in Reactor Coolant Flow Rate,
- Reactivity and Power Distribution Anomalies, and
- Events Resulting in Increasing/Decreasing RCS Inventory.

5.1.3.1 Increase in Heat Removal by the Secondary System

A malfunction which causes an increase in heat removal by the secondary system results in a decrease in the temperature of the primary coolant. In the presence of a negative Moderator Temperature Coefficient (MTC), this can result in an increase in the core power level and a reduction in the minimum DNBR. In addition, if the malfunction is due to an increase in feedwater flow, this can cause overfilling of the steam generator.

The events typically analyzed for CE plants are:

- Feedwater System Malfunctions,
- Increase in Secondary Steam Flow, and
- Steamline Depressurization/Steamline Break events.

These transients are primarily "system-driven" in that the system transient results are not dictated by specifics of the fuel assembly geometry, but rather by the response of the RCS to the transient conditions. The details of the fuel assembly and fuel rod design are not modeled in the system transient and are not critical parameters in the system response.

For Condition I and II events, the analyses of these events are performed to confirm that the primary coolant temperature reduction and associated insertion of positive reactivity does not result in an excessively large power increase that challenges the DNB limit for the plant. For Condition III and IV events, the extent of DNB and linear heat rate limit violations are examined. Although the DNB analysis of the fuel will be affected by this fuel change, the overall RCS statepoints (i.e., power, temperature, flow, pressure) will not be significantly different.

An evaluation will be performed to address the increase in vessel pressure drop and potential changes in core bypass flow and core stored energy. However, these changes will not have a significant effect on the results of the non-LOCA analyses.

With respect to DNB, the new Mid grid and IFM designs will improve the DNB performance of the fuel. An evaluation or analysis will be performed to quantify the effect of changes in the fuel assembly DNB performance on the results for the increased heat removal DNB analyses. These changes will be seen in the setpoint analysis, the fuel failure analysis and the DNB correlation (e.g., NGF) used in the analysis.

5.1.3.2 Decrease in Heat Removal by the Secondary System

A malfunction which causes a decrease in heat removal by the secondary system results in an increase in the temperature of the primary coolant. The heatup and expansion of the coolant can lead to a reduction in the DNBR, a primary or secondary system pressure increase, or pressurizer overfill.

The events typically analyzed for CE plants are:

- Loss of Electrical Load/Turbine Trip,
- Loss of Non-Emergency AC Power,
- Loss of Normal Feedwater,
- Feedwater System Pipe Break

As with the cool-down events, these events are primarily system-driven. The details of the fuel assembly and fuel rod are not modeled in the system transient and are not critical parameters.

For example, the Loss of Normal Feedwater/Feedwater Pipe Break events are driven by the heat transfer between the primary and secondary side and, in particular, the performance of the auxiliary feedwater system. The details of the fuel assembly and fuel rod are not modeled and are not critical parameters.

The analyses of these events are performed to confirm that limits on RCS pressure, pressurizer water volume, and secondary side pressure are met. For a plant-specific application, an evaluation will be performed to address the consequences of an increase in vessel pressure drop, potential changes in core bypass flow and stored energy in the fuel and RCS coolant. However, these changes will not have a significant effect on the results of the non-LOCA heatup events. With respect to DNB, the new Mid grid and IFM designs will improve the DNB performance of the fuel. These changes will be seen in the setpoint analysis and the DNB correlation (e.g., NGF) used in the analysis. An evaluation will be performed to confirm that DNB and linear heat rate limit violations do not occur for this class of events.

The Loss of Non-Emergency AC Power event can also result in a flow coastdown due a loss of power to the reactor coolant pumps. This is addressed in the section below.

5.1.3.3 Decrease in Reactor Coolant Flow Rate

A malfunction which causes a decrease in reactor coolant flow rate results in an increase in the temperature of the primary coolant in the core, and a decrease in the ability of the coolant to remove heat from the fuel. This can cause a reduction in the minimum DNBR.

The events typically analyzed for CE plants are:

- Partial/Complete Loss of Forced Reactor Coolant Flow,
- Reactor Coolant Pump (RCP) Shaft Seizure or Shaft Break.

For a plant-specific application, an evaluation will be performed to address the consequences of an increase in vessel pressure drop and potential changes in core bypass flow and core stored energy. New flow coastdown curves will be generated for the 4 pump loss of flow and Reactor Coolant Pump (RCP) Shaft Seizure or Shaft Break. If the coastdown curves are more adverse than current analyses of record, evaluations will be performed to confirm that the DNBR results for the Loss of Flow and Locked Rotor events remain valid.

With respect to DNB, the new Mid grid and IFM designs will improve the DNB performance of the fuel. These changes will be seen through the use of the DNB correlation associated with NGF fuel, Reference 4, used in the analysis.

5.1.3.4 Reactivity and Power Distribution Anomalies

Several non-LOCA transients are characterized by changes, either locally or globally, in core reactivity or power shape. The resulting increase in core power, or the core power peaking factor, could cause a reduction in the minimum DNBR. In the case of the CEA Ejection event, the concern is the post-DNB, pellet temperature and enthalpy increase.

The events typically analyzed for CE plants are:

- Uncontrolled Control Element Assembly (CEA) Withdrawal,
- Dropped/Misaligned CEA events,
- Uncontrolled Boron Dilution, and
- Spectrum of CEA Ejection events.

The Rod Withdrawal at Power, Uncontrolled CEA withdrawal from a Low Power and Subcritical Conditions, and Dropped/Misaligned CEA events are not expected to be significantly affected by the proposed fuel changes.

With respect to DNB, the new Mid grid and IFM designs will improve the DNB performance of the fuel. These changes will be seen in the setpoint process Core Thermal Limits and the DNB correlation (e.g., NGF) used in the analyses. An evaluation will be performed to confirm that the DNBR results for these events remain valid.

Changes in the overall RCS hydraulic parameters, such as core bypass flow and pressure drop, will also have to be evaluated but will not have a significant effect on the results of these analyses.

The Control Rod Ejection event is the result of the assumed mechanical failure of a control rod mechanism pressure housing such that the reactor coolant system pressure would eject the control rod and drive shaft to the fully withdrawn position. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

For plants which perform a DNBR analysis of the CEA Ejection event, improvements to the results is expected through the use of the NGF DNBR critical heat flux correlation, Reference 4.

Changes in the overall RCS hydraulic parameters will not significantly affect this analysis since the transient is over very quickly.

The Uncontrolled Boron Dilution event is the addition of unborated water to the RCS resulting in a positive reactivity insertion and erosion of plant shutdown margin. The proposed fuel changes will not affect this analysis since the details of the fuel are not modeled.

However, the cycle specific RCS initial boron concentration, critical boron concentration, and shutdown requirements must be reviewed against the analysis assumptions to ensure that the results remain valid. This will be performed as part of the normal reload process.

5.1.3.5 Events Resulting in Increasing/Decreasing RCS Inventory

These non-LOCA events are characterized by either an increase or decrease in RCS water inventory. The events typically analyzed for CE plants are:

- RCS Depressurization,
- Letdown Line Break,
- Steam Generator Tube Rupture, and
- Inadvertent Operation of the ECCS.

These transients are "system-driven" events and are not typically DNB limiting. Fuel details such as the cladding material, pellet density, and burnable absorber are not modeled in these analyses.

Therefore, the proposed fuel changes will not have a significant effect on the results of these analyses.

5.1.4 Conclusions

Based on the assessments provided above, the proposed fuel changes associated with CE 16x16 NGF will not have a significant effect on the non-LOCA analyses since the DNB performance of the fuel will improve due to the new Mid grid design, the addition of IFM grids and the use of the NGF critical heat flux correlation. Some evaluations will be performed to address changes in the DNB performance and RCS hydraulic parameters. These will be addressed in a plant specific application. These evaluations will demonstrate that implementation of the new fuel design does not result in any violations of the non-LOCA analysis acceptance criteria. In addition to the event-specific evaluations described above, the normal reload process will be followed to ensure that the fuel-related analysis assumptions remain bounding.

5.2 LOCA

5.2.1 LOCA Introduction and Overview

This section addresses the effect of the CE 16x16 NGF design on the LOCA-related analyses, including ECCS Performance and Blowdown Loads analyses. Referring to Section 2.2, the following new features associated with the CE 16x16 NGF designs need to be evaluated:

- Reduced fuel rod outer diameter and fuel pellet diameter
- Mid grid mixing vane design with an I-spring rod support
- Introduction of Intermediate Flow Mixing (IFM) grids

These design features primarily affect the following aspects of the LOCA-related analyses:

1) Core thermal-hydraulic calculations that are dependent on fuel assembly and fuel geometric parameters
2) Fuel assembly loss coefficient/pressure drop
3) Spacer (Mid grid) and IFM grid geometry (blocked area ratio, open area fraction, inner strap thickness and inner strap height)
4) Core flow redistribution during transition cycles

The CE 16x16 NGF design utilizes Optimized ZIRLO™, an advanced cladding alloy which has been approved by the NRC in References 5 and 6. CE 16x16 NGF design features and changes that impact fuel performance characteristics, which are initial conditions to LOCA analyses, are evaluated and described in Section 2.5. The LOCA analysis methodologies for CE plants explicitly interface with fuel performance initial conditions using design specific inputs and methodologies appropriate for the plant specific applications.

The following ECCS Performance-related analyses, which will use NRC-accepted models and methods, are addressed:

- Section 5.2.2 – Large Break LOCA
- Section 5.2.3 – Small Break LOCA
- Section 5.2.4 – Post-LOCA Long-Term Cooling
- Section 5.2.5 – Transition Core Evaluation

The LOCA Hydraulic Blowdown Loads analysis is addressed in Section 5.2.7.

The Appendix K steam cooling heat transfer component model in the Westinghouse LBLOCA Evaluation Model for CE plants has been modified to include spacer grid heat transfer effects. The details of this improvement to the Appendix K Evaluation Model are documented in Appendix A for NRC review and approval.

5.2.2 Large Break LOCA

For plants transitioning to the CE 16x16 NGF design, a Large Break LOCA (LBLOCA) analysis will be performed using either the Westinghouse best-estimate method or the Westinghouse Appendix K method for CE plants. The currently accepted Evaluation Models for these two methods are described in References 53 and 54, respectively. Future versions of these Evaluation Models may be utilized, however, for this Core Reference Report, the effects of CE 16x16 NGF designs are examined in the context of these two LBLOCA Evaluation Models.

5.2.2.1 Best Estimate Large Break LOCA

The Westinghouse best-estimate Large Break LOCA methods utilize the NRC-approved WCOBRA/TRAC computer code, which has explicit models for fuel assembly geometry, hydraulic resistance, and spacer grid heat transfer. As such, the changes in fuel assembly geometry, loss coefficient/pressure drop and grid geometry for the CE 16x16 NGF design can be handled through appropriate specification of the WCOBRA/TRAC input.

Due to the addition of IFM grids, the distance between grids in the corresponding spans is reduced relative to the standard 16x16 designs. Since the standard core axial noding in WCOBRA/TRAC uses two nodes between each structural (non-IFM) spacer grid, the core axial noding for analyzing a full core of the CE 16x16 NGF design will be the same as for a core without IFM grids. As with current designs, the continuity cell placement for a full core of the CE 16x16 NGF design will be determined using the basic approach described in Section 20-1-2 of Reference 55. Any effects of the fuel design differences between the 16x16 standard design and the CE 16x16 NGF designs will be reflected in the results of the full-core analyses.

Since the CE 16x16 NGF fuel assembly has a higher pressure drop, a transition core evaluation will also be performed to assess the effect of flow redistribution on the CE 16x16 NGF assemblies during the transition cycles. An explicit calculation of the transition core configuration will be performed to support this evaluation. The results of the evaluation will determine the transition core effect that will be applied to the full-core CE 16x16 NGF case to establish the overall results for the CE 16x16 NGF design during the transition cycles.

5.2.2.2 Appendix K Large Break LOCA

The Westinghouse ECCS Performance Appendix K Evaluation Model for CE plants is the 1999 Evaluation Model (1999 EM) for LBLOCA[54]. The 1999 EM for LBLOCA is augmented by CENPD-404-P-A for analysis of ZIRLO™ cladding[8] and by Addendum 1 to CENPD-404-P-A for analysis of Optimized ZIRLO™ cladding[5]. Also, the 1999 EM is supplemented by WCAP-16072-P-A[7] for implementation of Zirconium Diboride (ZrB_2) Integral Fuel Burnable Absorber (IFBA) fuel assembly designs.

The 1999 EM for LBLOCA includes the following computer codes: CEFLASH-4A and COMPERC-II perform the blowdown and refill/reflood hydraulic analyses, respectively. In addition, COMPERC-II calculates the minimum containment pressure and FLECHT-based reflood heat transfer coefficients. STRIKIN-II performs the hot rod heatup analysis. COMZIRC, which is a derivative of the COMPERC-II code, calculates the core-wide cladding oxidation percentage. The 1999 EM is NRC-accepted for ECCS performance analyses of CE plants fueled with Zircaloy-4, ZIRLO™, or Optimized ZIRLO™ clad fuel assemblies. All of the 1999 EM computer codes and methods have explicit inputs for representing the geometric features of the fuel rod and have explicit models for fuel assembly hydraulic resistance that are sufficient for the CE 16x16 NGF design.

The CE 16x16 NGF design changes that impact LOCA analyses and in particular ECCS performance analyses have been encountered in previous CE plant fuel design evaluations with the exception of IFM grids. For example, CE plant fuel design characteristics that have been implemented previously include changes in fuel rod diameter (14x14, 15x15, and 16x16), pellet diameter (value-added pellet), cladding type (ZIRLO™), spacer grids (Guardian™ and Turbo), integral burnable absorber (IFBA – Erbia, Gadolina, and ZrB_2), and axial blankets.

Previous experience with implementing fuel design changes relied on the commonality among CE fuel assembly designs and on the explicit representation of fuel design changes via normal computer code inputs. The same commonality of design and the same representation through normal computer code inputs exist for the implementation of CE 16x16 NGF design as for previous CE plant fuel assembly design changes. For example, CE plants with 14x14, 15x15, or 16x16 standard fuel assemblies have a fuel rod pitch to diameter ratio (P/D) of roughly 1.32. The CE 16x16 NGF design has a P/D ratio of 1.35. This CE 16x16 NGF value differs from the standard design value by a relatively small amount, only 2%, which translates into a difference in the core cross-sectional flow area of only 3-4%. The P/D ratio and core flow area are explicitly represented in various 1999 EM component models through computer code input parameters. These 1999 EM component models all calculate a large range of variation during the LBLOCA transient due to thermal-hydraulic effects compared to the small impact of the change in P/D ratio.

The following lists the computer code input parameters that represent the specific fuel assembly design aspects pertinent to the implementation of CE 16x16 NGF fuel in CE plants:

- Fuel performance parameters such as initial stored energy, initial cladding and pellet dimensions, initial fuel rod internal pressure and gas volume distribution versus burnup are input through the output from an approved fuel performance code and through other standard fuel specific computer code inputs.
- Similarly, physics parameters such as axial power shapes for representing blankets, radial peaking and pin power census are input through standard physics related computer code inputs.
- Cladding type is a specific option for selecting the appropriate physical models for ZIRLO™ and Optimized ZIRLO™ cladding including rupture, rupture strain, and assembly blockage models.
- Hydraulic pressure losses in the core are specifically represented in the blowdown and reflood transient systems codes using fuel design-specific thermal-hydraulics data.

- All fuel rod and fuel assembly geometric characteristics of CE 16x16 NGF that are pertinent to core-wide representation or single hot rod representation are specifically input to the computer codes.

The adequacy and the range of applicability of the NRC-accepted component models of the 1999 EM have been confirmed for the CE 16x16 NGF design. In particular, the CE 16x16 NGF fuel design characteristics can be handled through appropriate specification of the computer code input for the following list of 1999 EM fuel rod or core component models:

- Core blowdown and reflood thermal hydraulics for mass and energy release, core recovery, and steam venting
- Fuel rod pellet stored energy, gap conductance, and cladding ballooning and rupture
- Fuel assembly blockage using the NUREG-0630 methodology and CENPD-404-P-A, Addendum 1-A
- Reflood heat transfer using the FLECHT correlation adjusted to represent the CE 16x16 NGF coolant channel and axial power shape
- Blowdown hydraulics lateral flow for the three radial region representation of the core
- Steam cooling heat transfer for core reflood rates less than 1 in/sec including flow redistribution and recovery around the rupture region
- Rod-to-rod thermal radiation including specific geometric inputs for the limiting enclosure

As discussed above, any effects of the differences between the 16x16 standard design and the CE 16x16 NGF designs will be reflected in the plant-specific results of the full-core analyses. Since the CE 16x16 NGF fuel assembly has a higher pressure drop, a transition core evaluation will also be performed to assess the effect of flow redistribution on the CE 16x16 NGF assemblies during the transition cycles. The results of the evaluation will determine the transition core effect that will be covered by the bounding full-core CE 16x16 NGF analyses.

5.2.3 Small Break LOCA

For plants transitioning to the CE 16x16 NGF design, a Small Break LOCA (SBLOCA) analysis will be performed using the Westinghouse ECCS Performance Appendix K Evaluation Model for CE plants described in Reference 56. This Evaluation Model is referred to as the Supplement 2 Evaluation Model (S2M). In the future, other SBLOCA Evaluation Models may be utilized, however, for this Core Reference Report, the effects of CE 16x16 NGF designs are examined in the context of the S2M SBLOCA Evaluation Model. The S2M for SBLOCA is augmented by CENPD-404-P-A for analysis of ZIRLO™ cladding[8] and by Addendum 1 to CENPD-404-P-A for analysis of Optimized ZIRLO™ cladding[5]. Also, the S2M is supplemented by WCAP-16072-P-A[7] for implementation of ZrB_2 IFBA fuel assembly designs.

The S2M for SBLOCA uses the following computer codes: CEFLASH-4AS performs the hydraulic analysis prior to the time that the Safety Injection Tanks (SITs) begin to inject. After injection from the SITs begins, COMPERC-II is used to perform the hydraulic analysis. COMPERC-II is only used in the SBLOCA evaluation model for larger break sizes that exhibit prolonged periods of SIT flow and significant core voiding. The hot rod heatup analysis is performed by STRIKIN-II during the initial period of forced convection heat transfer and by PARCH during the subsequent period of pool boiling heat transfer. The S2M is NRC-accepted for ECCS performance analyses of CE plants fueled with Zircaloy-4, ZIRLO™, or Optimized ZIRLO™ clad fuel assemblies. All of the S2M computer codes and methods have explicit inputs for representing the geometric features of the fuel rod and have explicit models for fuel assembly hydraulic resistance that are sufficient for the CE 16x16 NGF design.

SBLOCA transients are characterized by a gradual top-down draining of the reactor coolant system, with low flow rates in the core relative to those occurring at steady-state or for LBLOCA transients. The hydraulic losses in the core due to frictional drag, form loss, and acceleration are small, and reasonable variations in the flow resistance would be expected to have a negligible effect on the SBLOCA analysis results. Spacer and IFM grids are not explicitly modeled in the S2M.

The effects of core level swell and phase separation in low flow core reflood conditions are represented with fundamentally based models in the S2M, where the CE 16x16 NGF core and fuel rod geometries are explicitly represented through computer code input. The phase separation and level swell models utilized in the S2M have no specific fuel rod geometry dependent inputs, with only pressure, temperature, and void fraction as the primary dependencies. These models are acceptable for application to the CE 16x16 NGF core.

As discussed above, any effects of the differences between the 16x16 standard design and the CE 16x16 NGF designs will be reflected in the plant-specific results of the full-core analyses. No SBLOCA mixed-core analysis is necessary during transition core cycles due to the negligible effect of variations in core hydraulic losses on SBLOCA analysis results.

