

Attachment i

SAFETY EVALUATION
FOR THE
BYRON/BRAIDWOOD STATIONS UNITS 1 AND 2
TRANSITION TO WESTINGHOUSE 17x17 VANTAGE 5 FUEL

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BYRON/BRAIDWOOD STATIONS UNITS 1 AND 2
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1.0 INTRODUCTION AND CONCLUSIONS

The Byron/Braidwood Stations Units 1 and 2 are currently operating with a Westinghouse 17x17 Optimized Fuel Assembly (OFA) core. For subsequent cycles, it is planned to refuel and operate the Byron/Braidwood Stations Units 1 and 2 with the Westinghouse VANTAGE 5 improved fuel design. As a result, future transition core loadings would range from approximately 50%-70% OFA, and 30%-50% VANTAGE 5, to eventually an all VANTAGE 5 fueled core. The VANTAGE 5 fuel assembly was designed as a modification to the current 17x17 OFA design, Reference 1.

The VANTAGE 5 design features were conceptually packaged to be licensed as a single entity. This was accomplished via the NRC review and approval of the "VANTAGE 5 Fuel Assembly Reference Core Report," WCAP-10444-P-A, Reference 2. The initial irradiation of a fuel region containing all the VANTAGE 5 design features occurred in the Callaway Plant in November 1987. The Callaway VANTAGE 5 licensing submittal was made to the NRC on March 31, 1987 (ULNRC-1470, Docket No. 50-483). NRC approval was received in October 1987. Several of the VANTAGE 5 design features, such as axial blankets, reconstitutable top nozzles, extended burnup modified fuel assemblies and Integral Fuel Burnable Absorbers have been successfully licensed as individual design features and are currently in operating Westinghouse plants. The Byron/Braidwood Stations Units 1 and 2, commencing with Byron Unit 1, Cycle 4 will be operating with a core containing the following VANTAGE 5 features: Integral Fuel Burnable Absorbers (IFBAs), Intermediate Flow Mixer (IFM) grids, Reconstitutable Top Nozzle (RTN currently operating in Byron 1, Cycle 3), fuel assemblies modified for extended burnup (currently operating in Byron 1, Cycle 3), and axial blankets. In addition, both the Byron and Braidwood Stations Units 1 and 2 VANTAGE 5 fuel assemblies contain the Debris Filter Bottom Nozzle (DFBN).

A brief summary of the VANTAGE 5 design features and major advantages of the improved fuel design are given below. These features and figures illustrating the design are presented in more detail in Section 2.0.

Integral Fuel Burnable Absorber (IFBA) - The IFBA features a zirconium diboride coating on the fuel pellet surface on the central portion of the enriched UO_2 pellet stack. In a typical reload core, approximately one third of the fuel rods in the feed region are expected to include IFBAs. IFBAs provide power peaking and moderator temperature coefficient control.

Intermediate Flow Mixer (IFM) Grids - Three IFM grids located between the three upper most Zircaloy grids provide increased DNB margin. Increased margin permits an increase in the design basis $F_{\Delta H}^N$ and F_Q .

Reconstitutable Top Nozzle - A mechanical disconnect feature facilitates the top nozzle removal. Changes in the design of both the top and bottom nozzles increase burnup margins by providing additional plenum space and room for fuel rod growth.

Extended Burnup - The VANTAGE 5 fuel design will be capable of achieving extended burnups. The basis for designing to extended burnup is contained in the approved Westinghouse extended burnup topical WCAP-10125-P-A, Reference 6.

Blankets - The axial blanket consists of a nominal six inches of natural UO_2 pellets at each end of the fuel stack to reduce neutron leakage and to improve uranium utilization. For VANTAGE 5 reload cores, low leakage loading patterns (burned blankets) are shown to further improve uranium utilization and provide additional pressurized thermal shock margin.

