ENCLOSURE

## SER ON WCAP-10444

### 1.0 INTRODUCTION

By letter dated December 30, 1983, Westinghouse requested NRC to review the topical report WCAP-10444, "Westinghouse Reference Core Report, VANTAGE 5 Fuel Assembly."

The VANTAGE 5 fuel assembly is a modification of the design of the Standard and Optimized 17 X 17 fuel assemblies. (It will also be offered in 14 X 14 and 15 X 15 configurations.) The new features of the VANTAGE 5 fuel assembly consist of (1) axial blankets for improved neutron utilization, (2) Integral Fuel Burnable Absorber (IFBA), (3) Intermediate Flow Mixer (IFM) grids, (4) Reconstitutable top and bottom nozzles, and (5) extended burnup. The impact of these design changes is evaluated from the standpoints of mechanical, nuclear and thermal hydraulic designs and transient and accident analyses. The analysis is performed with the 17 X 17 fuel assembly array for a 4-loop reference plant. The 14 X 14 and 15 X 15 application should be evaluated on a plant-specific basis.

Because our SER on Westinghouse's extended burnup topical report WCAP-10125 (Ref. 1) is not complete, the design bases/criteria and analysis methods review in this submittal have not been approved for application to extended burnup levels. Consequently, the approval of the VANTAGE 5 design for operation to extended burnup levels is contingent on NRC approval of Westinghouse's extended burnup topical report. It should be noted, however, that the review of WCAP-10125 is complete and there are no outstanding issues at this time. Our review, documented in this safety evaluation report, covers the VANTAGE 5 fuel designs to those burnups to which PWRs with Westinghouse fuel are presently operating.

## 2.0 MECHANICAL DESIGN

### 2.1 Fuel Assembly Design

WCAP-10444 indicates that the design bases and design limits for the VANTAGE-5 fuel assembly are essentially the same as those for the optimized fuel assembly (OFA) design described in WCAP-9500 (Ref. 2). We have therefore relied heavily on our review of Reference 2 and only the differences in design bases will be discussed in the following evaluation.

#### 2.1.1 Fuel Rod Growth Gap

WCAP-10444 states that the design basis for the axial clearance between core plates and nozzle end plates should allow sufficient margin for fuel assembly and fuel rod irradiation growth to design burnup established in WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel" (Ref. 1). WCAP-10125 indicates that the initial fuel rod-to-nozzle growth gap is designed based on a statistical convolution of the distribution of the measured gap data as a function of fluence and the distribution of rod average fluences for an assembly. This statistical treatment of uncertainties associated with the fuel rod/assembly irradiation growth effect is a deviation from the approved method. In response to a staff question (Ref. 3), however, Westinghouse indicates that the current approved design criterion has been used for the VANTAGE-5 design, i.e., the minimum gap is the required allowance for irradiation growth established to preclude fuel rod-to-nozzle interface during projected operation on the basis of the assumption of worst case fuel rod and fuel assembly growth combined with worst case fabrication tolerances. Therefore, the design for fuel rod growth gap for VANTAGE-5 is the same as that for the OFA design and is therefore acceptable.

For the VANTAGE-5 fuel assembly design, the overall assembly length has been adjusted to accommodate assembly growth and the distance between the top and bottom nozzle plates has been adjusted to accommodate the expected

increased fuel growth related to extended burnup operation. The growth predictions were based on Westinghouse incore experience including results from the high burnup demonstration program. Therefore, adequate shoulder gap is provided to accommodate rod growth.

### 2.1.2 Fuel Fretting Wear

The Westinghouse design guideline for VANTAGE 5 regarding fuel rod fretting wear is the same as that for OFA by limiting the fuel rod wear due to contact with the spacer grids to less than a certain percentage (proprietary) of clad thickness. That the VANTAGE 5 design meets this guideline is confirmed via hydraulic testing. The tests were performed in the Westinghouse Fuel Assembly Test System (FATS) with a VANTAGE 5 assembly adjacent to a 17 X 17 OFA assembly, adjacent to a 17 X 17 STD fuel assembly, and adjacent to another VANTAGE 5 assembly. The results of these tests confirm that the projected fuel rod wear of VANTAGE 5 is well within the design guideline. There was no indication of adverse fretting wear of the fuel rods by the standard structure or the IFM grids.

### 2.1.3 Fuel Assembly Structural Integrity

Section 2.2.3.3 of WCAP-10444 provides an evaluation of the VANTAGE 5 fuel assembly structural response to the externally applied forces such as LOCA and seismic forces. Appendix A.3.0 of the report describes the mechanical test program performed to confirm the structural integrity of the fuel assembly. The mechanical tests consist of lateral vibration and impact tests, and axial stiffness and impact tests. The results show that the VANTAGE 5 design changes represent a small improvement over the OFA design with respect to the structural dynamic response. This is attributable to the IFM grids providing a small stiffness effect near the top of the fuel assembly. The IFM grids also provide additional load sharing during Seismic/LOCA impact.

Analytical evaluations of the fuel assembly response to the most limiting LOCA and seismic forces are performed with the time history numerical technique in the same manner as the approved method described in WCAP-9401 (Ref. 4). The resulting LOCA and seismic induced grid impact forces are combined in accordance with the method specified in Section 4.2 of SRP. The results show that the maximum combined grid impact force for both the Zircaloy structural grids and IFM grids are within the allowable limits. The maximum fuel assembly deflection and the stress resulting from the deflection indicated substantial safety margins compared to the allowable value. Therefore, we conclude that structural damage from external forces is not a concern. However, as discussed in the staff safety evaluation report for WCAP-9401, for each plant application, it must be demonstrated that the applied seismic/LOCA loads considered in WCAP-9401 bound the plant in question or else additional analysis will be required.

### 2.1.4 Fuel Assembly Shipping and Handling Loads

The design requirement for shipping and handling loads for the VANTAGE 5 fuel assembly is reduced from 6g to 4g. This change allows for reduction in the thickness of nozzle end plates and therefore increases the axial space available for fuel assembly and rod design. Westinghouse has tested the handling acceleration at both the manufacturing facility and reactor sites, and determined that handling acceleration is well below the 4g limit. Westinghouse has also performed extensive over-the-road tests with shipping containers containing dummy fuel assemblies and found that insignificant g loads were communicated to the fuel assembly carriage in the container. At Columbia University, tests were also performed with a dummy assembly under the most accelerated crane movement possible with abrupt stops. Results of repeated tests confirm the maximum axial load to be far less than 4g. Therefore, based on the results of these tests, the 4g design criterion is acceptable.

### 2.1.5 Fuel Assembly Structural Components

The design bases for the VANTAGE 5 fuel assembly structural components are essentially the same as those described in WCAP-9500 for OFA except for the 4g structural requirement. However, there are a few changes in the VANTAGE 5 component designs. The bottom nozzle uses Inconel 718 instead of Type 304 stainless steel to permit the use of a thinner nozzle plate. The top nozzle has a groove in each thimble through-hole in the nozzle plate to facilitate removal and has a reduced nozzle plate thickness to provide additional axial space. The hold-down spring is a 4-leaf spring design instead of the standard 3-leaf design in order to compensate for the increase in pressure drop due to the IFM grids.

For both the top and bottom nozzles, Westinghouse verified through a finite element analysis that the maximum stresses at 4g's do not exceed the allowable limits. They are also tested to demonstrate that the strength exceeds the structural 4g functional requirement.

The 4-leaf hold-down spring has been hydraulically tested in the Westinghouse FATS facility to determine the assembly lift off flow rate and the assembly pressure loss coefficient. The results of the flow testing serve as the basis for optimizing the spring force. In response to a staff question (Ref. 5), Westinghouse indicated that the effects of spring relaxation and assembly growth are also accounted for in the spring design. The spring has been designed with a 10% higher BOL load than required by the hydraulic lift force to account for the net effect of the assembly growth and spring relaxation.

There are no changes involved in the guide thimbles, instrumentation tubes and the structural spacer grids from OFA design. The IFM grids are non-structural members whose primary function is to promote mid-span flow mixing. Therefore, the design bases for the IFM grids are to avoid cladding wear and interactive damage with the grids of the neighboring assemblies during fuel handling. The impact tests described previously have also been performed for the IFM grids.

The results show that the crush strength of the IFM grids exceeds the expected dynamic load during a seismic/LOCA event. Therefore, a coolable geometry is assured.

### 2.1.6 Irradiation Demonstration Program

The report indicates that an irradiation demonstration program will be performed to provide early confirmation performance data for the VANTAGE-5 design. This program will be performed consistent with the product demonstration effort employed for the introduction of the 17 X 17 STD fuel and OFA fuel design. We find this commitment acceptable.

## 2.2 Fuel Rod Design

The VANTAGE 5 fuel rod design is identical to the OFA design with the exceptions that (1) axial blanket pellets replace the top and bottom parts of the fuel stacks; (2) thin boride coating is added to the fuel pellets in the middle portion of the fuel rod to serve as integral fuel burnable absorber (IFBA), (3) an additional plenum space is included to accommodate gas release from the fuel and the thermal expansion differential between the cladding and fuel; and (4) a tapered bottom end plug is used to guide the fuel rod during top-end reconstitution. In addition, VANTAGE 5 is designed for a higher burnup and higher enthalpy rise factor ($F_{\Delta H}$) operation. However, the design bases for VANTAGE 5 are the same as OFA in that fuel rod damage should not occur due to excessive rod internal pressure, clad stress, clad strain, clad temperature, fuel temperature or clad strain fatigue, and that the fuel rod should not fail due to fretting wear on the outer cladding surface or by clad flattening. The analyses showing that these design bases are met were performed using the same methods as were used for OFA and the STD designs.

(a) Rod Internal Pressure - Since VANTAGE 5 fuel rods contain IFBA, additional helium gas is created and released from the depleted boride contained in the burnable boride coating. However, a Westinghouse evaluation has shown that the internal pressure of VANTAGE 5 meets the design criterion.

(b) Clad Strain and Stress - Design evaluations show tht the clad stresses and strains for both IFBA and non-IFBA fuel rods of VANTAGE 5 design meet the stress and strain design limits.

(c) Clad Temperature - In order to limit metal oxide formation to acceptable values, specific design limits at the metal/oxide interface temperature are specified for both condition I and II transients. Westinghouse calculations have shown that the clad temperature of VANTAGE 5 fuel rod meets the design limits.

(d) Fuel Temperature - To avoid $UO_2$ centerline melting during condition I and II transients, the fuel centerline temperature is limited to 4700°F. A Westinghouse design evaluation has shown that the fuel centerline temperature remains below this limit.

(e) Clad Fatigue - As a design basis, Westinghouse specified a value (proprietary) of a fatigue life usage factor limit. The design evaluations have shown that the cumulative fatigue life usage factor for the VANTAGE 5 cladding is far less than the specified limit value.

(f) Clad Wear - Westinghouse has a value (proprietary) of wall thickness as a general guideline in evaluating cladding imperfections including fretting wear. Since the structural grids and fuel tube for both VANTAGE 5 and OFA are identical with respect to grid cell forces acting on the fuel rods, the amount of clad wear for both fuel designs should be the same. As part of the hydraulic test performed at the FATS facility, the wear test results have verified that the VANTAGE 5 fuel wear characteristics are similar to that of the 17 X 17 OFA. Therefore, the conclusion can be

drawn that the VANTAGE 5 fuel rod wear will not exceed the maximum wear depth limit.

(g) Clad Flattening - If axial gaps in the fuel pellet columns were to occur due to densification, the cladding would have the potential of collapsing into a gap, i.e., clad flattening. Because of the large local strains that would result from collapse, the cladding is assumed to fail. Westinghouse calculations for VANTAGE 5 show that the predicted clad flattening time exceeds residence times expected for the extended burnup fuel management. Typical values of clad flattening times are in excess of 45000 effective full power hours.

## 2.3 Integral Fuel Burnable Absorber (IFBA)

Some VANTAGE 5 fuel pellets have a thin layer of boride coating on the surface to act as integral fuel burnable absorber. The design bases are the same for both IFBA and non-IFBA fuels. Evaluations are made to determine if the IFBA rods meet the design bases.

(a) Performance Test - The performance of the IFBA rods has been demonstrated with test fuel rods having various thickness of boride coating irradiated in both a test reactor (BR-3) and a commercial reactor (Turkey Point Unit 3). Post Irradiation Examinations (PIE) of these test rods irradiated up to 13000 MWD/MTU indicate that the boride coating performs well with no apparent loss of coating integrity. The monitoring of the reactivity and depletion characteristics shows that the absorber behaves as predicted. The discharged fuel assembly shows no leaking rods in sipping tests. Additional tests are planned to have IFBA rods inserted in the same reactor to provide additional confirmation performance data for the IFBA feature.

(b) Clad Hydriding - Hydriding as a cladding failure mechanism is precluded by controlling the level of moisture and other hydrogenous impurities during fabrication. The IFBA-containing fuel is required to meet the same hydrogen, fluorine and impurity limits set for the $UO_2$ fuel with the exception of the coating constituent. Clad hydrogen analysis of the irradiated test rods from BR-3 reactor has shown no difference in the cladding hydrogen absorption characteristics between the areas adjacent to both coated and non-coated fuel pellets. Based on PIE results and thermodynamics consideration, Westinghouse concludes that no chemical reaction between boride coated pellet and the Zircaloy-4 cladding during reactor life is expected. In addition, the clad is also protected by $ZrO_2$ which is also thermodynamically stable with respect to the boride coating.

(c) Clad Breaching - Though the design basis for VANTAGE 5 fuel is to maintain cladding integrity, the event of a sufficiently large Zircaloy cladding breaching is considered which would release fission products from fuel rod and would also likely result in the boride coating leached out by the coolant water. Since the boride loading is small in any individual rod and the IFBA rods are distributed across the core, Westinghouse states that the change in peaking factor would be small even if the clad breach and absorber loss occur early in life.

(d) Fuel Rod Waterside Corrosion - The IFBA coating on the fuel pellets has no effect on clad/oxide interface temperature or waterside corrosion compared to non-IFBA rods. The waterside corrosion criterion for the non-IFBA rod remains applicable for the IFBA rod.

(e) IFBA Coating -Fuel Compatibility - Westinghouse has evaluated the compatibility of IFBA coating and $UO_2$ fuel. Thermodynamic evaluations show no adverse chemical reaction occur between the boride coating and the $UO_2$ fuel. This is confirmed by electron microprobe analysis showing no adverse diffusion reaction between the boride coating and $UO_2$ when heated to 1750°C for 8 hours or 1600°C for 24 hours. PIE test results

from the BR-3 test reactor also confirm the compatibility of the absorber coating and $UO_2$ under operating conditions. Although a reaction has been observed due to excess oxygen and boride depletion, Westinghouse has found this to have no adverse impact on either the fuel or absorber performance.

(f) Boride Coating Axial Distribution - The boride coating is required to effectively remain in place throughout its functional lifetime to ensure no significant loss of neutron absorber uniformity. Thin coating integrity is confirmed by the PIE results of test rods from the BR-3 test reactor. A direct measurement of neutron absorption in a commercial reactor having boride coated pellets also confirms no absorber redistribution after one full cycle. Therefore, mechanical integrity of the absorber coating is reasonably assured.

## 2.4. Testing and Inspection Plans

As required by SRP Section 4.2, testing and inspection plans for new fuel should include verification of significant fuel design parameters. WCAP-10444 provides a brief discussion of the Westinghouse quality assurance program which provides for control over all activities affecting product quality including design, development, fabrication, testing and inspection, etc. The quality control program described in WCAP-9500 for OFA fuel is also applicable to VANTAGE-5 fuel. WCAP-10444 indicates that manufacturing control equipment for full scale production of VANTAGE 5 has not yet been finally designed. However, the equipment and process will be fully qualified by quality control to ensure that enrichment and stack length controls consistently meet design requirements, and that all other specification requirements such as moisture and hydrogen content are not adversely affected. We therefore conclude that the fuel testing and inspection program for VANTAGE 5 is acceptable.

## 3.0 NUCLEAR DESIGN

From the viewpoint of the nuclear design the VANTAGE 5 (17 X 17) fuel is for the most part essentially identical to the normal Westinghouse 17 X 17 Optimized Fuel Assembly (OFA). Thus, for the most part nuclear design related parameters and the methods for deriving them are the same as have been reviewed and approved for the OFA fuel (via WCAP-9500, Byron, Catawba and McGuire reviews) and requires no additional attention here. There are, however two principal direct design aspects of VANTAGE-5 different from OFA which directly affect the neutronics (and its review). These are the natural $UO_2$ end blanket ("axial blanket") design and the IFBA. Furthermore, there are additional aspects of the design or proposed operations with the fuel which are somewhat different (from WCAP-9500) and potentially affect the nuclear design review. These will also be discussed. The nuclear design bases for VANTAGE-5 are the same as for OFA, as stated in WCAP-9500, except for those related to extended burnup and a modification of the moderator temperature coefficient (MTC) basis. Thus, except for these modifications, the bases need not be discussed further. WCAP-10444 is primarily addressed to four loop 17 X 17 design, but the features may be applied to other cores and fuel bundle geometries (e.g., 14 X 14 or 15 X 15). This nuclear design review is similarly oriented, but the conclusions would likely also generally apply to other assembly arrays and core sizes.

### 3.1 Principal Nuclear Related Direct Design Changes

The use of natural enrichment $UO_2$ axial blanket zones at the ends of the fuel pins is not new in LWR fuel assembly design. BWRs have used this for some time. It is the first (extensive) use in Westinghouse design, however. The concept is derived largely from cycle economics considerations such as enrichment requirements and uranium utilization (via reduced end leakage), but the primary effect on nuclear aspects of operation is on axial power distribution. With all else the same the axial peaking factor increases as a result of the blankets, although the changes would be a function of burnup. The use of burnable poison with a length less than the initial enriched fuel length can, however, be designed to counteract the effects of the blanket, and

Westinghouse calculations indicate that they can produce distributions such that the peaks over the operating cycle are not significantly greater than normal. While neutronic analyses in this area of power distribution control may need to involve greater use of three dimensional (3D) methodologies, no new or novel problem is introduced. Axial xenon stability would be improved with just axial blanket addition, but is not significantly changed from normal with the combined blanket and part length burnable poison design.

The new burnable poison (IFBA) design involves the use of a boride region on the outer surface of the fuel pellets. This is the first extensive use by Westinghouse of burnable poison integral to the fuel pin rather than as discrete elements in control rod thimbles. It is also the first (extensive) use in LWRs of this type of design, since most previous fuel pin integral designs have used gadolinium mixed throughout the $UO_2$. The IFBA design will not in general have the boride the full length of the active fuel pin. The design length will depend on the specific reload application, but will generally be less than the enriched fuel length in order to improve the axial power distribution. This type of shorter and application-specific length design concept has been used by Westinghouse in the Wet Annular Burnable Absorber (WABA) design, has been approved in the review of that design and has been in use in operating reactors. The IFBA boride region is sufficiently thin neutronically that the flux attenuation through it is relatively small. The flux perturbations produced, therefore, are much less significant than with gadolinium designs which are essentially neutronically black at the effective radius corresponding to unburned gadolinium. Thus the neutronic calculation problems are much less significant for IFBA, and normal lattice methods can be used easily. The VANTAGE-5 design can use IFBA in as many fuel pins in an assembly as is needed to provide the required reactivity and (radial or axial) power distribution effects. Furthermore it does not preclude the use of more standard burnable poison pins in thimbles, although they would not normally be used, thus improving end of cycle parasitic capture and providing more moderation. The IFBA fuel pins can also be used in control rod assemblies where standard burnable poison can not.

The axial blanket and IFBA used together provide an only slightly altered, from standard, neutronic calculational problem. Normal design methods can be used. The standard methods have been compared against higher order transport theory calculations and have been found to be satisfactory for reactivity and power distribution for the blankets and IFBA. Incore measurements have been made of the effects of IFBA and its burnup, and standard methods have provided satisfactory comparison calculations. The new design does require a somewhat greater emphasis on 3D calculations (as does WABA use), but generally the same methodology as described in WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology" can and will be used. The input parameters calculation for the INCORE code calculations used in power distribution measurements in operating reactors will require the use of 3D methods. Experience with this aspect of analysis has been provided in cores using WABA burnable poisons.

Our review of the power distribution and reactivity characteristics of the VANTAGE-5 axial blanket and IFBA neutronic design indicate that no significant adverse characteristic has been introduced compared to the OFA design, and that parameters for cycle analysis and operation can be calculated with standard methods, augmented only with a somewhat greater emphasis on 3D methods, especially for INCORE data. Other parameters of the neutronic design appear not to be significantly affected by these changes. Thus the direct nuclear design changes of the VANTAGE-5 assembly and associated areas are acceptable.

### 3.2 Indirect Neutronics Related Core Changes

There are additional neutronic aspects of the VANTAGE-5 fuel design and proposed or potential operation parameters which are extensions beyond the standard OFA (WCAP-9500) parameters. These include extended burnup, increased design $F_{\Delta H}$, low radial neutron leakage fuel load design and less negative MTC design.