5.2.4 Post-LOCA Long-Term Cooling

Analyses performed with the Westinghouse post-LOCA long-term cooling evaluation model for CE plants (CENPD-254-P-A[(57)]) are not sensitive to the fuel assembly changes being introduced for the CE 16x16 NGF design. As a result, no plant-specific post-LOCA long-term cooling analyses are required to support the introduction of the CE 16x16 NGF fuel assembly.

5.2.5 Transition Core Evaluation

Sections 5.2.2.1 and 5.2.2.2 outline the transition core considerations for LBLOCA, and Section 5.2.3 indicates that no mixed-core analysis is necessary for SBLOCA. The post-LOCA long-term cooling analysis is not sensitive to mixed-core effects, so no further consideration is required.

5.2.6 Conclusions

With respect to the ECCS Performance-related analyses, the CE 16x16 NGF design features primarily affect the core, fuel assembly, and fuel rod geometric parameters, the fuel assembly loss coefficient/pressure drop, the spacer and IFM grid geometry, and the flow redistribution during transition core cycles. The adequacy and the range of applicability of the NRC-accepted component models of the Westinghouse ECCS Performance Evaluation Models have been confirmed for the CE 16x16 NGF design. For LBLOCA and SBLOCA, plant-specific calculations will be performed to determine the effect of the CE 16x16 NGF design on the analysis results. Post-LOCA long-term cooling analyses are not sensitive to the changes being introduced for the CE 16x16 NGF design, so plant-specific post-LOCA long term cooling analyses are not required. To address the assembly pressure drop differences, a transition core evaluation will be performed for LBLOCA, while no mixed-core analysis is necessary during transition core cycles for SBLOCA.

5.2.7 LOCA Hydraulic Blowdown Loads

The following discussion focuses on calculations of LOCA hydraulic forces, and their effects on fuel and vessel internals qualification. Although other factors are considered in fuel qualification, such as seismic loading and component weight, the following discussion is primarily constrained to the generation and effects of hydraulic loads resulting from a postulated pipe rupture.

The Westinghouse methodology for determining the hydraulic blowdown loads on the reactor vessel (RV) internals and the core in response to a LOCA in CE-designed PWRs is described in Reference 58.

Based on this methodology, for a given plant design, the parameters that can have a significant effect on the calculated blowdown loads on the RV internals and fuel are:

Parameter a: Coolant temperature (T_{COLD}),

Parameter b: Primary coolant flow rate,

Parameter c: Design changes in and around the core (e.g., grid design),

Parameter d: Steam generator tube plugging,

Parameter e: Break parameters (location, size and opening rate)

Assessment of the impact of the CE 16x16 NGF designs on the above parameters was performed. Consideration of the resulting structural dynamics in similar configurations indicated that the calculated hydraulic blowdown loads were impacted by CE 16x16 NGF in a manner that is computable via a limiting multiplier on the vertical forces on standard fuel. Therefore, plant-specific analyses of the blowdown loads on the RV internals and fuel are not warranted.

In place of plant-specific analyses of blowdown loads on the RV internals and fuel, the plant-specific assessments and calculations outlined below will be performed.

- An assessment will confirm that the break analyzed is a branch line pipe break, not a full size coolant line break.
- An assessment will confirm that significant effects of the CE 16x16 NGF design implementation being considered are limited to the core design data, with no significant change in the core flow rate or coolant temperatures.
- An assessment will confirm that the core design parameter changes due to CE 16x16 NGF designs are limited to approximately 20% higher pressure losses and 3% higher core flow area.
- The lateral forces on RV internals will be based on the hydraulic blowdown loads calculated for the standard fuel.
- The vertical forces on RV internals and fuel will be based on the hydraulic blowdown loads calculated for the standard fuel, and will be increased by a limiting multiplier of 1.15.

If the plant-specific assessments and calculations outlined above are not performed, then plant-specific analyses of blowdown loads on the RV internals and fuel will be performed using the Westinghouse methodology for CE-designed PWRs described in Reference 58.

5.3 Setpoints

The introduction of the CE 16x16 NGF design impacts the setpoint analysis area primarily in the areas of fuel modeling and the application of the NGF critical heat flux (CHF) correlations in the thermal hydraulics design and on-line computer codes. The thermal hydraulics design codes are used to determine or verify setpoints and uncertainties for DNBR-related monitoring and protection systems while the thermal hydraulics on-line codes are incorporated into on-line digital monitoring and protection systems such as the Core Operating Limit Supervisory System (COLSS[64]) and the Core Protection Calculator System (CPCS[65]).

As discussed in Section 4.0, the TORC[29][66] thermal hydraulics computer code can be used to model the CE 16x16 NGF design. In addition, a new CHF correlation for the CE 16x16 NGF design[4] and the ABB-NV CHF correlation [40] have been incorporated into the TORC code. Also as discussed in Section 4.0, the CETOP-D[31] thermal hydraulics computer code is typically used in setpoint analyses. The new CHF correlation [4] and the ABB-NV CHF correlation [40] have also been incorporated into the CETOP-D code. Adjustments are applied to the CETOP-D results based on benchmarking to TORC such that they are conservative relative to corresponding TORC results. Therefore, the current setpoint analysis methodology [36][37] using CETOP-D can be applied to reload cores with CE 16x16 NGF assemblies.

Furthermore, as discussed in Section 4.0, the new CHF correlation[4] and the ABB-NV CHF correlation[40] have been incorporated into the VIPRE-01 thermal hydraulics code which can be used to model the CE 16x16 NGF design. The VIPRE-01 thermal hydraulics computer code is used in the setpoint analyses for some CE plants, such as [ ] a, c, in which the methodology documented in WCAP-8745-P-A[70] is applied. WCAP-8745-P-A is part of the reload evaluation method[35] discussed in Section 4.0, which has been applied and approved by the NRC for the [ ] a, c(71). Therefore, the setpoint analysis methodology[70] using the VIPRE-01 code, for plants that have implemented the reload evaluation method[35], can be applied to reload cores with CE 16x16 NGF assemblies.

Certain CE 16x16 type plants utilize the on-line digital monitoring and protection systems, COLSS and CPCS. The current versions of COLSS and CPCS employ [ ] a,c to perform on-line DNBR and DNBR margin calculations. It is expected that the on-line codes [ ] a, c. The standard reload uncertainty analysis methodology[37] will provide appropriate uncertainty factors for the on-line systems such that the DNB design bases are maintained.

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6.0 Reactor Vessel and Reactor Vessel Internals (RVI) Evaluation

The reactor pressure vessel (RPV) system consists of the reactor vessel, reactor vessel internals (RVI), fuel and control element drive mechanisms (CEDM). The reactor internals function to support and orient the reactor core fuel assemblies and control element assemblies (CEA), absorb CEA dynamic loads, and transmit these and other loads to the reactor vessel. The RVI components also function to direct coolant flow through the fuel assemblies (core), to provide adequate cooling flow to the various internals structures, and to support in-core instrumentation. They are designed to withstand forces due to structure deadweight, preload of fuel assemblies, CEA dynamic loads, vibratory loads, earthquake accelerations, and pipe break loads.

Reloading a reactor core with fuel other than that for which the plant was originally designed requires that the RVI/fuel interface be thoroughly addressed to assure compatibility with the reactor vessel and RVI and to assure that the structural integrity of the reactor vessel and RVI are not adversely affected.

The areas affected by a change in fuel are:

1. RVI System Thermal-Hydraulic Performance
2. RVI System Structural Response to Seismic and Pipe Break Conditions
3. RVI Structural Analysis and Hold Down Ring Clamping
4. CEA scram performance

6.1 RVI System Thermal-Hydraulic Performance

6.1.1 Introduction and Overview

A key area in evaluation of core performance is the determination of hydraulic behavior of coolant flow within the RVI system, i.e. core pressure drop, core bypass flow, and hydraulic lift forces. The pressure loss data is necessary input to the safety analysis and to the Nuclear Steam Supply System (NSSS) performance calculations. The hydraulic forces are critical in the assessment of the structural integrity of the RVI and core clamping loads generated by the internals hold down ring.

6.1.2 Model and Methodology

The thermal-hydraulic analysis models the reactor vessel and internals system in a pressurized water reactors (PWR). The thermal-hydraulic analysis computes the reactor vessel pressure losses for various system flow rates, associated core bypass flows, interior region flow rates, hydraulic uplift forces, and hydraulic and geometrical data.

The reactor vessel pressure losses are calculated by classical analytical fluid mechanics. The thermal-hydraulic analysis solves the continuity and momentum equations for a flow system that represented the entire reactor vessel and internals system.

The thermal-hydraulic analysis utilizes the fuel assembly design loss coefficients and geometric data as inputs. The fuel assembly data is used to determine the pressure drop across the core, which is essential in determining the reactor vessel pressure losses, core bypass flow, and hydraulic uplift forces.

The thermal-hydraulic analysis methodology used in the thermal-hydraulic analysis of the reactor internals remains valid in the analysis of a pressurized water reactor implementing the 16x16 NGF fuel design. With the fuel change, a plant-specific thermal-hydraulic analysis would be performed to address the impact of the fuel change. This plant-specific thermal-hydraulic analysis is performed to assure compatibility with the reactor vessel and internals and to assure that the structural integrity of the reactor vessel and internals are not adversely affected.

Thermal-hydraulic evaluations have already been performed for some plants. The effect of the 16x16 NGF fuel design is a small increase in the pressure drop across the core. This small core pressure drop increase impacts the core bypass flow and the hydraulic lift forces. These example evaluations demonstrate that the reactor internals design criteria are met with the CE 16x16 NGF fuel.

6.1.3 Conclusions

The methodology used in the thermal-hydraulic analysis of the RVI has been used on Westinghouse pressurized water reactors implementing changes in fuel. The thermal-hydraulic analysis methodology remains valid for Westinghouse pressurized water reactor implementing the 16x16 NGF fuel design. For any CE NSSS pressurized water reactor implementing 16x16 NGF, a plant-specific thermal-hydraulic analysis would have to be performed to address the impact of the 16x16 NGF fuel design and verify that design criteria are met.

6.2 RVI System Structural Response to Seismic and Pipe Break Conditions

6.2.1 Introduction and Overview

Changes in fuel assembly properties generally impact the performance of the RVI under all modes of operation. It is, therefore, important that with a change of fuel, the mechanical response of the RVI be evaluated. This is done to assure compatibility of the fuel with the RVI and to assure that the structural integrity of the reactor pressure vessel (RPV) system is not adversely affected. The mechanical system evaluations consist of dynamic response due to seismic and pipe break excitations.

6.2.2 Model and Methodology

The method used in the dynamic analysis of the RVI and fuel is described in detail in Reference 11. The method addresses broadening of the seismic excitation in accordance with Reference 69. A short description of this method follows.

The first step in the dynamic loads analysis is to develop a model of the NGF assembly. This is done in a step-by-step manner using test data from a series of static and dynamic tests as building blocks. The NGF assembly model is then used in the coupled lateral RVI and fuel model and in the detailed core model. For the coupled RVI and fuel model seismic analysis, seismic acceleration time histories are applied to the Reactor Vessel. The CESHOCK code is used to perform the analysis. The equations of motion are integrated to determine the time history response of the RVI and fuel. The pipe break analysis is performed in a similar manner, except that the excitations include both vessel motion and pressure loads due to the blowdown. The results from these analyses include seismic and pipe break loads on RVI components and core boundary motions that are used to excite detailed core models.

Core models are developed to represent different core loading patterns. Full NGF core and several mixed core configurations are considered. The evaluation of the mixed core models and the all NGF model covers the transition from a core with Lead Test Assemblies (LTA), to full batch implementation, to a full NGF core. The detailed core model seismic and pipe break analyses provide spacer grid impact loads and fuel assembly displacement shapes at times of peak response. These results are used as input to the fuel assembly structural evaluation.

Additionally, a coupled axial RVI and fuel model is developed to reflect NGF properties and used for the seismic and pipe break analyses. These analyses provide axial loads on the RVI components and on the fuel.

6.2.3 Conclusion

The methodology used in the seismic and pipe break analysis of the reactor vessel, RVI, and fuel is based on NRC approved methodology. The methodology is valid in the analysis of a Westinghouse pressurized water reactor implementing the 16x16 NGF fuel design. With the fuel change, plant-specific RPV system seismic and pipe break analyses are performed to address the impact of the fuel change. The plant-specific RPV system seismic and pipe break analyses are performed to evaluate RVI and fuel response and to assure that the structural integrity of the RVI are not adversely affected by the change in fuel.

6.3 RVI Structural Analysis and Hold Down Ring Clamping Evaluation

6.3.1 Introduction and Overview

The thermal and hydraulic loads are combined with the fuel mechanical loads and seismic and pipe break dynamic response loads as appropriate to demonstrate that the stresses and deflections in the RVI meet design basis criteria. Additionally, the effects of the changes in the hydraulic loads and fuel mechanical loads on the ability of the hold down ring to adequately prevent the RVI from rocking or sliding during plant operation are assessed.

6.3.2 Methodology

The RVI components are evaluated to assess the impact of revised hydraulic, mechanical, seismic and pipe break input data due to fuel change on the Level A+B (normal operating plus upset condition) and Level D (faulted condition) structural evaluations documented in the analyses of record (AOR). The impact of the revised mechanical and hydraulic input data on the ability of the hold down ring to provide adequate RVI hold down force was also evaluated. Changes in thermal loading of the RVI components is also considered due to fuel change, however the thermal input for the 16x16 NGF fuel design is identical to the standard 16x16 fuel design.

The revised hydraulic input, in the form of hydraulic loads, moments and pressure differentials, reflects the 16x16 NGF fuel design. The revised mechanical input, in the form of core weights and fuel spring loads, also reflects the 16x16 NGF fuel design. The revised seismic and pipe break input, comprising loads and moments on RVI components, again reflects 16x16 NGF fuel design.

All RVI components, both core support structures and internal structures, are evaluated per design basis requirements.

All Level A+B stress intensities are evaluated against design basis criteria. This criteria must be determined on a plant-specific basis, however the criteria is generally consistent with or defined in Section III, Subsection NG of the ASME Boiler and Pressure Vessel Code. These criteria include limitations on primary membrane, primary membrane plus bending, and primary plus secondary stress intensities of $n \times 1 \times S_m$, $n \times 1.5 \times S_m$, and $3 \times S_m$, respectively, where S_m represents the design stress intensity and n represents the weld quality factor, if applicable.

A scoping fatigue evaluation of the RVI components is performed by demonstrating that the peak alternating stress required to achieve maximum allowable fatigue usage was greater than that calculated for any of the RVI components. This evaluation utilizes the appropriate fatigue curve provided in Section III, Appendix I of the ASME Code. Fatigue curves in early editions of the Code were limited to 10^6 cycles. The calculation of high-cycle (> 10^6 cycles) fatigue usage, normally associated with flow-induced vibration, was therefore not required. However, the dynamic hydraulic loads that cause flow-induced vibration are included in the revised hydraulic input described above, and are thus

accounted for in the Level A+B stress evaluation. Therefore, later editions of the ASME Code that employ fatigue curves out to 10^{11} cycles are used to ensure high-cycle fatigue will not adversely affect the RVI.

All Level D stress intensities are evaluated against design basis criteria. This criteria must de determined on a plant-specific basis, however the criteria is generally consistent with or defined in Section III, Appendix F of the ASME Boiler and Pressure Vessel Code. These criteria include limitations on primary membrane and primary membrane plus bending stress intensities of $n \times 2.4 \times S_m$ and $n \times 3.6 \times S_m$, respectively.

The hold down ring exerts a downward force on the CSB and UGS upper flanges; maintaining them in a clamped configuration to prevent rocking and sliding of the CSB and UGS assemblies relative to one another and to the reactor vessel. Excessive wear can develop at the Reactor Vessel and RVI interfacing surfaces if the RVI rocks or slides during normal operating or startup conditions and the rocking and sliding analyses ensure that wear doesn't develop.

The net hold down load is calculated using the hold down ring, dead weight, fuel spring and the vertical hydraulic loads. Sliding margin is defined as the ratio of the lateral (frictional) component of the net hold down load over the applied lateral hydraulic load. Rocking margin is defined as the ratio of the moment generated by the net hold down load over the applied hydraulic moment. Any margin greater than 1.0 will prevent rocking or sliding. Uncertainties and plant transient conditions are accounted for in the analysis.

6.3.3 Conclusion

The analyses performed to demonstrate that the stresses and deflections in the RVI meet design basis criteria is performed on a plant-specific basis. The evaluation of the hold down ring to adequately prevent the RVI from rocking or sliding during plant operation is also performed on a plant-specific basis.

6.4 Control Element Assembly (CEA) Scram Performance

6.4.1 Introduction and Overview

The Control Element Assemblies (CEA) represent one of the most critical interfaces between the fuel and the reactor internal components. Because of this critical interface it is necessary to ensure that the fuel does not adversely impact the operation of the control rods, either during accident conditions or normal operation.

6.4.2 Model and Methodology

The CE 16x16 NGF design maintains the same interface configurations with the CEA as the Standard CE 16x16 design. This includes maintaining the same inside diameters of the posts, guide thimbles, and wear sleeves, as well as maintaining the same number of flow holes, their size, and approximate location. The only aspect of the NGF design that influences the CEA scram times is its increased pressure drop compared to that of the standard design. Analyses performed with the standard CE methodology for a typical CE 16x16 plant have documented that sufficient margin exists to accommodate the slight increase in the CEA scram time due to the NGF pressure drop without violating applicable insertion time requirements.

6.4.3 Conclusion

The evaluation of the CEA scram times associated with the CE 16x16 NGF design for a typical CE 16x16 plant demonstrates the acceptability of the design. However, due to differences in the reactor flow conditions between plants, the implementation of CE 16x16 NGF will include a plant-specific CEA scram time analysis done as part of the standard reload process for the first time implementation of this fuel at a plant to confirm compliance with insertion time requirements.

7.0 Radiological Assessment

The fuel related radiological source terms used in the accident analysis are mainly dependent on the Uranium loading, burnup, and power history of the fuel in the core. Table 3-1 shows that both the fuel rod and fuel assembly Uranium loadings for the 16x16 NGF fuel rod and assembly is within those of previous CE 16x16 value added type fuel and not significantly different from the CE 16x16 Standard fuel. Likewise the power history for the limiting fuel rods is not expected to change significantly from current values. An evaluation of the radiological nuclide source terms used in the accident analyses has been performed to a peak rod average burnup of 62 MWd/kgU and all radiological consequences continue to be acceptable for CE 16x16 NGF (i.e., 10CFR100 limits continue to be met).

7.1 Design Bases

The design bases and functional requirements used for the radiological assessment of the CE 16x16 Next Generation Fuel (NGF) cores are the same as those employed in previous CE 16x16 fuel designs. The design bases are consistent with current NRC regulatory guides.

7.2 Design Methods

No changes to currently approved methods are required to design and analyze the radiological source term in cores containing the 16x16 NGF assemblies. The methodology and values of the source terms are documented in UFSARs. The values are updated if conditions such as power level, power history, or mass of uranium increase above the values assumed in the bounding analysis. The industry standard ORIGEN-II code is the main tool used for radionuclide analysis. This code uses as input the initial mass of U-235 and U-238 and the power operating history. ORIGEN-II performs a very detailed calculation of the evolution of all fission products and actinides, and provides a number of edits of the various concentrations and reaction rates as a function of irradiation time or decay time after shutdown.

Three types of radionuclide source terms are considered in the typical design analysis. These three sources are the MHA (e.g. LOCA), non-LOCA, and Fuel Mishandling source terms. The methodology used in each is discussed below.

7.2.1 Maximum Hypothetical Accident (e.g. LOCA) Source Term

The Maximum Hypothetical Accident (MHA) source term is used for several applications that calculate dose and consequences of a worst case accident scenario such as the LOCA. The calculation of this source terms assumes failure of all fuel rods in the core and subsequent release of all volatile and some solid radionuclides to the primary coolant. The magnitude of the source term is proportional to the power level and the mass of fuel in the core but depends slightly on the core average exposure since almost all of the radioactive isotopes saturate with time.