This submittal is to serve as a reference safety evaluation/analysis report for the region-by-region reload transition from the present Byron/Braidwood OFA fueled core to an all VANTAGE 5 fueled core. The submittal examines the differences between the VANTAGE 5 and the OFA fuel assembly designs and evaluates the effect of these differences on the cores during the transition to an all VANTAGE 5 core. The VANTAGE 5 core evaluation/analyses were performed at a core thermal power level of 3411 MWt with the following conservative assumptions made in the safety evaluations: a full power $F_{\Delta H}^N$ of 1.65 for the VANTAGE 5 fuel and 1.55 for the OFA fuel, an increase in the

maximum F_0 to 2.50 and 10% plant total steam generator tube plugging for both the Byron/Braidwood Stations Units 1 and 2. LOCA transients were analyzed at 15% steam generator tube plugging. Reduced length WABA rods will be introduced beginning with the Byron Unit 1 Cycle 4 VANTAGE 5 reload fuel. For the LOCA analysis thimble plug removal and 15% plant total steam generator tube plugging for both the Byron/Braidwood Stations Units 1 and 2 were assumed.

The standard reload design methods described in Reference 3 will be used as a basic reference document in support of future Byron/Braidwood Reload Safety Evaluations (RSE) for VANTAGE 5 fuel reloads. Sections 2.0 through 5.0 summarize the Mechanical, Nuclear, Thermal and Hydraulic, and Accident Evaluations, respectively. Section 6.0 gives a summary of the Technical Specifications changes needed. Attachments 2 and 3 contain the Technical Specification change pages and non-LOCA safety analyses results, respectively. Attachment 4 contains the large and small break LOCA safety analyses. Attachment 5 provides the radiological assessment supporting the safety analyses.

Consistent with the Westinghouse standard reload methodology, Reference 3, parameters are chosen to maximize the applicability of the safety evaluations for future cycles. The objective of subsequent cycle specific RSEs will be to verify that applicable safety limits are satisfied based on the reference evaluation/analyses established in this submittal.

In order to demonstrate early performance of the VANTAGE 5 design product features in a commercial reactor, four VANTAGE 5 demonstration assemblies (17x17) were loaded into the V. C. Summer Unit 1 Cycle 2 core and began power production in December of 1984. These assemblies completed one cycle of irradiation in October of 1985 with an average burnup of 11,357 MWD/MTU. Post-irradiation examinations showed all 4 demonstration assemblies were of good mechanical integrity. No mechanical damage or wear was evident on any of the VANTAGE 5 components. Likewise, the IFM grids on the VANTAGE 5

demonstration assemblies had no effect on the adjacent fuel assemblies. All four demonstration assemblies were reinserted into V. C. Summer 1 for a second cycle of irradiation. This cycle was completed in March of 1987, at which time the demonstration assemblies achieved an average burnup of about 30,000 MWD/MTU. The observed behavior of the four demonstration assemblies at the end of 2 cycles of irradiation was as good as that observed at the end of the first cycle of irradiation. The four assemblies were reinserted for a third cycle of irradiation which was completed in November of 1988 (EOC burnup 46,000 MWD/MTU).

In addition to V. C. Summer, individual VANTAGE 5 product features have been demonstrated at other nuclear plants. IFBA demonstration fuel rods have been irradiated in Turkey Point Units 3 and 4 for two reactor cycles. Unit 4 contains 112 fuel rods equally distributed in four demonstration assemblies. The IFBA coating performed well with no loss of coating integrity or adherence. The IFM grid feature has been demonstrated at McGuire Unit 1. The demonstration assembly at McGuire was irradiated for three reactor cycles and showed good mechanical integrity.

The results of evaluation/analysis described herein lead to the following conclusions:

1. The Westinghouse VANTAGE 5 reload fuel assemblies for the Byron/Braidwood Stations Units 1 and 2 are mechanically compatible with the current OFA fuel assemblies, control rods, and reactor internals interfaces. The VANTAGE 5/OFA fuel assemblies satisfy the current design bases for the Byron/Braidwood Stations Units 1 and 2.
2. The structural integrity of the 17x17 VANTAGE 5 fuel assembly design for seismic/LOCA loadings has been evaluated for the Byron/Braidwood Stations Units 1 and 2. Evaluation of the 17x17 VANTAGE 5 fuel assembly component stresses and grid impact forces due to postulated faulted condition accidents verified that the VANTAGE 5 fuel assembly design is structurally acceptable.