The design basis burnup for the VANTAGE-5 fuel is extended well beyond the 36,000 MWD/MTU basis for the OFA fuel. It is, however, within the range

examined in the extended fuel burnup reviews which have been recently conducted for various fuel vendor designs and in particular for Westinghouse (using the report WCAP-10125 as a base). The review for Westinghouse is expected to be published in the near future. It will conclude that the burnup range including that proposed for VANTAGE-5 fuel and the methodologies for neutronic analyses in that range are acceptable. Neutronically the extended burnup affects the relative U235 and Pu isotopes fission rates, fission product poison productions, end of cycle parameters for transient analyses, relative power in low and high burnup assemblies and related uncertainties in power distribution. However, the nuclear parameter changes from standard to extended burnup ranges are sufficiently small that the methodologies and their uncertainties are not significantly changed. The Pu isotopes increase and to some extent U235 decrease with burnup are at or near saturation for the standard burnup and remain at similar levels in the extended range. The extended kinetics parameters are generally bounded by the standard conservative design selection. The reactivity coefficients used in event analyses for the VANTAGE-5 fueled cores considered a suitable range of extreme cycle lengths and average burnups and are acceptably bounding for the burnups basis. The variation in assembly power, fission products and the changes in fissile isotope ratios are not sufficiently increased to affect the acceptability of the calculational methods or the uncertainty assigned to peaking factors. (It is noted that peak power assemblies are not directly involved in extended burnup ranges.)

The intermediate flow mixer grids for the VANTAGE-5 fuel have no significant direct neutronic effect but there is an indirect effect of permitting a higher design $F_{\Delta H}$ value in order to take advantage of the improved DNB margin from the grids. The increased $F_{\Delta H}$ in turn permits greater design freedom in accounting for, e.g., increased assembly to assembly power differences from increased burnup, or low radial neutron leakage design to improve pressure vessel neutron fluence. This is an acceptable nuclear design process, assuming safety analyses are done with the increased $F_{\Delta H}$ and the Technical Specifications reflect the increase. The transient and accident event analyses in WCAP-10444, using limiting parameters from a transition or equilibrium core, did include an

increased $F_{\Delta H}$. This included the equilibrium core LOCA analysis, which also used an (artificially) increased total peaking, $F_Q$, of 2.55 (compared to the standard 2.32) to indicate maximum $F_Q$ levels before exeeding LOCA 2200°F clad temperature limits for the full VANTAGE-5 core. (Transition core penalties would require a lower $F_Q$.) Other safety evaluations assumed standard axial power distributions and $F_Q$.

Low radial neutron leakage designs are not new and have been and are acceptable. But the increased design $F_{\Delta H}$ capability provides an easier design problem. Similarly the use of a less negative (more positive) MTC is not new. Several Westinghouse reactors have submitted safety analyses with positive MTC used throughout the non-LOCA analyses and have Technical Specifications providing for such a coefficient up to some power level, thus far not including full power. The VANTAGE-5 safety evaluations used a positive MTC for non-LOCA events, but not for the LOCA analyses. Thus the possibility of use for VANTAGE-5 (full and transitional) cores is acceptable with a Technical Specification similar to previously accepted models.

Other nuclear design characteristics and parameters not explicitly discussed here are sufficiently similar to OFA core design (WCAP-9500) that previous reviews for that fuel are applicable. Since VANTAGE-5 fuel will be used in transition cores from STD or OFA loaded cores to full VANTAGE-5 cores, Westinghouse has examined neutronic parameters from all three types and has provided limiting nuclear parameters (Table 3.2 of WCAP-10444) for use in safety analyses which should in general be applicable to transition and full core loadings. (Reload methodology requires that parameters be checked for the specific case.) The review has indicated that these parameters are acceptable for such use. The safety evaluations of WCAP-10444 used these values along with specific values for MTC and peaking factors which are generally loading dependent and are intended to be specific to a given reactor. It is expected that these latter parameters (e.g., MTC, ejected rod worth and peaking, $F_Q$, $F_{\Delta H}$) will be provided and justified for specific reactors.

## 4.0 THERMAL AND HYDRAULIC DESIGN

### 4.1 Core Thermal Hydraulic Analysis

The principal criterion for the thermal hydraulic design of reactor fuel is the avoidance of thermally induced fuel damage during normal operation and anticipated operational occurrences (AOO). To this end, the thermal-hydraulic design basis requires that there must be at least a 95% probability at 95% confidence level that departure from nucleate boiling (DNB) will not occur on the limiting power rod during normal operation and AOOs. This requirement is met by limiting the DNB ratio to the DNBR limit associated with the critical heat flux correlation used. This design basis is the same as that for the OFA fuel. However, since the VANTAGE-5 fuel has three intermediate flow mixer grids in addition to the existing structural grids in the OFA fuel, the thermal hydraulic characteristics have changed and the WRB-1 CHF correlation used in the OFA design is not applicable to VANTAGE-5. Westinghouse has therefore developed the WRB-2 CHF correlation based on CHF tests pertinent to VANTAGE-5 fuel. The DNBR limit for WRB-2 is 1.17, the same as for WRB-1. We have reviewed the WRB-2 correlation and the DNBR limit of 1.17 and found them acceptable. The evaluation of WRB-2 will be addressed in Section 4.2.

The thermal hydraulic calculation is performed using the THINC-IV (Ref. 6) open lattice thermal hydraulic code which takes into account the effects of interchannel cross flow and turbulent mixing. The fluid mechanics design and empirical correlations used in VANTAGE 5 is the same as those for OFA design described in WCAP-9500. The thermal diffusion coefficient (TDC) is used as a measure of the rate of heat exchange by fluid mixing between adjacent channels. The IFM grids used in VANTAGE-5 fuel have the same mixing characteristics as the OFA mixing vane grids. However, the VANTAGE-5 mixing characteristics are improved due to the presence of IFM grids which reduce the grid spacing. For thermal hydraulic analysis, the same TDC used in OFA is used for VANTAGE-5 and is therefore conservative.

Like OFA fuel, the VANTAGE-5 thermal hydraulic analysis uses the improved thermal design procedure (ITDP). The ITDP method is based upon a statistical combination of the effects on DNBR of uncertainties of the plant parameters such as reactor coolant flow rate, core power, core coolant inlet temperatures, system pressure and hot channel factors, etc. The nominal values of these parameters are used in the safety analysis and the effect of the uncertainties on DNBR is added to the 95/95 DNBR limit specified by the WRB-2 correlation to establish a design DNBR limit. The ITDP method described in WCAP-8567 (Ref. 7) has been approved for use in licensing applications subject to the certain restrictions. One of the restrictions requires that if the sensitivity factors are changed as a result of correlation change, then the use of an uncertainty allowance for application of Equation 3-2 (WCAP-8567) must be re-evaluated and the linearity assumption of WCAP-8567 must be validated. Westinghouse in response to a staff question (Ref. 5) has performed the required re-evaluation and validation using the same methods described in the staff safety evaluation report for WCAP-8567. The results show that no additional uncertainty allowance is required based on the typical sensitivity factors specified in Tables 4-4 and 4-5 of WCAP-10444, and that the linearity assumption is indeed conservative. Therefore, the ITDP method described in WCAP-8567 in conjunction with WRB-2 correlation is acceptable.

Tables 4-4 and 4-5 also provide typical values of the means, uncertainties and sensitivity factors of the plant parameters. A typical value of the design DNBR limits of 1.29 and 1.288, respectively, are derived for the typical cell and thimble cell. However, since the uncertainties of some parameters are plant-specific, those plants using ITDP are required to abide by the restrictions imposed on WCAP-8567 and perform an analysis to establish a plant-specific design DNBR limit. In addition, information regarding the measurement uncertainties for pressurizer pressure, power, coolant flow rate and temperature must be provided as specified below.

A block diagram depicting sensor, process equipment, computer and readout devices for each parameter channel used in the uncertainty analysis should be provided. Within each element of the block diagram, the accuracy, drift, range, span, operating limits and setpoints should be identified. The overall accuracy of each channel transmitter to final output and the minimum acceptable accuracy for use with the new procedure should also be identified. In addition, the overall accuracy of the final output value and maximum accuracy requirements for each input channel for this final output device should be identified.

In addition to the design DNBR which must be met in the plant safety analysis to ensure the specified acceptable fuel design limit on DNBR is not violated, a safety analysis DNBR limit is used in the safety analysis. This safety analysis DNBR limit is derived from the design DNBR limit with an additional specific allowance (e.g., 30%). The use of safety analysis DNBR in the thermal-hydraulic design provides certain safety margin which can be credited to compensate for certain required penalties, improved fuel management, or increased plant availability. However, the licensees referencing WCAP-10444 should incorporate in the bases of their plant Technical Specifications the plant-specific safety analysis DNBR limit, the DNBR allowance and the amount of allowance that has been used.

### 4.2 WRB-2 CHF Correlation

In conjunction with the development of the VANTAGE 5 fuel design, Westinghouse developed a new CHF correlation designated WRB-2 to predict DNB performance of this fuel design and other fuel designs which use the same mixing vane design as the 17 X 17 standard fuel mixing vane design. The VANTAGE 5 fuel design features an increase in DNB margin over the 17 X 17 OFA design by the addition of IFM grids. To quantify this increase in DNB performance, a test program was conducted at Columbia University using the VANTAGE 5 fuel design. As described in Appendix A.2.0, "DNB Test Program," of WCAP-10444, the test program consisted of two test series utilizing 5 X 5, 14-foot electrically heated rod

bundles. The detailed bundle geometry is similar to that of 17 X 17 OFA used in earlier DNB tests. A total of 684 data points from 11 test series consisting of the new data, the 5 X 5 OFA test bundle data and that from previous DNB tests involving 5 X 5 test bundles of various lengths but employing the standard fuel mixing vane design, were used as the data base for the development of WRB-2. The WRB-2 correlation is described in Figure A-9 of WCAP-10444 and has a minimum DNBR (95/95) limit of 1.17.

The review of WRB-2 consists of independent thermal hydraulic audit calculations to verify the correctness of the predicted local conditions and CHF predictions, and a statistical analysis to verify the correctness of the DNBR limit.

The local conditions and CHF audit calculations were performed with the COBRA-IV code (Ref. 8). The input data on the test bundle geometry, grid loss coefficients for the various grid designs and subchannel types, flow, heat flux and axial power distributions are obtained from the references provided in WCAP-10444 and the information provided by Westinghouse at staff request. For each data point, the ratio of the measured CHF to the predicted CHF is calculated. All 684 data points used in the WRB-2 development are calculated. Since Westinghouse analysis was performed with the THINC-IV thermal hydraulic code, the results of calculations using COBRA-IV/WRB-2 are not expected to be identical to those presented in WCAP-10444 but provide a general indication of the correctness of Westinghouse analysis. A summary of comparison of the measured to predicted CHF ratios by Westinghouse and the audit calculations is shown in Table 1 of this report which shows very good agreement.

The independent statistical analysis is performed with the measured to predicted CHF ratios from the thermal hydraulic calculation results. The hypothesis that data from all 11 test series come from a single population is tested in two ways. First a chi-square test of homogeneity is performed. The observed value $X^2$ = 111.6, with a degree of freedom of 60 is highly significant. This indicates that the 11 data sets do not all come from the same population. Second, an analysis of variance is performed to test the hypothesis that the

means of all 11 populations are equal. With the among-series degree of freedom of 10 and the within-series degree of freedom of 673, the F-statistic is 4.51. This value is also highly significant, indicating that not all means are equal. On the basis of these analyses, we conclude that the data do not all represent a single population. However, the estimated standard deviation from one-way analysis of variance is 0.0773, while the estimate obtained by treating the data as a single population is 0.0793. This indicates that the error introduced into variance estimates by assuming that the data are all from the same population is negligible.

The 95/95 probability/confidence DNBR limit is calculated using the formula

$$DNBR_{95/95} = \frac{1}{(M/P)_{avg} - KS}$$

where (M/P)avg is the population mean of measured to predicted CHF ratios, S is standard deviation of the M/P data and K is a tolerance multiplier which provides the 95/95 probability/confidence limit.

When the test assembly means are found to be unequal, the "total variance" is estimated by means of the formula

$$S^2_{total} = S^2_w + S^2_A,$$

where $S^2_w$ is the mean square within test series component of variance and $S^2_A$ is the "variance of the test series average." An effective degree of freedom $f_T$ is computed for $S^2_{total}$ from Satterthwaite's formula. The total variance and effective degree of freedom should be used in the DNBR limit calculates as has been done for WRB-1. This is also done for WRB-2. However, the calculation of the variance of the test series means $S^2_A$ was performed by Westinghouse with an assumption of an equal number of data points for each test series. In response to a staff question (Ref. 9), Westinghouse provided a correct formula for calculating $S^2_A$ without the assumption of equal number of data points. It also

shows that the equal data assumption results in higher total variance and fewer effective degrees of freedom. Therefore, the resulting DNBR limit of 1.17 is conservative. We have concluded that WRB-2 with a DNBR limit of 1.17 is acceptable for application to VANTAGE 5 with the following range of applicability.

RANGE OF VARIABLES

| | |
|---|---|
| Pressure | $1440 \leq P \leq 2490$ psia |
| Local Mass Velocity | $0.9 \leq G_{loc}/10^6 \leq 3.7$ lb/ft$^2$hr |
| Local Quality | $-0.1 \leq X_{loc} \leq 0.3$ |
| Heated Length, Inlet to CHF Location | $L_h \leq 14$ feet |
| Grid Spacing | $10 \leq g_{sp} \leq 26$ inches |
| Equivalent Hydraulic Diameter | $0.37 \leq d_e \leq 0.51$ inches |
| Equivalent Heated Hydraulic Diameter | $0.46 \leq d_h \leq 0.59$ inches |

TABLE 1

WRB-2 CHF CORRELATION - STATISTICAL RESULTS

| Rod O.D. (inch) | Length (feet) | Grids (in.) | Heat Flux Profile | Assembly Geometry | Data Points | WESTINGHOUSE M/P Ratio | WESTINGHOUSE Standard Deviation | AUDIT CALCULATIONS M/P Ratio | AUDIT CALCULATIONS Standard Deviation | Reference | Data Table |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | 14 | 10 | Cosine | TYP-5X5 | 51 | 0.9861 | 0.0758 | 0.9721 | 0.0705 | WCAP-10444 | A-2 |
| | 14 | 10 | Cosine | TYP-5X5 | 31 | 1.0097 | 0.0680 | 1.0181 | 0.0623 | WCAP-10444 | A-3 |
| | 14 | 20 | Cosine | TYP-5X5 | 63 | 0.9961 | 0.0946 | 0.9852 | 0.0952 | WCAP-9401 | A-4 |
| | 14 | 20 | Cosine | THM-5X5 | 38 | 0.9832 | 0.0599 | 0.9946 | 0.0600 | WCAP-9401 | A-5 |
| 0.374 | 8 | 22 | Uniform | TYP-5X5 | 67 | 1.0316 | 0.0897 | 1.0348 | 0.0753 | WCAP-8536 | A-6 |
| 0.374 | 14 | 22 | Uniform | TYP-5X5 | 71 | 1.0095 | 0.0664 | 0.9924 | 0.0626 | WCAP-8536 | A-7 |
| 0.374 | 14 | 22 | Cosine | TYP-5X5 | 74 | 0.9893 | 0.0822 | 0.9691 | 0.0773 | WCAP-8536 | A-8 |
| 0.374 | 14 | 22 | Cosine | THM-5X5 | 70 | 0.9884 | 0.0775 | 0.9729 | 0.0765 | WCAP-8536 | A-9 |
| 0.374 | 8 | 26 | Uniform | TYP-5X5 | 78 | 1.0198 | 0.0810 | 0.9890 | 0.0753 | WCAP-8926 | A-10 |
| 0.374 | 8 | 26 | Uniform | THM-5X5 | 68 | 1.0398 | 0.1062 | 1.0092 | 0.0942 | WCAP-8926 | A-11 |
| 0.374 | 14 | 26 | Uniform | TYP-5X5 | 73 | 0.9914 | 0.0823 | 0.9761 | 0.0785 | WCAP-8926 | A-12 |
| | | | All Data | | 684 | 1.0051 | 0.0847 | 0.9907 | 0.0793 | | |

## 4.3 Core Flow Design

WCAP-10444 specifies as a design basis that at least a certain percent of the reactor coolant flow should pass through the fuel rod region of the core for cooling. This design basis assures that for RCS flow rate specified in the plant Technical Specifications the core bypass flow should be accounted for. Core bypass flow is the sum of the flow through rod cluster control guide thimbles, head cooling flow, baffle leakage and leakage to the vessel outlet nozzle. The uncertainties associated with the calculation of bypass flow should be accounted for in the safety analysis. For the analysis using ITDP, the uncertainties are statistically combined with uncertainties of other parameters to obtain the design DNBR limit. For analysis not using ITDP, the uncertainties are accounted for by reducing the active core flow.

The bypass flow is dependent upon core geometry and is plant specific. Higher fuel assembly pressure drop results in larger bypass flow. The presence of the IFM grids in VANTAGE 5 increases the fuel assembly pressure drop as well as bypass flow. Westinghouse indicates that the total bypass flow used in the safety analysis of the reference plant is conservative. However, for licensees referencing WCAP-10444, a plant specific analysis is required to determine the correct bypass flow. Correct or conservative active core flow rate should be used in safety analysis.

## 4.4. Thermohydrodynamic Stability

One of the design bases for VANTAGE-5 is to ensure that the mode of operation with Conditions I and II events will not lead to thermohydrodynamic instability. The flow excursion (Ledinegg) instability generally does not exist in a PWR. This is due to the inherent characteristics of RCS pump head-capacity curve having a negative slope and system hydraulic pressure drop-flow curve having a positive slope in PWR. The staff has previously accepted the stability evaluation for OFA fuel based on the past operating

experience, flow stability experiments and the inherent thermal hydraulic characteristics of Westinghouse open channel core configuration. Use of VANTAGE-5 fuel will not change these characteristics. We therefore conclude that flow instability will not be a problem for VANTAGE 5.

## 4.5 Rod Bow Effect on DNBR

Fuel rod bowing results in flow channel gap closure and reduction of critical heat flux and DNBR. This CHF effect is accounted for through a rod bow DNBR penalty. The method described in WCAP-8691, Revision 1 (Ref. 10), has been approved for the rod bow penalty calculations. A scaling factor dependent on the cladding moment of inertia and span length between grids is used for the channel closure calculation when extrapolating from the channel closure data of different fuel assembly geometry. Since three IFM grids have been installed between the Zircaloy grids in the VANTAGE-5 fuel, the topical report asserts that the grid spacing and, therefore, the channel closure are reduced. In response to a staff question (Ref. 5) on whether the presence of the "non-structural" IFM grids provides the necessary support in the reduction of rod bowing. Westinghouse indicates that the term "non-structural" is a misnomer and that the IFM grids do provide a positive pin type support and have a small stiffening effect near the top of the fuel assembly. Using the approved scaling factor, the predicted channel closure in the limiting span of the VANTAGE 5 fuel is less than 50%. Experimental data (Ref. 11) have shown that there is no CHF reduction for channel closure less than 50%. Therefore no rod bow penalty is required for the VANTAGE 5 fuel.

## 4.6 Transitional Mixed Core DNBR Effect

As a reactor is reloaded with VANTAGE-5 fuel prior to a full core of VANTAGE-5 fuel, there are transitional cycles when both VANTAGE-5 and remaining fuel (such as standard fuel or OFA fuel) will co-exist in the core. The differences between the adjacent fuel assemblies in the hydraulic resistance

characteristics such as spacer grid designs, flow areas and the presence of VANTAGE-5 IFM grids result in local hydraulic mismatches. Such a hydraulic mismatch results in localized flow redistribution due to the open core configuration. While beneficial to STD or OFA fuel due to lower grid resistance, the interbundle cross flow is detrimental to VANTAGE-5 fuel. In the safety analysis, a full core of VANTAGE 5 fuel is assumed. A mixed core DNBR penalty is applied during the transitional mixed core configuration to account for the detrimental effect of interbundle cross flow on the VANTAGE-5 fuel. Westinghouse has performed sensitivity studies in the same way approved previously for the standard fuel/OFA fuel mixed core (Ref. 12) to determine the penalty factor required for the VANTAGE-5 fuel. Since the hydraulic mismatch between VANTAGE 5/OFA assemblies is less complex than that between the VANTAGE-5/STD fuel assemblies, the penalty factor determined from VANTAGE-5/STD mixed core is bounding for VANTAGE-5/OFA mixed core. The value of the penalty factor is described in WCAP-10444 and is later revised to 11% in a response to a staff question (Ref. 5). This value (11%) is generic for 17 X 17 fuel assemblies. For other fuel types such as 14 X 14 or 15 X 15 fuel assemblies, the mixed core penalty factor should be determined separately. As described in Section 4.1, when the ITDP is used in safety analysis, there is a generic margin between the design DNBR limit and plant safety analysis DNBR. This safety margin is enough to compensate for the penalty factor required for mixed core configuration. However, for these plants having the VANTAGE-5 fuel, both the plant specific safety margin and mixed core penalty factor should be addressed in Technical Specification bases.