The implementation of the 16x16 NGF will not change the mass of fuel in the core nor the core power level. The core average exposure is dependent on the cycle length and the number of feed fuel assemblies to the core. The MHA source terms used in the accident analysis has assumed bounding values for the cycle length and the feed batch size which is sufficient to accommodate these cycle to cycle variations. The cycle specific reload analysis confirms that the cycle length and feed batch size are within those assumed in the bounding safety analysis. The implementation of the NGF fuel design will not impact the MHA source term.

7.2.2 Source Terms for Non-LOCA Events

Failure of cladding of some the fuel rods may occur for the most limiting non-LOCA accidents. The potential failure mechanism is fuel centerline melt, which is primarily controlled by the LHGR, or the mechanism is DNB. In both cases, the fuel failures during the non-LOCA type events are limited to high power low burnup rods. ·During such fuel failure the volatile radioactive inventory (primarily krypton, iodine and xenon) is released to the primary coolant.

The non-LOCA source term is primarily dependent on the maximum rod power, the mass of fuel inside the fuel rod, and to a lesser extent the burnup of the fuel rod. The activities of the volatile radionuclides are calculated by ORIGEN-II for a range of hot rod fuel rod burnups and enrichments assuming that the rod has operated at the maximum allowable power for the duration of its residency. It is not necessary to evaluate activities at exposures greater than 40 MWd/kgU because the relative power of a fuel rod having a burnup larger than 40 MWd/kgU will be significantly below the failure threshold during non-LOCA accidents. Furthermore the activities of the short lived iodine, xenon and Kr-83m nuclides quickly assume reduced equilibrium values consistent at the new reduced power level associated with the higher burnup. (The activity of the long lived Kr-85 nuclide (10.7 years) will persist, but its contribution to the total activity is small.)

The implementation of the CE 16x16 NGF will have no significant impact on the non-LOCA source term since neither the maximum allowable rod power or the mass of fuel per fuel rod will be increased.

7.2.3 Source Terms for Fuel Mishandling Events

In the Fuel Mishandling event the fuel assembly is assumed to experience an impact force during fuel movement outside of the core which results in clad breach and subsequent release of the volatile fission products of several fuel rods to the spent fuel pool. Although the list of radionuclides released are the same as for the non-LOCA accident, one important difference is that the Fuel Mishandling accident may involve high burnup fuel as well as low the low burnup fuel that is considered in the generation of the non-LOCA source term. Because of this difference the calculation of the Fuel Mishandling source term must consider the potential higher release of radionuclide from the pellet to the gap in addition to the potential increase in inventory due to burnup.

The Fuel Mishandling source term is primarily dependent on the mass of fuel in the assembly, the power history of the fuel assembly, and the amount of time that has elapsed since shutdown of the reactor. The source terms for the Fuel Mishandling accident is calculated using the ORIGEN-II computer code in a manner consistent with Regulatory Guides 1.25 or 1.183. This calculation assumes that the fuel assembly has been operating at the maximum assembly power consistent with current safety limits.

The release of radionuclides from the pellet to the gap for cases where the fuel assembly burnup is less than 25 MWd/kgU is taken from the Regulatory Guide 1.25. For analysis of Fuel Mishandling events of fuel assemblies with burnups greater than 25 MWd/kgU the methodology described in Regulatory Guide 1.183 or ANSI/ANS Standard 5.4 is used to calculate the release fraction of radionuclides from the pellet. If ANSI/ANS Standard 5.4 is used then the release fraction is calculated as a function of burnup using conservative values of the power and fuel temperature histories. A conservative axial power distribution and power history is assumed and the radial fuel temperature distribution is conservatively calculated by an approved fuel performance code. Because of the reduced fuel temperatures associated with high burnup fuel, the maximum release fraction usually occurs at or just beyond the time of power falloff. Since the maximum radionuclide inventory also occurs at burnups earlier less than this point, there will be no impact on the limiting release fraction.

Since neither the mass of Uranium in the 16x16 CE NGF fuel rod or assembly nor the power history will significantly change from the current values there will be no impact on the FHA source term, and the current FHA methods and values will be appropriate for cores containing the CE 16x16 NGF assembly with peak rod burnups up to 62 MWd/kgU.

7.3 Conclusions

The radioactive source terms following LOCA, non-LOCA, or fuel mishandling events have been evaluated under extended power, burnup, or enrichment limits. It is concluded that the methodology described in the current licensing basis is applicable for evaluating source terms for MHA (e.g. LOCA), non-LOCA, and Fuel Mishandling events for CE 16x16 NGF fuel assemblies for burnups up to 62 MWd/kgU. For burnups significantly above 62 MWd/kgU, the radiological source terms must be reassessed for continued applicability.

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8.0 Conclusion

This topical report presents generic information relative to a combination of improved fuel design features being introduced by Westinghouse and referred to as the CE 16x16 Next Generation Fuel (CE 16x16 NGF) assembly design.

The driving forces and goals of the CE 16x16 NGF design include improving fuel reliability to resolve grid to rod fretting failures, improving fuel performance for high duty operation, and providing enhanced margin. The NGF design features a full complement of components to meet these goals for CE 16x16 plants.

This topical report provides a licensing basis for evaluating the CE 16x16 NGF fuel assembly design and, once approved, will serve as the basis for applications incorporating CE 16x16 NGF design features into any of the CE 16x16 plants. Minor variations in assembly configurations will be required for plant specific applications. These variations will be assessed using the methodology and licensing basis presented in this topical and all of the design bases will continue to be satisfied.

The CE 16x16 NGF design features, licensing bases, and criteria as described in this report have been reviewed with respect to the individual NSSS plant conditions where the CE 16x16 design may be utilized and the licensing bases and criteria have been found to be generically applicable. Plant specific analyses will be performed to confirm the acceptability of the NGF design prior to implementation.

This topical report presents the CE 16x16 NGF design evaluation in conformance with the content guide given in the NRC Standard Review Plan (NUREG 0800)[(2)], refer to Table 1-1. As appropriate, reference is made to any materials already approved by the NRC. The evaluations described herein confirm that CE 16x16 NGF fuel design is compatible with the Westinghouse CE reactor and fuel designs and that the requirements associated with the Standard Review Plan will be met.

Plant specific analyses/evaluations will be done as needed for each initial application of CE 16x16 NGF. The licensing for full region implementation of NGF fuel will require that each plant reference this topical in the COLR reference section as an administrative Technical Specification change and then will meet the requirements of a 10 CFR 50.59 evaluation. These analyses/evaluations will address the transition core effects from the co-resident fuel (referred to as CE 16x16 Standard Fuel) to a full core of CE 16x16 NGF. The licensing basis for the CE 16x16 Standard Fuel design is referenced herein. Changes to this licensing basis for implementing NGF in CE 16x16 plants were defined herein.

Fuel performance models and methods were used to evaluate the CE 16x16 NGF fuel assembly up to a peak rod average burnup of [ ]^{a,c}. However, Westinghouse is only requesting licensing approval of this design to 62 MWd/kgU peak rod average burnup for use in CE NSSS units with existing methodology.

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Appendix A

Improvement to the 1999 EM Steam Cooling Model for Less Than 1 in/sec Core Reflood

The consequences of ECCS performance calculated using the 1999 EM for the CE 16x16 NGF fuel design are adversely impacted by the increase in core hydraulic pressure loss, the increase in core cross-sectional flow area, and the decrease in fuel rod cladding outside diameter. In particular, the core reflood calculations during a LBLOCA are adversely impacted by the changes in the core from CE 16x16 NGF implementation and the core reflood rates that are used to calculate reflood heat transfer coefficients for the hot rod are decreased. The CE 16x16 NGF design changes are estimated to have an insignificant impact on the ECCS performance peak cladding temperature. However, the impact of CE 16x16 NGF design changes on the ECCS performance maximum cladding local oxidation percentage for the hot rod rupture node is estimated to be large enough to warrant specific consideration.

CE 16x16 NGF design changes related to spacer grids impact evaluations using the 1999 EM for CE plants through the impact on hydraulic pressure loss. The 1999 EM does not have NRC-accepted spacer grid heat transfer models available for licensing calculations. Currently, there is no impact from CE 16x16 NGF design changes related to the details of the spacer grid design, placement, or potential impact on heat transfer other than through the core pressure drop change. Therefore, to improve ECCS performance calculated by the 1999 EM, a component model improvement is made to include the effects of spacer grids. The component model being improved is the 1999 EM steam cooling model for less than 1 in/sec core reflood. This improvement to the existing 1999 EM component model is intended to be an optional feature of the 1999 EM that is applicable to the CE 16x16 NGF design changes including Mid grids and IFM grids as well as to any other CE fuel design and will be used in future applications if deemed appropriate.

Spacer grids have an important effect on several key phenomena during the reflood period, including droplet breakup, interfacial heat transfer, and dispersed flow convective heat transfer. For the 1999 EM, these aspects of reflood heat transfer are covered by the use of the empirically-based, Appendix K required, FLECHT correlation. The FLECHT correlation does not explicitly consider spacer grids, and is based on test measurements taken at mid-span locations, which are away from the direct effects of spacer grids. The FLECHT correlation, nevertheless, is considered here as having included the effects of spacer grids, even though the egg-crate grids used in those tests are not like the spacer grids for the CE 16x16 NGF assembly design.

As required by Appendix K for core reflood rates less than 1 in/sec, heat transfer calculations must be based on the assumption that cooling is only by steam. As described below, the 1999 EM component model for steam cooling on the rupture node and above for reflood rates less than 1 in/sec is being improved to include the effects of spacer grids, including IFM grids. This improvement is designed to more accurately model the steam flow rate and the steam cooling heat transfer coefficients on the hot rod rupture node and above. However to maintain a conservative bias for the impact of the improvement, the current NRC-specified EM constraint and limitation for this component model will be maintained;

namely that, the 1999 EM steam cooling model for reflood rates less than 1 in/sec may not yield a heat transfer coefficient greater than determined by the FLECHT correlation.

<u>1999 EM Steam Cooling Model for Core Reflood Rate Less Than 1 in/sec</u>

The 1999 EM NRC-accepted steam cooling model is documented in Reference A.1 Section S III.D.6.b, Reference A.2, and Reference A.3, Section 2.7. To summarize its current configuration, the 1999 EM steam cooling model for core reflood rates less than (<) 1 in/sec is characterized by the following features and methodology constraints:

- The 1999 EM steam cooling model is an Appendix K required model, which is applied to the hot rod rupture node elevation and above when the core reflood rate is < 1 in/sec
- COMPERC-II reflood thermal-hydraulic calculations provide [ ] a, c
- The steam cooling model includes [ ] a, c
- HCROSS calculates single phase steam flow diversion from the hot rod rupture node blocked subchannel to unblocked adjacent subchannels; including flow recovery above the blockage
- PARCH calculates steam cooling heat transfer coefficients through the rupture node blockage and above; including the effect of steam superheating
- STRIKIN-II calculates rod-to-rod radiation heat transfer for the hot rod enclosure, which is also used by PARCH to calculate hot rod cladding temperatures needed for the steam cooling analysis
- The PARCH hot rod-to-coolant energy balance for calculating the steam temperature includes heat from cladding oxidation and decay heat
- The steam cooling model has imposed a FLECHT correlation upper bound that is required by an NRC-specified model constraint

<u>Improved Model for Steam Cooling for Core Reflood Rate < 1 in/sec</u>

The basis for the improved model for steam cooling includes no changes to the current model described above. An approach for improving the steam cooling heat transfer model has been developed utilizing the beneficial aspects of the CE 16x16 NGF spacer grids (both Mid grid and IFM grids) that are not included in the current model. The 1999 EM spacer grid improvements are patterned after models included in the Westinghouse BELOCA methodology[A.4]. The Westinghouse BELOCA spacer grid models have been NRC-accepted for and generically applied to many different spacer grid designs and fuel assembly lattice configurations. To summarize the improved model, the 1999 EM improved steam cooling model for core reflood rates < 1 in/sec includes the following features and methodology constraints:

- The revised steam cooling model considers only the spacer grids above the core two-phase level (both Mid grid and IFM grids)
- PARCH steam cooling heat transfer coefficients on the rupture node and above are augmented by the Westinghouse spacer grid heat transfer enhancement model, Reference A.4 Section 6-2-8
- Below the rupture node and above the core two-phase level, the steam flow rate [ ] a, c

- The FLECHT correlation upper bound required by NRC model constraint is also applied to the spacer grid model improvement, that is, the result of the grid model enhancement can not give a heat transfer coefficient greater than the FLECHT correlation
- Required physical characteristics of the Westinghouse spacer grid heat transfer enhancement model include
    - o Maximum flow area reduction or spacer grid blockage fraction
    - o Fuel lattice hydraulic diameter
    - o Height of the spacer grid, used to estimate wetted surface area
    - o Elevation of top edge of each spacer grid, relative to bottom of core

Model Basis

As described in Reference A.4, Sections 4-6-5 and 5-2-10, spacer grids are structural members of the fuel assembly, which support the fuel rods at a prescribed rod-to-rod pitch. With the exception of CE 16x16 NGF IFM grids in transition cores, all fuel assemblies have spacer grids at the same elevations across the core. Because the grids are at the same elevations, no flow bypass or flow redistribution occurs. Since the grid reduces the fuel assembly flow area, the flow is contracted and accelerated, and then expands downstream of each gridded layer in the core. As the flow is accelerated within the grid and then expands downstream, it re-establishes the thermal boundary layer on the fuel rod, which increases local heat transfer within and downstream of the grid. When the flow is a two-phase dispersed droplet flow, characteristic of PWR blowdown or reflood, the grids promote additional heat transfer effects. Since the grids are unpowered and have a large surface area to volume ratio, they quench before the fuel rods. When the grids quench, they create additional liquid surface area, which helps core cooling conditions by adding additional steam to the vapor stream by evaporation. Because the spacer grid blocks a portion of the fuel assembly flow area, the velocity of the vapor passing through the grid is higher than velocities nearby in the fuel bundle. As a result, the vapor-film relative velocity at the grid is larger, so that a wetted grid below the rupture node elevation has a higher interfacial heat transfer coefficient compared to nearby droplets. A thermal radiation heat transfer model is used to calculate the heat transfer from the adjacent fuel rods to the spacer grid:

[ ] a, c

$$(A-1)$$

where

[ ] a, c

The temperature of the fuel rod in the above representation is taken to be the STRIKIN-II calculated cladding temperature of the average rod of the hot assembly on the axial node adjacent to the spacer grid. The average rod of the hot assembly is used instead of the hot rod, because the hot assembly average conditions are [

] a, c

In order to calculate the spacer grid temperature, the grid is [

] a, c. That is,

[ ] a, c (A-2)

where

[ ] a, c

The grid temperature from this equation is

[ ] a, c (A-3)

The spacer grid heat transfer model provides [

] a, c for use on the rupture node and above, when the reflood rate is < 1 in/sec. Only spacer grids located above the two-phase mixture level and below the rupture node elevation are used for this calculation and the spacer grid temperature must be less than the rewet temperature. That is,

[ ] a, c (A-4)

where

[ ] a, c

Several single-phase experiments show that the continuous phase heat transfer downstream of a spacer grid can be modeled on entrance effect phenomena where the abrupt contraction and expansion result in establishment of a new thermal boundary layer on the heated surface downstream of the grid. The entrance effect heat transfer decays exponentially downstream of the spacer grid and the local Nusselt number decreases exponentially downstream of the grid. Chiou, Hochreiter, and Young (1991)[A.5] summarized the single phase and two-phase experiments that demonstrated the grid convective enhancement effect, and provided a description of the effects of grids on the flow. [

]^{a,c}, which is given by:

[ ]^{a,c} (A-5)

where

[ ]^{a,c}

[

]^{a,c}

The convective heat transfer coefficient from the spacer grid to the vapor is represented by the Condie-Bengston IV correlation using a [

]^{a,c} The use of this correlation is consistent with the existing 1999 EM film boiling model in the CEFLASH-4A and STRIKIN-II codes (Reference A.3, Section 2.2 Equation (2.2.1-1)).

[ ]^{a,c} (A-6)

where

[ ]^{a,c}

Combining these two equations, where the spacer grid itself is located at $Z = 0$, the interfacial heat transfer coefficient for the wetted spacer grid becomes

[ ] a, c $\qquad (A-7)$

Model as Coded

The emissivities of the fuel rod and spacer grid are given by the following from the PARCH code (Reference A.7, Section 3.4.1, Equation 3.4.1-5)

[ ] a, c $\qquad (A-8)$

where

[ ] a, c

The equivalent spacer grid cell diameter is defined as follows

[ ] a, c $\qquad (A-9)$

where

P_{rod} = Assembly fuel rod pitch (ft)

The spacer grid liquid film interfacial surface area for heat transfer is estimated to be the grid metal surface area as follows:

$$A_{grid} = 4(P_{rod})H_{grid}N_{fuelrods} \qquad (A-10)$$

where

H_{grid} = Height of spacer grid (ft)

$N_{fuelrods}$ = Number of fuel rods in the core

The radiative heat flux to the spacer grid is calculated explicitly using the grid temperature from the previous time step. After the grid temperature for the current time step is calculated, the spacer grid temperature is numerically damped to prevent rapid changes as follows:

[ ] a, c $\qquad (A-11)$

where

[ ] a, c

The steam cooling convective heat transfer coefficients on the rupture node and above for reflood rates < 1 in/sec are based on the PARCH steam cooling model, as described above. To include the impact of the spacer grids on this heat transfer coefficient, the Westinghouse spacer grid heat transfer enhancement model is linearly averaged for the nodes located between spacer grid spans at and above the rupture node. This average representation is used because the PARCH and STRIKIN-II nodalizations are equal axial segments that are not specifically located with respect to the spacer grid locations. This nodalization is coordinated with the 1999 EM axial power shape methodology, which is characterized by axially dependent conditions selected for overall conservatism. Use of an average spacer grid enhancement model avoids continuity issues that would be introduced with an explicit axial dependent spacer grid model.

Model Impact

In most calculations with the 1999 EM, the limiting node for peak cladding temperature is generally either the FLECHT cooled node below the rupture node or the steam cooled node immediately above the rupture node. The limiting condition occurs during the time period of the transient when the core reflood rates are calculated to be < 1 in/sec. The rupture node is not usually the limiting node for peak cladding temperature. The impact of the improved steam cooling model for reflood rates < 1 in/sec based on spacer grid heat transfer effects is summarized as follows:

- Below the rupture node, the peak cladding temperature of the FLECHT cooled node is not impacted by the model changes with spacer grid heat transfer effects.
- Above the rupture node, the steam cooled node will experience a decrease in cladding temperature due to implementing the spacer grid heat transfer model effects. Figure A-1 shows this effect on the calculated cladding temperature for the node above the rupture node beginning after roughly 250 seconds. These results are a representative example of the performance of the revised model due to the spacer grid effects. The change in heat transfer coefficient at this elevation above the rupture node is shown in Figure A-2. Note that before 250 seconds in Figure A-2, the FLECHT heat transfer coefficients bound the steam cooling heat transfer coefficients. The magnitude of the reduction in cladding temperature depends on the plant-specific spacer grid arrangement and physical characteristics.
- On the rupture node, for the heat transfer conditions where the steam cooling heat transfer model is being used, the spacer grid model improves the heat transfer coefficient and lower rupture node temperatures are calculated.
- On the rupture node, when the FLECHT heat transfer coefficients are relatively low, the heat transfer calculation is limited by FLECHT and the steam cooling model may not be used. In this case, the spacer grid heat transfer model increases the time interval of FLECHT heat transfer being used to cool the rupture node until such time when the steam cooling heat transfer coefficient becomes less than the FLECHT heat transfer coefficient. This increased time interval for FLECHT cooling also lowers the calculated rupture node temperatures. Figure A-3 shows this effect beginning after roughly 300 seconds in the example case. The change in heat transfer coefficient on the rupture node is shown in Figure A-4. The magnitude of the reduction in cladding temperature depends on the plant-specific spacer grid arrangement and physical characteristics.