3. Changes in the nuclear characteristics due to the transition from OFA to VANTAGE 5 fuel will be within the range normally seen from cycle to cycle due to fuel management effects.
4. The reload VANTAGE 5 fuel assemblies are hydraulically compatible with the OFA fuel assemblies from previous cycles of operation.
5. The core design and safety analyses results documented in this report show the core's capability for operating safely for the rated Byron/Braidwood Stations Units 1 and 2 design thermal power with $F_{\Delta H}^N$ of 1.65, and 1.55, $F_Q = 2.50$, and steam generator tube plugging levels up to 10% (15% for LOCA).
6. Previously reviewed and licensed safety limits continue to be met when the Byron/Braidwood Stations Units 1 and 2 are reloaded with VANTAGE 5 fuel. Plant operating limitations given in the Technical Specifications will be satisfied with the proposed changes noted in Section 6.0 of this report. A reference is established upon which to base Westinghouse reload safety evaluations for future reloads with VANTAGE 5 fuel.
7. The staff reviewed the VANTAGE 5 reference core report, WCAP-10444, Reference 2, and concluded that the report is acceptable for reference for the Westinghouse VANTAGE 5 fuel design report subject to specific conditions. These conditions summarized in Section 6.0 of the SER for WCAP-10444 have been considered in the Byron/Braidwood Stations Units 1 and 2 specific safety evaluations contained in this submittal.

2.0 MECHANICAL EVALUATION

This Section evaluates the mechanical design and the compatibility of the 17x17 VANTAGE 5 fuel assembly with the current 17x17 OFA fuel assemblies during the transition through mixed-fuel cores to all VANTAGE 5 fuel cores in the Byron/Braidwood Stations Units 1 and 2. The VANTAGE 5 fuel assembly has been designed to be compatible with the OFA fuel assembly, reactor internals interfaces, the fuel handling equipment, and refueling equipment. The VANTAGE 5 design is intended to replace and be compatible with fuel cores containing fuel of the OFA design. The VANTAGE 5 design dimensions as shown on Figure 2.1 are essentially equivalent to the OFA design from an exterior assembly envelope and reactor internals interface standpoint. References in this Section are made to WCAP-9500-A, "Reference Core Report 17x17 Optimized Fuel Assembly," Reference 1, and WCAP-10444-P-A, "VANTAGE 5 Fuel Assembly Reference Core Report," Reference 2. Where similarities in the mechanical design between the VANTAGE 5 and the OFA designs are described, the design bases and evaluations given in these reference reports are directly applicable.

The significant new mechanical features of the VANTAGE 5 design relative to the current OFA fuel design include the following:

- o Integral Fuel Burnable Absorber (IFBA)
- o Intermediate Flow Mixer (IFM) Grids
- o Reconstitutable Top Nozzle (RTN)
- o Reconstitutable Debris Filter Bottom Nozzle (DFBN)
- o Extended Burnup Capability
- o Axial Blankets

The DFBN (initially to be introduced in the Region 4 Braidwood Station Unit 1 Cycle 2 fuel design) is used instead of the bottom nozzle described in the Reference 1 report. Table 2.1 provides a comparison of the VANTAGE 5 and OFA fuel assembly design parameters.

TABLE 2.1

Comparison of 17x17 OFA
and
17x17 VANTAGE 5 Fuel Assembly Design Parameters

<u>PARAMETER</u>	<u>17x17 OFA DESIGN</u>	<u>17x17 VANTAGE 5 DESIGN</u>
Fuel Assy Length, in	159.765	159.975
Fuel Rod Length, in	151.560	152.285
Assembly Envelope, in	8.426	8.426
Compatible with Core Internals	Yes	Yes
Fuel Rod Pitch, in	.496	.496
Number of Fuel Rods/Assy	264	264
Number/Guide Thimble Tubes/Assy	24	24
Number/Instrumentation Tube/Assy	1	1
Fuel Tube Material	Zircaloy 4	Zircaloy 4
Fuel Rod Clad OD, in	0.360	0.360
Fuel Rod Clad Thickness, in	.0225	.0225
Fuel/Clad Gap, mil	6.2	6.2
Fuel Pellet Diameter, in	.3088	.3088
Fuel Pellet Length, in	.370	.370