### 4.7 Low Leakage Loading Core Design

In order to improve nuclear design flexibility and fuel economy, and to reduce pressure vessel fluence, some cores containing VANTAGE 5 fuel will be designed with burned blankets, i.e., the burned fuel assemblies will be placed in the core periphery. To facilitate this, the hot channel enthalpy rise factor $F_{\Delta H}$ is raised from the current 1.55 to a higher value (Westinghouse Proprietary). The increase in $F_{\Delta H}$ is made possible because of the thermal margin gained from

the use of the IFM grids resulting in higher CHF for VANTAGE 5 fuel. For the OFA and STD fuel, thermal margin could be gained from the rod bow penalty reduction as a result of change of rod bow calculation from the NRC interim method (Ref. 13) to the method of WCAP-8691, Revision 1. This thermal margin gain from rod bow penalty reduction could be used to increase the normal operation $F_{\Delta H}$. However, plant specific analyses are required with appropriate $F_{\Delta H}$ to ensure that the specified acceptable fuel design limits will not be violated during normal operation and anticipated operational occurrences.

### 4.8 Other VANTAGE 5 Design Feature Effects

Two VANTAGE-5 design features, namely, axial blanket and IFBA, result in axial zoning in the fuel rod which perturbs core axial power distribution and affects the DNBR evaluation. The axial blanket reduces power at the ends of the rod and causes a slight increase in axial power peaking. Though IFBA flattens the axial power distribution and reduces the local peaking, the IFBA burns out during irradiation. Therefore, the net result of axial blanket and IFBA is a slight increase in power peaking during core operation. Westinghouse has examined this effect and found that the design axial power distribution used in the determination of the overtemperature $\Delta T$ setpoint is still conservative.

Another VANTAGE 5 design feature is the reconstitutable top and bottom nozzles which results in changes in hydraulic resistance at the inlet and outlet of the active fuel region from the previous standard and OFA designs. However, the hydraulic tests performed by Westinghouse have shown that the changes in nozzle design have a negligible impact on the inlet flow and outlet pressure distributions. Therefore, the effect on DNBR is negligible.

## 5.0 SAFETY EVALUATION

A safety evaluation is necessary to assess the impact of the VANTAGE 5 design features such as axial blankets, IFBA, IFM grid and reconstitutable top nozzle, etc., and the core reload strategy having a higher $F_{\Delta H}$ and a positive moderator

coefficient. Table 5-1 of WCAP-10444 lists the non-LOCA accidents which are most sensitive to these changes. They are:

(a) Main Steamline Rupture;
(b) Loss of Load/Turbine Trip;
(c) Complete Loss of Forced Reactor Coolant Flow;
(d) Single Reactor Coolant Pump Locked Rotor;
(e) Uncontrolled RCCA Bank Withdrawal at Power; and
(f) RCCA Ejection.

As an evaluation of the impact of the VANTAGE 5 design features, analyses of these accidents and loss of coolant accident (LOCA) are performed using the methods described in WCAP-9272, "Westinghouse Reload Safety Analysis Methodology." The analyses are performed with a typical 4-loop reference plant. All analytical procedures and computer codes used in the non-LOCA safety analysis are the same as those described in WCAP-9500 for OFA, except that the WRB-2 CHF correlation is used for VANTAGE 5. For most accidents which are DNB limited, the improved thermal design procedure is used in the analyses. Table 5-3 of WCAP-10444 summarizes the initial condition and computer codes used in the non-LOCA accident analyses and also shows which accidents employ a DNB analysis using ITDP. For the large break LOCA, the analysis is performed with a modified version of the 1981 Westinghouse ECCS evaluation model (Ref. 14). The modified version of ECCS evaluation model uses the BART code (Ref. 15) to calculated the reflood heat transfer coefficient in place of the FLECHT reflood heat transfer correlation normally used in the LOCTA code (Ref. 16). BART is a mechanistic core heat transfer code. The code with the grid rewet model has been approved for LOCA analysis. However, the drop break up model is not approved. In response to a staff question (Ref. 5), Westinghouse confirms that in every application of BART supporting WCAP-10444, a version of the code was used that explicitly excluded the drop breakup model.

For a transitional mixed core where the VANTAGE 5 fuel co-exists with either STD or OFA fuel, the local hydraulic resistance mismatch between the adjacent

assemblies of different design has a detrimental effect on VANTAGE 5. The flow redistribution due to hydraulic resistance mismatch results in a reduction of reflood steam flow rate and a increase in peak cladding temperature in the VANTAGE 5 fuel . For a transitional core, the LOCA analysis is performed with a full core of VANTAGE 5 fuel. The resulting PCT is then increased by adding a mixed core penalty. This approach has been reviewed and approved during the review of WCAP-9500. Westinghouse has performed the same analysis for VANTAGE 5 and the conservative nature of the required mixed core penalty on PCT was determined for VANTAGE 5. We conclude that this approach is acceptable.

The results of analyses as presented in WCAP-10444 generally demonstrate that the SAFDLs are not violated and the criteria specified in 10 CFR 50.46 can be met. Since the safety analyses of WCAP-10444 are not intended to be directly applicable to specific plants but were submitted for demonstrating acceptability of the fuel in a generic sense, we find the safety analyses presented in WCAP-10444 acceptable. The licensees referencing WCAP-10444 are required to perform plant-specific safety analyses to ensure the safety criteria are met.

There are, however, several deficiencies in the safety analyses of WCAP-10444. These are discussed below and should be corrected in the plant-specific safety analyses.

(1) For the steam system piping failure accident, the analysis assumes that the offsite power is available. This assumption is not in compliance with the SRP which requires that assumptions as to the loss of offsite power and the time of loss should be made to study their effects on the consequence of the accident. This requirement is also specified in General Design Criterion 17, 10 CFR 50 Appendix A.

(2) With regard to the reactor coolant pump seizure (locked rotor) accident, WCAP-10444 states that although some rods may enter DNB for a short period

of time, fuel failure for the locked rotor/shaft seizure incident is treated on a mechanistic basis, thus eliminating the need to automatically equate DNB with fuel failure. WCAP-10444 also indicates that the mechanistic methods are allowed by SRP and that the temperature-time criteria of NUREG-0562 (Ref. 17) will be used in the fuel failure calculation on a plant specific basis. However, NUREG-0562 prepared by the Office of Nuclear Reactor Research, has not been applied in licensing. The temperature-time criteria noted in the report are from another reference (Pawel, Journal of Nuclear Materials, Volume 50, Page 247-258) which the NRC has not endorsed for this type of application.

It has been and continues to be the NRC position that cladding failure is assumed to occur when the fuel rod DNBR is less than the safety limit. This position provides a degree of conservatism which varies with each anticipated operational occurrence and accident. This conservatism covers such analyses uncertainties as:

(i) Variation in history of an actual event from that idealized in the accident analysis

(ii) Uncertainties in the thermal hydraulics such as

- (a) prediction of heat transfer coefficient in film boiling
- (b) prediction of subchannel flow in film boiling
- (c) prediction of extent and location of DNB
- (d) multiple DNB on the same rod
- (e) changes in fuel bundle geometry during event due to overheating
- (f) uncertainties in prediction of transients, non-isothermal zirconium/water reaction kinetics, etc., (see the above reference by Pawel).

In addition to these, there are questions related to the prototypicality of the NUREG-0562 data. EPRI report NP 1999 which addresses post-DNB effects in PWRs has identified several areas which would be appropriate subjects for future research before application to licensing (see Table 5-1 of this report). While we are not taking the position that all items in Table 5-1 must be addressed, it is clear from this industry report that more review would be necessary before approval of a post-DNB fuel failure criterion.

Therefore, the licensees referencing WCAP-10444 should assume that all fuel which experiences a DNBR of less than the DNBR limit fails and calculate the offsite dose consequences. In the offsite dose analysis the licensee should assume maximum Technical Specification pre-accident coolant activity and steam generator leakage. Single failures should be considered including a stuck open secondary relief valve. Loss of offsite power should be assumed per GDC-17. The effect of steam generator tube uncovery on the offsite dose consequences caused by operator action to isolate the affected steam generator should also be considered in the analysis.

(3) The large break LOCA analysis performed for a full core of VANTAGE 5 fuel using a total peaking factor $F_Q$ of 2.55 has shown a peak cladding temperature of 2194°F which is below the acceptance criterion of 2200°F. However, for a transitional mixed core configuration, addition of the mixed core penalty on PCT will result in the PCT exceeding the 2200°F acceptance criterion per 10 CFR 50.46. The plant specific analysis must be done to ensure that with appropriate value of $F_Q$, the 2200°F criterion can be met even for the transitional mixed core.

## 6.0 SUMMARY AND CONCLUSION

The staff has reviewed the topical report WCAP-10444 and concludes that the report is acceptable for reference for the Westinghouse VANTAGE 5 fuel design. This acceptability is subject to the following conditions:

1. The statistical convolution method described in WCAP-10125 for the evaluation of initial fuel rod to nozzle growth gap has not been approved. This method should not be used in VANTAGE 5.

2. For each plant application, it must be demonstrated that the LOCA/seismic loads considered in WCAP-9401 bound the plant in question; otherwise additional analysis will be required to demonstrate the fuel assembly structural integrity.

3 An irradiation demonstration program should be performed to provide early confirmation performance data for the VANTAGE 5 design.

4. For those plants using the ITDP, the restrictions enumerated in Section 4.1 of this report must be addressed and information regarding measurement uncertainties must be provided.

5. The WRB-2 correlation with a DNBR limit of 1.17 is acceptable for application to 17 X 17 VANTAGE 5 fuel. Additional data and analysis are required when applied to 14 X 14 or 15 X 15 fuel with an appropriate DNBR limit. The applicability range of WRB-2 is specified in Section 4.2.

6. For 14 X 14 and 15 X 15 VANTAGE 5 fuel designs, separate analyses will be required to determine a transitional mixed core penalty. The mixed core penalty and plant specific safety margin to compensate for the penalty should be addressed in the plant Technical Specifications Bases.

7. Plant specific analysis should be performed to show that the DNBR limit will not be violated with the higher value of $F_{\Delta H}$.

8. The plant-specific safety analysis for the steam system piping failure event should be performed with the assumption of loss of offsite power if that is the most conservative case.

9. With regard to the RCS pump shaft seizure accident, the fuel failure criterion should be the 95/95 DNBR limit. The mechanistic method mentioned in WCAP-10444 is not acceptable.

10. If a positive MTC is intended for VANTAGE 5, the same positive MTC consistent with the plant Technical Specifications should be used in the plant specific safety analysis.

11. The LOCA analysis performed for the reference plant with higher $F_Q$ of 2.55 has shown that the PCT limit of 2200°F is violated during transitional mixed core configuration. Plant specific LOCA analysis must be done to show that with the appropriate value of $F_Q$, the 2200°F criterion can be met during use of transitional mixed core.

12. Our SER on Westinghouse's extended burnup topical report WCAP-10125 is not yet complete; the approval of the VANTAGE 5 design for operation to extended burnup levels is contingent on NRC approval of WCAP-10125. However, VANTAGE 5 fuel may be used to those burnups to which Westinghouse fuel is presently operating. Our review of the Westinghouse extended burnup topical report has not identified any safety issues with operation to the burnup value given in the extended burnup report.

13. Recently, a vibration problem has been reported in a French reactor having 14-foot fuel assemblies; vibration below the fuel assemblies in the lower portion of the reactor vessel is damaging the movable incore instrumentation probe thimbles. The staff is currently evaluating the implications of this problem to other cores having 14 foot long fuel bundle assemblies. Any limitations to the 14-foot core design resulting from the staff evaluation must be addressed in plant-specific evaluations.

## 7.0 REFERENCES


1. WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel," dated July 1982.

2. WCAP-9500-A, "Reference Core Report 17 X 17 Optimized Fuel Assembly," dated May 1982.

3. Letter from E. P. Rahe, Jr. (Westinghouse) to C. O. Thomas (NRC), "Reference to NRC Request Number 3 on WCAP-10125," NS-NRC-85-3018, dated March 14, 1985.

4. WCAP-9401-P-A, "Reference Core Report 17 X 17 Optimized Fuel Assembly," dated May 1982.

5. Letter from E. P. Rahe, Jr. to C. O. Thomas, "Response to Request Number 1 for Additional Information on WCAP-10444 entitled, 'VANTAGE-5 Fuel Assembly'" (Proprietary), NS-NRC-85-3014, dated March 1, 1985.

6. WCAP-7956, "THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," 1973.

7. WCAP-8567, "Improved Thermal Design Procedure," dated July 1975.

8. BNWL-1962 - UC-32, "COBRA-IV-1: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores," Battelle Pacific Northwest Laboratories, dated March 1976.

9. Letter from E. P. Rahe, Jr. (Westinghouse) to C. O. Thomas (NRC), "Follow-up Response to Request Number 1 for Additional Information for WCAP-10444 entitled, 'VANTAGE 5 Fuel Assembly - Reference Core Report'," (Proprietary), NS-NRC-85-3033, dated May 8, 1985.

10. WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," dated July 1979.

11. E. S. Markowski, et al., "Effect of Rod Bowing on CHF in PWR Fuel Assemblies," ASME Peper 77-HT-91.

12. Letter from E. P. Rahe, Jr. (Westinghouse) to J. R. Miller (NRC), "Supplement to WCAP-9500 and WCAP-9401/9402 NRC Safety Evaluation Report (SER) Mixed Core Compatibility Items - Supplemental Information," NS-EPR-2643, dated August 17, 1982.

13. Memorandum from D. F. Ross and D. G. Eisenhut to D. B. Vassallo and K. R. Goller, "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," dated February 16, 1977.

14. WCAP-9220-A, Revision 1, "Westinghouse ECCS Evaluation Model, 1981 Version," dated February 1981.

15. WCAP-9561, "BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients," dated January 1980.

16. WCAP-8301, "LOCTA-IV Program: Loss of Coolant Transient Analyses," dated June 1974.

17. WCAP-8170, "Calculated Model for Core Reflood After a Loss of Coolant Accident (Westinghouse Reflood Code)," dated June 1974.

18. R. Van Houten, "Fuel Rod Failure As a Consequence of Department from Nucleate Boiling or Dryout," NUREG-0562, 1979.

Mr. E. P. Rahe, Jr., Manager
Nuclear Safety Department
Westinghouse Electric Corporation
Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT WCAP-10444, "VANTAGE 5 FUEL ASSEMBLY"

We have completed our review of the subject topical report submitted by Westinghouse Electric Corporation (Westinghouse) letter dated December 30, 1983. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that Westinghouse publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, Westinghouse and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Cecil O. Thomas, Chief
Standardization and Special
Projects Branch
Division of Licensing

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ENCLOSURE

SER ON WCAP-10444

## 1.0 INTRODUCTION

By letter dated December 30, 1983, Westinghouse requested NRC to review the topical report WCAP-10444, "Westinghouse Reference Core Report, VANTAGE 5 Fuel Assembly."

The VANTAGE 5 fuel assembly is a modification of the design of the Standard and Optimized 17 X 17 fuel assemblies. (It will also be offered in 14 X 14 and 15 X 15 configurations.) The new features of the VANTAGE 5 fuel assembly consist of (1) axial blankets for improved neutron utilization, (2) Integral Fuel Burnable Absorber (IFBA), (3) Intermediate Flow Mixer (IFM) grids, (4) Reconstitutable top and bottom nozzles, and (5) extended burnup. The impact of these design changes is evaluated from the standpoints of mechanical, nuclear and thermal hydraulic designs and transient and accident analyses. The analysis is performed with the 17 X 17 fuel assembly array for a 4-loop reference plant. The 14 X 14 and 15 X 15 application should be evaluated on a plant-specific basis.

Because our SER on Westinghouse's extended burnup topical report WCAP-10125 (Ref. 1) is not complete, the design bases/criteria and analysis methods review in this submittal have not been approved for application to extended burnup levels. Consequently, the approval of the VANTAGE 5 design for operation to extended burnup levels is contingent on NRC approval of Westinghouse's extended burnup topical report. It should be noted, however, that the review of WCAP-10125 is complete and there are no outstanding issues at this time. Our review, documented in this safety evaluation report, covers the VANTAGE 5 fuel designs to those burnups to which PWRs with Westinghouse fuel are presently operating.

## 2.0 MECHANICAL DESIGN

### 2.1 Fuel Assembly Design

WCAP-10444 indicates that the design bases and design limits for the VANTAGE-5 fuel assembly are essentially the same as those for the optimized fuel assembly (OFA) design described in WCAP-9500 (Ref. 2). We have therefore relied heavily on our review of Reference 2 and only the differences in design bases will be discussed in the following evaluation.

#### 2.1.1 Fuel Rod Growth Gap

WCAP-10444 states that the design basis for the axial clearance between core plates and nozzle end plates should allow sufficient margin for fuel assembly and fuel rod irradiation growth to design burnup established in WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel" (Ref. 1). WCAP-10125 indicates that the initial fuel rod-to-nozzle growth gap is designed based on a statistical convolution of the distribution of the measured gap data as a function of fluence and the distribution of rod average fluences for an assembly. This statistical treatment of uncertainties associated with the fuel-rod/assembly irradiation growth effect is a deviation from the approved method. In response to a staff question (Ref. 3), however, Westinghouse indicates that the current approved design criterion has been used for the VANTAGE-5 design, i.e., the minimum gap is the required allowance for irradiation growth established to preclude fuel rod-to-nozzle interface during projected operation on the basis of the assumption of worst case fuel rod and fuel assembly growth combined with worst case fabrication tolerances. Therefore, the design for fuel rod growth gap for VANTAGE-5 is the same as that for the OFA design and is therefore acceptable.

For the VANTAGE-5 fuel assembly design, the overall assembly length has been adjusted to accommodate assembly growth and the distance between the top and bottom nozzle plates has been adjusted to accommodate the expected

increased fuel growth related to extended burnup operation. The growth predictions were based on Westinghouse incore experience including results from the high burnup demonstration program. Therefore, adequate shoulder gap is provided to accommodate rod growth.

### 2.1.2 Fuel Fretting Wear

The Westinghouse design guideline for VANTAGE 5 regarding fuel rod fretting wear is the same as that for OFA by limiting the fuel rod wear due to contact with the spacer grids to less than a certain percentage (proprietary) of clad thickness. That the VANTAGE 5 design meets this guideline is confirmed via hydraulic testing. The tests were performed in the Westinghouse Fuel Assembly Test System (FATS) with a VANTAGE 5 assembly adjacent to a 17 X 17 OFA assembly, adjacent to a 17 X 17 STD fuel assembly, and adjacent to another VANTAGE 5 assembly. The results of these tests confirm that the projected fuel rod wear of VANTAGE 5 is well within the design guideline. There was no indication of adverse fretting wear of the fuel rods by the standard structure or the IFM grids.

### 2.1.3 Fuel Assembly Structural Integrity

Section 2.2.3.3 of WCAP-10444 provides an evaluation of the VANTAGE 5 fuel assembly structural response to the externally applied forces such as LOCA and seismic forces. Appendix A.3.0 of the report describes the mechanical test program performed to confirm the structural integrity of the fuel assembly. The mechanical tests consist of lateral vibration and impact tests, and axial stiffness and impact tests. The results show that the VANTAGE 5 design changes represent a small improvement over the OFA design with respect to the structural dynamic response. This is attributable to the IFM grids providing a small stiffness effect near the top of the fuel assembly. The IFM grids also provide additional load sharing during Seismic/LOCA impact.

Analytical evaluations of the fuel assembly response to the most limiting LOCA and seismic forces are performed with the time history numerical technique in the same manner as the approved method described in WCAP-9401 (Ref. 4). The resulting LOCA and seismic induced grid impact forces are combined in accordance with the method specified in Section 4.2 of SRP. The results show that the maximum combined grid impact force for both the Zircaloy structural grids and IFM grids are within the allowable limits. The maximum fuel assembly deflection and the stress resulting from the deflection indicated substantial safety margins compared to the allowable value. Therefore, we conclude that structural damage from external forces is not a concern. However, as discussed in the staff safety evaluation report for WCAP-9401, for each plant application, it must be demonstrated that the applied seismic/LOCA loads considered in WCAP-9401 bound the plant in question or else additional analysis will be required.

### 2.1.4 Fuel Assembly Shipping and Handling Loads

The design requirement for shipping and handling loads for the VANTAGE 5 fuel assembly is reduced from 6g to 4g. This change allows for reduction in the thickness of nozzle end plates and therefore increases the axial space available for fuel assembly and rod design. Westinghouse has tested the handling acceleration at both the manufacturing facility and reactor sites, and determined that handling acceleration is well below the 4g limit. Westinghouse has also performed extensive over-the-road tests with shipping containers containing dummy fuel assemblies and found that insignificant g loads were communicated to the fuel assembly carriage in the container. At Columbia University, tests were also performed with a dummy assembly under the most accelerated crane movement possible with abrupt stops. Results of repeated tests confirm the maximum axial load to be far less than 4g. Therefore, based on the results of these tests, the 4g design criterion is acceptable.

### 2.1.5 Fuel Assembly Structural Components

The design bases for the VANTAGE 5 fuel assembly structural components are essentially the same as those described in WCAP-9500 for OFA except for the 4g structural requirement. However, there are a few changes in the VANTAGE 5 component designs. The bottom nozzle uses Inconel 718 instead of Type 304 stainless steel to permit the use of a thinner nozzle plate. The top nozzle has a groove in each thimble through-hole in the nozzle plate to facilitate removal and has a reduced nozzle plate thickness to provide additional axial space. The hold-down spring is a 4-leaf spring design instead of the standard 3-leaf design in order to compensate for the increase in pressure drop due to the IFM grids.

For both the top and bottom nozzles, Westinghouse verified through a finite element analysis that the maximum stresses at 4g's do not exceed the allowable limits. They are also tested to demonstrate that the strength exceeds the structural 4g functional requirement.

