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CE 16x16 Next Generation Fuel Core Reference Report

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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
    - 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.
    - The standard design Mid grids have cantilever springs, while the NGF grids have vertical "I-springs" (Figure 2-9), which are designed [

]^{a,c}

  - 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}:
  - 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.
  - 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):
  - 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.
  - Cladding material has been changed from OPTIN™ to Optimized ZIRLO™ to take advantage of ZIRLO™'s improved corrosion resistance and dimensional stability under irradiation.
  - 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 Holddown Springs	$P_m \leq S_m$ $P_m + P_b \leq F_s S_m$ Shear stress $\leq$ Minimum Yield Stress in Shear
Condition III	All components except Holddown Springs Holddown Springs	$P_m \leq 1.5\ S_m$ $P_m + P_b \leq 1.5\ F_s S_m$ 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

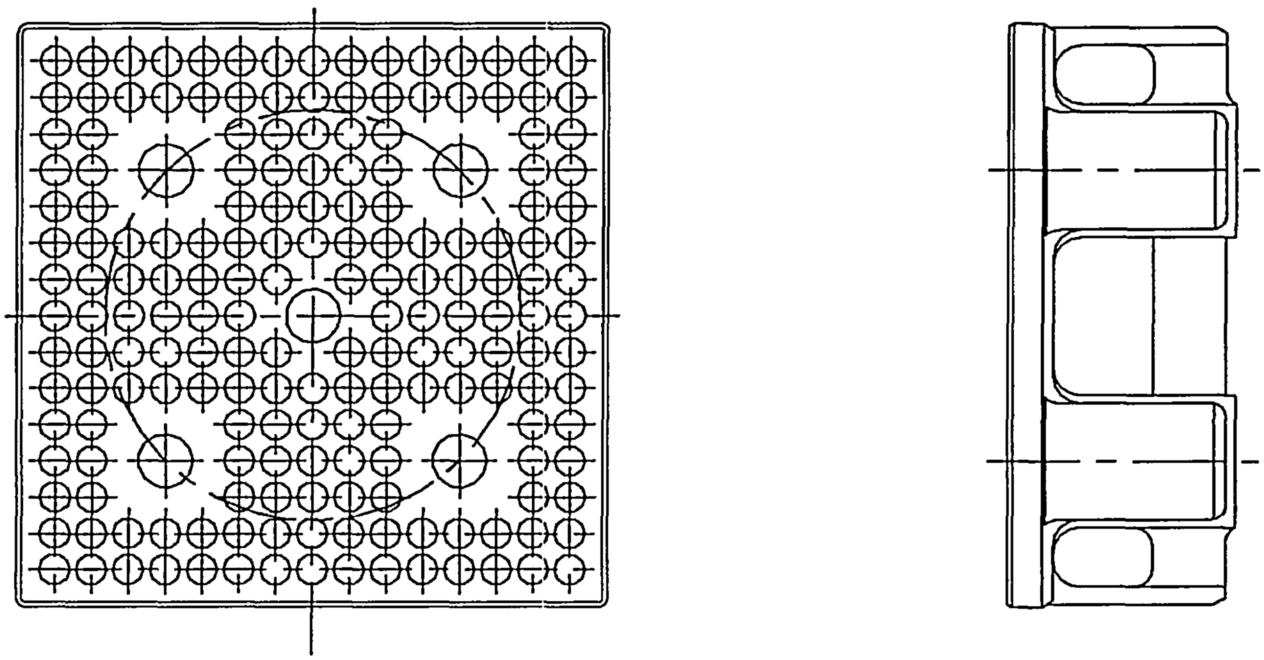


Figure 2-3
16 CE NGF Top Nozzle Design

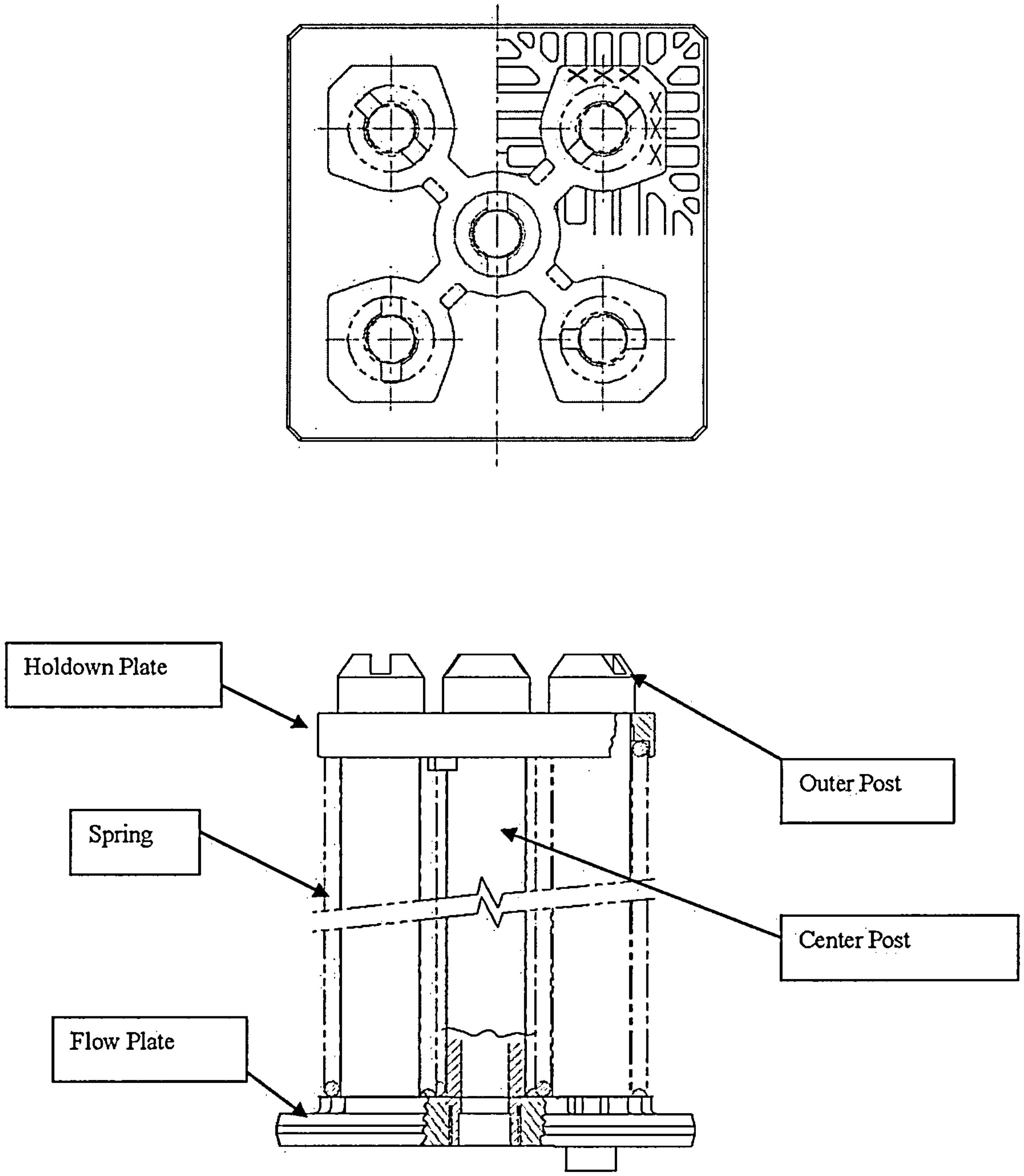


Figure 2-4
16 CE NGF Guide Thimble Flange to Upper Nozzle Flow Plate 3-D Interface