Fuel Rod Performance

Fuel rod performance for all Byron/Braidwood fuel is shown to satisfy the NRC Standard Review Plan (SRP) fuel rod design bases on a region by region basis. These same bases are applicable to all fuel rod designs, including the Westinghouse OFA and VANTAGE 5 fuel designs, with the only difference being that the VANTAGE 5 fuel is designed to achieve a higher burnup and VANTAGE 5 fuel is designed to operate with a higher $F_{\Delta H}^N$ limit. The design bases for Westinghouse VANTAGE 5 fuel are discussed in Reference 2.

There is no affect from a fuel rod design standpoint due to having fuel with more than one type of geometry simultaneously residing in the core during the transition cycles. The mechanical fuel rod design evaluation for each region incorporates all appropriate design features of the region, including any changes to the fuel rod or pellet geometry from that of previous fuel regions (such as the presence of axial blankets or changes in the fuel rod and plenum length, for example). Analysis of Integral Fuel Burnable Absorber (IFBA) rods includes any geometry changes necessary to model the presence of the burnable absorber, and conservatively models the helium gas release from the ZrB_2 coating. Fuel performance evaluations are completed for each fuel region to demonstrate that the design criteria will be satisfied for all fuel rod types in the core under the planned operating conditions. Any changes from the plant operating conditions originally evaluated for the mechanical design of a fuel region (for example, a power uprating or an increase in the peaking factors) are addressed for all affected fuel regions when the plant change is to be implemented.

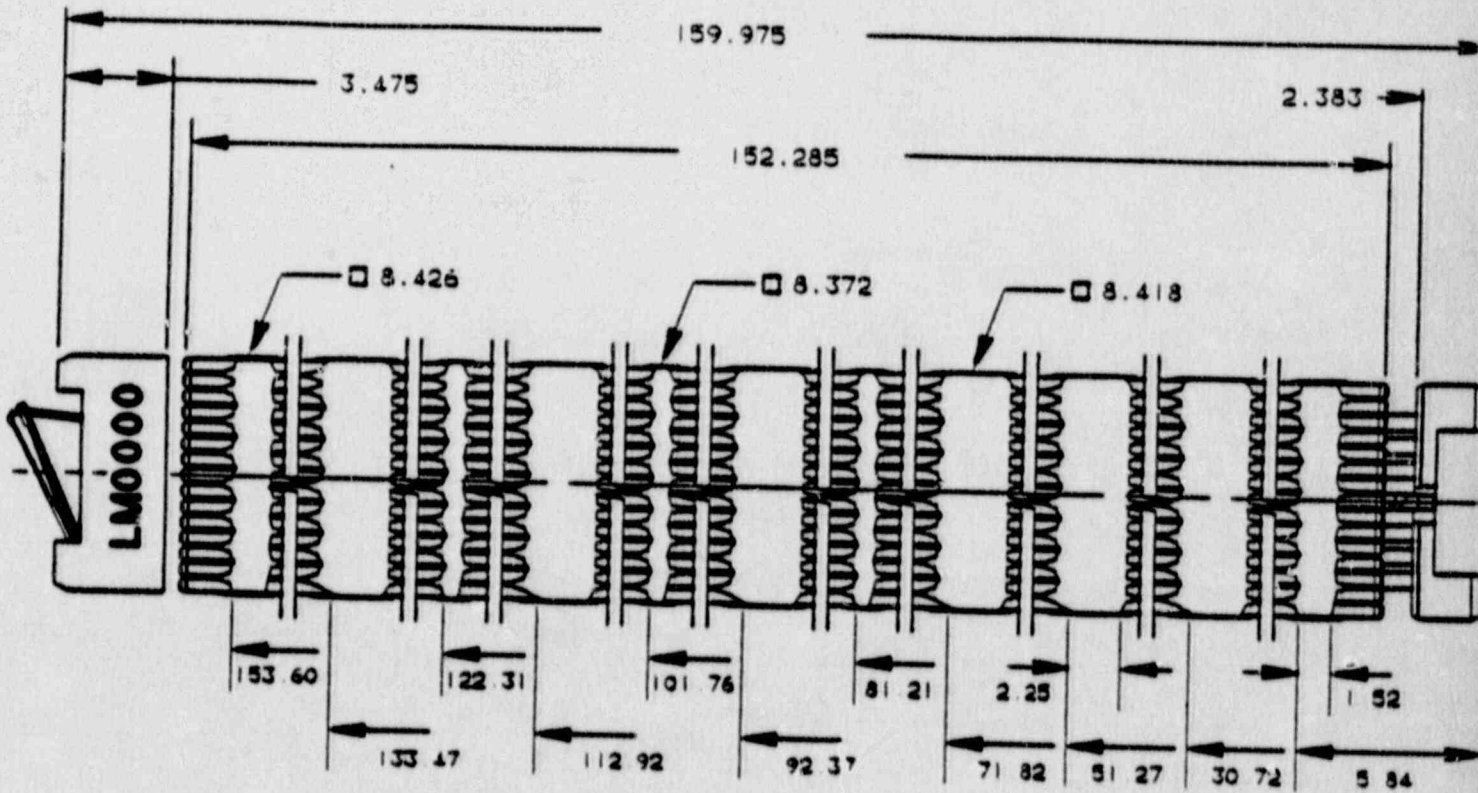
Fuel rod design evaluations for the Byron/Braidwood VANTAGE 5 transition core fuel were performed using the NRC approved models in References 4 and 5 and the NRC approved extended burnup design methods in Reference 6 to demonstrate that all of the SRP fuel rod design bases are satisfied.