The 4-leaf hold-down spring has been hydraulically tested in the Westinghouse FATS facility to determine the assembly lift off flow rate and the assembly pressure loss coefficient. The results of the flow testing serve as the basis for optimizing the spring force. In response to a staff question (Ref. 5), Westinghouse indicated that the effects of spring relaxation and assembly growth are also accounted for in the spring design. The spring has been designed with a 10% higher BOL load than required by the hydraulic lift force to account for the net effect of the assembly growth and spring relaxation.

There are no changes involved in the guide thimbles, instrumentation tubes and the structural spacer grids from OFA design. The IFM grids are non-structural members whose primary function is to promote mid-span flow mixing. Therefore, the design bases for the IFM grids are to avoid cladding wear and interactive damage with the grids of the neighboring assemblies during fuel handling. The impact tests described previously have also been performed for the IFM grids.

The results show that the crush strength of the IFM grids exceeds the expected dynamic load during a seismic/LOCA event. Therefore, a coolable geometry is assured.

### 2.1.6 Irradiation Demonstration Program

The report indicates that an irradiation demonstration program will be performed to provide early confirmation performance data for the VANTAGE-5 design. This program will be performed consistent with the product demonstration effort employed for the introduction of the 17 X 17 STD fuel and OFA fuel design. We find this commitment acceptable.

## 2.2 Fuel Rod Design

The VANTAGE 5 fuel rod design is identical to the OFA design with the exceptions that (1) axial blanket pellets replace the top and bottom parts of the fuel stacks; (2) thin boride coating is added to the fuel pellets in the middle portion of the fuel rod to serve as integral fuel burnable absorber (IFBA), (3) an additional plenum space is included to accommodate gas release from the fuel and the thermal expansion differential between the cladding and fuel; and (4) a tapered bottom end plug is used to guide the fuel rod during top-end reconstitution. In addition, VANTAGE 5 is designed for a higher burnup and higher enthalpy rise factor ($F_{\Delta H}$) operation. However, the design bases for VANTAGE 5 are the same as OFA in that fuel rod damage should not occur due to excessive rod internal pressure, clad stress, clad strain, clad temperature, fuel temperature or clad strain fatigue, and that the fuel rod should not fail due to fretting wear on the outer cladding surface or by clad flattening. The analyses showing that these design bases are met were performed using the same methods as were used for OFA and the STD designs.

(a) Rod Internal Pressure - Since VANTAGE 5 fuel rods contain IFBA, additional helium gas is created and released from the depleted boride contained in the burnable boride coating. However, a Westinghouse evaluation has shown that the internal pressure of VANTAGE 5 meets the design criterion.

(b) Clad Strain and Stress - Design evaluations show tht the clad stresses and strains for both IFBA and non-IFBA fuel rods of VANTAGE 5 design meet the stress and strain design limits.

(c) Clad Temperature - In order to limit metal oxide formation to acceptable values, specific design limits at the metal/oxide interface temperature are specified for both condition I and II transients. Westinghouse calculations have shown that the clad temperature of VANTAGE 5 fuel rod meets the design limits.

(d) Fuel Temperature - To avoid $UO_2$ centerline melting during condition I and II transients, the fuel centerline temperature is limited to 4700°F. A Westinghouse design evaluation has shown that the fuel centerline temperature remains below this limit.

(e) Clad Fatigue - As a design basis, Westinghouse specified a value (proprietary) of a fatigue life usage factor limit. The design evaluations have shown that the cumulative fatigue life usage factor for the VANTAGE 5 cladding is far less than the specified limit value.

(f) Clad Wear - Westinghouse has a value (proprietary) of wall thickness as a general guideline in evaluating cladding imperfections including fretting wear. Since the structural grids and fuel tube for both VANTAGE 5 and OFA are identical with respect to grid cell forces acting on the fuel rods, the amount of clad wear for both fuel designs should be the same. As part of the hydraulic test performed at the FATS facility, the wear test results have verified that the VANTAGE 5 fuel wear characteristics are similar to that of the 17 X 17 OFA. Therefore, the conclusion can be

drawn that the VANTAGE 5 fuel rod wear will not exceed the maximum wear depth limit.

(g) Clad Flattening - If axial gaps in the fuel pellet columns were to occur due to densification, the cladding would have the potential of collapsing into a gap, i.e., clad flattening. Because of the large local strains that would result from collapse, the cladding is assumed to fail. Westinghouse calculations for VANTAGE 5 show that the predicted clad flattening time exceeds residence times expected for the extended burnup fuel management. Typical values of clad flattening times are in excess of 45000 effective full power hours.

## 2.3 Integral Fuel Burnable Absorber (IFBA)

Some VANTAGE 5 fuel pellets have a thin layer of boride coating on the surface to act as integral fuel burnable absorber. The design bases are the same for both IFBA and non-IFBA fuels. Evaluations are made to determine if the IFBA rods meet the design bases.

(a) Performance Test - The performance of the IFBA rods has been demonstrated with test fuel rods having various thickness of boride coating irradiated in both a test reactor (BR-3) and a commercial reactor (Turkey Point Unit 3). Post Irradiation Examinations (PIE) of these test rods irradiated up to 13000 MWD/MTU indicate that the boride coating performs well with no apparent loss of coating integrity. The monitoring of the reactivity and depletion characteristics shows that the absorber behaves as predicted. The discharged fuel assembly shows no leaking rods in sipping tests. Additional tests are planned to have IFBA rods inserted in the same reactor to provide additional confirmation performance data for the IFBA feature.

(b) Clad Hydriding - Hydriding as a cladding failure mechanism is precluded by controlling the level of moisture and other hydrogenous impurities during fabrication. The IFBA-containing fuel is required to meet the same hydrogen, fluorine and impurity limits set for the $UO_2$ fuel with the exception of the coating constituent. Clad hydrogen analysis of the irradiated test rods from BR-3 reactor has shown no difference in the cladding hydrogen absorption characteristics between the areas adjacent to both coated and non-coated fuel pellets. Based on PIE results and thermodynamics consideration, Westinghouse concludes that no chemical reaction between boride coated pellet and the Zircaloy-4 cladding during reactor life is expected. In addition, the clad is also protected by $ZrO_2$ which is also thermodynamically stable with respect to the boride coating.

(c) Clad Breaching - Though the design basis for VANTAGE 5 fuel is to maintain cladding integrity, the event of a sufficiently large Zircaloy cladding breaching is considered which would release fission products from fuel rod and would also likely result in the boride coating leached out by the coolant water. Since the boride loading is small in any individual rod and the IFBA rods are distributed across the core, Westinghouse states that the change in peaking factor would be small even if the clad breach and absorber loss occur early in life.

(d) Fuel Rod Waterside Corrosion - The IFBA coating on the fuel pellets has no effect on clad/oxide interface temperature or waterside corrosion compared to non-IFBA rods. The waterside corrosion criterion for the non-IFBA rod remains applicable for the IFBA rod.

(e) IFBA Coating -Fuel Compatibility - Westinghouse has evaluated the compatibility of IFBA coating and $UO_2$ fuel. Thermodynamic evaluations show no adverse chemical reaction occur between the boride coating and the $UO_2$ fuel. This is confirmed by electron microprobe analysis showing no adverse diffusion reaction between the boride coating and $UO_2$ when heated to 1750°C for 8 hours or 1600°C for 24 hours. PIE test results

from the BR-3 test reactor also confirm the compatibility of the absorber coating and $UO_2$ under operating conditions. Although a reaction has been observed due to excess oxygen and boride depletion, Westinghouse has found this to have no adverse impact on either the fuel or absorber performance.

(f) Boride Coating Axial Distribution - The boride coating is required to effectively remain in place throughout its functional lifetime to ensure no significant loss of neutron absorber uniformity. Thin coating integrity is confirmed by the PIE results of test rods from the BR-3 test reactor. A direct measurement of neutron absorption in a commercial reactor having boride coated pellets also confirms no absorber redistribution after one full cycle. Therefore, mechanical integrity of the absorber coating is reasonably assured.

## 2.4. Testing and Inspection Plans

As required by SRP Section 4.2, testing and inspection plans for new fuel should include verification of significant fuel design parameters. WCAP-10444 provides a brief discussion of the Westinghouse quality assurance program which provides for control over all activities affecting product quality including design, development, fabrication, testing and inspection, etc. The quality control program described in WCAP-9500 for OFA fuel is also applicable to VANTAGE-5 fuel. WCAP-10444 indicates that manufacturing control equipment for full scale production of VANTAGE 5 has not yet been finally designed. However, the equipment and process will be fully qualified by quality control to ensure that enrichment and stack length controls consistently meet design requirements, and that all other specification requirements such as moisture and hydrogen content are not adversely affected. We therefore conclude that the fuel testing and inspection program for VANTAGE 5 is acceptable.

## 3.0 NUCLEAR DESIGN

From the viewpoint of the nuclear design the VANTAGE 5 (17 X 17) fuel is for the most part essentially identical to the normal Westinghouse 17 X 17 Optimized Fuel Assembly (OFA). Thus, for the most part nuclear design related parameters and the methods for deriving them are the same as have been reviewed and approved for the OFA fuel (via WCAP-9500, Byron, Catawba and McGuire reviews) and requires no additional attention here. There are, however two principal direct design aspects of VANTAGE-5 different from OFA which directly affect the neutronics (and its review). These are the natural $UO_2$ end blanket ("axial blanket") design and the IFBA. Furthermore, there are additional aspects of the design or proposed operations with the fuel which are somewhat different (from WCAP-9500) and potentially affect the nuclear design review. These will also be discussed. The nuclear design bases for VANTAGE-5 are the same as for OFA, as stated in WCAP-9500, except for those related to extended burnup and a modification of the moderator temperature coefficient (MTC) basis. Thus, except for these modifications, the bases need not be discussed further. WCAP-10444 is primarily addressed to four loop 17 X 17 design, but the features may be applied to other cores and fuel bundle geometries (e.g., 14 X 14 or 15 X 15). This nuclear design review is similarly oriented, but the conclusions would likely also generally apply to other assembly arrays and core sizes.

### 3.1 Principal Nuclear Related Direct Design Changes

The use of natural enrichment $UO_2$ axial blanket zones at the ends of the fuel pins is not new in LWR fuel assembly design. BWRs have used this for some time. It is the first (extensive) use in Westinghouse design, however. The concept is derived largely from cycle economics considerations such as enrichment requirements and uranium utilization (via reduced end leakage), but the primary effect on nuclear aspects of operation is on axial power distribution. With all else the same the axial peaking factor increases as a result of the blankets, although the changes would be a function of burnup. The use of burnable poison with a length less than the initial enriched fuel length can, however, be designed to counteract the effects of the blanket, and

Westinghouse calculations indicate that they can produce distributions such that the peaks over the operating cycle are not significantly greater than normal. While neutronic analyses in this area of power distribution control may need to involve greater use of three dimensional (3D) methodologies, no new or novel problem is introduced. Axial xenon stability would be improved with just axial blanket addition, but is not significantly changed from normal with the combined blanket and part length burnable poison design.

The new burnable poison (IFBA) design involves the use of a boride region on the outer surface of the fuel pellets. This is the first extensive use by Westinghouse of burnable poison integral to the fuel pin rather than as discrete elements in control rod thimbles. It is also the first (extensive) use in LWRs of this type of design, since most previous fuel pin integral designs have used gadolinium mixed throughout the $UO_2$. The IFBA design will not in general have the boride the full length of the active fuel pin. The design length will depend on the specific reload application, but will generally be less than the enriched fuel length in order to improve the axial power distribution. This type of shorter and application-specific length design concept has been used by Westinghouse in the Wet Annular Burnable Absorber (WABA) design, has been approved in the review of that design and has been in use in operating reactors. The IFBA boride region is sufficiently thin neutronically that the flux attenuation through it is relatively small. The flux perturbations produced, therefore, are much less significant than with gadolinium designs which are essentially neutronically black at the effective radius corresponding to unburned gadolinium. Thus the neutronic calculation problems are much less significant for IFBA, and normal lattice methods can be used easily. The VANTAGE-5 design can use IFBA in as many fuel pins in an assembly as is needed to provide the required reactivity and (radial or axial) power distribution effects. Furthermore it does not preclude the use of more standard burnable poison pins in thimbles, although they would not normally be used, thus improving end of cycle parasitic capture and providing more moderation. The IFBA fuel pins can also be used in control rod assemblies where standard burnable poison can not.

The axial blanket and IFBA used together provide an only slightly altered, from standard, neutronic calculational problem. Normal design methods can be used. The standard methods have been compared against higher order transport theory calculations and have been found to be satisfactory for reactivity and power distribution for the blankets and IFBA. Incore measurements have been made of the effects of IFBA and its burnup, and standard methods have provided satisfactory comparison calculations. The new design does require a somewhat greater emphasis on 3D calculations (as does WABA use), but generally the same methodology as described in WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology" can and will be used. The input parameters calculation for the INCORE code calculations used in power distribution measurements in operating reactors will require the use of 3D methods. Experience with this aspect of analysis has been provided in cores using WABA burnable poisons.

Our review of the power distribution and reactivity characteristics of the VANTAGE-5 axial blanket and IFBA neutronic design indicate that no significant adverse characteristic has been introduced compared to the OFA design, and that parameters for cycle analysis and operation can be calculated with standard methods, augmented only with a somewhat greater emphasis on 3D methods, especially for INCORE data. Other parameters of the neutronic design appear not to be significantly affected by these changes. Thus the direct nuclear design changes of the VANTAGE-5 assembly and associated areas are acceptable.

### 3.2 Indirect Neutronics Related Core Changes

There are additional neutronic aspects of the VANTAGE-5 fuel design and proposed or potential operation parameters which are extensions beyond the standard OFA (WCAP-9500) parameters. These include extended burnup, increased design $F_{\Delta H}$, low radial neutron leakage fuel load design and less negative MTC design.

The design basis burnup for the VANTAGE-5 fuel is extended well beyond the 36,000 MWD/MTU basis for the OFA fuel. It is, however, within the range

examined in the extended fuel burnup reviews which have been recently conducted for various fuel vendor designs and in particular for Westinghouse (using the report WCAP-10125 as a base). The review for Westinghouse is expected to be published in the near future. It will conclude that the burnup range including that proposed for VANTAGE-5 fuel and the methodologies for neutronic analyses in that range are acceptable. Neutronically the extended burnup affects the relative U235 and Pu isotopes fission rates, fission product poison productions, end of cycle parameters for transient analyses, relative power in low and high burnup assemblies and related uncertainties in power distribution. However, the nuclear parameter changes from standard to extended burnup ranges are sufficiently small that the methodologies and their uncertainties are not significantly changed. The Pu isotopes increase and to some extent U235 decrease with burnup are at or near saturation for the standard burnup and remain at similar levels in the extended range. The extended kinetics parameters are generally bounded by the standard conservative design selection. The reactivity coefficients used in event analyses for the VANTAGE-5 fueled cores considered a suitable range of extreme cycle lengths and average burnups and are acceptably bounding for the burnups basis. The variation in assembly power, fission products and the changes in fissile isotope ratios are not sufficiently increased to affect the acceptability of the calculational methods or the uncertainty assigned to peaking factors. (It is noted that peak power assemblies are not directly involved in extended burnup ranges.)

The intermediate flow mixer grids for the VANTAGE-5 fuel have no significant direct neutronic effect but there is an indirect effect of permitting a higher design $F_{\Delta H}$ value in order to take advantage of the improved DNB margin from the grids. The increased $F_{\Delta H}$ in turn permits greater design freedom in accounting for, e.g., increased assembly to assembly power differences from increased burnup, or low radial neutron leakage design to improve pressure vessel neutron fluence. This is an acceptable nuclear design process, assuming safety analyses are done with the increased $F_{\Delta H}$ and the Technical Specifications reflect the increase. The transient and accident event analyses in WCAP-10444, using limiting parameters from a transition or equilibrium core, did include an

increased $F_{\Delta H}$. This included the equilibrium core LOCA analysis, which also used an (artificially) increased total peaking, $F_Q$, of 2.55 (compared to the standard 2.32) to indicate maximum $F_Q$ levels before exeeding LOCA 2200°F clad temperature limits for the full VANTAGE-5 core. (Transition core penalties would require a lower $F_Q$.) Other safety evaluations assumed standard axial power distributions and $F_Q$.

Low radial neutron leakage designs are not new and have been and are acceptable. But the increased design $F_{\Delta H}$ capability provides an easier design problem. Similarly the use of a less negative (more positive) MTC is not new. Several Westinghouse reactors have submitted safety analyses with positive MTC used throughout the non-LOCA analyses and have Technical Specifications providing for such a coefficient up to some power level, thus far not including full power. The VANTAGE-5 safety evaluations used a positive MTC for non-LOCA events, but not for the LOCA analyses. Thus the possibility of use for VANTAGE-5 (full and transitional) cores is acceptable with a Technical Specification similar to previously accepted models.

Other nuclear design characteristics and parameters not explicitly discussed here are sufficiently similar to OFA core design (WCAP-9500) that previous reviews for that fuel are applicable. Since VANTAGE-5 fuel will be used in transition cores from STD or OFA loaded cores to full VANTAGE-5 cores, Westinghouse has examined neutronic parameters from all three types and has provided limiting nuclear parameters (Table 3.2 of WCAP-10444) for use in safety analyses which should in general be applicable to transition and full core loadings. (Reload methodology requires that parameters be checked for the specific case.) The review has indicated that these parameters are acceptable for such use. The safety evaluations of WCAP-10444 used these values along with specific values for MTC and peaking factors which are generally loading dependent and are intended to be specific to a given reactor. It is expected that these latter parameters (e.g., MTC, ejected rod worth and peaking, $F_Q$, $F_{\Delta H}$) will be provided and justified for specific reactors.

## 4.0 THERMAL AND HYDRAULIC DESIGN

### 4.1 Core Thermal Hydraulic Analysis

The principal criterion for the thermal hydraulic design of reactor fuel is the avoidance of thermally induced fuel damage during normal operation and anticipated operational occurrences (AOO). To this end, the thermal-hydraulic design basis requires that there must be at least a 95% probability at 95% confidence level that departure from nucleate boiling (DNB) will not occur on the limiting power rod during normal operation and AOOs. This requirement is met by limiting the DNB ratio to the DNBR limit associated with the critical heat flux correlation used. This design basis is the same as that for the OFA fuel. However, since the VANTAGE-5 fuel has three intermediate flow mixer grids in addition to the existing structural grids in the OFA fuel, the thermal hydraulic characteristics have changed and the WRB-1 CHF correlation used in the OFA design is not applicable to VANTAGE-5. Westinghouse has therefore developed the WRB-2 CHF correlation based on CHF tests pertinent to VANTAGE-5 fuel. The DNBR limit for WRB-2 is 1.17, the same as for WRB-1. We have reviewed the WRB-2 correlation and the DNBR limit of 1.17 and found them acceptable. The evaluation of WRB-2 will be addressed in Section 4.2.

The thermal hydraulic calculation is performed using the THINC-IV (Ref. 6) open lattice thermal hydraulic code which takes into account the effects of interchannel cross flow and turbulent mixing. The fluid mechanics design and empirical correlations used in VANTAGE 5 is the same as those for OFA design described in WCAP-9500. The thermal diffusion coefficient (TDC) is used as a measure of the rate of heat exchange by fluid mixing between adjacent channels. The IFM grids used in VANTAGE-5 fuel have the same mixing characteristics as the OFA mixing vane grids. However, the VANTAGE-5 mixing characteristics are improved due to the presence of IFM grids which reduce the grid spacing. For thermal hydraulic analysis, the same TDC used in OFA is used for VANTAGE-5 and is therefore conservative.

Like OFA fuel, the VANTAGE-5 thermal hydraulic analysis uses the improved thermal design procedure (ITDP). The ITDP method is based upon a statistical combination of the effects on DNBR of uncertainties of the plant parameters such as reactor coolant flow rate, core power, core coolant inlet temperatures, system pressure and hot channel factors, etc. The nominal values of these parameters are used in the safety analysis and the effect of the uncertainties on DNBR is added to the 95/95 DNBR limit specified by the WRB-2 correlation to establish a design DNBR limit. The ITDP method described in WCAP-8567 (Ref. 7) has been approved for use in licensing applications subject to the certain restrictions. One of the restrictions requires that if the sensitivity factors are changed as a result of correlation change, then the use of an uncertainty allowance for application of Equation 3-2 (WCAP-8567) must be re-evaluated and the linearity assumption of WCAP-8567 must be validated. Westinghouse in response to a staff question (Ref. 5) has performed the required re-evaluation and validation using the same methods described in the staff safety evaluation report for WCAP-8567. The results show that no additional uncertainty allowance is required based on the typical sensitivity factors specified in Tables 4-4 and 4-5 of WCAP-10444, and that the linearity assumption is indeed conservative. Therefore, the ITDP method described in WCAP-8567 in conjunction with WRB-2 correlation is acceptable.

Tables 4-4 and 4-5 also provide typical values of the means, uncertainties and sensitivity factors of the plant parameters. A typical value of the design DNBR limits of 1.29 and 1.288, respectively, are derived for the typical cell and thimble cell. However, since the uncertainties of some parameters are plant-specific, those plants using ITDP are required to abide by the restrictions imposed on WCAP-8567 and perform an analysis to establish a plant-specific design DNBR limit. In addition, information regarding the measurement uncertainties for pressurizer pressure, power, coolant flow rate and temperature must be provided as specified below.