- On all nodes, lower temperatures lead to lower calculated local cladding oxidation percentages. The magnitude of the reduction in maximum cladding local oxidation depends on the plant-specific spacer grid arrangement and physical characteristics.

Model Conclusion

An improvement is made to the 1999 EM steam cooling model for < 1 in/sec core reflood rates by utilizing the beneficial aspects of the CE 16x16 NGF spacer grids (both Mid grid and IFM grids). The amount of evaporated liquid that is calculated for the steam flow rate is increased by [

] [a,c] Increasing the steam flow rate leads to improved steam cooling heat transfer coefficients on the rupture node and above provided the FLECHT correlation is not more limiting. The spacer grid model is fundamentally based and applied in an overall conservative manner. The impact of the improved model will depend on the spacer grid arrangement and physical characteristics, which will be reflected in the plant-specific results of the full-core analyses.

References for Appendix A

A.1. CENPD-132P Supplement 1, "Calculational Methods for the C-E Large Break LOCA Evaluation Model," February 1975.

A.2. LD-81-095 Enclosure 1-P-A, "C-E ECCS Evaluation Model, Flow Blockage Analysis," December 1981.

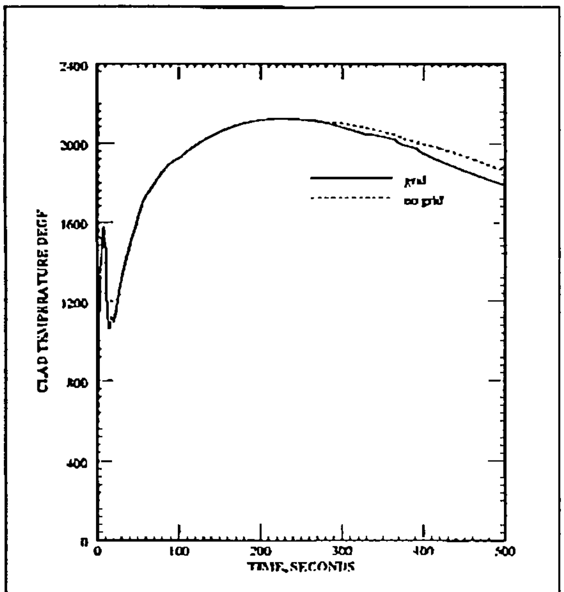
A.3. CENPD-132 Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

A.4. WCAP-12945-P-A, Volume 1 (Revision 2) and Volumes 2 through 5 (Revision 1), "Code Qualification Document for Best Estimate LOCA Analysis," March 1998.

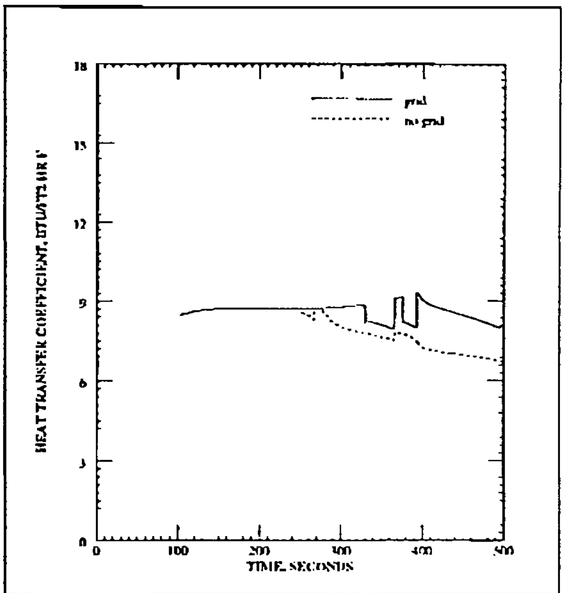
A.5. WCAP-10484-P-A, "Spacer Grid Heat Transfer Effects During Reflood," March 1991.

A.6. Yao, S. C., Hochreiter, L. E., and Leech, J. J., 1982, "Heat Transfer Augmentation in Rod Bundles Near Grid Spacers," J. Heat Transfer, Vol. 104, pp. 76-81.

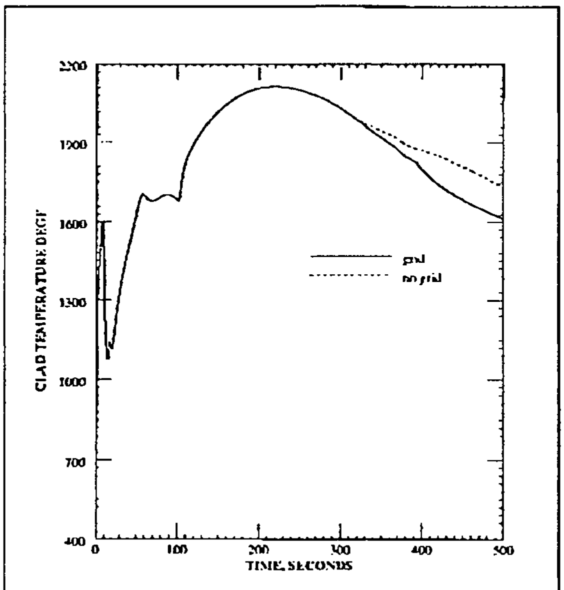
A.7. CENPD-138-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974.



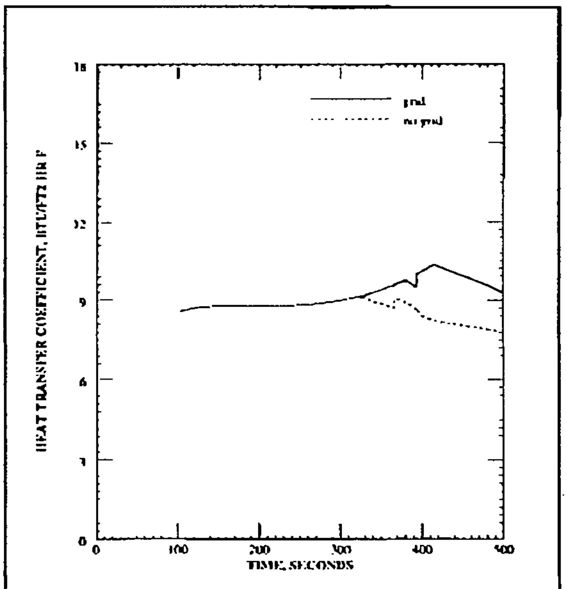
	CLAD TEMPERATURE Node 14 (Above Rupture Node)	Figure A-1



	HEAT TRANSFER COEFFICIENT Node 14 (Above Rupture Node)	Figure A-2



	CLAD TEMPERATURE Node 13 (Rupture Node)	Figure A-3



	HEAT TRANSFER COEFFICIENT Node 13 (Rupture Node)	Figure A-4

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Section C

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Our ref: LTR-NRC-06-50

August 18, 2006

Subject: Response to NRC's Draft Request for Additional Information By the Office Of Nuclear Reactor Regulation Topical Report WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report" (TAC No. MD0560) (Proprietary/Non-Proprietary)

Enclosed are copies of the Proprietary and Non-Proprietary responses for NRC's Draft Request for Additional Information for WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report".

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-06-2191 (Non-proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-proprietary).

This submittal contains proprietary information of Westinghouse Electric Company, LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to this affidavit or Application for Withholding should reference AW-06-2191 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

Attachments

cc: F. M. Akstulewicz, NRR
P. M. Clifford, NRR
J. H. Thompson, NRR

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Our ref: AW-06-2191

August 18, 2006

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-06-50 P-Attachment, Response to NRC's Draft Request for Additional Information By the Office Of Nuclear Reactor Regulation Topical Report WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report" (TAC No. MD0560) (Proprietary)

Reference: Letter from J. A. Gresham to NRC, LTR-NRC-06-50, dated August 18, 2006

The application for withholding is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-06-2191 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-06-2191 and should be addressed to J. A. Gresham, Manager of 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

AW-06-2191

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

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

Sworn to and subscribed
before me this 18th day
of August, 2006.

Sharon L. Fiori
Notary Public

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

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse) and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking 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 which 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 LTR-NRC-06-50 P-Attachment, Response to NRC's Draft Request for Additional Information By the Office Of Nuclear Reactor Regulation Topical Report WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report" (TAC No. MD0560) (Proprietary), for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-06-50) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company is responses to NRC's draft Request for Additional Information.

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

(a) Demonstrate the acceptability of the CE 16x16 Next Generation Fuel.

(b) Assist customers in implementing an improved fuel product.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this fuel design to further enhance their licensing position over their competitors.

(b) Assist customers to obtain license changes.

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 fuel design 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 for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

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Response to NRC's Draft Request for Additional Information By the Office Of Nuclear Reactor Regulation Topical Report WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report"

This document is the property of and contains Proprietary Information owned by Westinghouse Electric Company LLC and/or its subcontractors and suppliers. It is transmitted to you in confidence and trust, and you agree to treat this document in strict accordance with the terms and conditions of the agreement under which it was provided to you.

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NRC's Draft Request for Additional Information
By the Office Of Nuclear Reactor Regulation
Topical Report WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report" (TAC No. MD0560)(Proprietary)

Question: Please provide information as noted in the following table and also provide the power history and axial power profiles to run a case in FRAPCON.

Spring Dimensions
spring outer diameter
spring wire diameter
number of spring turns

Pellet Shape
Pellet Height
Central Hole Radius
Dish Radius
Dish Depth

Pellet Isotopics
Fuel U-235 Enrichment
UO_2 or MOX?

Pellet Fabrication
pellet density
open porosity
pellet surface roughness
expected density increase
sintering temperature

Cladding Fabrication
Cladding type
Cladding cold work
Cladding surface roughness
cladding texture factor
Hydrogen in cladding

Rod Fill Conditions
Fill gas pressure
Fill Gas

Code Operation
number of radial nodes
number of radial gas nodes
number of axial nodes
internal creep step
fission gas atoms/100 fissions
limit on swelling
end node heat to plenum

Response:

In response to the request from NRC on CE 16x16 NGF fuel performance input data, the fuel rod design information, and model options are listed in Table 1, the rod average power is provided in Table 2, and the axial power distribution is provided in Table 3.

These data will be used by NRC to create FRAPCON audit cases. Certain requested parameters (i.e., sintering temperature, cladding cold work, etc.) are not used by Westinghouse methodology for CE fuel and therefore are not readily available; however, the default values in FRAPCON are judged to be appropriate to model the related properties of CE 16x16 NGF fuel.

Data provided in this response are for CE 16x16 NGF fuel and are based on Westinghouse CE fuel performance methodology, but the data are not applicable for plant specific implementation.

Table 1
Fuel Performance Input Data for
CE 16x16 Next Generation Fuel UO_2 and ZrB_2 Fuel Rod

a, c

a, c

Table 2
Normalized Rod Average Power for UO_2 and ZrB_2 Rods*

a, c

a, c

Table 3 Axial Power Shape for UO_2 and ZrB_2 Rods

a, c

Section D

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Direct fax: (412) 374-4011
e-mail: greshaja@westinghouse.com

Our ref: LTR-NRC-06-66
November 29, 2006

Subject: Response to NRC's Request for Additional Information By the Office Of Nuclear Reactor Regulation Topical Report WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report" (TAC No. MD0560)(Proprietary/Non-proprietary)

Enclosed are copies of the Proprietary and Non-Proprietary responses for NRC's Request for Additional Information for WCAP-16500-P/WCAP-16500-NP "CE 16x16 Next Generation Fuel Core Reference Report".

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-06-2220 (Non-proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-proprietary).

This submittal contains proprietary information of Westinghouse Electric Company, LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or Application for Withholding should reference AW-06-2220 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: R. Landry, NRR
P. Clifford, NRR
H. Cruz, NRR
J. Thompson, NRR

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Our ref: AW-06-2220
November 29, 2006

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-06-66 P-Attachment, Response to NRC's Request for Additional Information By the Office Of Nuclear Reactor Regulation Topical Report WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report" (TAC No. MD0560) (Proprietary)

Reference: Letter from J. A. Gresham to NRC, LTR-NRC-06-66, dated November 29, 2006

The application for withholding is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-06-2220 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-06-2220 and should be addressed to J. A. Gresham, Manager of 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

Cc: R. Landry, NRR
P. Clifford, NRR
H. Cruz, NRR
J. Thompson, NRR

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

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

Sworn to and subscribed
before me this 29th day
of November, 2006.

Sharon L. Fiori
Notary Public

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

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse) and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking 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 which 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 "LTR-NRC-06-66 P-Attachment, Response to NRC's Request for Additional Information By the Office Of Nuclear Reactor Regulation Topical Report WCAP-16500-P, 'CE 16x16 Next Generation Fuel Core Reference Report' (TAC No. MD0560) (Proprietary)," November 29, 2006, for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-06-66) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company is responses to NRC's Request for Additional Information.

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

(a) Demonstrate the acceptability of the CE 16x16 Next Generation Fuel and corresponding correlation.

(b) Assist customers in implementing an improved fuel product.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this fuel design with its associated correlation to further enhance their licensing position over their competitors.

(b) Assist customers to obtain license changes.

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 fuel design 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 for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

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Response to NRC's Request for Additional Information By the Office Of Nuclear Reactor Regulation Topical Report WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report" (TAC No. MD0560) (Non-Proprietary)

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

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Request for Additional Information
CE 16x16 Next Generation Fuel Core Reference Report
WCAP-16500-P

Question 1: *The staff's approval of Optimized ZIRLO™ (Addendum 1 to WCAP-12610-P-A and CENPD-404-P-A) included a condition whereby Westinghouse demonstrate the continued applicability of their fuel performance models based upon measured LTA data.*

a. *Provide a schedule for this demonstration which shows that it remains ahead of the burnups achieved by batch implementation.*

b. *Figure 2-15 of WCAP-16500-P provides a comparison of adjusted SIGREEP growth predictions against a limited set of measured assembly growth data. Additional measured data (especially up to fluence levels expected at 62,000 MWd/MTU) are required to validate the adjusted SIGREEP calculations. Update Figure 2-15 with additional measured data.*

Response 1a: The terms 'Optimized ZIRLO™' and 'Low-tin ZIRLO™' as used in this document and in the past documents refer to material with tin levels of 0.6 to 0.79% which are lower than the lowest allowed tin limit of ZIRLO of 0.8%. All other alloying element levels in the materials referred to as Optimized ZIRLO™ and low-tin ZIRLO™ are same as the originally licensed ZIRLO™. The first use of Optimized ZIRLO™ cladding in Byron LTAs was in a [ ]a,c. Subsequent applications of Optimized ZIRLO for cladding are in a [ ]a,c which is currently approved by the NRC for cladding application. In the future, references to low-tin ZIRLO™ will be replaced with Optimized ZIRLO™.

The four listed LTA programs are at different stages of their execution. While the Byron LTA program has concluded, the Calvert-Cliffs, Catawba and Millstone LTA programs are still on-going. The Byron LTA program included both [ ]a,c Optimized ZIRLO™ while the other three LTA programs only included the current [ ]a,c Optimized ZIRLO™. The table below, Table 1-1, provides a summary of the status of the various LTA programs. It should be noted that data and plans associated with future dates are projections and depend on the operation of the plants and thus may change in the future. Data analysis reports will be written in about 9 to 12 months after the LTA inspection data become available.

The listed burnups for the LTAs in Table 1-1 are leading in calendar year of exposure compared to the batchwide exposure of Optimized ZIRLO™. The appropriate LTA data will be checked with the ZIRLO™ model predictions as the data become available.

The projected maximum assembly average burnups for NGF batch implementation at ANO-2 and Waterford 3 are:

ANO-2	Cycle 20	Cycle 21	Cycle 22
Outage Date	Fall 2009	Spring 2011	Fall 2012
Maximum Assembly Average Burnup, MWD/T	27032	48400	-

Waterford 3	Cycle 16	Cycle 17	Cycle 18
Outage Date	Fall 09	Spring 2011	Fall 2012
Maximum Assembly Average Burnup, MWD/T	26697	47438	52844

Response 1b: The data plotted in Figure 2-15 of WCAP-16500-P includes all available measured growth data for guide thimbles fabricated with [ ]^{a,c} ZIRLO™ tubing. The data presented are from two non-NGF LTAs in a foreign CE NSSS reactor after two cycles of operation. In addition to these LTAs, there are four CE 16x16 NGF LTAs with [ ]^{a,c} ZIRLO™ guide thimbles that are about to complete their first cycle of operation in Waterford-3. PIE inspections will include guide thimble growth measurements of the two non-NGF LTAs after their third cycle of operation (Spring '07) and the four NGF LTAs after each cycle of operation (Fall '06, Spring '08, and Fall '09). As these data become available, Westinghouse will compare the PIE data to predictions using the technique described in Section 2.3.1.1 of WCAP-16500-P and modify the correlation if necessary. Westinghouse will keep the NRC informed of progress on these activities through the semi-annual update meetings.

Table 1-1
Optimized ZIRLO™ LTA irradiation and examination status

a, b, c

Question 2: *Westinghouse's FCEP fuel design change process is not currently applicable to CE fuel assembly designs. As such, the staff's approval of the CE 16x16 NGF bundle design must specifically include all variations of the fuel rod and fuel assembly design.*

a. *Figure 1-1 of WCAP-16500-P illustrates 5 possible variances of the CE 16x16 NGF assembly design. Specify, in detail, variations in assembly and component design included in the regulatory envelope for the CE 16x16 NGF design.*

b. *Figure 2-14 of WCAP-16500-P illustrates a single possible variance of the CE 16x16 NGF fuel rod design (Plant B). Specify, in detail, variations in fuel rod design included in the regulatory envelope for the CE 16x16 NGF design.*

Response 2a and 2b: The NGF design was primarily developed for Plant B. Variations in the NGF design are needed due to plant differences. The key differences are summarized below relative to Plant B.

Plant	**Difference relative to Plant B**
A	Requires an additional mid-grid since existing fuel always utilized an additional grid
B	No difference (Reference Design)
C	Higher seismic requirements at this plant require a modification to selected mid-grids to increase grid strength
D	Top and bottom nozzles are different due to different reactor internals, possible use of higher strength mid-grids at selected locations
E	Fuel assembly length is reduced due to a shorter core

As a result of these differences the distribution of grids with and without mixing vanes and the location of IFM grids change slightly for the different plant designs. Figure 1-1 in WCAP-16500-P summarizes these differences. Table 2-1 provides further detail on the differences.

Table 2-1
NGF Differences for CE 16x16 Plants

	Plant B	Plant A	Plant C	Plant D	Plant E
Fuel Assembly Overall Length	Reference	Same	Same	Longer (consistent with current design)	Shorter (consistent with current design)
Top Grid	Reference	Same	Same	Same	Same
Mid Grids	Reference	Same grid design, 1 additional grid (consistent with current design)	Same number of grids, same grid design at some elevations, higher strength grids at other elevations	Same number of grids, same grid design, except possibly higher strength grids at some elevations	Same grid design, 1 grid less (consistent with current design)
IFM Grid	Reference, number may vary depending upon thermal hydraulic requirements	Same	Same	Same	Same
Bottom Grid	Reference	Same	Same	Same	Same
Top Nozzle	Reference	Same	Same	Similar (differences as necessary due to different internals)	Same except shorter (consistent with reactor internals)
Bottom Nozzle	Reference	Same	Same	Similar (differences as necessary due to different internals)	Same
Outer Guide Thimbles	Reference	Same	Same	Same, except possible enlarged diameter at upper end (like current design)	Same except shorter length (consistent with shorter fuel assembly)
Instrumentation Tube	Reference	Same	Same	Same except for provision for centering ICI	Same except shorter length (consistent with shorter fuel assembly)
Wear Sleeve	Reference	Same	Same	None (consistent with current design)	Same except shorter length (consistent with shorter fuel assembly
Fuel Rod	Reference	Same	Same	Same	Same, except shorter overall and active lengths (active length consistent with current design) and different plenum spring

Question 3: *In many instances, both the Westinghouse suite of models and methods and the CE suite of models and methods have been discussed. The staff's review of CE 16x16 NGF assembly design and supporting implementation methodology addresses both the Westinghouse and CE methods. However, approval of this topical report does not provide a licensee the ability to migrate to reload methodologies beyond their currently approved methods without further NRC review. It is expected that licensees implementing CE 16x16 NGF maintain their currently approved models and methods except where specifically delineated within WCAP-16500-P. Please clarify any deviations from currently approved methods or design criteria being requested as part of the staff's review of WCAP-16500-P.*

Response 3: The deviations from currently approved methods or design criteria in plant licensing basis for WCAP-16500-P are summarized below. If current methodology is not part of plant licensing basis, a plant specific LAR will be provided.