Grid Assemblies

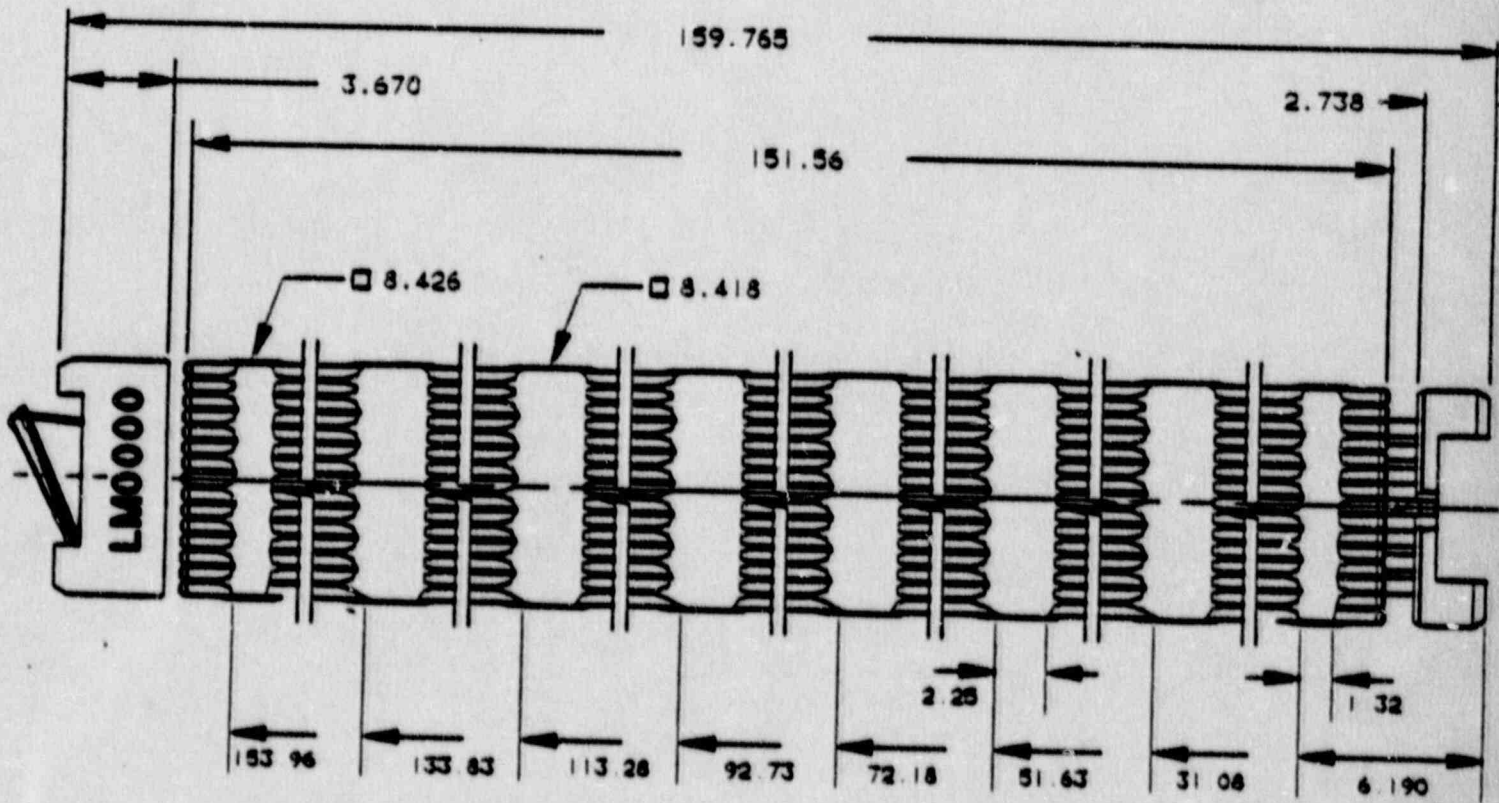
The top and bottom Inconel (non-mixing vane) grids of the OFA and VANTAGE 5 are nearly identical in design. The only differences are: 1) the top and bottom grids have a snag-resistant design which minimizes assembly interactions during core loading/unloading, 2) the top and bottom grids have dimples which are rotated 90 degrees to minimize fuel rod fretting and dimple cocking, and 3) the top and bottom grid heights have been increased to 1.522 inches. Both OFA and VANTAGE 5 grid designs have reduced grid spring forces to minimize rod bow. The six intermediate (mixing vane) grids are made of Zircaloy material and are identical in the OFA and VANTAGE 5 designs. The Zircaloy grids incorporate the same grid cell support configuration as the Inconel grids (six support locations per cell, four dimples, and two springs). The Zircaloy grid interlocking strap joints and grid/sleeve joints are fabricated by laser welding, whereas the Inconel grid joints are brazed.