A block diagram depicting sensor, process equipment, computer and readout devices for each parameter channel used in the uncertainty analysis should be provided. Within each element of the block diagram, the accuracy, drift, range, span, operating limits and setpoints should be identified. The overall accuracy of each channel transmitter to final output and the minimum acceptable accuracy for use with the new procedure should also be identified. In addition, the overall accuracy of the final output value and maximum accuracy requirements for each input channel for this final output device should be identified.

In addition to the design DNBR which must be met in the plant safety analysis to ensure the specified acceptable fuel design limit on DNBR is not violated, a safety analysis DNBR limit is used in the safety analysis. This safety analysis DNBR limit is derived from the design DNBR limit with an additional specific allowance (e.g., 30%). The use of safety analysis DNBR in the thermal-hydraulic design provides certain safety margin which can be credited to compensate for certain required penalties, improved fuel management, or increased plant availability. However, the licensees referencing WCAP-10444 should incorporate in the bases of their plant Technical Specifications the plant-specific safety analysis DNBR limit, the DNBR allowance and the amount of allowance that has been used.

### 4.2 WRB-2 CHF Correlation

In conjunction with the development of the VANTAGE 5 fuel design, Westinghouse developed a new CHF correlation designated WRB-2 to predict DNB performance of this fuel design and other fuel designs which use the same mixing vane design as the 17 X 17 standard fuel mixing vane design. The VANTAGE 5 fuel design features an increase in DNB margin over the 17 X 17 OFA design by the addition of IFM grids. To quantify this increase in DNB performance, a test program was conducted at Columbia University using the VANTAGE 5 fuel design. As described in Appendix A.2.0, "DNB Test Program," of WCAP-10444, the test program consisted of two test series utilizing 5 X 5, 14-foot electrically heated rod

bundles. The detailed bundle geometry is similar to that of 17 X 17 OFA used in earlier DNB tests. A total of 684 data points from 11 test series consisting of the new data, the 5 X 5 OFA test bundle data and that from previous DNB tests involving 5 X 5 test bundles of various lengths but employing the standard fuel mixing vane design, were used as the data base for the development of WRB-2. The WRB-2 correlation is described in Figure A-9 of WCAP-10444 and has a minimum DNBR (95/95) limit of 1.17.

The review of WRB-2 consists of independent thermal hydraulic audit calculations to verify the correctness of the predicted local conditions and CHF predictions, and a statistical analysis to verify the correctness of the DNBR limit.

The local conditions and CHF audit calculations were performed with the COBRA-IV code (Ref. 8). The input data on the test bundle geometry, grid loss coefficients for the various grid designs and subchannel types, flow, heat flux and axial power distributions are obtained from the references provided in WCAP-10444 and the information provided by Westinghouse at staff request. For each data point, the ratio of the measured CHF to the predicted CHF is calculated. All 684 data points used in the WRB-2 development are calculated. Since Westinghouse analysis was performed with the THINC-IV thermal hydraulic code, the results of calculations using COBRA-IV/WRB-2 are not expected to be identical to those presented in WCAP-10444 but provide a general indication of the correctness of Westinghouse analysis. A summary of comparison of the measured to predicted CHF ratios by Westinghouse and the audit calculations is shown in Table 1 of this report which shows very good agreement.

The independent statistical analysis is performed with the measured to predicted CHF ratios from the thermal hydraulic calculation results. The hypothesis that data from all 11 test series come from a single population is tested in two ways. First a chi-square test of homogeneity is performed. The observed value $X^2$ = 111.6, with a degree of freedom of 60 is highly significant. This indicates that the 11 data sets do not all come from the same population. Second, an analysis of variance is performed to test the hypothesis that the

means of all 11 populations are equal. With the among-series degree of freedom of 10 and the within-series degree of freedom of 673, the F-statistic is 4.51. This value is also highly significant, indicating that not all means are equal. On the basis of these analyses, we conclude that the data do not all represent a single population. However, the estimated standard deviation from one-way analysis of variance is 0.0773, while the estimate obtained by treating the data as a single population is 0.0793. This indicates that the error introduced into variance estimates by assuming that the data are all from the same population is negligible.

The 95/95 probability/confidence DNBR limit is calculated using the formula

$$DNBR_{95/95} = \frac{1}{(M/P)_{avg} - KS}$$

where (M/P)avg is the population mean of measured to predicted CHF ratios, S is standard deviation of the M/P data and K is a tolerance multiplier which provides the 95/95 probability/confidence limit.

When the test assembly means are found to be unequal, the "total variance" is estimated by means of the formula

$$S^2_{total} = S^2_w + S^2_A,$$

where $S^2_w$ is the mean square within test series component of variance and $S^2_A$ is the "variance of the test series average." An effective degree of freedom $f_T$ is computed for $S^2_{total}$ from Satterthwaite's formula. The total variance and effective degree of freedom should be used in the DNBR limit calculates as has been done for WRB-1. This is also done for WRB-2. However, the calculation of the variance of the test series means $S^2_A$ was performed by Westinghouse with an assumption of an equal number of data points for each test series. In response to a staff question (Ref. 9), Westinghouse provided a correct formula for calculating $S^2_A$ without the assumption of equal number of data points. It also

shows that the equal data assumption results in higher total variance and fewer effective degrees of freedom. Therefore, the resulting DNBR limit of 1.17 is conservative. We have concluded that WRB-2 with a DNBR limit of 1.17 is acceptable for application to VANTAGE 5 with the following range of applicability.

RANGE OF VARIABLES

| | |
|---|---|
| Pressure | $1440 \leq P \leq 2490$ psia |
| Local Mass Velocity | $0.9 \leq G_{loc}/10^6 \leq 3.7$ lb/ft$^2$hr |
| Local Quality | $-0.1 \leq X_{loc} \leq 0.3$ |
| Heated Length, Inlet to CHF Location | $L_h \leq 14$ feet |
| Grid Spacing | $10 \leq g_{sp} \leq 26$ inches |
| Equivalent Hydraulic Diameter | $0.37 \leq d_e \leq 0.51$ inches |
| Equivalent Heated Hydraulic Diameter | $0.46 \leq d_h \leq 0.59$ inches |

TABLE 1

WRB-2 CHF CORRELATION - STATISTICAL RESULTS

| Rod O.D. (inch) | Length (feet) | Grids (in.) | Heat Flux Profile | Assembly Geometry | Data Points | WESTINGHOUSE M/P Ratio | WESTINGHOUSE Standard Deviation | AUDIT CALCULATIONS M/P Ratio | AUDIT CALCULATIONS Standard Deviation | Reference | Data Table |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | 14 | 10 | Cosine | TYP-5X5 | 51 | 0.9861 | 0.0758 | 0.9721 | 0.0705 | WCAP-10444 | A-2 |
| | 14 | 10 | Cosine | TYP-5X5 | 31 | 1.0097 | 0.0680 | 1.0181 | 0.0623 | WCAP-10444 | A-3 |
| | 14 | 20 | Cosine | TYP-5X5 | 63 | 0.9961 | 0.0946 | 0.9852 | 0.0952 | WCAP-9401 | A-4 |
| | 14 | 20 | Cosine | THM-5X5 | 38 | 0.9832 | 0.0599 | 0.9946 | 0.0600 | WCAP-9401 | A-5 |
| 0.374 | 8 | 22 | Uniform | TYP-5X5 | 67 | 1.0316 | 0.0897 | 1.0348 | 0.0753 | WCAP-8536 | A-6 |
| 0.374 | 14 | 22 | Uniform | TYP-5X5 | 71 | 1.0095 | 0.0664 | 0.9924 | 0.0626 | WCAP-8536 | A-7 |
| 0.374 | 14 | 22 | Cosine | TYP-5X5 | 74 | 0.9893 | 0.0822 | 0.9691 | 0.0773 | WCAP-8536 | A-8 |
| 0.374 | 14 | 22 | Cosine | THM-5X5 | 70 | 0.9884 | 0.0775 | 0.9729 | 0.0765 | WCAP-8536 | A-9 |
| 0.374 | 8 | 26 | Uniform | TYP-5X5 | 78 | 1.0198 | 0.0810 | 0.9890 | 0.0753 | WCAP-8926 | A-10 |
| 0.374 | 8 | 26 | Uniform | THM-5X5 | 68 | 1.0398 | 0.1062 | 1.0092 | 0.0942 | WCAP-8926 | A-11 |
| 0.374 | 14 | 26 | Uniform | TYP-5X5 | 73 | 0.9914 | 0.0823 | 0.9761 | 0.0785 | WCAP-8926 | A-12 |
| | | | All Data | | 684 | 1.0051 | 0.0847 | 0.9907 | 0.0793 | | |

## 4.3 Core Flow Design

WCAP-10444 specifies as a design basis that at least a certain percent of the reactor coolant flow should pass through the fuel rod region of the core for cooling. This design basis assures that for RCS flow rate specified in the plant Technical Specifications the core bypass flow should be accounted for. Core bypass flow is the sum of the flow through rod cluster control guide thimbles, head cooling flow, baffle leakage and leakage to the vessel outlet nozzle. The uncertainties associated with the calculation of bypass flow should be accounted for in the safety analysis. For the analysis using ITDP, the uncertainties are statistically combined with uncertainties of other parameters to obtain the design DNBR limit. For analysis not using ITDP, the uncertainties are accounted for by reducing the active core flow.

The bypass flow is dependent upon core geometry and is plant specific. Higher fuel assembly pressure drop results in larger bypass flow. The presence of the IFM grids in VANTAGE 5 increases the fuel assembly pressure drop as well as bypass flow. Westinghouse indicates that the total bypass flow used in the safety analysis of the reference plant is conservative. However, for licensees referencing WCAP-10444, a plant specific analysis is required to determine the correct bypass flow. Correct or conservative active core flow rate should be used in safety analysis.

## 4.4. Thermohydrodynamic Stability

One of the design bases for VANTAGE-5 is to ensure that the mode of operation with Conditions I and II events will not lead to thermohydrodynamic instability. The flow excursion (Ledinegg) instability generally does not exist in a PWR. This is due to the inherent characteristics of RCS pump head-capacity curve having a negative slope and system hydraulic pressure drop-flow curve having a positive slope in PWR. The staff has previously accepted the stability evaluation for OFA fuel based on the past operating

experience, flow stability experiments and the inherent thermal hydraulic characteristics of Westinghouse open channel core configuration. Use of VANTAGE-5 fuel will not change these characteristics. We therefore conclude that flow instability will not be a problem for VANTAGE 5.

### 4.5 Rod Bow Effect on DNBR

Fuel rod bowing results in flow channel gap closure and reduction of critical heat flux and DNBR. This CHF effect is accounted for through a rod bow DNBR penalty. The method described in WCAP-8691, Revision 1 (Ref. 10), has been approved for the rod bow penalty calculations. A scaling factor dependent on the cladding moment of inertia and span length between grids is used for the channel closure calculation when extrapolating from the channel closure data of different fuel assembly geometry. Since three IFM grids have been installed between the Zircaloy grids in the VANTAGE-5 fuel, the topical report asserts that the grid spacing and, therefore, the channel closure are reduced. In response to a staff question (Ref. 5) on whether the presence of the "non-structural" IFM grids provides the necessary support in the reduction of rod bowing. Westinghouse indicates that the term "non-structural" is a misnomer and that the IFM grids do provide a positive pin type support and have a small stiffening effect near the top of the fuel assembly. Using the approved scaling factor, the predicted channel closure in the limiting span of the VANTAGE 5 fuel is less than 50%. Experimental data (Ref. 11) have shown that there is no CHF reduction for channel closure less than 50%. Therefore no rod bow penalty is required for the VANTAGE 5 fuel.

### 4.6 Transitional Mixed Core DNBR Effect

As a reactor is reloaded with VANTAGE-5 fuel prior to a full core of VANTAGE-5 fuel, there are transitional cycles when both VANTAGE-5 and remaining fuel (such as standard fuel or OFA fuel) will co-exist in the core. The differences between the adjacent fuel assemblies in the hydraulic resistance

characteristics such as spacer grid designs, flow areas and the presence of VANTAGE-5 IFM grids result in local hydraulic mismatches. Such a hydraulic mismatch results in localized flow redistribution due to the open core configuration. While beneficial to STD or OFA fuel due to lower grid resistance, the interbundle cross flow is detrimental to VANTAGE-5 fuel. In the safety analysis, a full core of VANTAGE 5 fuel is assumed. A mixed core DNBR penalty is applied during the transitional mixed core configuration to account for the detrimental effect of interbundle cross flow on the VANTAGE-5 fuel. Westinghouse has performed sensitivity studies in the same way approved previously for the standard fuel/OFA fuel mixed core (Ref. 12) to determine the penalty factor required for the VANTAGE-5 fuel. Since the hydraulic mismatch between VANTAGE 5/OFA assemblies is less complex than that between the VANTAGE-5/STD fuel assemblies, the penalty factor determined from VANTAGE-5/STD mixed core is bounding for VANTAGE-5/OFA mixed core. The value of the penalty factor is described in WCAP-10444 and is later revised to 11% in a response to a staff question (Ref. 5). This value (11%) is generic for 17 X 17 fuel assemblies. For other fuel types such as 14 X 14 or 15 X 15 fuel assemblies, the mixed core penalty factor should be determined separately. As described in Section 4.1, when the ITDP is used in safety analysis, there is a generic margin between the design DNBR limit and plant safety analysis DNBR. This safety margin is enough to compensate for the penalty factor required for mixed core configuration. However, for these plants having the VANTAGE-5 fuel, both the plant specific safety margin and mixed core penalty factor should be addressed in Technical Specification bases.

### 4.7 Low Leakage Loading Core Design

In order to improve nuclear design flexibility and fuel economy, and to reduce pressure vessel fluence, some cores containing VANTAGE 5 fuel will be designed with burned blankets, i.e., the burned fuel assemblies will be placed in the core periphery. To facilitate this, the hot channel enthalpy rise factor $F_{\Delta H}$ is raised from the current 1.55 to a higher value (Westinghouse Proprietary). The increase in $F_{\Delta H}$ is made possible because of the thermal margin gained from

the use of the IFM grids resulting in higher CHF for VANTAGE 5 fuel. For the OFA and STD fuel, thermal margin could be gained from the rod bow penalty reduction as a result of change of rod bow calculation from the NRC interim method (Ref. 13) to the method of WCAP-8691, Revision 1. This thermal margin gain from rod bow penalty reduction could be used to increase the normal operation $F_{\Delta H}$. However, plant specific analyses are required with appropriate $F_{\Delta H}$ to ensure that the specified acceptable fuel design limits will not be violated during normal operation and anticipated operational occurrences.

### 4.8 Other VANTAGE 5 Design Feature Effects

Two VANTAGE-5 design features, namely, axial blanket and IFBA, result in axial zoning in the fuel rod which perturbs core axial power distribution and affects the DNBR evaluation. The axial blanket reduces power at the ends of the rod and causes a slight increase in axial power peaking. Though IFBA flattens the axial power distribution and reduces the local peaking, the IFBA burns out during irradiation. Therefore, the net result of axial blanket and IFBA is a slight increase in power peaking during core operation. Westinghouse has examined this effect and found that the design axial power distribution used in the determination of the overtemperature $\Delta T$ setpoint is still conservative.

Another VANTAGE 5 design feature is the reconstitutable top and bottom nozzles which results in changes in hydraulic resistance at the inlet and outlet of the active fuel region from the previous standard and OFA designs. However, the hydraulic tests performed by Westinghouse have shown that the changes in nozzle design have a negligible impact on the inlet flow and outlet pressure distributions. Therefore, the effect on DNBR is negligible.

## 5.0 SAFETY EVALUATION

A safety evaluation is necessary to assess the impact of the VANTAGE 5 design features such as axial blankets, IFBA, IFM grid and reconstitutable top nozzle, etc., and the core reload strategy having a higher $F_{\Delta H}$ and a positive moderator

coefficient. Table 5-1 of WCAP-10444 lists the non-LOCA accidents which are most sensitive to these changes. They are:

(a) Main Steamline Rupture;
(b) Loss of Load/Turbine Trip;
(c) Complete Loss of Forced Reactor Coolant Flow;
(d) Single Reactor Coolant Pump Locked Rotor;
(e) Uncontrolled RCCA Bank Withdrawal at Power; and
(f) RCCA Ejection.

As an evaluation of the impact of the VANTAGE 5 design features, analyses of these accidents and loss of coolant accident (LOCA) are performed using the methods described in WCAP-9272, "Westinghouse Reload Safety Analysis Methodology." The analyses are performed with a typical 4-loop reference plant. All analytical procedures and computer codes used in the non-LOCA safety analysis are the same as those described in WCAP-9500 for OFA, except that the WRB-2 CHF correlation is used for VANTAGE 5. For most accidents which are DNB limited, the improved thermal design procedure is used in the analyses. Table 5-3 of WCAP-10444 summarizes the initial condition and computer codes used in the non-LOCA accident analyses and also shows which accidents employ a DNB analysis using ITDP. For the large break LOCA, the analysis is performed with a modified version of the 1981 Westinghouse ECCS evaluation model (Ref. 14). The modified version of ECCS evaluation model uses the BART code (Ref. 15) to calculated the reflood heat transfer coefficient in place of the FLECHT reflood heat transfer correlation normally used in the LOCTA code (Ref. 16). BART is a mechanistic core heat transfer code. The code with the grid rewet model has been approved for LOCA analysis. However, the drop break up model is not approved. In response to a staff question (Ref. 5), Westinghouse confirms that in every application of BART supporting WCAP-10444, a version of the code was used that explicitly excluded the drop breakup model.

For a transitional mixed core where the VANTAGE 5 fuel co-exists with either STD or OFA fuel, the local hydraulic resistance mismatch between the adjacent

assemblies of different design has a detrimental effect on VANTAGE 5. The flow redistribution due to hydraulic resistance mismatch results in a reduction of reflood steam flow rate and a increase in peak cladding temperature in the VANTAGE 5 fuel . For a transitional core, the LOCA analysis is performed with a full core of VANTAGE 5 fuel. The resulting PCT is then increased by adding a mixed core penalty. This approach has been reviewed and approved during the review of WCAP-9500. Westinghouse has performed the same analysis for VANTAGE 5 and the conservative nature of the required mixed core penalty on PCT was determined for VANTAGE 5. We conclude that this approach is acceptable.

The results of analyses as presented in WCAP-10444 generally demonstrate that the SAFDLs are not violated and the criteria specified in 10 CFR 50.46 can be met. Since the safety analyses of WCAP-10444 are not intended to be directly applicable to specific plants but were submitted for demonstrating acceptability of the fuel in a generic sense, we find the safety analyses presented in WCAP-10444 acceptable. The licensees referencing WCAP-10444 are required to perform plant-specific safety analyses to ensure the safety criteria are met.

There are, however, several deficiencies in the safety analyses of WCAP-10444. These are discussed below and should be corrected in the plant-specific safety analyses.

(1) For the steam system piping failure accident, the analysis assumes that the offsite power is available. This assumption is not in compliance with the SRP which requires that assumptions as to the loss of offsite power and the time of loss should be made to study their effects on the consequence of the accident. This requirement is also specified in General Design Criterion 17, 10 CFR 50 Appendix A.

(2) With regard to the reactor coolant pump seizure (locked rotor) accident, WCAP-10444 states that although some rods may enter DNB for a short period

of time, fuel failure for the locked rotor/shaft seizure incident is treated on a mechanistic basis, thus eliminating the need to automatically equate DNB with fuel failure. WCAP-10444 also indicates that the mechanistic methods are allowed by SRP and that the temperature-time criteria of NUREG-0562 (Ref. 17) will be used in the fuel failure calculation on a plant specific basis. However, NUREG-0562 prepared by the Office of Nuclear Reactor Research, has not been applied in licensing. The temperature-time criteria noted in the report are from another reference (Pawel, Journal of Nuclear Materials, Volume 50, Page 247-258) which the NRC has not endorsed for this type of application.

It has been and continues to be the NRC position that cladding failure is assumed to occur when the fuel rod DNBR is less than the safety limit. This position provides a degree of conservatism which varies with each anticipated operational occurrence and accident. This conservatism covers such analyses uncertainties as:

(i) Variation in history of an actual event from that idealized in the accident analysis

(ii) Uncertainties in the thermal hydraulics such as

- (a) prediction of heat transfer coefficient in film boiling
- (b) prediction of subchannel flow in film boiling
- (c) prediction of extent and location of DNB
- (d) multiple DNB on the same rod
- (e) changes in fuel bundle geometry during event due to overheating
- (f) uncertainties in prediction of transients, non-isothermal zirconium/water reaction kinetics, etc., (see the above reference by Pawel).

In addition to these, there are questions related to the prototypicality of the NUREG-0562 data. EPRI report NP 1999 which addresses post-DNB effects in PWRs has identified several areas which would be appropriate subjects for future research before application to licensing (see Table 5-1 of this report). While we are not taking the position that all items in Table 5-1 must be addressed, it is clear from this industry report that more review would be necessary before approval of a post-DNB fuel failure criterion.