a, c

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

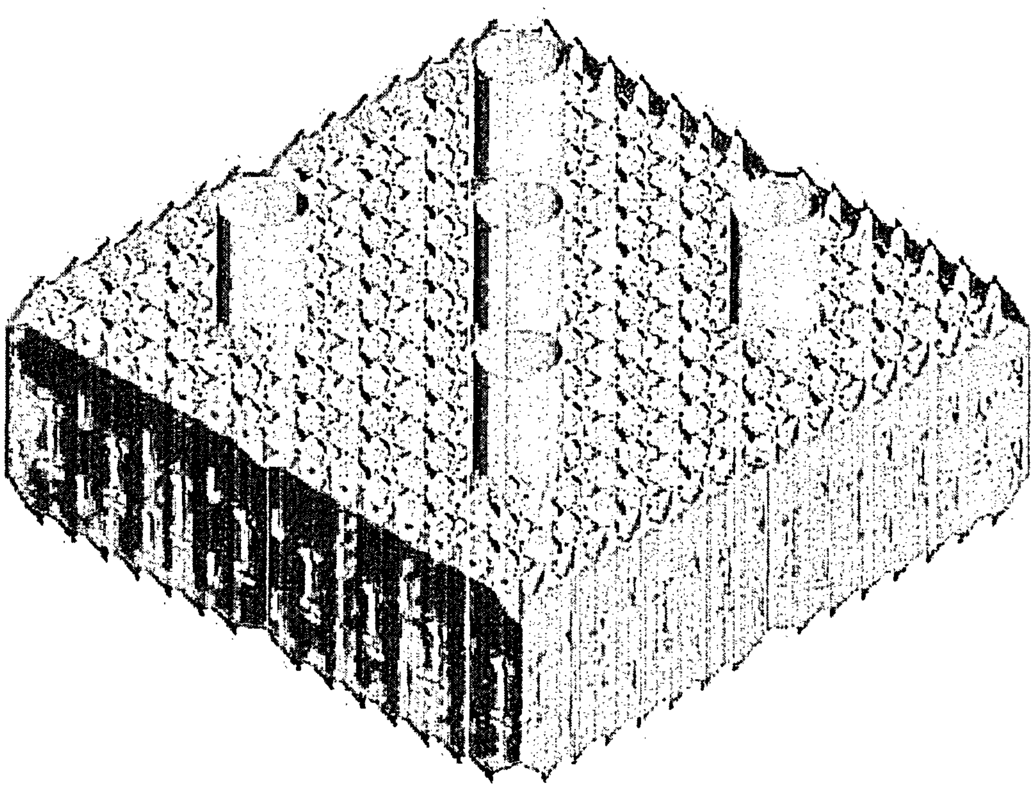


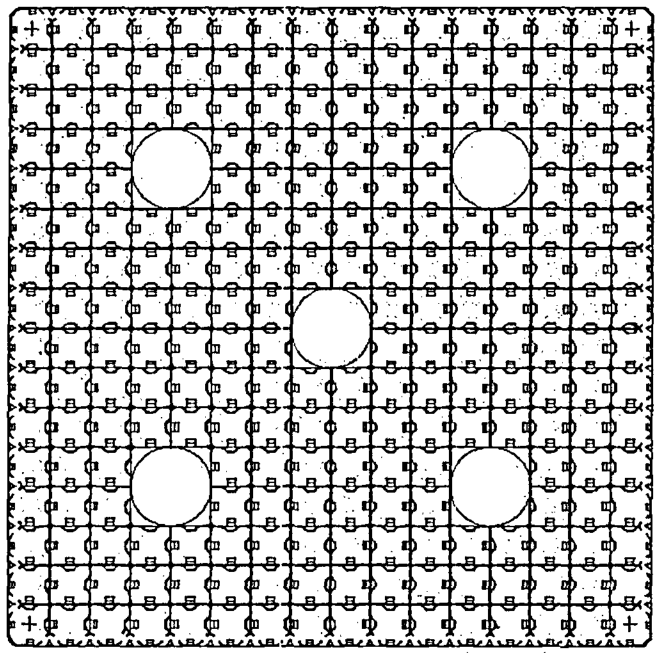
Figure 2-9
16 CE NGF Mid Grid "I'Spring" Design