Functional Area **Deviations to current method or design criteria**

Mechanical & Fuel Rod Design

- Use of [ ]^{a,c} on guide thimble growth for [ ]^{a,c} ZIRLO™ guide thimbles
- Use of low tin ZIRLO™ growth data to support grid growth evaluations
- Use of Westinghouse fuel rod growth correlation for shoulder gap evaluations of rods with ZIRLO™ cladding (Reference 1)
- Use of Optimized ZIRLO™ properties (Reference 1) for Optimized ZIRLO™ Mechanical and Fuel rod performance evaluations
- License NGF fuel to a peak rod burnup of 62 MWD/kgU

Nuclear Design

- No deviations
- T-H Design
- Use DNB correlation topical WCAP-16523-P (Reference 2) for NGF fuel
- Application of WSSV & ABB-NV DNB correlations for NGF fuel in VIPRE code
- Application of WSSV-T & ABB-NV DNB correlations for NGF fuel in TORC & CETOP-D codes

Non-LOCA

- Application of WSSV & ABB-NV DNB correlations for NGF fuel in VIPRE code
- Application of WSSV-T & ABB-NV DNB correlations for NGF fuel in TORC & CETOP-D codes
- Possible use of RETRAN, FACTRAN, VIPRE and TWINKLE codes in Non-LOCA safety evaluations

LOCA

- Use of Optimized ZIRLO™ Addendum to CENPD-404-P-A (Reference 1) for Optimized ZIRLO™ LOCA performance evaluations
- Possible use of Best Estimate Large Break LOCA methods defined in WCAP-16009-P-A (Reference 3)
- Grid heat transfer model defined in CENPD-132 Supplement 4-P-A (Reference 4)

Setpoints

- Application of WSSV & ABB-NV DNB correlations for NGF fuel in VIPRE code together with reload methodology defined in WCAP-8745-P-A (Reference 5)
- Application of WSSV-T & ABB-NV DNB correlations for NGF fuel in TORC & CETOP-D codes
- Future use of WSSV, WSSV-T and ABB-NV correlations in the BEACON-COLSS core monitoring system (Reference 6)

Structural

- No Deviations

Radiological

- No Deviations

References:

1. WCAP-12610-P-A and CENPD-404-P-A Addendum 1, "Addendum 1 to WCAP-12610-P-A and CENPD-404-P-A Optimized ZIRLO™," February 2003.
2. WCAP-16523-P, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," March 2006.
3. WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," January 2005.
4. CENPD-132 Supplement 4-P-A, Addendum 1-P, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model Improvement to 1999 Large Break LOCA EM Steam Cooling Model for Less than 1in/sec Core Reflood," May 2006.
5. WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Over-temperature ΔT Trip Functions," September 1986.
6. WCAP-12472-P Addendum 3-A, "BEACON™ Core Monitoring and Operation Support System," June 2006.

<u>*Question 4:*</u> *Flow induced vibration (FIV) testing is required to quantify flow induced assembly vibration and grid-to-rod fretting. Figure 1-1 of WCAP-16500-P illustrates 5 different combinations of grid strip designs and elevations.*

a. Provide the FIV test results for the different CE 16x16 NGF assembly designs.

b. Provide the FIV test results supporting the mixed-core evaluations.

Response 4a: The fuel assembly hydraulic stability is evaluated using vibration [

] ^{a,c}. The single bundle Flow Induced Vibration (FIV) tests for the CE 16x16 NGF and CE 16x16 Standard prototypical assemblies were conducted in the Westinghouse Fuel Assembly Compatibility Test System (FACTS) test loop. Based on previous testing, [

] ^{a,c}.

The CE 16x16 NGF design has a mix of non-vaned grids and mixing vane grids that have the side-supported vane (SSV), [

] ^{a,c}. To verify that the CE 16x16 NGF design does not have unacceptable fuel assembly flow induced vibration, vibration data were taken during the FACTS hydraulic tests [

] ^{a,c}. Plots of the assembly amplitude [
] ^{a,c}, Figures 4-1 and 4-2, show very low assembly motion, [ ] ^{a,c}, at all elevations for the wide test flow range for both assemblies. For these plots the first non-vaned grid in Figure 1-1 is identified as grid 1. Due to the addition of two IFM grids, Grid 11 for the CE 16x16 NGF assembly is at the same elevation as Grid 9 in the CE 16x16 Standard assembly. These results confirm the excitation source for FIV is not present with the SSV mixing vane for the CE 16x16 NGF fuel designs. Without the excitation source, FIV would not occur for any of the five assembly designs shown in Figure 1-1 for full core conditions.

Response 4b: Following the single bundle tests, a dual test was performed in the Westinghouse Vibration Investigation and Pressure-drop Experimental Research (VIPER) test loop to demonstrate acceptable performance for mixed-core applications. The flow induced vibration test was repeated over a wide range of flows prior to an endurance test to confirm that flow induced fuel assembly vibration was not present at mixing core environment. The [

] ^{a,c}.

An endurance test was then performed to confirm improved margin for grid-to-rod fretting with the CE 16x16 NGF grid design. As stated above, [

] [a,c]. The wear results confirmed the greater margin for grid-to-rod fretting wear for [

] [a,b,c].

Since the test results have been acceptable for the [

] [a,b,c].

Reference:

1. WCAP-16006, "Examination of Calvert Cliffs Unit 1 Batch R Lead Fuel Assemblies at EOC-15," April 2003.

Figure 4-1

a, b, c

Figure 4-2

a, b, c

Figure 4-3

a, b, c

Figure 4-4

a, b, c

Question 5: *WCAP-16500-P seeks approval of the CE 16x16 NGF fuel assembly design up to 62,000 MWd/MTU. Included in this design is the flexibility to employ different burnable absorbers (e.g. Gd, Er, ZrB2) and axial zoning (e.g. annular, enrichment). Demonstrate that the currently approved nuclear design and fuel performance analytical models and methods are valid up to 62,000 MWd/MTU for the different fuel rod configurations.*

Response 5: The NGF assembly can employ any of the burnable absorbers, axial (and radial) enrichment zoning, and annular fuel pellets at the top and bottom of the active fuel region. Currently approved Nuclear design and Fuel performance models and methods are applicable, and predicted performance parameters must satisfy performance criteria using these models and methods.

The models and algorithms used in the Westinghouse and CE nuclear design codes are based on widely accepted theoretical first principles of reactor physics. The solutions produced by these physics codes do not contain any empirical methods. (Although benchmarks to experiment and plant measurements are used to establish the uncertainty of the predictions.) Consequently, the small extension in burnup from 60 to 62 MWd/kgU will not significantly affect the accuracy of the predictions from these codes.

Further, the Westinghouse physics methodology (ANC, PARAGON, and PHOENIX) has been used for many years for design analysis of Westinghouse NSSS cores with peak fuel rod burnups of up to 62 MWd/kgU. These cores have included both ZrB_2 IFBA and gadolinia burnable absorber types.

The CE physics methodology (ROCS and DIT) has been used for many years for analysis of CE NSSS type cores with peak fuel rod burnups of up to 60 MWd/kgU. These cores have contained both erbia and gadolinia burnable absorbers. In addition ROCS and DIT were also used for the analysis of CE cores containing LTAs to demonstrate acceptability to 62 MWd/kgU and above.

Based on these considerations Westinghouse concludes that the physics models will remain valid for peak fuel rod burnups well beyond 62 MWd/kgU.

The approved fuel performance model, FATES3B, has the capability to accurately treat the impact of burnable absorbers and annular fuel features. Standard UO_2 fuel rods were demonstrated to be acceptably modeled in the design and licensing calculations in Section 2.6 of WCAP-16500-P for CE plants.

The effects of burnable absorbers are discussed in approved topical reports. The addition of gadolinia (CENPD-275-P Revision 1-P Supplement 1-P-A, Reference 1) or erbia (CENPD-382-P-A, Reference 2) will impact the thermal conductivity and melt temperature, but only incrementally from BOL and not burnup dependent. No model or methods changes are required to extend these fuels to 62 MWd/kgU. Similarly, the ZrB_2 burnable absorber model generates helium which is released to the fuel rod plenum and is applicable to 62 MWd/kgU. The geometric effects (thickness) of the ZrB_2 absorber and the model for annular fuel have been reviewed and approved in WCAP-16072-P-A (Reference 3).

Therefore, the models and methodology for the burnable absorbers Gd, Er, and ZrB_2 are applicable to slightly higher burnups, and the fuel rod configurations containing these materials and geometry can be extended to 62 MWd/kgU.

References:

1. CENPD-275-P, Revision 1-P, Supplement 1-P-A, "C-E Methodology for PWR Core Designs Containing Gadolinia-Urania Burnable Absorbers," April 1999.
2. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.
3. WCAP-16072-P-A, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," August 2004.

Question 6: *Section 1.2 of WCAP-16500-P states: "The transition core DNBR penalty is more than offset by the available margin from the mixing vane grids". Provide evidence that the DNB margin associated with the mixing vanes always offsets mixed core effects for the CE 16x16 fleet.*

Response 6: The NGF assembly has a higher hydraulic resistance than the co-resident fuel without mixing vanes so flow will be diverted from the NGF assembly resulting in a loss in DNB margin in some locations in the transition core relative to a uniform core. The NGF DNB correlation described in WCAP-16523-P (Reference 1) provides significant DNB overpower margin relative to the current fuel design without mixing vanes (CE-1 or ABB-NV correlation). This increased DNB overpower margin from the NGF DNB correlation will more than offset a DNB penalty associated with flow redistribution in the transition cycle. Utilizing the methodology defined in Section 7 of CENPD-387-P-A (Reference 2) for analyzing transition cores there is at least [ ]^{a,c} DNB overpower margin available after the transition core penalty for flow redistribution is applied to Plant B. These transition core effects will be evaluated and DNB overpower margin will be confirmed in the reload analyses for each plant.

References:

1. WCAP-16523-P, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," March 2006.
2. CENPD-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel", May, 2000.

Question 7: *The fuel assembly designs illustrated in Figure 1-1 of WCAP-16500-P would exhibit axially dependent CHF correlations.*

a. Discuss the methodology for incorporating multiple, axially-dependent CHF correlations into the analog and digital setpoints analysis.

b. Is the application of multiple CHF correlations within the bounds of the approved methodologies and within the scope of the staff's original review?

c. COLSS provides an on-line DNB margin calculation based upon a single, fixed CHF correlation. Discuss the impact of the NGF's multiple CHF correlations on on-line DNB margin calculations. Include a discussion of "actual" versus "indicated" DNB margin as axial power distribution migrates from top peaked to bottom peaked.

d. Describe the impact of NGF's multiple CHF correlations on the analytical calculations of transient DNBR degradation (e.g. required over-power margin).

e. CPCS provides a Low DNBR reactor trip based upon a single, fixed CHF correlation. Describe the impact of NGF's multiple CHF correlations on CPC-calculated DNBR and on how this trip function is credited in safety analysis.

f. Discuss any changes to the analytical approach, inputs, or assumptions within the transient analyses necessitated by the NGF's multiple CHF correlations (e.g. limiting initial axial shape, transient power redistribution).

g. For each of the above items, address mixed cores where co-resident fuel may have different axially-dependent CHF correlations.

Response 7a: Section 6.1 of WCAP-16523-P (Reference 1) described application of axially-dependent CHF correlations for CE-PWR reload analyses, including the analog or digital setpoint analysis. The WSSV/WSSV-T correlation in WCAP-16523-P (Reference 1) is used for the fuel region above the first side-supported vane grids. For the fuel region below the vane grid, or the non-mixing vane region, the ABB-NV correlation in CENPD-387-P-A (Reference 2) is used.

Both WSSV and ABB-NV correlations are incorporated into the VIPRE code (Reference 3) for CE-PWR analysis using the WCAP-9272-P-A approach (Reference 4). The WSSV-T and ABB-NV correlations are incorporated into the TORC code (Reference 5) for CE-PWR analysis using the traditional method. A fast running tool CETOP-D (Reference 6) containing the required DNB correlations can also be used for the setpoint analysis. The CETOP-D DNBR results are conservative as compared to results from more detailed subchannel calculations using TORC or VIPRE.

Response 7b: The licensed methodology for implementing multiple, axially-dependent CHF correlations is described in Section 7 of the ABB-NV/ABB-TV DNB correlation topical (CENPD-387-P-A, Reference 2) for Turbo fuel. The methodology for multiple CHF correlations for NGF fuel is the same as Turbo fuel as described in Section 6 of WCAP-16523-P (Reference 1) and in Section 4 of WCAP-16500-P.

Response 7c: The response to question 7a discusses how multiple, axially-dependent CHF correlations are incorporated into the design thermal hydraulics (TH) codes for CE-PWRs. This discussion applies to the offline TH analysis for CE-PWRs with COLSS. The on-line DNBR algorithm in COLSS will continue to use the CE-1 CHF correlation. The impact of the WSSV-T and ABB-NV correlations on the COLSS margin calculations will be addressed for each plant and cycle in the uncertainty analysis as described in CEN-356(V)-P-A Revision 01-P-A (Reference 7). The plant specific scoping calculations are ongoing and will be available for audit/review in the Westinghouse offices.

Response 7d: The Thermal Hydraulic codes TORC and CETOP-D (used to determine DNBR, minimum DNBR and CHF) include the capability of modeling different CHF correlations for different regions of the hot fuel rod simultaneously. Depending on the placement of differing grid types, the user, via code input, specifies which CHF correlation is applicable for which fuel rod span. For the NGF design, the non-mixing grid region near the bottom of the fuel rod would specify the ABB-NV correlation and the approximate upper two thirds of the fuel rod would specify the WSSV-T correlation. Consequently, irrespective of where the node of minimum DNBR occurs, the correct CHF correlation is applied by TORC or CETOP-D and therefore the correct and accurate DNBR is calculated. This is true whether it's absolute DNBR that's of interest or if it's required overpower margin (ROPM) that's of interest (since Q_{SAFDL} will be based on the CHF correlation where the node of MDNBR occurs) even if the node of MDNBR shifts CHF correlation region.

Based on this, no impact or changes to current analytical methods, calculational approaches, or assumptions are planned with respect to transient analysis of ROPM or DNBR.

Response 7e: Same response as 7c for CPC instead of COLSS.

Response 7f: See response to question 7.d above.

Response 7g: The typical DNB impact of mixed cores was described in response to question 6 and the

licensed methodology for evaluating multiple CHF correlations was discussed in response to question 7b. The multiple CHF correlations will be explicitly utilized in the TH codes, VIPER or TORC. The component hydraulic loss coefficients for NGF and co-resident fuel will also be explicitly modeled in the TH codes therefore DNB analyses in transition cores will be accurately evaluated for all operating conditions and axial power shapes.

In the first transition cycle where NGF fuel is implemented the DNB margin gains associated with mixing vane grids will not be fully accounted for in Safety Analyses, therefore many analyses will not be affected. The CETOP-D model for the first transition cycle will be unchanged relative to the current cycle model and will be verified to be conservative relative to TORC code evaluations where transition core effects and new DNB correlations are explicitly accounted for. In subsequent cycles the DNB benefit for mixing vane grids will be fully accounted for in Setpoints and Safety Analyses so the CETOP-D code and model will be modified to explicitly model the NGF assembly and utilize the WSSV-T/ABB-NV DNB correlations.

References:

1. WCAP-16523-P, "Westinghouse Correlations WSSV and WSSVT for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," March 2006.
2. CENPD-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel," May, 2000.
3. WCAP-14565-P-A Addendum 1-A, "Addendum 1 to 14565-P-A, Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," August 2004.
4. Letter from B. T. Moroney (NRC) to J. A. Stall (FP&L), "St. Lucie Plant, Unit No. 2 – Issuance of Amendment Regarding Change in Reload Methodology and Increase in Steam Generator Tube Plugging Limit (TAG No. MC1566)," January 2005.
5. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986.
6. CEN-214(A)-P, "CETOP-D Code Structure and Modeling Methods for Arkansas Nuclear One – Unit 2," July 1982.
7. CEN-356(V)-P-A Revision 1-P-A, "Modified Statistical Combination of Uncertainties," May 1988.

Question 8: *Section 2.3.1.3 of WCAP-16500-P describes component and assembly mechanical testing performed to demonstrate that the CE 16x16 NGF design satisfies design criteria.*

a. Provide a more detailed discussion of the number and types of tests performed along with results. Compare these results to the fuel design criteria.

b. Discuss the impact of irradiation induced spring relaxation on measured grid crush strength testing and the associated Seismic/LOCA load analysis.

Response 8a: Mechanical tests were performed for two reasons: a) to determine fuel assembly mechanical properties that were then modeled in analyses that calculated fuel assembly spacer grid impact loads and component stresses for comparison to the stress limits given in Table 2-2 of WCAP-16500-P; and b) to determine the load carrying capabilities of spacer grids and fuel assembly joints for direct comparison to predicted loads. As indicated in the table below, the tests associated with the first design basis (fuel assembly structural integrity) fit in the first category, and the remaining tests fit into the second category.

Design Basis	Test	Result
The fuel assembly must maintain its structural integrity under all operating conditions.	The following tests were used to establish parameters used in the fuel assembly model in the seismic/LOCA core analysis: 1. Skeleton lateral load-deflection test 2. Fuel assembly lateral-load deflection test 3. Fuel assembly pluck and forced vibrations tests 4. Fuel assembly pluck impact test 5. Fuel assembly axial load-deflection test 6. Grid (Mid grid, top grid, IFM) static buckling strength tests	Fuel assembly model used in CE licensed methodology to determine stresses in the fuel assembly for comparison to applicable stress limits. 1. Strain gage measurements used to benchmark computer code for calculating stresses in guide tubes, fuel rod, and top and bottom nozzles. 2. Stiffness and deflected shapes used to establish fuel assembly model parameters. 3. Natural frequency and damping used to establish fuel assembly model parameters. 4. Used to determine spacer grid one-sided impact stiffness in fuel assembly model. 5. Used to determine axial stiffness of fuel assembly. 6. Used to determine spacer grid thru-grid stiffnesses in fuel assembly model.
The strength of the bulged connections between the guide thimble and the grid sleeves or the guide thimble flange must exceed the loads applied to the connection under all operating conditions.	Mid-grid, IFM grid, and top grid sleeve-to-guide thimble bulge joint strength tests Flange-to-guide thimble bulge joint strength test	All bulged joint strengths exceeded requirements.
Welded connections between the grids and their respective sleeves/inserts must not fail under all operating conditions.	Mid grid, IFM grid, top grid and bottom grid sleeve-to-grid joint strength tests	All welded and brazed joint strengths exceeded requirements.
The lateral strength of the spacer grids must be sufficient to withstand seismic and LOCA events with no channel closure greater than that which would significantly impair the coolability of the fuel rod array or insertability of the CEAs.	Mid-grid and IFM grid one-sided impact strength test Mid-grid and IFM grid long pulse through grid impact strength test Top grid static buckling strength test	The tested grid strengths exceeded requirements.

Response 8b: Grid crush strength testing for the CE 16x16 NGF grids was performed in accordance with Standard Review Plan Section 4.2, Appendix A, which states that unirradiated production grids at (or corrected to) operating temperature shall be tested. In accordance with this direction, BOL spring settings were used.