Therefore, the licensees referencing WCAP-10444 should assume that all fuel which experiences a DNBR of less than the DNBR limit fails and calculate the offsite dose consequences. In the offsite dose analysis the licensee should assume maximum Technical Specification pre-accident coolant activity and steam generator leakage. Single failures should be considered including a stuck open secondary relief valve. Loss of offsite power should be assumed per GDC-17. The effect of steam generator tube uncovery on the offsite dose consequences caused by operator action to isolate the affected steam generator should also be considered in the analysis.

(3) The large break LOCA analysis performed for a full core of VANTAGE 5 fuel using a total peaking factor $F_Q$ of 2.55 has shown a peak cladding temperature of 2194°F which is below the acceptance criterion of 2200°F. However, for a transitional mixed core configuration, addition of the mixed core penalty on PCT will result in the PCT exceeding the 2200°F acceptance criterion per 10 CFR 50.46. The plant specific analysis must be done to ensure that with appropriate value of $F_Q$, the 2200°F criterion can be met even for the transitional mixed core.

## 6.0 SUMMARY AND CONCLUSION

The staff has reviewed the topical report WCAP-10444 and concludes that the report is acceptable for reference for the Westinghouse VANTAGE 5 fuel design. This acceptability is subject to the following conditions:

1. The statistical convolution method described in WCAP-10125 for the evaluation of initial fuel rod to nozzle growth gap has not been approved. This method should not be used in VANTAGE 5.

2. For each plant application, it must be demonstrated that the LOCA/seismic loads considered in WCAP-9401 bound the plant in question; otherwise additional analysis will be required to demonstrate the fuel assembly structural integrity.

3 An irradiation demonstration program should be performed to provide early confirmation performance data for the VANTAGE 5 design.

4. For those plants using the ITDP, the restrictions enumerated in Section 4.1 of this report must be addressed and information regarding measurement uncertainties must be provided.

5. The WRB-2 correlation with a DNBR limit of 1.17 is acceptable for application to 17 X 17 VANTAGE 5 fuel. Additional data and analysis are required when applied to 14 X 14 or 15 X 15 fuel with an appropriate DNBR limit. The applicability range of WRB-2 is specified in Section 4.2.

6. For 14 X 14 and 15 X 15 VANTAGE 5 fuel designs, separate analyses will be required to determine a transitional mixed core penalty. The mixed core penalty and plant specific safety margin to compensate for the penalty should be addressed in the plant Technical Specifications Bases.

7. Plant specific analysis should be performed to show that the DNBR limit will not be violated with the higher value of $F_{\Delta H}$.

8. The plant-specific safety analysis for the steam system piping failure event should be performed with the assumption of loss of offsite power if that is the most conservative case.

9. With regard to the RCS pump shaft seizure accident, the fuel failure criterion should be the 95/95 DNBR limit. The mechanistic method mentioned in WCAP-10444 is not acceptable.

10. If a positive MTC is intended for VANTAGE 5, the same positive MTC consistent with the plant Technical Specifications should be used in the plant specific safety analysis.

11. The LOCA analysis performed for the reference plant with higher $F_Q$ of 2.55 has shown that the PCT limit of 2200°F is violated during transitional mixed core configuration. Plant specific LOCA analysis must be done to show that with the appropriate value of $F_Q$, the 2200°F criterion can be met during use of transitional mixed core.

12. Our SER on Westinghouse's extended burnup topical report WCAP-10125 is not yet complete; the approval of the VANTAGE 5 design for operation to extended burnup levels is contingent on NRC approval of WCAP-10125. However, VANTAGE 5 fuel may be used to those burnups to which Westinghouse fuel is presently operating. Our review of the Westinghouse extended burnup topical report has not identified any safety issues with operation to the burnup value given in the extended burnup report.

13. Recently, a vibration problem has been reported in a French reactor having 14-foot fuel assemblies; vibration below the fuel assemblies in the lower portion of the reactor vessel is damaging the movable incore instrumentation probe thimbles. The staff is currently evaluating the implications of this problem to other cores having 14 foot long fuel bundle assemblies. Any limitations to the 14-foot core design resulting from the staff evaluation must be addressed in plant-specific evaluations.

## 7.0 REFERENCES


1. WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel," dated July 1982.

2. WCAP-9500-A, "Reference Core Report 17 X 17 Optimized Fuel Assembly," dated May 1982.

3. Letter from E. P. Rahe, Jr. (Westinghouse) to C. O. Thomas (NRC), "Reference to NRC Request Number 3 on WCAP-10125," NS-NRC-85-3018, dated March 14, 1985.

4. WCAP-9401-P-A, "Reference Core Report 17 X 17 Optimized Fuel Assembly," dated May 1982.

5. Letter from E. P. Rahe, Jr. to C. O. Thomas, "Response to Request Number 1 for Additional Information on WCAP-10444 entitled, 'VANTAGE-5 Fuel Assembly'" (Proprietary), NS-NRC-85-3014, dated March 1, 1985.

6. WCAP-7956, "THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," 1973.

7. WCAP-8567, "Improved Thermal Design Procedure," dated July 1975.

8. BNWL-1962 - UC-32, "COBRA-IV-1: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores," Battelle Pacific Northwest Laboratories, dated March 1976.

9. Letter from E. P. Rahe, Jr. (Westinghouse) to C. O. Thomas (NRC), "Follow-up Response to Request Number 1 for Additional Information for WCAP-10444 entitled, 'VANTAGE 5 Fuel Assembly - Reference Core Report'," (Proprietary), NS-NRC-85-3033, dated May 8, 1985.

10. WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," dated July 1979.

11. E. S. Markowski, et al., "Effect of Rod Bowing on CHF in PWR Fuel Assemblies," ASME Peper 77-HT-91.

12. Letter from E. P. Rahe, Jr. (Westinghouse) to J. R. Miller (NRC), "Supplement to WCAP-9500 and WCAP-9401/9402 NRC Safety Evaluation Report (SER) Mixed Core Compatibility Items - Supplemental Information," NS-EPR-2643, dated August 17, 1982.

13. Memorandum from D. F. Ross and D. G. Eisenhut to D. B. Vassallo and K. R. Goller, "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," dated February 16, 1977.

14. WCAP-9220-A, Revision 1, "Westinghouse ECCS Evaluation Model, 1981 Version," dated February 1981.

15. WCAP-9561, "BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients," dated January 1980.

16. WCAP-8301, "LOCTA-IV Program: Loss of Coolant Transient Analyses," dated June 1974.

17. WCAP-8170, "Calculated Model for Core Reflood After a Loss of Coolant Accident (Westinghouse Reflood Code)," dated June 1974.

18. R. Van Houten, "Fuel Rod Failure As a Consequence of Department from Nucleate Boiling or Dryout," NUREG-0562, 1979.

Mr. E. P. Rahe, Jr., Manager
Nuclear Safety Department
Westinghouse Electric Corporation
Box 355
Pittsburgh, Pennsylvania 15230

Dear Mr. Rahe:

SUBJECT: ACCEPTANCE FOR REFERENCING OF LICENSING TOPICAL REPORT WCAP-10444, "VANTAGE 5 FUEL ASSEMBLY"

We have completed our review of the subject topical report submitted by Westinghouse Electric Corporation (Westinghouse) letter dated December 30, 1983. We find the report to be acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

We do not intend to repeat our review of the matters described in the report and found acceptable when the report appears as a reference in license applications, except to assure that the material presented is applicable to the specific plant involved. Our acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, it is requested that Westinghouse publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should our criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, Westinghouse and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation, or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Cecil O. Thomas, Chief
Standardization and Special
Projects Branch
Division of Licensing

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ENCLOSURE

SER ON WCAP-10444

## 1.0 INTRODUCTION

By letter dated December 30, 1983, Westinghouse requested NRC to review the topical report WCAP-10444, "Westinghouse Reference Core Report, VANTAGE 5 Fuel Assembly."

The VANTAGE 5 fuel assembly is a modification of the design of the Standard and Optimized 17 X 17 fuel assemblies. (It will also be offered in 14 X 14 and 15 X 15 configurations.) The new features of the VANTAGE 5 fuel assembly consist of (1) axial blankets for improved neutron utilization, (2) Integral Fuel Burnable Absorber (IFBA), (3) Intermediate Flow Mixer (IFM) grids, (4) Reconstitutable top and bottom nozzles, and (5) extended burnup. The impact of these design changes is evaluated from the standpoints of mechanical, nuclear and thermal hydraulic designs and transient and accident analyses. The analysis is performed with the 17 X 17 fuel assembly array for a 4-loop reference plant. The 14 X 14 and 15 X 15 application should be evaluated on a plant-specific basis.

Because our SER on Westinghouse's extended burnup topical report WCAP-10125 (Ref. 1) is not complete, the design bases/criteria and analysis methods review in this submittal have not been approved for application to extended burnup levels. Consequently, the approval of the VANTAGE 5 design for operation to extended burnup levels is contingent on NRC approval of Westinghouse's extended burnup topical report. It should be noted, however, that the review of WCAP-10125 is complete and there are no outstanding issues at this time. Our review, documented in this safety evaluation report, covers the VANTAGE 5 fuel designs to those burnups to which PWRs with Westinghouse fuel are presently operating.

## 2.0 MECHANICAL DESIGN

### 2.1 Fuel Assembly Design

WCAP-10444 indicates that the design bases and design limits for the VANTAGE-5 fuel assembly are essentially the same as those for the optimized fuel assembly (OFA) design described in WCAP-9500 (Ref. 2). We have therefore relied heavily on our review of Reference 2 and only the differences in design bases will be discussed in the following evaluation.

#### 2.1.1 Fuel Rod Growth Gap

WCAP-10444 states that the design basis for the axial clearance between core plates and nozzle end plates should allow sufficient margin for fuel assembly and fuel rod irradiation growth to design burnup established in WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel" (Ref. 1). WCAP-10125 indicates that the initial fuel rod-to-nozzle growth gap is designed based on a statistical convolution of the distribution of the measured gap data as a function of fluence and the distribution of rod average fluences for an assembly. This statistical treatment of uncertainties associated with the fuel rod/assembly irradiation growth effect is a deviation from the approved method. In response to a staff question (Ref. 3), however, Westinghouse indicates that the current approved design criterion has been used for the VANTAGE-5 design, i.e., the minimum gap is the required allowance for irradiation growth established to preclude fuel rod-to-nozzle interface during projected operation on the basis of the assumption of worst case fuel rod and fuel assembly growth combined with worst case fabrication tolerances. Therefore, the design for fuel rod growth gap for VANTAGE-5 is the same as that for the OFA design and is therefore acceptable.

For the VANTAGE-5 fuel assembly design, the overall assembly length has been adjusted to accommodate assembly growth and the distance between the top and bottom nozzle plates has been adjusted to accommodate the expected

increased fuel growth related to extended burnup operation. The growth predictions were based on Westinghouse incore experience including results from the high burnup demonstration program. Therefore, adequate shoulder gap is provided to accommodate rod growth.

### 2.1.2 Fuel Fretting Wear

The Westinghouse design guideline for VANTAGE 5 regarding fuel rod fretting wear is the same as that for OFA by limiting the fuel rod wear due to contact with the spacer grids to less than a certain percentage (proprietary) of clad thickness. That the VANTAGE 5 design meets this guideline is confirmed via hydraulic testing. The tests were performed in the Westinghouse Fuel Assembly Test System (FATS) with a VANTAGE 5 assembly adjacent to a 17 X 17 OFA assembly, adjacent to a 17 X 17 STD fuel assembly, and adjacent to another VANTAGE 5 assembly. The results of these tests confirm that the projected fuel rod wear of VANTAGE 5 is well within the design guideline. There was no indication of adverse fretting wear of the fuel rods by the standard structure or the IFM grids.

### 2.1.3 Fuel Assembly Structural Integrity

Section 2.2.3.3 of WCAP-10444 provides an evaluation of the VANTAGE 5 fuel assembly structural response to the externally applied forces such as LOCA and seismic forces. Appendix A.3.0 of the report describes the mechanical test program performed to confirm the structural integrity of the fuel assembly. The mechanical tests consist of lateral vibration and impact tests, and axial stiffness and impact tests. The results show that the VANTAGE 5 design changes represent a small improvement over the OFA design with respect to the structural dynamic response. This is attributable to the IFM grids providing a small stiffness effect near the top of the fuel assembly. The IFM grids also provide additional load sharing during Seismic/LOCA impact.

Analytical evaluations of the fuel assembly response to the most limiting LOCA and seismic forces are performed with the time history numerical technique in the same manner as the approved method described in WCAP-9401 (Ref. 4). The resulting LOCA and seismic induced grid impact forces are combined in accordance with the method specified in Section 4.2 of SRP. The results show that the maximum combined grid impact force for both the Zircaloy structural grids and IFM grids are within the allowable limits. The maximum fuel assembly deflection and the stress resulting from the deflection indicated substantial safety margins compared to the allowable value. Therefore, we conclude that structural damage from external forces is not a concern. However, as discussed in the staff safety evaluation report for WCAP-9401, for each plant application, it must be demonstrated that the applied seismic/LOCA loads considered in WCAP-9401 bound the plant in question or else additional analysis will be required.

### 2.1.4 Fuel Assembly Shipping and Handling Loads

The design requirement for shipping and handling loads for the VANTAGE 5 fuel assembly is reduced from 6g to 4g. This change allows for reduction in the thickness of nozzle end plates and therefore increases the axial space available for fuel assembly and rod design. Westinghouse has tested the handling acceleration at both the manufacturing facility and reactor sites, and determined that handling acceleration is well below the 4g limit. Westinghouse has also performed extensive over-the-road tests with shipping containers containing dummy fuel assemblies and found that insignificant g loads were communicated to the fuel assembly carriage in the container. At Columbia University, tests were also performed with a dummy assembly under the most accelerated crane movement possible with abrupt stops. Results of repeated tests confirm the maximum axial load to be far less than 4g. Therefore, based on the results of these tests, the 4g design criterion is acceptable.

2.1.5 Fuel Assembly Structural Components

The design bases for the VANTAGE 5 fuel assembly structural components are essentially the same as those described in WCAP-9500 for OFA except for the 4g structural requirement. However, there are a few changes in the VANTAGE 5 component designs. The bottom nozzle uses Inconel 718 instead of Type 304 stainless steel to permit the use of a thinner nozzle plate. The top nozzle has a groove in each thimble through-hole in the nozzle plate to facilitate removal and has a reduced nozzle plate thickness to provide additional axial space. The hold-down spring is a 4-leaf spring design instead of the standard 3-leaf design in order to compensate for the increase in pressure drop due to the IFM grids.

For both the top and bottom nozzles, Westinghouse verified through a finite element analysis that the maximum stresses at 4g's do not exceed the allowable limits. They are also tested to demonstrate that the strength exceeds the structural 4g functional requirement.

The 4-leaf hold-down spring has been hydraulically tested in the Westinghouse FATS facility to determine the assembly lift off flow rate and the assembly pressure loss coefficient. The results of the flow testing serve as the basis for optimizing the spring force. In response to a staff question (Ref. 5), Westinghouse indicated that the effects of spring relaxation and assembly growth are also accounted for in the spring design. The spring has been designed with a 10% higher BOL load than required by the hydraulic lift force to account for the net effect of the assembly growth and spring relaxation.

There are no changes involved in the guide thimbles, instrumentation tubes and the structural spacer grids from OFA design. The IFM grids are non-structural members whose primary function is to promote mid-span flow mixing. Therefore, the design bases for the IFM grids are to avoid cladding wear and interactive damage with the grids of the neighboring assemblies during fuel handling. The impact tests described previously have also been performed for the IFM grids.

The results show that the crush strength of the IFM grids exceeds the expected dynamic load during a seismic/LOCA event. Therefore, a coolable geometry is assured.

### 2.1.6 Irradiation Demonstration Program

The report indicates that an irradiation demonstration program will be performed to provide early confirmation performance data for the VANTAGE-5 design. This program will be performed consistent with the product demonstration effort employed for the introduction of the 17 X 17 STD fuel and OFA fuel design. We find this commitment acceptable.

## 2.2 Fuel Rod Design

The VANTAGE 5 fuel rod design is identical to the OFA design with the exceptions that (1) axial blanket pellets replace the top and bottom parts of the fuel stacks; (2) thin boride coating is added to the fuel pellets in the middle portion of the fuel rod to serve as integral fuel burnable absorber (IFBA), (3) an additional plenum space is included to accommodate gas release from the fuel and the thermal expansion differential between the cladding and fuel; and (4) a tapered bottom end plug is used to guide the fuel rod during top-end reconstitution. In addition, VANTAGE 5 is designed for a higher burnup and higher enthalpy rise factor ($F_{\Delta H}$) operation. However, the design bases for VANTAGE 5 are the same as OFA in that fuel rod damage should not occur due to excessive rod internal pressure, clad stress, clad strain, clad temperature, fuel temperature or clad strain fatigue, and that the fuel rod should not fail due to fretting wear on the outer cladding surface or by clad flattening. The analyses showing that these design bases are met were performed using the same methods as were used for OFA and the STD designs.

(a) Rod Internal Pressure - Since VANTAGE 5 fuel rods contain IFBA, additional helium gas is created and released from the depleted boride contained in the burnable boride coating. However, a Westinghouse evaluation has shown that the internal pressure of VANTAGE 5 meets the design criterion.

(b) Clad Strain and Stress - Design evaluations show tht the clad stresses and strains for both IFBA and non-IFBA fuel rods of VANTAGE 5 design meet the stress and strain design limits.

(c) Clad Temperature - In order to limit metal oxide formation to acceptable values, specific design limits at the metal/oxide interface temperature are specified for both condition I and II transients. Westinghouse calculations have shown that the clad temperature of VANTAGE 5 fuel rod meets the design limits.

(d) Fuel Temperature - To avoid $UO_2$ centerline melting during condition I and II transients, the fuel centerline temperature is limited to 4700°F. A Westinghouse design evaluation has shown that the fuel centerline temperature remains below this limit.

(e) Clad Fatigue - As a design basis, Westinghouse specified a value (proprietary) of a fatigue life usage factor limit. The design evaluations have shown that the cumulative fatigue life usage factor for the VANTAGE 5 cladding is far less than the specified limit value.

(f) Clad Wear - Westinghouse has a value (proprietary) of wall thickness as a general guideline in evaluating cladding imperfections including fretting wear. Since the structural grids and fuel tube for both VANTAGE 5 and OFA are identical with respect to grid cell forces acting on the fuel rods, the amount of clad wear for both fuel designs should be the same. As part of the hydraulic test performed at the FATS facility, the wear test results have verified that the VANTAGE 5 fuel wear characteristics are similar to that of the 17 X 17 OFA. Therefore, the conclusion can be

drawn that the VANTAGE 5 fuel rod wear will not exceed the maximum wear depth limit.

(g) Clad Flattening - If axial gaps in the fuel pellet columns were to occur due to densification, the cladding would have the potential of collapsing into a gap, i.e., clad flattening. Because of the large local strains that would result from collapse, the cladding is assumed to fail. Westinghouse calculations for VANTAGE 5 show that the predicted clad flattening time exceeds residence times expected for the extended burnup fuel management. Typical values of clad flattening times are in excess of 45000 effective full power hours.

## 2.3 Integral Fuel Burnable Absorber (IFBA)

Some VANTAGE 5 fuel pellets have a thin layer of boride coating on the surface to act as integral fuel burnable absorber. The design bases are the same for both IFBA and non-IFBA fuels. Evaluations are made to determine if the IFBA rods meet the design bases.

(a) Performance Test - The performance of the IFBA rods has been demonstrated with test fuel rods having various thickness of boride coating irradiated in both a test reactor (BR-3) and a commercial reactor (Turkey Point Unit 3). Post Irradiation Examinations (PIE) of these test rods irradiated up to 13000 MWD/MTU indicate that the boride coating performs well with no apparent loss of coating integrity. The monitoring of the reactivity and depletion characteristics shows that the absorber behaves as predicted. The discharged fuel assembly shows no leaking rods in sipping tests. Additional tests are planned to have IFBA rods inserted in the same reactor to provide additional confirmation performance data for the IFBA feature.

(b) Clad Hydriding - Hydriding as a cladding failure mechanism is precluded by controlling the level of moisture and other hydrogenous impurities during fabrication. The IFBA-containing fuel is required to meet the same hydrogen, fluorine and impurity limits set for the $UO_2$ fuel with the exception of the coating constituent. Clad hydrogen analysis of the irradiated test rods from BR-3 reactor has shown no difference in the cladding hydrogen absorption characteristics between the areas adjacent to both coated and non-coated fuel pellets. Based on PIE results and thermodynamics consideration, Westinghouse concludes that no chemical reaction between boride coated pellet and the Zircaloy-4 cladding during reactor life is expected. In addition, the clad is also protected by $ZrO_2$ which is also thermodynamically stable with respect to the boride coating.

(c) Clad Breaching - Though the design basis for VANTAGE 5 fuel is to maintain cladding integrity, the event of a sufficiently large Zircaloy cladding breaching is considered which would release fission products from fuel rod and would also likely result in the boride coating leached out by the coolant water. Since the boride loading is small in any individual rod and the IFBA rods are distributed across the core, Westinghouse states that the change in peaking factor would be small even if the clad breach and absorber loss occur early in life.

(d) Fuel Rod Waterside Corrosion - The IFBA coating on the fuel pellets has no effect on clad/oxide interface temperature or waterside corrosion compared to non-IFBA rods. The waterside corrosion criterion for the non-IFBA rod remains applicable for the IFBA rod.