a, c

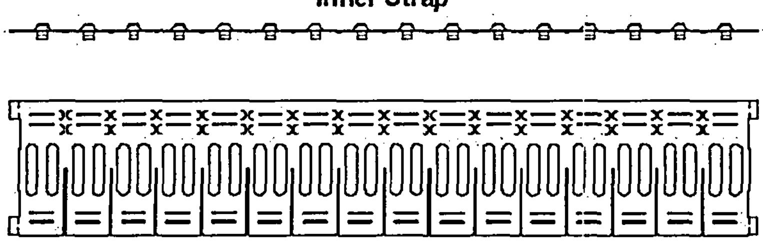
Figure 2-10

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

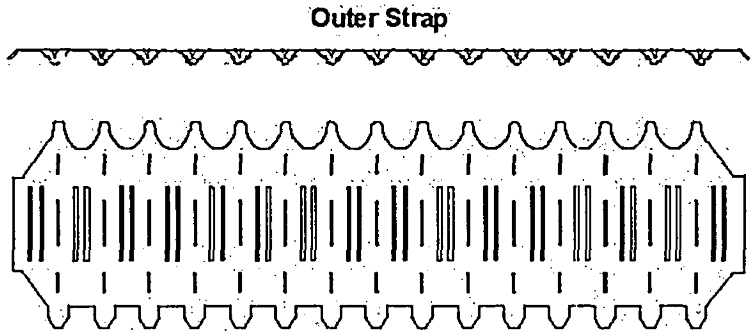
Grid Assembly

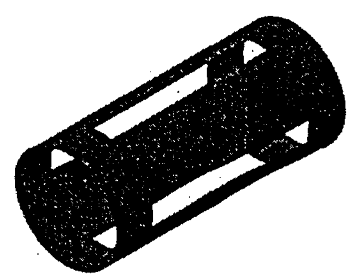


Inner Strap



Outer Strap





Sleeve

Figure 2-11

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