The Intermediate Flow Mixer (IFM) grids shown in Figure 2.1 are located in the three uppermost spans between the Zircaloy mixing vane structural grids and incorporate a similar mixing vane array. Their prime function is mid-span flow mixing in the hottest fuel assembly spans. Each IFM grid cell contains four dimples which are designed to prevent mid-span channel closure in the spans containing IFMs and fuel rod contact with the mixing vanes. This simplified cell arrangement allows short grid cells so that the IFM grid can accomplish its flow mixing objective with minimal pressure drop.

The IFM grids are not intended to be structural members. The outer strap configuration was designed similar to current fuel designs to preclude grid hang-up and damage during fuel handling. Additionally, the grid envelope is smaller which further minimizes the potential for damage and reduces calculated forces during seismic/LOCA events. A coolable geometry is, therefore, assured at the IFM grid elevation, as well as at the structural grid elevation.



17X17 VANTAGE 5 FUEL ASSEMBLY



17X17 RECONSTITUTABLE OFA FUEL ASSEMBLY

Reconstitutable Top Nozzle and Bottom Nozzle

The reconstitutable top nozzle for the VANTAGE 5 fuel assembly differs from the OFA design in two ways: a groove is provided in each thimble thru-hole in the nozzle plate to facilitate attachment and removal; and the nozzle plate thickness is reduced to provide additional axial space for fuel rod growth.

To remove the top nozzle, a tool is first inserted through a lock tube and expanded radially to engage the bottom edge of the tube. An axial force is then exerted on the tool which overrides local lock tube deformations and withdraws the lock tube from the insert. After the lock tubes have been withdrawn, the nozzle is removed by raising it off the upper slotted ends of the nozzle inserts which deflect inwardly under the axial lift load.