(e) IFBA Coating -Fuel Compatibility - Westinghouse has evaluated the compatibility of IFBA coating and $UO_2$ fuel. Thermodynamic evaluations show no adverse chemical reaction occur between the boride coating and the $UO_2$ fuel. This is confirmed by electron microprobe analysis showing no adverse diffusion reaction between the boride coating and $UO_2$ when heated to 1750°C for 8 hours or 1600°C for 24 hours. PIE test results

from the BR-3 test reactor also confirm the compatibility of the absorber coating and $UO_2$ under operating conditions. Although a reaction has been observed due to excess oxygen and boride depletion, Westinghouse has found this to have no adverse impact on either the fuel or absorber performance.

(f) Boride Coating Axial Distribution - The boride coating is required to effectively remain in place throughout its functional lifetime to ensure no significant loss of neutron absorber uniformity. Thin coating integrity is confirmed by the PIE results of test rods from the BR-3 test reactor. A direct measurement of neutron absorption in a commercial reactor having boride coated pellets also confirms no absorber redistribution after one full cycle. Therefore, mechanical integrity of the absorber coating is reasonably assured.

## 2.4. Testing and Inspection Plans

As required by SRP Section 4.2, testing and inspection plans for new fuel should include verification of significant fuel design parameters. WCAP-10444 provides a brief discussion of the Westinghouse quality assurance program which provides for control over all activities affecting product quality including design, development, fabrication, testing and inspection, etc. The quality control program described in WCAP-9500 for OFA fuel is also applicable to VANTAGE-5 fuel. WCAP-10444 indicates that manufacturing control equipment for full scale production of VANTAGE 5 has not yet been finally designed. However, the equipment and process will be fully qualified by quality control to ensure that enrichment and stack length controls consistently meet design requirements, and that all other specification requirements such as moisture and hydrogen content are not adversely affected. We therefore conclude that the fuel testing and inspection program for VANTAGE 5 is acceptable.

## 3.0 NUCLEAR DESIGN

From the viewpoint of the nuclear design the VANTAGE 5 (17 X 17) fuel is for the most part essentially identical to the normal Westinghouse 17 X 17 Optimized Fuel Assembly (OFA). Thus, for the most part nuclear design related parameters and the methods for deriving them are the same as have been reviewed and approved for the OFA fuel (via WCAP-9500, Byron, Catawba and McGuire reviews) and requires no additional attention here. There are, however two principal direct design aspects of VANTAGE-5 different from OFA which directly affect the neutronics (and its review). These are the natural $UO_2$ end blanket ("axial blanket") design and the IFBA. Furthermore, there are additional aspects of the design or proposed operations with the fuel which are somewhat different (from WCAP-9500) and potentially affect the nuclear design review. These will also be discussed. The nuclear design bases for VANTAGE-5 are the same as for OFA, as stated in WCAP-9500, except for those related to extended burnup and a modification of the moderator temperature coefficient (MTC) basis. Thus, except for these modifications, the bases need not be discussed further. WCAP-10444 is primarily addressed to four loop 17 X 17 design, but the features may be applied to other cores and fuel bundle geometries (e.g., 14 X 14 or 15 X 15). This nuclear design review is similarly oriented, but the conclusions would likely also generally apply to other assembly arrays and core sizes.

### 3.1 Principal Nuclear Related Direct Design Changes

The use of natural enrichment $UO_2$ axial blanket zones at the ends of the fuel pins is not new in LWR fuel assembly design. BWRs have used this for some time. It is the first (extensive) use in Westinghouse design, however. The concept is derived largely from cycle economics considerations such as enrichment requirements and uranium utilization (via reduced end leakage), but the primary effect on nuclear aspects of operation is on axial power distribution. With all else the same the axial peaking factor increases as a result of the blankets, although the changes would be a function of burnup. The use of burnable poison with a length less than the initial enriched fuel length can, however, be designed to counteract the effects of the blanket, and

Westinghouse calculations indicate that they can produce distributions such that the peaks over the operating cycle are not significantly greater than normal. While neutronic analyses in this area of power distribution control may need to involve greater use of three dimensional (3D) methodologies, no new or novel problem is introduced. Axial xenon stability would be improved with just axial blanket addition, but is not significantly changed from normal with the combined blanket and part length burnable poison design.

The new burnable poison (IFBA) design involves the use of a boride region on the outer surface of the fuel pellets. This is the first extensive use by Westinghouse of burnable poison integral to the fuel pin rather than as discrete elements in control rod thimbles. It is also the first (extensive) use in LWRs of this type of design, since most previous fuel pin integral designs have used gadolinium mixed throughout the $UO_2$. The IFBA design will not in general have the boride the full length of the active fuel pin. The design length will depend on the specific reload application, but will generally be less than the enriched fuel length in order to improve the axial power distribution. This type of shorter and application-specific length design concept has been used by Westinghouse in the Wet Annular Burnable Absorber (WABA) design, has been approved in the review of that design and has been in use in operating reactors. The IFBA boride region is sufficiently thin neutronically that the flux attenuation through it is relatively small. The flux perturbations produced, therefore, are much less significant than with gadolinium designs which are essentially neutronically black at the effective radius corresponding to unburned gadolinium. Thus the neutronic calculation problems are much less significant for IFBA, and normal lattice methods can be used easily. The VANTAGE-5 design can use IFBA in as many fuel pins in an assembly as is needed to provide the required reactivity and (radial or axial) power distribution effects. Furthermore it does not preclude the use of more standard burnable poison pins in thimbles, although they would not normally be used, thus improving end of cycle parasitic capture and providing more moderation. The IFBA fuel pins can also be used in control rod assemblies where standard burnable poison can not.

The axial blanket and IFBA used together provide an only slightly altered, from standard, neutronic calculational problem. Normal design methods can be used. The standard methods have been compared against higher order transport theory calculations and have been found to be satisfactory for reactivity and power distribution for the blankets and IFBA. Incore measurements have been made of the effects of IFBA and its burnup, and standard methods have provided satisfactory comparison calculations. The new design does require a somewhat greater emphasis on 3D calculations (as does WABA use), but generally the same methodology as described in WCAP-9272, "Westinghouse Reload Safety Evaluation Methodology" can and will be used. The input parameters calculation for the INCORE code calculations used in power distribution measurements in operating reactors will require the use of 3D methods. Experience with this aspect of analysis has been provided in cores using WABA burnable poisons.

Our review of the power distribution and reactivity characteristics of the VANTAGE-5 axial blanket and IFBA neutronic design indicate that no significant adverse characteristic has been introduced compared to the OFA design, and that parameters for cycle analysis and operation can be calculated with standard methods, augmented only with a somewhat greater emphasis on 3D methods, especially for INCORE data. Other parameters of the neutronic design appear not to be significantly affected by these changes. Thus the direct nuclear design changes of the VANTAGE-5 assembly and associated areas are acceptable.

## 3.2 Indirect Neutronics Related Core Changes

There are additional neutronic aspects of the VANTAGE-5 fuel design and proposed or potential operation parameters which are extensions beyond the standard OFA (WCAP-9500) parameters. These include extended burnup, increased design $F_{\Delta H}$, low radial neutron leakage fuel load design and less negative MTC design.

The design basis burnup for the VANTAGE-5 fuel is extended well beyond the 36,000 MWD/MTU basis for the OFA fuel. It is, however, within the range

examined in the extended fuel burnup reviews which have been recently conducted for various fuel vendor designs and in particular for Westinghouse (using the report WCAP-10125 as a base). The review for Westinghouse is expected to be published in the near future. It will conclude that the burnup range including that proposed for VANTAGE-5 fuel and the methodologies for neutronic analyses in that range are acceptable. Neutronically the extended burnup affects the relative U235 and Pu isotopes fission rates, fission product poison productions, end of cycle parameters for transient analyses, relative power in low and high burnup assemblies and related uncertainties in power distribution. However, the nuclear parameter changes from standard to extended burnup ranges are sufficiently small that the methodologies and their uncertainties are not significantly changed. The Pu isotopes increase and to some extent U235 decrease with burnup are at or near saturation for the standard burnup and remain at similar levels in the extended range. The extended kinetics parameters are generally bounded by the standard conservative design selection. The reactivity coefficients used in event analyses for the VANTAGE-5 fueled cores considered a suitable range of extreme cycle lengths and average burnups and are acceptably bounding for the burnups basis. The variation in assembly power, fission products and the changes in fissile isotope ratios are not sufficiently increased to affect the acceptability of the calculational methods or the uncertainty assigned to peaking factors. (It is noted that peak power assemblies are not directly involved in extended burnup ranges.)

The intermediate flow mixer grids for the VANTAGE-5 fuel have no significant direct neutronic effect but there is an indirect effect of permitting a higher design $F_{\Delta H}$ value in order to take advantage of the improved DNB margin from the grids. The increased $F_{\Delta H}$ in turn permits greater design freedom in accounting for, e.g., increased assembly to assembly power differences from increased burnup, or low radial neutron leakage design to improve pressure vessel neutron fluence. This is an acceptable nuclear design process, assuming safety analyses are done with the increased $F_{\Delta H}$ and the Technical Specifications reflect the increase. The transient and accident event analyses in WCAP-10444, using limiting parameters from a transition or equilibrium core, did include an

increased $F_{\Delta H}$. This included the equilibrium core LOCA analysis, which also used an (artificially) increased total peaking, $F_Q$, of 2.55 (compared to the standard 2.32) to indicate maximum $F_Q$ levels before exeeding LOCA 2200°F clad temperature limits for the full VANTAGE-5 core. (Transition core penalties would require a lower $F_Q$.) Other safety evaluations assumed standard axial power distributions and $F_Q$.

Low radial neutron leakage designs are not new and have been and are acceptable. But the increased design $F_{\Delta H}$ capability provides an easier design problem. Similarly the use of a less negative (more positive) MTC is not new. Several Westinghouse reactors have submitted safety analyses with positive MTC used throughout the non-LOCA analyses and have Technical Specifications providing for such a coefficient up to some power level, thus far not including full power. The VANTAGE-5 safety evaluations used a positive MTC for non-LOCA events, but not for the LOCA analyses. Thus the possibility of use for VANTAGE-5 (full and transitional) cores is acceptable with a Technical Specification similar to previously accepted models.

Other nuclear design characteristics and parameters not explicitly discussed here are sufficiently similar to OFA core design (WCAP-9500) that previous reviews for that fuel are applicable. Since VANTAGE-5 fuel will be used in transition cores from STD or OFA loaded cores to full VANTAGE-5 cores, Westinghouse has examined neutronic parameters from all three types and has provided limiting nuclear parameters (Table 3.2 of WCAP-10444) for use in safety analyses which should in general be applicable to transition and full core loadings. (Reload methodology requires that parameters be checked for the specific case.) The review has indicated that these parameters are acceptable for such use. The safety evaluations of WCAP-10444 used these values along with specific values for MTC and peaking factors which are generally loading dependent and are intended to be specific to a given reactor. It is expected that these latter parameters (e.g., MTC, ejected rod worth and peaking, $F_Q$, $F_{\Delta H}$) will be provided and justified for specific reactors.

## 4.0 THERMAL AND HYDRAULIC DESIGN

### 4.1 Core Thermal Hydraulic Analysis

The principal criterion for the thermal hydraulic design of reactor fuel is the avoidance of thermally induced fuel damage during normal operation and anticipated operational occurrences (AOO). To this end, the thermal-hydraulic design basis requires that there must be at least a 95% probability at 95% confidence level that departure from nucleate boiling (DNB) will not occur on the limiting power rod during normal operation and AOOs. This requirement is met by limiting the DNB ratio to the DNBR limit associated with the critical heat flux correlation used. This design basis is the same as that for the OFA fuel. However, since the VANTAGE-5 fuel has three intermediate flow mixer grids in addition to the existing structural grids in the OFA fuel, the thermal-hydraulic characteristics have changed and the WRB-1 CHF correlation used in the OFA design is not applicable to VANTAGE-5. Westinghouse has therefore developed the WRB-2 CHF correlation based on CHF tests pertinent to VANTAGE-5 fuel. The DNBR limit for WRB-2 is 1.17, the same as for WRB-1. We have reviewed the WRB-2 correlation and the DNBR limit of 1.17 and found them acceptable. The evaluation of WRB-2 will be addressed in Section 4.2.

The thermal hydraulic calculation is performed using the THINC-IV (Ref. 6) open lattice thermal hydraulic code which takes into account the effects of interchannel cross flow and turbulent mixing. The fluid mechanics design and empirical correlations used in VANTAGE 5 is the same as those for OFA design described in WCAP-9500. The thermal diffusion coefficient (TDC) is used as a measure of the rate of heat exchange by fluid mixing between adjacent channels. The IFM grids used in VANTAGE-5 fuel have the same mixing characteristics as the OFA mixing vane grids. However, the VANTAGE-5 mixing characteristics are improved due to the presence of IFM grids which reduce the grid spacing. For thermal hydraulic analysis, the same TDC used in OFA is used for VANTAGE-5 and is therefore conservative.

Like OFA fuel, the VANTAGE-5 thermal hydraulic analysis uses the improved thermal design procedure (ITDP). The ITDP method is based upon a statistical combination of the effects on DNBR of uncertainties of the plant parameters such as reactor coolant flow rate, core power, core coolant inlet temperatures, system pressure and hot channel factors, etc. The nominal values of these parameters are used in the safety analysis and the effect of the uncertainties on DNBR is added to the 95/95 DNBR limit specified by the WRB-2 correlation to establish a design DNBR limit. The ITDP method described in WCAP-8567 (Ref. 7) has been approved for use in licensing applications subject to the certain restrictions. One of the restrictions requires that if the sensitivity factors are changed as a result of correlation change, then the use of an uncertainty allowance for application of Equation 3-2 (WCAP-8567) must be re-evaluated and the linearity assumption of WCAP-8567 must be validated. Westinghouse in response to a staff question (Ref. 5) has performed the required re-evaluation and validation using the same methods described in the staff safety evaluation report for WCAP-8567. The results show that no additional uncertainty allowance is required based on the typical sensitivity factors specified in Tables 4-4 and 4-5 of WCAP-10444, and that the linearity assumption is indeed conservative. Therefore, the ITDP method described in WCAP-8567 in conjunction with WRB-2 correlation is acceptable.

Tables 4-4 and 4-5 also provide typical values of the means, uncertainties and sensitivity factors of the plant parameters. A typical value of the design DNBR limits of 1.29 and 1.288, respectively, are derived for the typical cell and thimble cell. However, since the uncertainties of some parameters are plant-specific, those plants using ITDP are required to abide by the restrictions imposed on WCAP-8567 and perform an analysis to establish a plant-specific design DNBR limit. In addition, information regarding the measurement uncertainties for pressurizer pressure, power, coolant flow rate and temperature must be provided as specified below.

A block diagram depicting sensor, process equipment, computer and readout devices for each parameter channel used in the uncertainty analysis should be provided. Within each element of the block diagram, the accuracy, drift, range, span, operating limits and setpoints should be identified. The overall accuracy of each channel transmitter to final output and the minimum acceptable accuracy for use with the new procedure should also be identified. In addition, the overall accuracy of the final output value and maximum accuracy requirements for each input channel for this final output device should be identified.

In addition to the design DNBR which must be met in the plant safety analysis to ensure the specified acceptable fuel design limit on DNBR is not violated, a safety analysis DNBR limit is used in the safety analysis. This safety analysis DNBR limit is derived from the design DNBR limit with an additional specific allowance (e.g., 30%). The use of safety analysis DNBR in the thermal-hydraulic design provides certain safety margin which can be credited to compensate for certain required penalties, improved fuel management, or increased plant availability. However, the licensees referencing WCAP-10444 should incorporate in the bases of their plant Technical Specifications the plant-specific safety analysis DNBR limit, the DNBR allowance and the amount of allowance that has been used.

### 4.2 WRB-2 CHF Correlation

In conjunction with the development of the VANTAGE 5 fuel design, Westinghouse developed a new CHF correlation designated WRB-2 to predict DNB performance of this fuel design and other fuel designs which use the same mixing vane design as the 17 X 17 standard fuel mixing vane design. The VANTAGE 5 fuel design features an increase in DNB margin over the 17 X 17 OFA design by the addition of IFM grids. To quantify this increase in DNB performance, a test program was conducted at Columbia University using the VANTAGE 5 fuel design. As described in Appendix A.2.0, "DNB Test Program," of WCAP-10444, the test program consisted of two test series utilizing 5 X 5, 14-foot electrically heated rod

bundles. The detailed bundle geometry is similar to that of 17 X 17 OFA used in earlier DNB tests. A total of 684 data points from 11 test series consisting of the new data, the 5 X 5 OFA test bundle data and that from previous DNB tests involving 5 X 5 test bundles of various lengths but employing the standard fuel mixing vane design, were used as the data base for the development of WRB-2. The WRB-2 correlation is described in Figure A-9 of WCAP-10444 and has a minimum DNBR (95/95) limit of 1.17.

The review of WRB-2 consists of independent thermal hydraulic audit calculations to verify the correctness of the predicted local conditions and CHF predictions, and a statistical analysis to verify the correctness of the DNBR limit.

The local conditions and CHF audit calculations were performed with the COBRA-IV code (Ref. 8). The input data on the test bundle geometry, grid loss coefficients for the various grid designs and subchannel types, flow, heat flux and axial power distributions are obtained from the references provided in WCAP-10444 and the information provided by Westinghouse at staff request. For each data point, the ratio of the measured CHF to the predicted CHF is calculated. All 684 data points used in the WRB-2 development are calculated. Since Westinghouse analysis was performed with the THINC-IV thermal hydraulic code, the results of calculations using COBRA-IV/WRB-2 are not expected to be identical to those presented in WCAP-10444 but provide a general indication of the correctness of Westinghouse analysis. A summary of comparison of the measured to predicted CHF ratios by Westinghouse and the audit calculations is shown in Table 1 of this report which shows very good agreement.

The independent statistical analysis is performed with the measured to predicted CHF ratios from the thermal hydraulic calculation results. The hypothesis that data from all 11 test series come from a single population is tested in two ways. First a chi-square test of homogeneity is performed. The observed value $X^2 = 111.6$, with a degree of freedom of 60 is highly significant. This indicates that the 11 data sets do not all come from the same population. Second, an analysis of variance is performed to test the hypothesis that the

means of all 11 populations are equal. With the among-series degree of freedom of 10 and the within-series degree of freedom of 673, the F-statistic is 4.51. This value is also highly significant, indicating that not all means are equal. On the basis of these analyses, we conclude that the data do not all represent a single population. However, the estimated standard deviation from one-way analysis of variance is 0.0773, while the estimate obtained by treating the data as a single population is 0.0793. This indicates that the error introduced into variance estimates by assuming that the data are all from the same population is negligible.

The 95/95 probability/confidence DNBR limit is calculated using the formula

$$DNBR_{95/95} = \frac{1}{(M/P)_{avg} - KS}$$

where (M/P)avg is the population mean of measured to predicted CHF ratios, S is standard deviation of the M/P data and K is a tolerance multiplier which provides the 95/95 probability/confidence limit.

When the test assembly means are found to be unequal, the "total variance" is estimated by means of the formula

$$S^2_{total} = S^2_w + S^2_A,$$

where $S^2_w$ is the mean square within test series component of variance and $S^2_A$ is the "variance of the test series average." An effective degree of freedom $f_T$ is computed for $S^2_{total}$ from Satterthwaite's formula. The total variance and effective degree of freedom should be used in the DNBR limit calculates as has been done for WRB-1. This is also done for WRB-2. However, the calculation of the variance of the test series means $S^2_A$ was performed by Westinghouse with an assumption of an equal number of data points for each test series. In response to a staff question (Ref. 9), Westinghouse provided a correct formula for calculating $S^2_A$ without the assumption of equal number of data points. It also

shows that the equal data assumption results in higher total variance and fewer effective degrees of freedom. Therefore, the resulting DNBR limit of 1.17 is conservative. We have concluded that WRB-2 with a DNBR limit of 1.17 is acceptable for application to VANTAGE 5 with the following range of applicability.

