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
P.O. Box 355
Pittsburgh, PA 15230-0355

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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^{TM}. 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 <u>CONCLUSIONS</u>

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 <u>REFERENCES</u>

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.”

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.