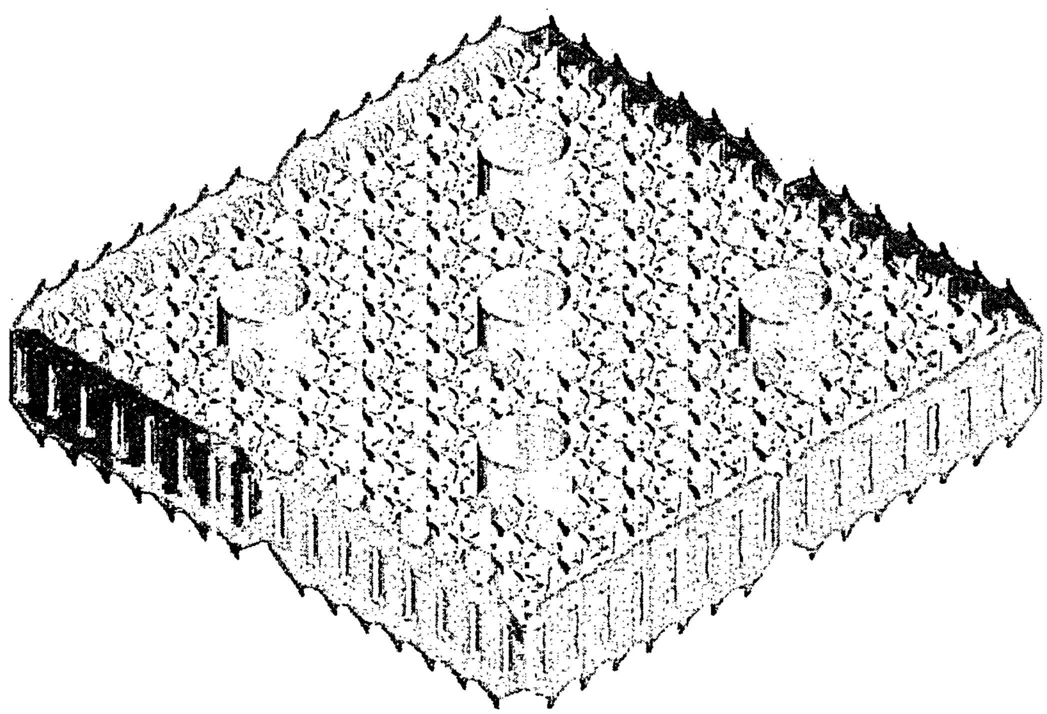


Figure 2-12
16 CE NGF Intermediate Flow Mixing (IFM) Grid Design

a, c

Figure 2-13

16 CE NGF Guardian™ Grid Assembly with Inserts

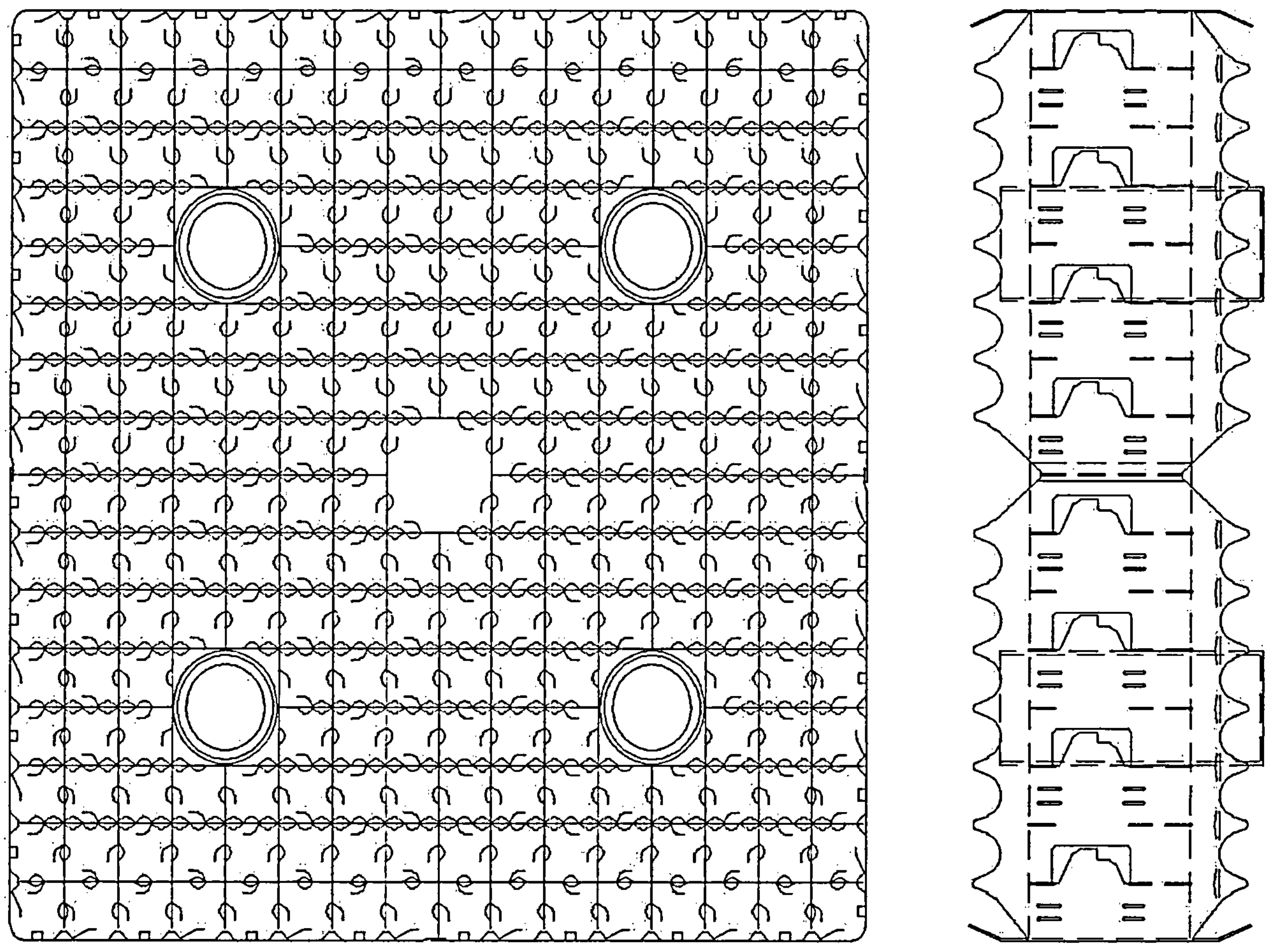


Figure 2-14
Typical 16 CE NGF Fuel Rod Design (Plant B)

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.