RANGE OF VARIABLES

| | |
|---|---|
| Pressure | $1440 \leq P \leq 2490$ psia |
| Local Mass Velocity | $0.9 \leq G_{loc}/10^6 \leq 3.7$ lb/ft$^2$hr |
| Local Quality | $-0.1 \leq X_{loc} \leq 0.3$ |
| Heated Length, Inlet to CHF Location | $L_h \leq 14$ feet |
| Grid Spacing | $10 \leq g_{sp} \leq 26$ inches |
| Equivalent Hydraulic Diameter | $0.37 \leq d_e \leq 0.51$ inches |
| Equivalent Heated Hydraulic Diameter | $0.46 \leq d_h \leq 0.59$ inches |

## TABLE 1

### WRB-2 CHF CORRELATION - STATISTICAL RESULTS

| Rod O.D. (inch) | Length (feet) | Grids (in.) | Heat Flux Profile | Assembly Geometry | Data Points | WESTINGHOUSE M/P Ratio | WESTINGHOUSE Standard Deviation | AUDIT CALCULATIONS M/P Ratio | AUDIT CALCULATIONS Standard Deviation | Reference | Data Table |
|---|---|---|---|---|---|---|---|---|---|---|---|
| | 14 | 10 | Cosine | TYP-5X5 | 51 | 0.9861 | 0.0758 | 0.9721 | 0.0705 | WCAP-10444 | A-2 |
| | 14 | 10 | Cosine | TYP-5X5 | 31 | 1.0097 | 0.0680 | 1.0181 | 0.0623 | WCAP-10444 | A-3 |
| | 14 | 20 | Cosine | TYP-5X5 | 63 | 0.9961 | 0.0946 | 0.9852 | 0.0952 | WCAP-9401 | A-4 |
| | 14 | 20 | Cosine | THM-5X5 | 38 | 0.9832 | 0.0599 | 0.9946 | 0.0600 | WCAP-9401 | A-5 |
| 0.374 | 8 | 22 | Uniform | TYP-5X5 | 67 | 1.0316 | 0.0897 | 1.0348 | 0.0753 | WCAP-8536 | A-6 |
| 0.374 | 14 | 22 | Uniform | TYP-5X5 | 71 | 1.0095 | 0.0664 | 0.9924 | 0.0626 | WCAP-8536 | A-7 |
| 0.374 | 14 | 22 | Cosine | TYP-5X5 | 74 | 0.9893 | 0.0822 | 0.9691 | 0.0773 | WCAP-8536 | A-8 |
| 0.374 | 14 | 22 | Cosine | THM-5X5 | 70 | 0.9884 | 0.0775 | 0.9729 | 0.0765 | WCAP-8536 | A-9 |
| 0.374 | 8 | 26 | Uniform | TYP-5X5 | 78 | 1.0198 | 0.0810 | 0.9890 | 0.0753 | WCAP-8926 | A-10 |
| 0.374 | 8 | 26 | Uniform | THM-5X5 | 68 | 1.0398 | 0.1062 | 1.0092 | 0.0942 | WCAP-8926 | A-11 |
| 0.374 | 14 | 26 | Uniform | TYP-5X5 | 73 | 0.9914 | 0.0823 | 0.9761 | 0.0785 | WCAP-8926 | A-12 |
| | | | All Data | | 684 | 1.0051 | 0.0847 | 0.9907 | 0.0793 | | |

### 4.3 Core Flow Design

WCAP-10444 specifies as a design basis that at least a certain percent of the reactor coolant flow should pass through the fuel rod region of the core for cooling. This design basis assures that for RCS flow rate specified in the plant Technical Specifications the core bypass flow should be accounted for. Core bypass flow is the sum of the flow through rod cluster control guide thimbles, head cooling flow, baffle leakage and leakage to the vessel outlet nozzle. The uncertainties associated with the calculation of bypass flow should be accounted for in the safety analysis. For the analysis using ITDP, the uncertainties are statistically combined with uncertainties of other parameters to obtain the design DNBR limit. For analysis not using ITDP, the uncertainties are accounted for by reducing the active core flow.

The bypass flow is dependent upon core geometry and is plant specific. Higher fuel assembly pressure drop results in larger bypass flow. The presence of the IFM grids in VANTAGE 5 increases the fuel assembly pressure drop as well as bypass flow. Westinghouse indicates that the total bypass flow used in the safety analysis of the reference plant is conservative. However, for licensees referencing WCAP-10444, a plant specific analysis is required to determine the correct bypass flow. Correct or conservative active core flow rate should be used in safety analysis.

### 4.4. Thermohydrodynamic Stability

One of the design bases for VANTAGE-5 is to ensure that the mode of operation with Conditions I and II events will not lead to thermohydrodynamic instability. The flow excursion (Ledinegg) instability generally does not exist in a PWR. This is due to the inherent characteristics of RCS pump head-capacity curve having a negative slope and system hydraulic pressure drop-flow curve having a positive slope in PWR. The staff has previously accepted the stability evaluation for OFA fuel based on the past operating

experience, flow stability experiments and the inherent thermal hydraulic characteristics of Westinghouse open channel core configuration. Use of VANTAGE-5 fuel will not change these characteristics. We therefore conclude that flow instability will not be a problem for VANTAGE 5.

## 4.5 Rod Bow Effect on DNBR

Fuel rod bowing results in flow channel gap closure and reduction of critical heat flux and DNBR. This CHF effect is accounted for through a rod bow DNBR penalty. The method described in WCAP-8691, Revision 1 (Ref. 10), has been approved for the rod bow penalty calculations. A scaling factor dependent on the cladding moment of inertia and span length between grids is used for the channel closure calculation when extrapolating from the channel closure data of different fuel assembly geometry. Since three IFM grids have been installed between the Zircaloy grids in the VANTAGE-5 fuel, the topical report asserts that the grid spacing and, therefore, the channel closure are reduced. In response to a staff question (Ref. 5) on whether the presence of the "non-structural" IFM grids provides the necessary support in the reduction of rod bowing. Westinghouse indicates that the term "non-structural" is a misnomer and that the IFM grids do provide a positive pin type support and have a small stiffening effect near the top of the fuel assembly. Using the approved scaling factor, the predicted channel closure in the limiting span of the VANTAGE 5 fuel is less than 50%. Experimental data (Ref. 11) have shown that there is no CHF reduction for channel closure less than 50%. Therefore no rod bow penalty is required for the VANTAGE 5 fuel.

## 4.6 Transitional Mixed Core DNBR Effect

As a reactor is reloaded with VANTAGE-5 fuel prior to a full core of VANTAGE-5 fuel, there are transitional cycles when both VANTAGE-5 and remaining fuel (such as standard fuel or OFA fuel) will co-exist in the core. The differences between the adjacent fuel assemblies in the hydraulic resistance

characteristics such as spacer grid designs, flow areas and the presence of VANTAGE-5 IFM grids result in local hydraulic mismatches. Such a hydraulic mismatch results in localized flow redistribution due to the open core configuration. While beneficial to STD or OFA fuel due to lower grid resistance, the interbundle cross flow is detrimental to VANTAGE-5 fuel. In the safety analysis, a full core of VANTAGE 5 fuel is assumed. A mixed core DNBR penalty is applied during the transitional mixed core configuration to account for the detrimental effect of interbundle cross flow on the VANTAGE-5 fuel. Westinghouse has performed sensitivity studies in the same way approved previously for the standard fuel/OFA fuel mixed core (Ref. 12) to determine the penalty factor required for the VANTAGE-5 fuel. Since the hydraulic mismatch between VANTAGE 5/OFA assemblies is less complex than that between the VANTAGE-5/STD fuel assemblies, the penalty factor determined from VANTAGE-5/STD mixed core is bounding for VANTAGE-5/OFA mixed core. The value of the penalty factor is described in WCAP-10444 and is later revised to 11% in a response to a staff question (Ref. 5). This value (11%) is generic for 17 X 17 fuel assemblies. For other fuel types such as 14 X 14 or 15 X 15 fuel assemblies, the mixed core penalty factor should be determined separately. As described in Section 4.1, when the ITDP is used in safety analysis, there is a generic margin between the design DNBR limit and plant safety analysis DNBR. This safety margin is enough to compensate for the penalty factor required for mixed core configuration. However, for these plants having the VANTAGE-5 fuel, both the plant specific safety margin and mixed core penalty factor should be addressed in Technical Specification bases.

### 4.7 Low Leakage Loading Core Design

In order to improve nuclear design flexibility and fuel economy, and to reduce pressure vessel fluence, some cores containing VANTAGE 5 fuel will be designed with burned blankets, i.e., the burned fuel assemblies will be placed in the core periphery. To facilitate this, the hot channel enthalpy rise factor $F_{\Delta H}$ is raised from the current 1.55 to a higher value (Westinghouse Proprietary). The increase in $F_{\Delta H}$ is made possible because of the thermal margin gained from

the use of the IFM grids resulting in higher CHF for VANTAGE 5 fuel. For the OFA and STD fuel, thermal margin could be gained from the rod bow penalty reduction as a result of change of rod bow calculation from the NRC interim method (Ref. 13) to the method of WCAP-8691, Revision 1. This thermal margin gain from rod bow penalty reduction could be used to increase the normal operation $F_{\Delta H}$. However, plant specific analyses are required with appropriate $F_{\Delta H}$ to ensure that the specified acceptable fuel design limits will not be violated during normal operation and anticipated operational occurrences.

### 4.8 Other VANTAGE 5 Design Feature Effects

Two VANTAGE-5 design features, namely, axial blanket and IFBA, result in axial zoning in the fuel rod which perturbs core axial power distribution and affects the DNBR evaluation. The axial blanket reduces power at the ends of the rod and causes a slight increase in axial power peaking. Though IFBA flattens the axial power distribution and reduces the local peaking, the IFBA burns out during irradiation. Therefore, the net result of axial blanket and IFBA is a slight increase in power peaking during core operation. Westinghouse has examined this effect and found that the design axial power distribution used in the determination of the overtemperature $\Delta T$ setpoint is still conservative.

Another VANTAGE 5 design feature is the reconstitutable top and bottom nozzles which results in changes in hydraulic resistance at the inlet and outlet of the active fuel region from the previous standard and OFA designs. However, the hydraulic tests performed by Westinghouse have shown that the changes in nozzle design have a negligible impact on the inlet flow and outlet pressure distributions. Therefore, the effect on DNBR is negligible.

## 5.0 SAFETY EVALUATION

A safety evaluation is necessary to assess the impact of the VANTAGE 5 design features such as axial blankets, IFBA, IFM grid and reconstitutable top nozzle, etc., and the core reload strategy having a higher $F_{\Delta H}$ and a positive moderator

coefficient. Table 5-1 of WCAP-10444 lists the non-LOCA accidents which are most sensitive to these changes. They are:

(a) Main Steamline Rupture;
(b) Loss of Load/Turbine Trip;
(c) Complete Loss of Forced Reactor Coolant Flow;
(d) Single Reactor Coolant Pump Locked Rotor;
(e) Uncontrolled RCCA Bank Withdrawal at Power; and
(f) RCCA Ejection.

As an evaluation of the impact of the VANTAGE 5 design features, analyses of these accidents and loss of coolant accident (LOCA) are performed using the methods described in WCAP-9272, "Westinghouse Reload Safety Analysis Methodology." The analyses are performed with a typical 4-loop reference plant. All analytical procedures and computer codes used in the non-LOCA safety analysis are the same as those described in WCAP-9500 for OFA, except that the WRB-2 CHF correlation is used for VANTAGE 5. For most accidents which are DNB limited, the improved thermal design procedure is used in the analyses. Table 5-3 of WCAP-10444 summarizes the initial condition and computer codes used in the non-LOCA accident analyses and also shows which accidents employ a DNB analysis using ITDP. For the large break LOCA, the analysis is performed with a modified version of the 1981 Westinghouse ECCS evaluation model (Ref. 14). The modified version of ECCS evaluation model uses the BART code (Ref. 15) to calculated the reflood heat transfer coefficient in place of the FLECHT reflood heat transfer correlation normally used in the LOCTA code (Ref. 16). BART is a mechanistic core heat transfer code. The code with the grid rewet model has been approved for LOCA analysis. However, the drop break up model is not approved. In response to a staff question (Ref. 5), Westinghouse confirms that in every application of BART supporting WCAP-10444, a version of the code was used that explicitly excluded the drop breakup model.

For a transitional mixed core where the VANTAGE 5 fuel co-exists with either STD or OFA fuel, the local hydraulic resistance mismatch between the adjacent

assemblies of different design has a detrimental effect on VANTAGE 5. The flow redistribution due to hydraulic resistance mismatch results in a reduction of reflood steam flow rate and a increase in peak cladding temperature in the VANTAGE 5 fuel . For a transitional core, the LOCA analysis is performed with a full core of VANTAGE 5 fuel. The resulting PCT is then increased by adding a mixed core penalty. This approach has been reviewed and approved during the review of WCAP-9500. Westinghouse has performed the same analysis for VANTAGE 5 and the conservative nature of the required mixed core penalty on PCT was determined for VANTAGE 5. We conclude that this approach is acceptable.

The results of analyses as presented in WCAP-10444 generally demonstrate that the SAFDLs are not violated and the criteria specified in 10 CFR 50.46 can be met. Since the safety analyses of WCAP-10444 are not intended to be directly applicable to specific plants but were submitted for demonstrating acceptability of the fuel in a generic sense, we find the safety analyses presented in WCAP-10444 acceptable. The licensees referencing WCAP-10444 are required to perform plant-specific safety analyses to ensure the safety criteria are met.

There are, however, several deficiencies in the safety analyses of WCAP-10444. These are discussed below and should be corrected in the plant-specific safety analyses.

(1) For the steam system piping failure accident, the analysis assumes that the offsite power is available. This assumption is not in compliance with the SRP which requires that assumptions as to the loss of offsite power and the time of loss should be made to study their effects on the consequence of the accident. This requirement is also specified in General Design Criterion 17, 10 CFR 50 Appendix A.

(2) With regard to the reactor coolant pump seizure (locked rotor) accident, WCAP-10444 states that although some rods may enter DNB for a short period

of time, fuel failure for the locked rotor/shaft seizure incident is treated on a mechanistic basis, thus eliminating the need to automatically equate DNB with fuel failure. WCAP-10444 also indicates that the mechanistic methods are allowed by SRP and that the temperature-time criteria of NUREG-0562 (Ref. 17) will be used in the fuel failure calculation on a plant specific basis. However, NUREG-0562 prepared by the Office of Nuclear Reactor Research, has not been applied in licensing. The temperature-time criteria noted in the report are from another reference (Pawel, Journal of Nuclear Materials, Volume 50, Page 247-258) which the NRC has not endorsed for this type of application.

It has been and continues to be the NRC position that cladding failure is assumed to occur when the fuel rod DNBR is less than the safety limit. This position provides a degree of conservatism which varies with each anticipated operational occurrence and accident. This conservatism covers such analyses uncertainties as:

(i) Variation in history of an actual event from that idealized in the accident analysis

(ii) Uncertainties in the thermal hydraulics such as

- (a) prediction of heat transfer coefficient in film boiling
- (b) prediction of subchannel flow in film boiling
- (c) prediction of extent and location of DNB
- (d) multiple DNB on the same rod
- (e) changes in fuel bundle geometry during event due to overheating
- (f) uncertainties in prediction of transients, non-isothermal zirconium/water reaction kinetics, etc., (see the above reference by Pawel).

In addition to these, there are questions related to the prototypicality of the NUREG-0562 data. EPRI report NP 1999 which addresses post-DNB effects in PWRs has identified several areas which would be appropriate subjects for future research before application to licensing (see Table 5-1 of this report). While we are not taking the position that all items in Table 5-1 must be addressed, it is clear from this industry report that more review would be necessary before approval of a post-DNB fuel failure criterion.

Therefore, the licensees referencing WCAP-10444 should assume that all fuel which experiences a DNBR of less than the DNBR limit fails and calculate the offsite dose consequences. In the offsite dose analysis the licensee should assume maximum Technical Specification pre-accident coolant activity and steam generator leakage. Single failures should be considered including a stuck open secondary relief valve. Loss of offsite power should be assumed per GDC-17. The effect of steam generator tube uncovery on the offsite dose consequences caused by operator action to isolate the affected steam generator should also be considered in the analysis.

(3) The large break LOCA analysis performed for a full core of VANTAGE 5 fuel using a total peaking factor $F_Q$ of 2.55 has shown a peak cladding temperature of 2194°F which is below the acceptance criterion of 2200°F. However, for a transitional mixed core configuration, addition of the mixed core penalty on PCT will result in the PCT exceeding the 2200°F acceptance criterion per 10 CFR 50.46. The plant specific analysis must be done to ensure that with appropriate value of $F_Q$, the 2200°F criterion can be met even for the transitional mixed core.

## 6.0 SUMMARY AND CONCLUSION

The staff has reviewed the topical report WCAP-10444 and concludes that the report is acceptable for reference for the Westinghouse VANTAGE 5 fuel design. This acceptability is subject to the following conditions:

1. The statistical convolution method described in WCAP-10125 for the evaluation of initial fuel rod to nozzle growth gap has not been approved. This method should not be used in VANTAGE 5.

2. For each plant application, it must be demonstrated that the LOCA/seismic loads considered in WCAP-9401 bound the plant in question; otherwise additional analysis will be required to demonstrate the fuel assembly structural integrity.

3 An irradiation demonstration program should be performed to provide early confirmation performance data for the VANTAGE 5 design.

4. For those plants using the ITDP, the restrictions enumerated in Section 4.1 of this report must be addressed and information regarding measurement uncertainties must be provided.

5. The WRB-2 correlation with a DNBR limit of 1.17 is acceptable for application to 17 X 17 VANTAGE 5 fuel. Additional data and analysis are required when applied to 14 X 14 or 15 X 15 fuel with an appropriate DNBR limit. The applicability range of WRB-2 is specified in Section 4.2.

6. For 14 X 14 and 15 X 15 VANTAGE 5 fuel designs, separate analyses will be required to determine a transitional mixed core penalty. The mixed core penalty and plant specific safety margin to compensate for the penalty should be addressed in the plant Technical Specifications Bases.

7. Plant specific analysis should be performed to show that the DNBR limit will not be violated with the higher value of $F_{\Delta H}$.

8. The plant-specific safety analysis for the steam system piping failure event should be performed with the assumption of loss of offsite power if that is the most conservative case.

9. With regard to the RCS pump shaft seizure accident, the fuel failure criterion should be the 95/95 DNBR limit. The mechanistic method mentioned in WCAP-10444 is not acceptable.

10. If a positive MTC is intended for VANTAGE 5, the same positive MTC consistent with the plant Technical Specifications should be used in the plant specific safety analysis.

11. The LOCA analysis performed for the reference plant with higher $F_Q$ of 2.55 has shown that the PCT limit of 2200°F is violated during transitional mixed core configuration. Plant specific LOCA analysis must be done to show that with the appropriate value of $F_Q$, the 2200°F criterion can be met during use of transitional mixed core.

12. Our SER on Westinghouse's extended burnup topical report WCAP-10125 is not yet complete; the approval of the VANTAGE 5 design for operation to extended burnup levels is contingent on NRC approval of WCAP-10125. However, VANTAGE 5 fuel may be used to those burnups to which Westinghouse fuel is presently operating. Our review of the Westinghouse extended burnup topical report has not identified any safety issues with operation to the burnup value given in the extended burnup report.

13. Recently, a vibration problem has been reported in a French reactor having 14-foot fuel assemblies; vibration below the fuel assemblies in the lower portion of the reactor vessel is damaging the movable incore instrumentation probe thimbles. The staff is currently evaluating the implications of this problem to other cores having 14 foot long fuel bundle assemblies. Any limitations to the 14-foot core design resulting from the staff evaluation must be addressed in plant-specific evaluations.

## 7.0 REFERENCES


1. WCAP-10125, "Extended Burnup Evaluation of Westinghouse Fuel," dated July 1982.

2. WCAP-9500-A, "Reference Core Report 17 X 17 Optimized Fuel Assembly," dated May 1982.

3. Letter from E. P. Rahe, Jr. (Westinghouse) to C. O. Thomas (NRC), "Reference to NRC Request Number 3 on WCAP-10125," NS-NRC-85-3018, dated March 14, 1985.

4. WCAP-9401-P-A, "Reference Core Report 17 X 17 Optimized Fuel Assembly," dated May 1982.

5. Letter from E. P. Rahe, Jr. to C. O. Thomas, "Response to Request Number 1 for Additional Information on WCAP-10444 entitled, 'VANTAGE-5 Fuel Assembly'" (Proprietary), NS-NRC-85-3014, dated March 1, 1985.

6. WCAP-7956, "THINC-IV - An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," 1973.

7. WCAP-8567, "Improved Thermal Design Procedure," dated July 1975.

8. BNWL-1962 - UC-32, "COBRA-IV-1: An Interim Version of COBRA for Thermal-Hydraulic Analysis of Rod Bundle Nuclear Fuel Elements and Cores," Battelle Pacific Northwest Laboratories, dated March 1976.

9. Letter from E. P. Rahe, Jr. (Westinghouse) to C. O. Thomas (NRC), "Follow-up Response to Request Number 1 for Additional Information for WCAP-10444 entitled, 'VANTAGE 5 Fuel Assembly - Reference Core Report'," (Proprietary), NS-NRC-85-3033, dated May 8, 1985.

10. WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation," dated July 1979.

11. E. S. Markowski, et al., "Effect of Rod Bowing on CHF in PWR Fuel Assemblies," ASME Peper 77-HT-91.

12. Letter from E. P. Rahe, Jr. (Westinghouse) to J. R. Miller (NRC), "Supplement to WCAP-9500 and WCAP-9401/9402 NRC Safety Evaluation Report (SER) Mixed Core Compatibility Items - Supplemental Information," NS-EPR-2643, dated August 17, 1982.

13. Memorandum from D. F. Ross and D. G. Eisenhut to D. B. Vassallo and K. R. Goller, "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors," dated February 16, 1977.

14. WCAP-9220-A, Revision 1, "Westinghouse ECCS Evaluation Model, 1981 Version," dated February 1981.

15. WCAP-9561, "BART-A1: A Computer Code for the Best Estimate Analysis of Reflood Transients," dated January 1980.

16. WCAP-8301, "LOCTA-IV Program: Loss of Coolant Transient Analyses," dated June 1974.

17. WCAP-8170, "Calculated Model for Core Reflood After a Loss of Coolant Accident (Westinghouse Reflood Code)," dated June 1974.

18. R. Van Houten, "Fuel Rod Failure As a Consequence of Department from Nucleate Boiling or Dryout," NUREG-0562, 1979.