The effect of spring relaxation, combined with grid growth and rod creepdown, is to create small gaps between the fuel rods and the grid rod support features. The effect of gaps, as well as other irradiation induced effects, on grid seismic capability, is documented in Addendum 1 to WCAP-12488-A (Reference 1), where it was concluded that the continued use of grid crush data from unirradiated production grids to perform seismic/LOCA analysis was validated for Westinghouse grids.

The CE 16x16 NGF spacer grids are similar to Westinghouse grids with respect to the parameters that influence grid strength (material, strap thicknesses, pitch, and straight strap). Therefore, the conclusion from WCAP-12488-A discussed above is also applicable to CE 16x16 NGF spacer grids.

Reference:

1. WCAP-12488-A Addendum 1-A, Revision 1, "Addendum 1 to WCAP-12488-A Revision to Design Criteria," January 2002.

Question 9: *Section 2.4.4 of WCAP-16500-P states that "yield and ultimate strengths of the two materials are almost identical". Provide unirradiated and irradiated YS and UTS for* ZIRLO™ *and current Zircaloy-4 tubing in order to quantify "almost identical".*

Response 9: A comparison of the unirradiated properties is given in the table below. Irradiated properties are not given because stress limits are based on unirradiated properties. Irradiated properties are not used since no credit is included for increased strength due to irradiation.

[ ] a, c

Question 10: *Section 2.5.2 of WCAP-16500-P provides the design basis for fuel rod cladding stress and strain. The staff's SE for Optimized* ZIRLO™ *(WCAP-12610-P-A and CENPD-404-P-A, Addendum 1) [*

] a, c. Clarify the cladding strain design basis.

Response 10: The 1% strain criteria applied to CE plants are based on ensuring that the cladding strain limit is less than the cladding strain capability determined by tensile ductility measurements. The strain evaluation addresses normal operation and application of a limiting AOO.

The cladding's plastic strain and total strain capabilities for Optimized ZIRLO™ decrease with burnup. However, the [

] a, c. Thus, the CE strain criteria presented in the report, as well as in Addendum 1 to CENPD-404-P-A (Reference 1), apply a [

] a, c.

Based on the above discussion, Westinghouse believes the CE strain criteria as stated in the report is appropriate.

Reference:

1. WCAP-12610-P-A and CENPD-404-P-A, Addendum 1, "Addendum 1 to WCAP-12610-P-A and CENPD-404-P-A Optimized ZIRLO™," February 2003.

Question 11: *Section 2.5.7 of WCAP-16500-P describes the fuel rod cladding flattening analysis. The plenum spring radial support capacity is credited in the design analysis. Therefore, the fuel rod plenum spring design is part of the fuel rod design basis and needs to be specifically defined.*

a. Provide the plenum spring specifications for each of the fuel rod designs identified in response to RAI #2.b.

b. Provide justification of each spring's radial support capacity based upon testing.

Response 11a: The Plant B NGF plenum spring design will be used in all plants except Plant E. Nominal spring parameters of the Plant B NGF spring include [

]^{a,c}. The design of the Plant E NGF plenum spring will differ, due to the plants' shorter reactor internals that result in shorter fuel assemblies, fuel rods, and plenum lengths. As a result, some of the Plant E NGF plenum spring parameters will differ from those presented above. The justification of the radial support capability of the Plant E spring design will be established by the technique described below. Any evolutionary changes to the Plant B NGF spring design would be evaluated by this same technique.

Response 11b: The topic of crediting the plenum spring's radial support of the cladding was addressed in WCAP-16072-P (Reference 1) and its related RAIs. Specifically, Round #3 RAI #4 discussed the autoclave testing that serves as the justification of this approach, while the SER on this topic finds the approach acceptable based on, among other things, "a commitment to validate adequate plenum spring support in future applications".

The validation of a particular plenum spring design consists of either an assessment of the spring design relative to previously justified designs or performing additional autoclave tests with the particular plenum spring. The NGF plenum spring design operating in the Waterford-3 LTAs was justified by demonstrating that the results of prior autoclave testing were conservative for the LTA design. Relative to designs that had successfully passed the autoclave tests, the LTA design has the same clad ID, clad thickness, and spring outside diameter, and it has a conservative spring wire diameter (larger), and a conservative spring coil pitch (shorter).

Reference:

1. WCAP-16072-P-A, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs," August 2004.

Question 12: *Section 2.6 of WCAP-16500-P describes the applicability of the currently approved fuel performance models up to 62,000 MWd/MTU. This section states, [*

]^{a,c}. These two modeling issues may be related since lower fuel temperature will usually result in less fission gas release. Provide further evidence that the currently approved fuel performance models and analytical methodology remain conservative up to 62,000 MWd/MTU.

Response 12: Section 2.6 of WCAP-16500-P provides a description of both fission gas release and temperature predictions relative to measured data. These two models are inter-related since fission gas release is directly dependent on fuel temperatures. Lower fuel temperature would normally result in lower fission gas release. However, the fission gas

release data comparisons of Section 2.6 demonstrate that in the FATES3B design and licensing calculations, where power ramps (simulating transients) are applied, conservatively high fission gas release predictions result. Steady-state fission gas release, which is occurring at lower temperatures, is still observed to be generally conservative.

It was also noted in Section 2.6 that while FATES3B under-prediction of temperatures did occur, the under-prediction occurred at such low temperatures that they are not of interest or concern in typical design and licensing. These are low LHGR and fuel temperatures, atypical of plant operation. The fuel thermal conductivity change with temperature is also larger at low LHGR and fuel temperature, sensitizing the predictions. At higher LHGR's the FATES3B trend is to predict the data well. The higher LHGR's are more typical for design and licensing.

Thus, it is concluded that the fuel performance model and methods are applicable to 62 MWd/kgU.

Section E

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Our ref: LTR-NRC-07-6
January 29, 2007

Subject: Slide Presentation in Support of NRC Audit on WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report" and Clarification of RAI Responses to Questions 1a, 4a and 10 (TAC No. MD0560) (Proprietary/Non-proprietary)

Enclosed are copies of the Proprietary and Non-Proprietary slide presentation in support of the NRC audit on WCAP-16500-P and Proprietary and Non-Proprietary Clarifications of RAI Responses to Question 1, 4a, and 10 as was requested by the NRC. The NRC audit is scheduled for January 3-31, 2007 in the Westinghouse Rockville Office.

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-07-2236 (Non-proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-proprietary).

This submittal contains proprietary information of Westinghouse Electric Company, LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or Application for Withholding should reference AW-07-2236 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: F. M. Akstulewicz, NRR
R. Landry, NRR
P. Clifford, NRR
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Our ref: AW-07-2236
January 29, 2007

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-07-6 P-Attachment Enclosures 1 and 3, Slide Presentation in Support of NRC Audit on WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report" and Clarification of RAI Responses to Questions 1a, 4a and 10 (TAC No. MD0560) (Proprietary)

Reference: Letter from J. A. Gresham to NRC, LTR-NRC-07-6, dated January 29, 2007

The application for withholding is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-07-2236 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-07-2236 and should be addressed to J. A. Gresham, Manager of 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

Cc: F. M. Akstulewicz, NRR
R. Landry, NRR
P. Clifford, NRR
H. Cruz, NRR
J. Thompson, NRR

AW-07-2236

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

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

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

Sworn to and subscribed
before me this 29th day
of January, 2007.

Sharon L. Markle
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

(1) I am Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse) and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking 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 which 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 "LTR-NRC-07-6 P-Attachment Enclosures 1 and 3, Slide Presentation in Support of NRC Audit on WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report" and Clarification of RAI Responses to Questions 1a, 4a and 10 (TAC No. MD0560) (Proprietary)," January 29, 2007, for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-07-6) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company is responses to NRC's Request for Additional Information.

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

(a) Demonstrate the acceptability of the CE 16x16 Next Generation Fuel Design.

(b) Assist customers in implementing an improved fuel product.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this fuel design with its associated correlation to further enhance their licensing position over their competitors.

(b) Assist customers to obtain license changes.

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 fuel design 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 for developing the enclosed improved core thermal performance methodology.

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).

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The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

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Enclosure 2

Slide Presentation in Support of NRC Audit on WCAP-16500-P, "CE 16x16 Next Generation Fuel Core Reference Report" (TAC No. MD0560) (Non-Proprietary)

Westinghouse Electric Company
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Implementation of Multiple CHF Correlations for NGF

Westinghouse Rockville Office
Audit with US NRC
January 30 & 31, 2007

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Agenda

- Purpose of Meeting
- Background on use of Multiple CHF correlations
- Overview of Reload Process
- Implementation of Multiple CHF Correlations in Reload Process for NGF
  - TH
  - Transients
  - Setpoints

- Summary of First Time Implementation Calculations
- Summary of Supporting Topicals

Purpose of Meeting

- Describe how ABB-NV and WSSV-T CHF correlations are implemented in TH, Setpoints and Transient reload areas for transition and full cores
- Provide details of the method applied in First Time Engineering Analysis
- Answer any questions from NRC on review of implementation of multiple CHF correlations in reload process

Background on use of Multiple CHF Correlations

- The NGF assembly contains grid spans with and without mixing vanes

- As a result of the two different grid spans, two CHF correlations are used for NGF:
  - ABB-NV for non-mixing vane spans and,
  - WSSV or WSSV-T for mixing vane spans

- These CHF correlations are placed in VIPRE, TORC and CETOP codes

Background on use of Multiple CHF Correlations

- Multiple CHF correlations have been implemented before
  - Turbo fuel using TORC & CETOP codes (ABB-NV & ABB-TV CHF correlations), also licensed in VIPRE
  - All Westinghouse fuel using THINC and VIPRE codes (WRB-series and W-3 CHF correlations)

- The methodology applied in reload process for one CHF correlation is essentially the same as applied for two CHF correlations

Mid-Grid with Mixing Vanes

a, c

IFM Grid with Mixing Vanes

a, c

Mid-Grid without Mixing Vanes

a, c

Westinghouse

Fuel Assembly Comparisons

a, c

Westinghouse

NGF Implementation Plan

a, c

Overview of Reload Process

a, c

Overview of Reload Process

a, c

Overview of Reload Process

a, c

Overview of Reload Process - Setpoints

a, c

Overview of Reload Process - Setpoints

a, c

Implementation of Multiple CHF Correlations in Reload Process for Thermal-Hydraulics

a, c

TORC MODELING

a, c

Westinghouse

TORC MODELING

a, c

Westinghouse

TORC MODELING

a, c

Implementation of Multiple CHF Correlations in Reload Process for Thermal-Hydraulics

a, c

Implementation of Multiple CHF Correlations in Reload Process for Thermal-Hydraulics

a, c

Westinghouse

CETOP MODELING

a, c

Westinghouse

CETOP/TORC BENCHMARKING

a, c

CETOP Benchmarking

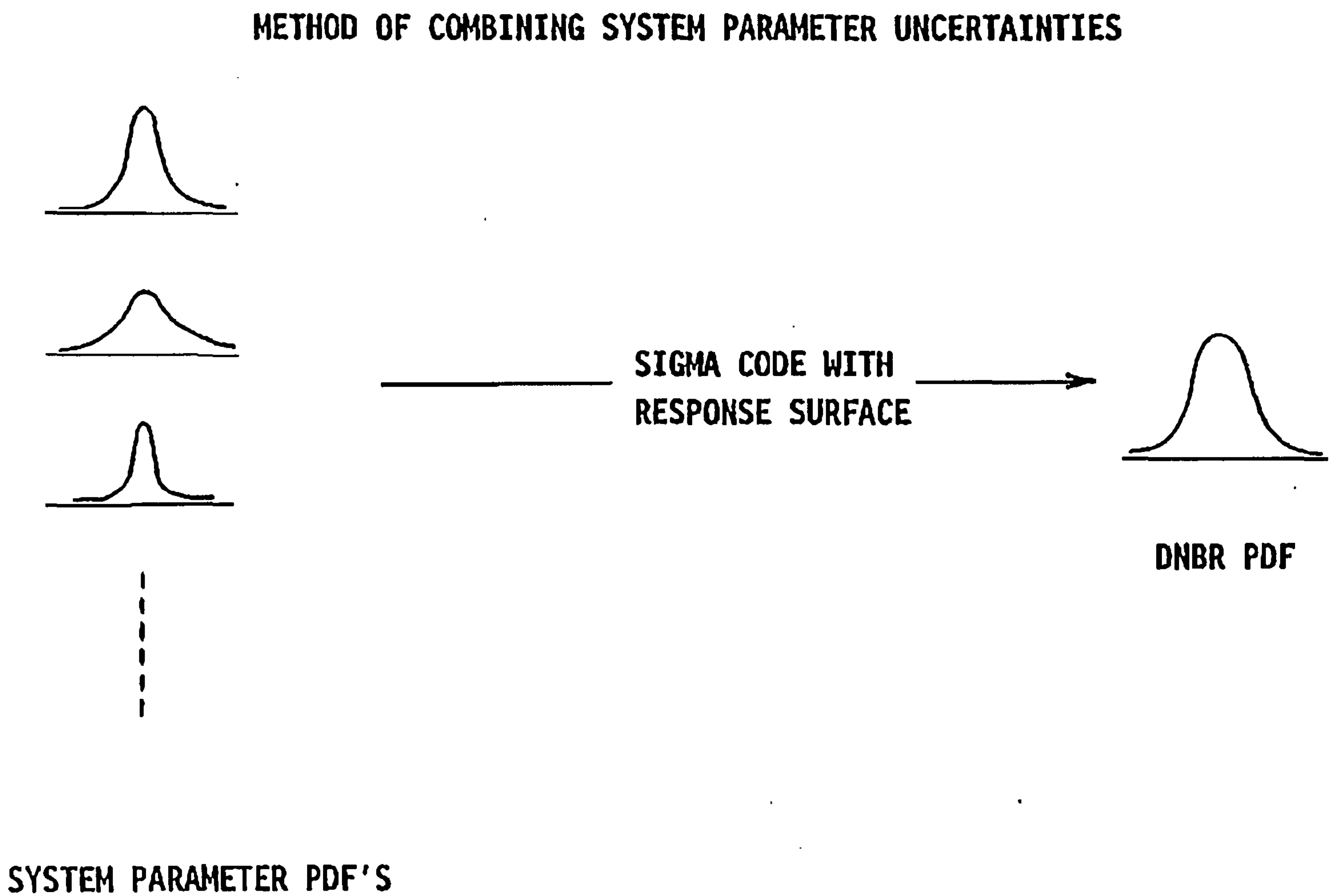
a, c

Implementation of Multiple CHF Correlations in Reload Process for Thermal-Hydraulics

a, c

Westinghouse

METHODOLOGY - MDNBR Limit and it's Probability Distribution Function (PDF)



Implementation of Multiple CHF Correlations in Reload Process for Transients

a, c

Implementation of Multiple CHF Correlations in Reload Process for Transients

a, c

Implementation of Multiple CHF Correlations in Reload Process for Transients

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Implementation of Multiple CHF Correlations in Reload Process for Transients

a, c

Implementation of Multiple CHF Correlations in Reload Process for Setpoints

a, c

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Implementation of Multiple CHF Correlations in Reload Process for Setpoints

a, c

Implementation of Multiple CHF Correlations in Reload Process for Setpoints

a, c

Summary of First Time Implementation Calculations

a, c

Summary of First Time Implementation Calculations

a, c

Summary of Supporting Topicals

- Thermal-Hydraulic
  1. WCAP-16523-P, "Westinghouse Correlations WSSV and WSSVT for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," March 2006.
  2. CENPD-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel," May, 2000.
  3. WCAP-14565-P-A Addendum 1-A, "Addendum 1 to 14565-P-A, Qualification of ABB Critical Heat Flux Correlations with VIPRE-01 Code," August 2004.
  4. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986.
  5. CEN-214(A)-P, "CETOP-D Code Structure and Modeling Methods for Arkansas Nuclear One – Unit 2," July 1982.
  7. CEN-356(V)-P-A Revision 1-P-A, "Modified Statistical Combination of Uncertainties," May 1988.

Summary of Supporting Topicals

- Transients
  1. CENPD-282-P-A, "Technical Manual for the CENTS Code," February 1991.
  2. CENPD-188-A, "HERMITE, A Multi-Dimensional Space-Time Kinetics Code for PWR Transients," March 1976.
  3. CENPD-183-A, "Loss of Flow, CE Methods of Loss of Flow Analysis," July 1975.
  4. CENPD-190, "C-E Method for Control Element Assembly Ejection Analysis," January 1976.
  5. CEN-308-P, "CPC/CEAC Software Modifications for the CPC Improvement Program," April 1986.
  6. CEN-310-P, "CPC and Methodology Changes for the CPC Improvement Program," April 1986.
  7. CEN-214(A)-P, "CETOP-D Code Structure and Modeling Methods for Arkansas Nuclear One – Unit 2," July 1982.

Summary of Supporting Topicals

- Setpoints
  1. CEN-356(V)-P-A Revision 1-P-A, "Modified Statistical Combination of Uncertainties," May 1988.
  2. CEN-312-P Revision 02-P, "Overview Description of the Core Operating Limit Supervisory System," November 1990.
  3. CEN-305-P Revision 02-P, "Functional Design Requirements for a Core Protection Calculator," May 1988.
  4. CEN-304-P Revision 02-P, "Functional Design Requirement for a Control Element Assembly Calculator," May 1988.

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Enclosure 4

Clarification of RAI Responses to Questions 1a, 4a and 10 (TAC No. MD0560) (Non-Proprietary)

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

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Clarification to RAI Responses 1a, 4a and 10 for CE 16x16 Next Generation Fuel Core Reference Report WCAP-16500-P

RAI 1a Clarification:	*Provide a summary of reactor operating experience for NGF features defined in Table 1-2 of WCAP-16500-P.*
Clarification Response 1a:	An additional column was provided to Table 1-2 of WCAP-16500-P where reactor operating experience is summarized for all NGF features. This is attached here as Table 1-1.

Table 1-1
Comparison of Standard and NGF Designs

a, c

Feature	Standard Fuel Design Feature Description	NGF Fuel Design Feature Description	Reactor Operating Experience of NGF Features
Top Grid	Zr-4 Wavy Strip grid or Inconel Straight Strip grid, both with L-shaped outer strap	Inconel, Straight Strip grid with corner weld outer strap	
Upper Nozzle	CE Std Nozzle	Same but tabs added to guide tube flange & keyways in flow plate	
Top GT flange joint	Zr-4	Zr-4	
Mid-grids	Zr-4 Wavy grid, no mixing vanes, alternating rod supports	Low tin ZIRLO™ "I" spring grid with Side Supported Vanes on selected grids, alternating rod supports & ZIRLO™ Sleeves	
IFM Grids	None	Low tin ZIRLO™ grid with Side Supported Vanes, non-contacting arches with ZIRLO™ sleeves	
Mid & IFM Grid Outer Strap Design	Zr-4 strap	Low tin ZIRLO™ strap	
Top, Mid & IFM Grid to GT Joints	Welded	Sleeves Bulged to GT above and below Mid grids and below IFM grid	

Table 1-1 (cont.)
Comparison of Standard and NGF Designs

Feature	Standard Fuel Design Feature Description	NGF Fuel Design Feature Description	Reactor Operating Experience of NGF Features
GTs and Wear Sleeves	Zr-4 GTs with dashpot and SS Inner Wear Sleeve	Same but use ZIRLO™ GTs and Short SS Inner wear sleeves	
Guardian™ Grid and joint with lower nozzle	Inconel grid with skirt	Same, but uses perimeter strap modified for no welding to lower nozzle and added SS sleeves in GT openings	
Bottom Nozzle	CE Std Nozzle	Same except features for welding Guardian™ grid not required	
Fuel Rod	0.382" OPTIN™ Zr-4 rod with Std Plenum Spring and Guardian™ solid end cap	Westinghouse 0.374" Optimized ZIRLO™ rod with low volume plenum spring and Guardian™ solid end plug	

a, c

* ( ), numbers in parentheses is the number of years the feature has in-reactor experience.