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

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

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

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
    - Maximum flow area reduction or spacer grid blockage fraction
    - Fuel lattice hydraulic diameter
    - Height of the spacer grid, used to estimate wetted surface area
    - 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

$$(A-7)$$

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

$$(A-8)$$

where

[ ] a, c

The equivalent spacer grid cell diameter is defined as follows

[ ] a, c

$$(A-9)$$

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

$$(A-11)$$

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.

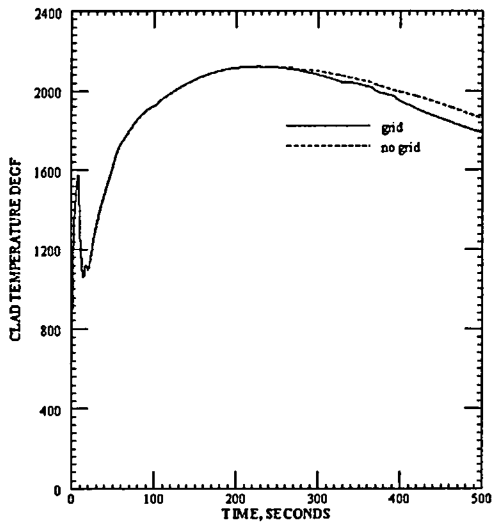
A.3. CENPD-132 Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," March 2001.

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.

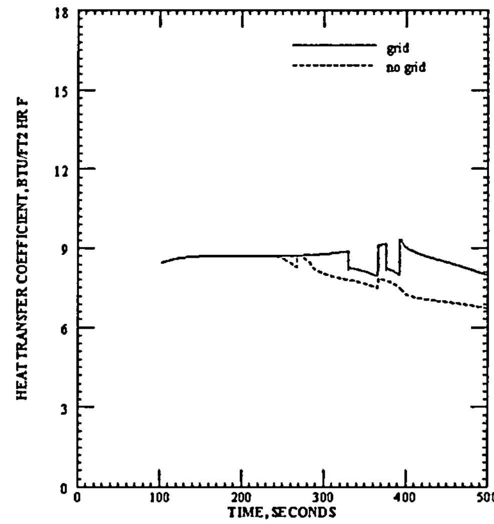
A.5. WCAP-10484-P-A, "Spacer Grid Heat Transfer Effects During Reflood," March 1991.

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.

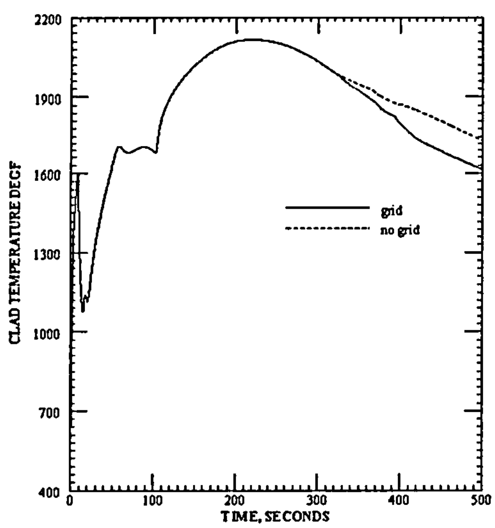
A.7. CENPD-138-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974.



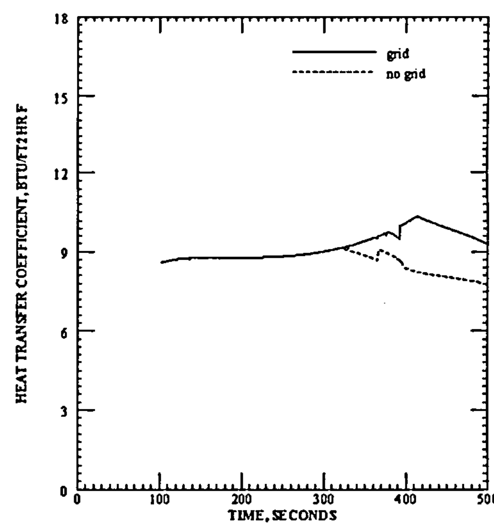
CLAD TEMPERATURE
Node 14 (Above Rupture Node)
Figure A-1



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