** [ ]^{a,c} contain Optimized ZIRLO™ cladding with [ ]^{a,c} material where other cladding for NGF and other LTA programs is [ ]^{a,c}

RAI 4a Clarification:

Figure 1-1 of WCAP-16500-P illustrates 5 different combinations of grid strip designs and elevations.

a. Provide evidence that the FIV test results for the Plant B are applicable to the CE 16x16 NGF assembly designs with different bundle length and grid spacing.

Clarification Response 4a:

For the five fuel assemblies shown in Figure 1-1 of WCAP-16500-P, only two plants have different grid spacing or bundle length. Plant A has shorter grid spacing in the non-IFM region due to the extra grid. [

] a, c for the other plants. As shown in Figure 1-1, Plant E is shorter than the remaining plants. For all plants except Plant E, [

] a, c.

Westinghouse has performed numerous hydraulic tests on [

] a, c.

For the fuel designs with [

] a, c. These tests provide strong evidence that without the excitation source, FIV would not occur for any of the fuel designs with the same grid design or grid orientation.

As stated in the response to 4a, the CE 16x16 NGF design has a mix of non-vaned grids and mixing vane grids that have the side-supported vane (SSV), [

] a, c. The results of the CE 16x16 NGF FIV test on the plant B design confirm the excitation source for FIV is not present with the SSV mixing vane for the CE 16x16 NGF fuel designs. The results matched expectation and were also consistent with the 14x14 fuel assembly FIV results with the side-supported mixing vane grid. Based upon [
] a, c and the testing with the side-supported mixing vane design, without the excitation source, FIV would not occur for any of the five assembly designs shown in Figure 1-1 for full core conditions.

Long-Term Wear or endurance tests have been performed on [

] a, c fretting wear results from the plant B endurance test, performed at a conservative flow for all plants, are applicable to the five fuel assembly designs shown in Figure 1-1 in WCAP-16500-P. Also, the change in bundle length between plants in Figure 1-1 will have no impact on fretting wear based on past testing of 17x17 standard and XL fuel designs.

Figure 4-5
Test Flow vs. Modal Response Frequency Map for 17XL Atom Grid Design

a, b, c

[ ]

RAI 10 Clarification: *Provide additional clarification of cladding strain design basis.*

Clarification Response 10: As stated in the initial response to RAI #10 of Reference 1, the CE cladding strain criterion imposes only a [

]^{a,c}.

The Optimized ZIRLO™ and CE 16x16 NGF topical reports (Reference 2 and 1, respectively) apply only the [

]^{a,c}. Per Reference 2, (specifically, Section 3.2.9 of the SER of Reference 2), the strain capability of Optimized ZIRLO™ after irradiation is similar to, or slightly higher than, that of ZIRLO™ cladding. Therefore, irradiated ZIRLO™ cladding data are used to evaluate the strain criterion.

Irradiated plastic strain data for ZIRLO™ are available from the [

]^{a,c}.

Since the irradiated data are for burnups well in excess of [

] a, c. As additional irradiated plastic strain data become available, it will be assessed to confirm the continued applicability of this criterion.

Table 10-1
ZIRLO™ Cladding Plastic Strain at 350 °C (662 °F)

a, b, c

[ ]

References:

1. WCAP-16500-P, "CE 16x16 NGF Next Generation Fuel Core Reference Report", February 2006.

2. WCAP-12610-P-A & CENPD-404-P-A Addendum 1-A, "Optimized ZIRLO™", July 2006.

Section F

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Direct fax: (412) 374-4011
e-mail: greshaja@westinghouse.com

Our ref: LTR-NRC-07-8
February 15, 2007

Subject: Further Clarification to RAI Response 7 and 10 for CE 16x16 Next Generation Fuel Core Reference Report WCAP-16500-P (TAC No. MD0560) (Proprietary/Non-proprietary)

Enclosed are copies of the Proprietary and Non-Proprietary Clarifications of RAI Responses to Questions 7 and 10 (Enclosure 1 and 2) and the slide presentation that supported RAI #7 subsequent discussion with the NRC on February 12, 2007 (Enclosure 3 and 4).

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-07-2243 (Non-proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-proprietary).

This submittal contains proprietary information of Westinghouse Electric Company, LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or Application for Withholding should reference AW-07-2243 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,

BFMaurer /for

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

Enclosures

cc: F. M. Akstulewicz, NRR
R. Landry, NRR
P. Clifford, NRR
H. Cruz, NRR
J. Thompson, NRR
L. M. Feizollahi, NRR

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Our ref: AW-07-2243
February 15, 2007

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-07-8 P-Enclosure 1, "Further Clarification to RAI Response 7 and 10 for CE 16x16 Next Generation Fuel Core Reference Report WCAP-16500-P" and LTR-NRC-07-8 P-Enclosure 3, "Clarification of RAI Responses to Question 7 Slide Presentation from February 12, 2007" (TAC No. MD0560) (Proprietary)

Reference: Letter from J. A. Gresham to NRC, LTR-NRC-07-8, dated February 15, 2007

The application for withholding is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-07-2243 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-07-2243 and should be addressed to J. A. Gresham, Manager of 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

Cc: F. M. Akstulewicz, NRR
R. Landry, NRR
P. Clifford, NRR
H. Cruz, NRR
J. Thompson, NRR

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared B. F. Maurer, 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:

BFMaurer

B. F. Maurer, Principal Engineer
Regulatory Compliance and Plant Licensing

Sworn to and subscribed
before me this 15th day
of February, 2007.

Sharon L. Markle
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

(1) I am Principal 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 rulemaking 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 which 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 LTR-NRC-07-8 P-Enclosure 1, "Further Clarification to RAI Response 7 and 10 for CE 16x16 Next Generation Fuel Core Reference Report WCAP-16500-P" and LTR-NRC-07-8 P-Enclosure 3, "Clarification of RAI Responses to Question 7 Slide Presentation from February 12, 2007 (TAC No. MD0560) (Proprietary)," February 15, 2007, for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-07-8) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company is responses to NRC's Request for Additional Information.

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

(a) Demonstrate the acceptability of the CE 16x16 Next Generation Fuel Design.

(b) Assist customers in implementing an improved fuel product.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this fuel design with its associated correlation to further enhance their licensing position over their competitors.

(b) Assist customers to obtain license changes.

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 fuel design 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 for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

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Enclosure 2
Further Clarification to RAI Response 7 and 10 for CE 16x16 Next Generation Fuel Core Reference Report WCAP-16500-P (Non-Proprietary)

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Further Clarification to RAI Response 7 and 10 for CE 16x16 Next Generation Fuel Core Reference Report WCAP-16500-P

RAI #7: *The fuel assembly designs illustrated in Figure 1-1 of WCAP-16500-P would exhibit axially-dependent CHF correlations.*

- *a. Discuss the methodology for incorporating multiple, axially-dependent CHF correlations into the analog and digital setpoints analysis.*
- *b. Is the application of multiple CHF correlations within the bounds of the approved methodologies and within the scope of the staff's original review?*
- *c. COLSS provides an on-line DNB margin calculation based upon a single, fixed CHF correlation. Discuss the impact of the NGF's multiple CHF correlations on on-line DNB margin calculations. Include a discussion of "actual" versus "indicated" DNB margin as axial power distribution migrates from top peaked to bottom peaked.*
- *d. Describe the impact of NGF's multiple CHF correlations on the analytical calculations of transient DNBR degradation (e.g., required over-power margin).*
- *e. CPCS provides a Low DNBR reactor trip based upon a single, fixed CHF correlation. Describe the impact of NGF's multiple CHF correlations on CPC-calculated DNBR and on how this trip function is credited in safety analysis.*
- *f. Discuss any changes to the analytical approach, inputs, or assumptions within the transient analyses necessitated by the NGF's multiple CHF correlations (e.g., limiting initial axial shape, transient power redistribution).*
- *g. For each of the above items, address mixed cores where co-resident fuel may have different axially-dependent CHF correlations.*

Response to RAI #7: Response to RAI #7 was provided to the NRC Staff during an audit conducted by the Staff at the Westinghouse Rockville Office on January 30-31, 2007. Slides supporting the audit were previously supplied to the NRC in LTR-NRC-07-6, dated January 29, 2007. Based on this audit, the staff requested a follow on discussion to clarify previous responses. This discussion occurred on February 12, 2007. The NRC Staff requested that the clarification response slide presentation be provided. Refer to Enclosure 3 and 4 for the Proprietary and Non-Proprietary versions of the slide presentation.

RAI 10 Clarification: *Provide additional clarification of cladding strain design basis.*

Clarification to RAI 10: Upon further review of available strain data, Westinghouse has decided not to use the criterion of [

]^{a,c} as defined in WCAP-16500-P for the application of Optimized ZIRLO™ cladding in CE 16 NGF assemblies. Instead, Westinghouse will apply a 1% total strain limit (elastic plus uniform plastic) for all burnups, consistent with the criterion defined in Section 4.2 of the SRP. Application of the strain limit also brings the CE criterion in line with the Westinghouse criterion. Therefore, instead of the CE strain criterion defined in WCAP-16500-P the Westinghouse strain criterion for Optimized ZIRLO™ from Addendum 1-A to WCAP-12610-P-A/CENPD-404-P-A will be applied to the CE 16 NGF assemblies. Specifically the cladding strain criterion is:

> "The design limit for the fuel rod clad strain is the total plastic tensile creep strain due to uniform cladding creep and uniform cylindrical fuel pellet expansion due to swelling and thermal expansion is less than 1% from the unirradiated condition, and that the total tensile strain due to uniform cylindrical pellet thermal expansion during a transient is less than 1% from the pre-transient value."

If adequate strain data become available for Optimized ZIRLO™ cladding in the future Westinghouse will submit these data to NRC to support a change to the strain limit criterion.

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Enclosure 4
Clarification of RAI Responses to Question 7
Slide Presentation from February 12, 2007
(TAC No. MD0560) (Non-Proprietary)

(44 Slides Enclosed)

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Setpoint Analysis Process for NGF Implementation

Presentation to NRC

February 12, 2007

Response to RAI #7 for WCAP-16500-P

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Outline

a, c

Outline

a, c

History of Uncertainty Analysis Methodology for CPC & COLSS

a, c

History of Uncertainty Analysis Methodology for CPC & COLSS

a, c

History of Uncertainty Analysis Methodology for CPC & COLSS

a, c

History of Uncertainty Analysis Methodology for CPC & COLSS

[ ]a, c

Components of Overall Uncertainty Analysis

a, c

Components of Overall Uncertainty Analysis

a, c

Overview of Methodology & Process

a, c

Overview of Methodology & Process

a, c

Overview of Methodology & Process

a, c

Overview of Methodology & Process

a, c

Process Refinements Required to Implement NGF

a, c

Process Refinements Required to Implement NGF

a, c

Statistical Evaluation

a, c

Statistical Evaluation

a, c

Example Comparison of Correlations vs ASI

a, c

Statistical Evaluation

a, c

Example Comparison of Correlations vs ASI (with NGF DNBR pdf sampling)

a, c

Example Uncertainty Factors vs ASI Range

a, c

Overview of Methodology & Process

a, c

Example Uncertainty Factors vs ASI Range

a, c

Statistical Evaluation

a, c

Example ASI-Dependent Multiplicative Factor

a, c

Example Uncertainty Factors vs ASI Range After Application of ASI-Dependent Multipliers

a, c

Statistical Evaluation

a, c

Correlation of DNB POL uncertainty with pressure

a, c

Correlation of DNB POL uncertainty with pressure

a, c

Correlation of DNB POL uncertainty with temperature

a, c

Correlation of DNB POL uncertainty with temperature

a, c

Correlation of DNB POL uncertainty with flow

a, c

Correlation of DNB POL uncertainty with flow

a, c

Correlations of DNB POL uncertainty with temperature, pressure & RCS flow

[ ]a, c

Statistical Evaluation

a, c

Statistical Evaluation

a, c

Implementation Questions

a, c

Implementation Questions

a, c

Implementation Questions

a, c

Implementation Questions

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Implementation Questions

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Implementation Questions

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Implementation Questions

a, c

Summary

a, c

Section G

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Our ref: LTR-NRC-07-13
March 16, 2007

Subject: Supplement 1-P to WCAP-16500-P, "Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)" (TAC No. MD0560) (Proprietary/Non-proprietary)

Enclosed are copies of the Proprietary and Non-Proprietary versions of Supplement 1-P to WCAP-16500-P, "Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)" which provides clarification of the CE Setpoint Methodology. This Supplement provides additional details over the responses provided to RAI 7 that was provided at the January 30-31, February 1, 2007, NRC audit and the slide presentation to the NRC on February 12, 2007.

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-07-2251 (Non-proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-proprietary).

This submittal contains proprietary information of Westinghouse Electric Company, LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or Application for Withholding should reference AW-07-2251 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,

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

Enclosures

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P. Clifford, NRR
H. Cruz, NRR
J. Thompson, NRR
L. M. Feizollahi, NRR

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Our ref: AW-07-2251
March 16, 2007

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: Supplement 1-P to WCAP-16500-P, "Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)" (TAC No. MD0560) (Proprietary)

Reference: Letter from B. F. Maurer to NRC, LTR-NRC-07-13, dated March 16, 2007

The application for withholding is submitted by Westinghouse Electric Company LLC (Westinghouse) pursuant to the provisions of paragraph (b)(1) of Section 2.390 of the Commission's regulations. It contains commercial strategic information proprietary to Westinghouse and customarily held in confidence.

The proprietary material for which withholding is being requested is identified in the proprietary version of the subject report. In conformance with 10 CFR Section 2.390, Affidavit AW-07-2251 accompanies this application for withholding, setting forth the basis on which the identified proprietary information may be withheld from public disclosure.

Accordingly, it is respectfully requested that the subject information which is proprietary to Westinghouse be withheld from public disclosure in accordance with 10 CFR Section 2.390 of the Commission's regulations.

Correspondence with respect to this application for withholding or the accompanying affidavit should reference AW-07-2251 and should be addressed to J. A. Gresham, Manager of Regulatory Compliance and Plant Licensing, Westinghouse Electric Company LLC, P. O. Box 355, Pittsburgh, Pennsylvania 15230-0355.

Very truly yours,

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

Cc: A. Mendiola, NRR
P. Clifford, NRR
H. Cruz, NRR
J. Thompson, NRR

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared B. F. Maurer, Acting Manager, 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:

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

Sworn to and subscribed
before me this 16th day
of March, 2007.

Sharon L. Markle
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

(1) I am Acting Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse) and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking 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 which 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 "Supplement 1-P to WCAP-16500-P, 'Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)' (TAC No. MD0560) (Proprietary)," March 16, 2007, for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-07-13) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company is in response to NRC's request for additional clarification.

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

(a) Demonstrate the acceptability of the CE 16x16 Next Generation Fuel Design.

(b) Assist customers in implementing an improved fuel product.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this fuel design with its associated correlation to further enhance their licensing position over their competitors.

(b) Assist customers to obtain license changes.

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 fuel design 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 for developing the enclosed improved core thermal performance methodology.

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).

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The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

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WCAP-16500-NP
Supplement 1-NP

Application of CE Setpoint Methodology for CE 16x16 Next Generation Fuel (NGF)

March 2007

Authors:

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Electronically Approved Records Are Authenticated In the Electronic Document Management System

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Table of Contents

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List of Figures

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1.0 Introduction and Background

1.1 Introduction

1.1.1 Purpose

The purpose of this supplement is to describe the application of the CE setpoint methodology to plants with CE 16x16 Next Generation Fuel (NGF). The CE setpoint methodology involves the modified statistical combination of uncertainties (MSCU) methodology as described in Reference 1. MSCU calculates setpoints for the digital protection and monitoring systems employed at several CE designed Nuclear Steam Supply Systems (NSSS). Application of the MSCU methodology to plants with NGF requires process and input changes in order to model the NGF design and address its thermal hydraulic characteristics.

1.1.2 Introduction to CPCS & COLSS

The Core Protection Calculator System (CPCS) is part of the reactor protection system (RPS). It consists of Core Protection Calculators (CPCs) and Control Element Assembly Calculators (CEACs). The CPC functional design is described in Reference 2 and the CEAC functional design is described in Reference 3. The CPCS initiates the low DNBR and high Local Power Density (LPD) trips of the RPS in order to assure that fuel design limits on DNBR and centerline fuel melting are not exceeded during Anticipated Operational Occurrences (AOOs) and to assist the Engineered Safety Features Actuation System (ESFAS) in limiting the consequences of certain postulated accidents. Each CPC channel receives safety grade sensor inputs and calculates DNBR, LPD and other quantities. The CEACs receive safety grade CEA position inputs and provide single CEA position-related penalty factors to each CPC channel such that the CPCs respond appropriately to single CEA–related AOOs which require CPC protection.

The Core Operating Limit Supervisory System (COLSS) is a digital computer based on-line monitoring system that is used issue alarm signals to the plant computer and to provide information to aid the operator in complying with Technical Specification operating limits on total core power, peak linear heat rate (LHR), DNBR, axial shape index (ASI) and azimuthal power tilt. An overview description of COLSS is provided in Reference 4.

The CPCS and COLSS include [

] ^{a,c} in order to decide whether to issue a trip signal to the RPS. The algorithm in COLSS [

] ^{a,c} in order to decide whether to issue an alarm signal to the plant computer.

1.1.3 Purpose of Setpoint Analysis

The setpoint analysis for CE NSSS with digital monitoring and protection systems is performed every reload cycle in order to calculate addressable constants for the CPCS and COLSS. Addressable constants are coefficients of the CPCS or COLSS algorithms which can be changed readily during startup or operation. Addressable constants include calibration coefficients, measurement results, uncertainty factors, adjustment factors, time delays and trip setpoints. The primary purpose of the cycle specific setpoint analysis is to calculate the CPCS and COLSS uncertainty factors.

1.1.4 How NGF Impacts Setpoint Analysis Methodology & Process

Implementation of NGF impacts the setpoint analysis methodology and process in four areas:

1. [

   ]^{a,c}.
2. [

   ]^{a,c}.
3. [ ]^{a,c}.
4. [

   ]^{a,c}.

The process for addressing these areas is described in this supplement.

1.2 Background

1.2.1 History of Uncertainty Analysis Methodology

The original uncertainty analysis methodology for CPC was documented in Reference 10, referenced in the ANO-2 Cycle 1 FSAR and approved in the ANO-2 Cycle 1 SERs. This methodology included three areas of statistical treatment and/or combination:

1. Statistical treatment of DNBR uncertainties resulting from the power distribution synthesis in CPC and the radial peaking factor measurement errors using the INCA code.
2. Statistical treatment of the DNBR uncertainties resulting from the error in the DNBR on-line algorithm CPCTH, a curve fit of DNBR vs thermal hydraulic conditions, compared to the design code COSMO.
3. Statistical combination of these synthesis, algorithm and radial peaking factor measurement uncertainties.

Other uncertainties, including those for thermal hydraulic parameter measurement, system parameters and the CHF correlation, were treated deterministically in the original CPC uncertainty analysis methodology.

Initially approved for ANO-2 Cycle 2 and SONGS-2 Cycle 1, the statistical combination of uncertainties methodology was documented in separate topicals for each plant (Reference 11) and approved in plant specific SERs. This methodology consisted of [

]^{a,c}. In addition, power measurement uncertainties were also calculated statistically but applied deterministically.

The Modified Statistical Combination of Uncertainties (MSCU) methodology was submitted and approved for PVNGS (Reference 1) and applied generically for all CE plants using the CPCS and COLSS via plant specific submittals and approvals. This methodology [

]^{a,c}. It also allows for determination and implementation of burnup, ASI and power dependent uncertainty factors. This methodology is in use at all CE plants using the CPCS and COLSS.

1.2.2 Overview of Methodology & Process

Figure 1 is a flow chart of the MSCU overall uncertainty analysis process as documented and approved in Reference 1 and in use at all CE plants using the CPCS and COLSS. The process consists of the following steps:

1. [ ]^{a,c}.

2. [ ]^{a,c}.

3. [ ]^{a,c}.

4. [ ]^{a,c}

5. [ ]^{a,c}.

The uncertainty analysis results consist of COLSS uncertainty factors for DNB POL, LHR POL, ASI and secondary calorimetric power and CPC uncertainty factors for DNBR, LPD, power and ASI. The key DNBR-related results are the EPOL addressable constants for COLSS and BERR1 for CPC.

2.0 Application of MSCU Methodology to CE 16x16 NGF

2.1 Introduction

The MSCU methodology is documented in Reference 1 and the process for performing an uncertainty analysis using the MSCU methodology is illustrated in Figure 1. This section describes the process for applying the MSCU methodology to plants with CE 16 x 16 NGF assemblies. The NGF design is described in Reference 12.

2.2 Methodology, Process and Input

The MSCU methodology and process were reviewed to determine what changes would be required for implementation of NGF. Four areas of potential impact, as listed in Section 1.1.4 are:

1. [ ]^{a,c}.
2. [ ]^{a,c}.
3. [ ]^{a,c}.
4. [ ]^{a,c}.

Items 1 through 3 are addressed automatically in the MSCU process. On Figure 1, the inputs to the box [ ]^{a,c}. In addition, the correction factors between TORC and CETOP-D reflect the fact that both TORC and CETOP-D model the NGF design and contain the NGF CHF correlations. [ ]^{a,c}. Calculating the DNBOPM error using the equation in step 4 of Section 1.2.2 (per Section 3.4 of Reference 1) automatically accounts for the difference between the [ ]^{a,c}.

The COLSS DNB POL and CPCS DNBR uncertainty factors, EPOL and BERR1, respectively, calculated using the methodology documented and approved in Reference 1 and adjustments in the MSCU process and inputs to implement and model NGF, will automatically reflect the impact of the NGF design, CHF correlations, DNBR limit and [ ]^{a,c}.

2.3 Statistical Evaluation

The NGF design includes grids with mixing vanes for only the top two-thirds portion of the axial height of the active fuel. These mixing grids improve CHF performance relative to grids without mixing vanes, resulting in increased DNB margin. [ ]^{a,c}.

2.3.1 Dependence on ASI

The dependence of the uncertainty factors on ASI can best be illustrated and investigated by a pure CETOP-D comparison of the NGF model and CHF correlations with the standard fuel model and CE-1

correlation for the hot pin power distributions used to calculate the uncertainty factors. Figure 2 is an example of such a comparison showing typical ASI dependent characteristics. The two populations of the POL ratio vs ASI with a transition at approximately ASI = +0.3 is clearly caused by the two grid types. This figure shows that the CETOP-D model for standard fuel with CE-1 yields conservative DNB POL results relative to the CETOP-D model for NGF with the NGF correlations. The conservatism decreases noticeably at ASI more positive than approximately +0.3. Therefore, the COLSS DNB POL and CPC DNBR uncertainty factors calculated for NGF should have a break point at approximately +0.3 ASI.

Figure 2 shows the potential for ASI dependence of the uncertainty factors most clearly since it is a clean comparison between the NGF model and the standard fuel model. [

] a, c. The ASI dependence in Figure 3 is not as strong as that in Figure 2.

Based on the comparisons illustrated in Figures 2 and 3, the ASI dependence of the uncertainty factors was investigated by calculating the CPC DNBR addressable constant uncertainty factor BERR1 for the full CPC ASI range (-0.6 to +0.6), bottom peaked shapes only (+0.3 to +0.6) and middle and top peaked shapes only (-0.6 to +0.1). For a typical set of power distributions, the full range BERR1 value was 1.0525, the bottom peaked shapes BERR1 value was 1.1021 and the middle and top peaked shapes BERR1 value was 0.9781. These results show that the full range BERR1 value would be non-conservative for bottom peaked shapes and too conservative for middle and top peaked shapes. The effect on the COLSS uncertainty factor is expected to be less since the ASI range for COLSS is narrower.

Both COLSS and CPC algorithms contain ASI dependent multiplicative factors (e.g. see Section 4.3.7 of Reference 2) which can be used to penalize portions of the ASI range to compensate for the ASI dependence of the uncertainty factors. Using ASI dependent multiplicative factors which ramp in from 1.0 at +0.2 to 1.12 at +0.3, the full range, bottom peaked shapes and middle and top peaked shapes in the above example, resultant BERR1 values are within 1% of each other. Appropriate ASI dependent multiplicative factors will be chosen each cycle so that the uncertainty factors calculated over the full ASI range will be valid. This process is consistent with the methodology for implementing ASI dependent uncertainty factors as documented in Reference 1.

2.3.2 Dependence on Temperature, Pressure and Flow

There are small but statistically significant correlations of the DNB POL and DNBR uncertainty factors with temperature and pressure in addition to the ASI dependence. These correlations are caused by [

] a, c. As a result, the uncertainty factors increase with reduced temperature or increased pressure. The correlation with flow was found to be insignificant.

Figure 4 is a scatter plot of the COLSS DNB POL uncertainty for an NGF cycle as a function of pressure showing the trend and correlation coefficient of +0.21. Figure 5 is a similar plot for a non-NGF cycle showing a statistically insignificant correlation coefficient of +0.03. Figures 6 and 7 are similar plots as a function of temperature where the NGF correlation coefficient is +0.13 and the non-NGF correlation coefficient is +0.03. Figures 8 and 9 show that the correlation coefficients for NGF and non-NGF cycles as a function of flow are essentially 0.0.

In order to evaluate the impact of the correlations of DNB POL and DNBR uncertainty with temperature, pressure and flow, the operating space was divided into 4 x 4 x 4 = 64 hypercubes composed of 25% of the full range for each parameter. The CPC addressable constant BERR1 was calculated for the full range of thermal hydraulic conditions, the "most favorable to DNB" hypercube (i.e. highest pressure, lowest

temperature and highest flow) and the "least favorable to DNB" hypercube (i.e. lowest pressure, highest temperature and lowest flow). It was determined that the "most favorable" hypercube yields the most conservative BERR1 value.

The cycle specific analysis will test for correlations with temperature, pressure and flow and, if present, utilize the "most favorable" portion of the full range of each parameter in order to calculate the uncertainty factors. These uncertainty factors will then be applied conservatively over all of operating space. Since the "most favorable" hypercube is composed of 1/64 or 1.5625% of the thermal hydraulic operating space, this approach maintains the 95/95 probability/confidence level of the resultant uncertainty factors.

2.4 Implementation

[

]^{a,c}.

3.0 Conclusions

The MSCU methodology described and approved in Reference 1 has the flexibility to address cores with CE 16x16 NGF. The COLSS and CPCS setpoint analysis which uses the MSCU methodology will incorporate the NGF design, input data and CHF correlations in the CETOP-D calculations. The MSCU process will be adjusted to account for ASI dependence and temperature, pressure or flow correlations.

The overall uncertainty factors determined using the MSCU methodology described in Reference 1 and the MSCU process as modified to reflect the NGF design and CHF correlations continue to ensure that the COLSS DNB POL calculations and the CPCS DNBR calculations will be conservative to at least a 95% probability and 95% confidence level.

4.0 References

1. CEN-356(V)-P-A Revision 01-P-A, "Modified Statistical Combination of Uncertainties," May 1988.

2. CEN-305-P Revision 02-P, "Functional Design Requirements for a Core Protection Calculator," May 1988.*

3. CEN-304-P Revision 02-P, "Functional Design Requirement for a Control Element Assembly Calculator," May 1988.*

4. CEN-312-P Revision 02-P, "Overview Description of the Core Operating Limit Supervisory System (COLSS)," November 1990.*

5. CEN-160(S)-P Revision 1-P, "CETOP-D Code Structure and Modeling Methods for San Onofre Nuclear Generating Station Units 2 and 3," September 1981.*

6. CENPD-162-P-A, "C-E Critical Heat Flux," September 1976; Supplement 1-A, February 1977.

7. CENPD-207-P-A, "C-E Critical Heat Flux Part 2 Nonuniform Axial Power Distribution," December 1984.

8. CENPD-387-P-A Revision 000, "ABB Critical Heat Flux Correlations for PWR Fuel," May 2000.

9. WCAP-16523-P Revision 0, "Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes," March 2006.

10. CENPD-170-P, "Assessment of the Accuracy of PWR Safety System Actuation as Performed by the Core Protection Calculators (CPC)," July 1975; Supplement 1-P, November 1975.

11. CEN-139(A)-P, "Statistical Combination of Uncertainties," November 1980 [Also, CEN-283(S)-P, October 1984; CEN-338(C)-P, August 1986; CEN-343(C)-P, October 1986.]

12. WCAP-16500-P Revision 0, "CE 16 x 16 Next Generation Fuel Core Reference Report," February 2006.

13. CENPD-161-P-A, "TORC Code - A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986.

* References 2 through 5 were submitted to the NRC for information in order to aid in the review of other documents or plant specific submittals. Reference 5 is a plant specific document which is typical of similar documents for other plants. Changes have been made to CPC, CEAC, COLSS and CETOP-D since these documents were written. However, the general description of the systems and codes remain applicable.

GLOSSARY

Acronym/Abbreviation	Definition
ABB-NV	Critical heat flux correlation for non-vaned fuel
ANC	Westinghouse neutronics computer code
ANO-2	Arkansas Nuclear One Unit 2
ASI	Axial shape index [(L-U)/(L+U)]
BERR1	CPC addressable constant - multiplicative power adjustment factor for DNBR
CEAC	Control Element Assembly Calculator
CE-1	CE CHF correlation present in on-line algorithms
CETOP-D	Thermal margin algorithm and computer code
CHF	Critical Heat Flux
COLSS	Core Operating Limit Supervisory System
COSMO	Old design thermal hydraulics code
CPC	Core Protection Calculator
CPCS	Core Protection Calculator System
CPCTH	DNBR algorithm in original CPCS design
DNB	Departure from Nucleate Boiling
DNBOPM	DNB Overpower Margin
DNBR	Departure from Nucleate Boiling Ratio
E1, E2	CPC region-dependent algorithm uncertainty allowances for DNBR
EPOL (EPOL2, EPOL4)	COLSS DNB POL addressable constant adjustment factors
FLAIR	Old three-dimensional neutronics code
INCA	Old power distribution measurement computer code
LHR	Linear Heat Rate
LPD	Local Power Density
MSCU	Modified Statistical Combination of Uncertainties
NGF	Next Generation Fuel
OPM	Over Power Margin
pdf	Probability Density Function
POL	Power Operating Limit
PVNGS	Palo Verde Nuclear Generating Station
ROCS	CE neutronics computer code
RPS	Reactor Protection System
SCU	Statistical Combination of Uncertainties
SONGS-2	San Onofre Nuclear Generating Station Unit 2
TORC	Detailed design thermal hydraulics code
WSSV-T	NGF CHF correlation for side supported mixing vaned fuel

Figure 1

Modified Statistical Combination of Uncertainties Process
(As documented and approved in CEN-356(V)-P-A Revision 1-P-A)

a, c

Figure 2

CETOP-D POL Ratio vs ASI

a, c

Figure 3

CETOP-D POL Ratio vs ASI
with NGF p.d.f.

a, c

Figure 4

a, c

DNB POL Uncertainty vs CETOP-D Pressure

Figure 5

a, c

DNB POL Uncertainty vs CETOP-D Pressure

Figure 6

a, c

DNB POL Uncertainty vs CETOP-D Temperature

Figure 7

a, c

DNB POL Uncertainty vs CETOP-D Temperature

Figure 8

a, c

DNB POL Uncertainty vs CETOP-D Flow

Figure 9

a, c

DNB POL Uncertainty vs CETOP-D Flow

Section H

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Our ref: LTR-NRC-07-20
April 5, 2007

Subject: Presentation Material on Audit Responses to Questions on Setpoints Supplement 1-P to WCAP-16500-P (TAC No. MD0560) (Proprietary/Non-proprietary)

Enclosed are copies of the Proprietary and Non-Proprietary presentation material on Audit Responses to Questions on Setpoints Supplement 1-P to WCAP-16500-P discussed with the NRC on March 29, 2007 (Enclosure 1 and 2).

Also enclosed is:

1. One (1) copy of the Application for Withholding, AW-07-2265 (Non-proprietary) with Proprietary Information Notice.
2. One (1) copy of Affidavit (Non-proprietary).

This submittal contains proprietary information of Westinghouse Electric Company, LLC. In conformance with the requirements of 10 CFR Section 2.390, as amended, of the Commission's regulations, we are enclosing with this submittal an Application for Withholding from Public Disclosure and an affidavit. The affidavit sets forth the basis on which the information identified as proprietary may be withheld from public disclosure by the Commission.

Correspondence with respect to the affidavit or Application for Withholding should reference AW-07-2265 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,

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

Enclosures
cc: A. Mendiola, NRR
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Our ref: AW-07-2265
April 5, 2007

APPLICATION FOR WITHHOLDING PROPRIETARY INFORMATION FROM PUBLIC DISCLOSURE

Subject: LTR-NRC-07-20, Enclosure 1, "Presentation Material on Audit Responses to Questions on Setpoints Supplement 1-P to WCAP-16500-P (TAC No. MD0560) (Proprietary)"

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Very truly yours,

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H. Cruz, NRR
J. Thompson, NRR

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

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COUNTY OF ALLEGHENY:

Before me, the undersigned authority, personally appeared B. F. Maurer, 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:

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

Sworn to and subscribed
before me this 5th day
of April, 2007.

Sharon L. Markle
Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Sharon L. Markle, Notary Public
Monroeville Boro, Allegheny County
My Commission Expires Jan. 29, 2011
Member, Pennsylvania Association of Notaries

(1) I am Acting Manager, Regulatory Compliance and Plant Licensing, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse) and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rulemaking 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 which 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 LTR-NRC-07-20 Enclosure 1, on "Presentation Material on Audit Responses to Questions on Setpoints Supplement 1-P to WCAP-16500-P (TAC No. MD0560) (Proprietary)," for submittal to the Commission, being transmitted by Westinghouse letter (LTR-NRC-07-20) and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse Electric Company is responses to NRC's Request for Additional Information conducted during an audit on March 29, 2007.

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

(a) Demonstrate the acceptability of the CE 16x16 Next Generation Fuel Design.

(b) Assist customers in implementing an improved fuel product.

Further this information has substantial commercial value as follows:

(a) Westinghouse can use this fuel design with its associated correlation to further enhance their licensing position over their competitors.

(b) Assist customers to obtain license changes.

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 fuel design 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 for developing the enclosed improved core thermal performance methodology.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

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

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

COPYRIGHT NOTICE

The reports transmitted herewith each bear a Westinghouse copyright notice. The NRC is permitted to make the number of copies of the information contained in these reports which are necessary for its internal use in connection with generic and plant-specific reviews and approvals as well as the issuance, denial, amendment, transfer, renewal, modification, suspension, revocation, or violation of a license, permit, order, or regulation subject to the requirements of 10 CFR 2.390 regarding restrictions on public disclosure to the extent such information has been identified as proprietary by Westinghouse, copyright protection notwithstanding. With respect to the non-proprietary versions of these reports, the NRC is permitted to make the number of copies beyond those necessary for its internal use which are necessary in order to have one copy available for public viewing in the appropriate docket files in the public document room in Washington, DC and in local public document rooms as may be required by NRC regulations if the number of copies submitted is insufficient for this purpose. Copies made by the NRC must include the copyright notice in all instances and the proprietary notice if the original was identified as proprietary.

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Enclosure 2
Presentation Material on Audit Responses to Questions on Setpoints Supplement 1-P to WCAP-16500-P (TAC No. MD0560) (Non-Proprietary)

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

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Response to Questions on Setpoints
Supplement 1-P to WCAP-16500-P

One White Flint, Rockville, MD
Presentation to NRC
3/29/2007

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Question 1

- Section 2.3.1 of Supplement 1-P documents an evaluation of ASI dependence which is illustrated in Figure 2. Examination of Figure 2 reveals a transition between +0.2 and +0.4 ASI. Please repeat the CPC sensitivity study, redefining the subset "bottom peaked shapes only" to avoid this transition (e.g. +0.4 to +0.6 ASI).

Response 1

- Changing the ASI range to +0.4 to +0.6 changes the resultant BERR1 value from 1.1021 to 1.1025

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- ASI dependent multiplicative factors will be chosen each reload to assure final BERR1 value covers entire range of ASIs at 95/95

Question 2

- Section 2.3 of Supplement 1-P documents a statistical evaluation documenting a dependency on ASI, temperature, and pressure; but concluding that any correlation with flow was found to be insignificant. Please explain further as to why the CHF correlation based on mixing vanes would not introduce a dependency on flow when compared to the non-vaned CE-1 correlation.

Response 2

- Figure 8 in Supplement 1-P is a plot of the ratio of the difference between COLSS with CE-1 correlation & CETOP-D with NGF correlations with all uncertainties included as applied in a COLSS uncertainty analysis

- A comparison of CHF data between SSV and Non-vane grid shows the relative gain in CHF performance for the SSV grid is nearly constant as a function of flow (this is consistent with Figure 8)

- This constant gain in CHF performance is probably due to the strong swirl generated in the subchannel by the Side Supported Vanes

Figure 8 from Supplement 1-P

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CHF Performance Comparison Between SSV and Non-vaned grids

a, c

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Question 3

- Section 2.3 of Supplement 1-P documents a statistical evaluation. Please repeat these sensitivity cases removing all monitoring and modeling uncertainties to highlight the effects of the different CHF correlations and DNBR pdfs.

Response 3

- Only way to remove all monitoring & modeling uncertainties is with CETOP-D to CETOP-D comparisons
  - See figures

- "Most favorable to DNB" hypercube yields most conservative BERR1 with or without statistically combined uncertainties

Response 3 (continued)

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Response 3 (continued)

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Response 3 (continued)

a, c

Response 3 (continued)

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Response 3 (continued)

a, c

Response 3 (continued)

a, c

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Question 4

- Section 2.4 of Supplement 1-P documents the implementation of the CPC factors using E1 and E2 to capture the NGF credits (since BERR1 must remain greater than 1.0). Are these constants (E1, E2) used for any other purpose and, if so, how will these combined effects be captured ?

Response 4

- E1 (narrow range) is not currently used for any other purpose (set to 1.0)
- [ ]a,c
- [ ]a,c
- [ ]a,c

Question 5

- Identify the statistical tests that will be used to assure poolability in the final uncertainty analyses.

Response 5

- We expect to use the F-Test and the two sample t-test to check for poolability of the DNB POL errors vs ASI
  - The F-Test checks to see if the sample variances of two subpopulations are close enough to indicate that it is likely that they come from the same population
  - The two sample t-test is then used to see if the sample means of the two subpopulations are close enough to indicate that it is likely that they come from the same population

- Visual evaluations can also be used as a confirmation

Question 6

- Outline the steps that will be used in the uncertainty analysis to assure that the final ASI dependent penalties and uncertainty factors are conservative at 95/95 for the entire cycle.

Response 6

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Response 6 (continued)

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Response 6 (continued)

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Response 6 (continued)

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Westinghouse

Response 6 (continued)

- Note that these steps have not been tested in detail.
- The steps may have to be adjusted in order to assure conservative results at 95/95.
- There may be some variation in the steps to accommodate computer code limitations.

Question 7

- Please confirm the validity of the bias and uncertainties for the nuclear design of NGF?

Response 7

- ANC used for analysis of a wide variety of PWR lattice configurations
  - Rod diameters 0.360 to 0.440
- Benchmarks reported in PARAGON Topical (WCAP-16045-P-A) show no significant difference in measured-predicted errors between plants with different rod diameters
- Uncertainties will continue to be updated as necessary to maintain accuracy with measurements

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