With the top nozzle removed, direct access is provided for fuel rod examinations or replacement. Reconstitution is completed by the remounting of the nozzle and the insertion of lock tubes. Additional details of this design feature, the design bases and evaluation of the reconstitutable top nozzle are given in Section 2.3.2 in Reference 2. As noted in Reference 7 the VANTAGE 5 bottom nozzle will continue to be fabricated from stainless steel which differs from the VANTAGE 5 Inconel bottom nozzle described in Reference 2. The stainless steel bottom nozzle meets all design requirements.

The Debris Filter Bottom Nozzle design is similar to the current OFA design except it is shorter and has a thinner top plate to allow for fuel rod growth. The design bases and evaluation of the VANTAGE 5 bottom nozzle are given in Section 2.3.1 in Reference 2.

Axial Blankets

Although noted as a new mechanical feature of the VANTAGE 5 design and licensed in Reference 2, axial blankets have been and are currently operating in Westinghouse plants. A description and design application of this feature are contained in Reference 2, Section 3.0. The Byron/Braidwood axial blanket

design differs from that described in Reference 2. Recent changes utilize a chamfered pellet physically different than the enriched pellet in the fuel stack to help prevent accidental mixing with the enriched pellet.

Mechanical Compatibility of Fuel Assemblies

Based on the evaluation of the VANTAGE 5/OFA design differences and hydraulic test results (References 1, 2), it is concluded that the two designs are mechanically compatible with each other. The VANTAGE 5 fuel rod mechanical design bases remain unchanged from that used for the OFA fuel assemblies.

Rod Bow

The amount of fuel rod bow for the VANTAGE 5 fuel is predicted to be no greater than that for the OFA rods, since both fuel designs have the same fuel rod diameter, similar Zircaloy grid spacings and grid designs. The current NRC approved methodology for comparing rod bow for two different fuel assembly designs is given in Reference 8.

Rod bow in fuel rods containing IFBAs is not expected to differ in magnitude or frequency from that currently observed in Westinghouse OFA fuel rods under similar operating conditions. No indications of abnormal rod bow have been observed on visual or dimensional inspections performed on the test IFBA rods. Rod growth measurements were also within predicted bounds.

Fuel Rod Wear

Fuel rod wear is dependent on both the support provided by the fuel assembly grids and the flow environment to which it is subjected. Due to the VANTAGE 5 fuel assembly design employing the IFM grids, there is an unequal axial pressure distribution between the OFA and VANTAGE 5 fuel assemblies resulting in inter-assembly crossflow. The VANTAGE 5 fuel assembly was flow tested adjacent to a 17x17 OFA, since vibration test results indicated that the crossflow effects produced by this fuel assembly combination would have the most detrimental effect on fuel rod wear.

Results of the wear inspection and analysis discussed in Reference 2, Appendix A.1.4, revealed that the VANTAGE 5 fuel assembly wear characteristic was similar to that of the 17x17 OFA when both sets of data were normalized to the test duration time. It was concluded that the VANTAGE 5 fuel rod wear would be less than the maximum wear depth established, Reference 9, for the 17x17 OFA at EOL.

Seismic/LOCA Impact on Fuel Assemblies

An evaluation of the VANTAGE 5 fuel assembly structural integrity considering the lateral effects of a LOCA and a seismic loadings has been performed.

The applied force input given in WCAP-9401 (Reference 9) bounds the Byron/Braidwood Stations Units 1 and 2. The seismic/LOCA results documented in WCAP-9401 are applicable to the previous OFA cycle fuel.

The OFA design is mechanically and structurally equivalent to the VANTAGE 5 design, except for the addition of three intermediate flow mixers for the VANTAGE 5 fuel design. The safe shutdown earthquake and LOCA comparative analyses between the OFA and VANTAGE 5 fuel assemblies indicated that the VANTAGE 5 fuel assembly experienced the lower grid loads due to load sharing among the structural grids and flow mixers. The grid load comparison study results show that the VANTAGE 5 fuel assembly has more capability of withstanding the faulted condition transients than the OFA. The VANTAGE 5 fuel assemblies would provide improved seismic/LOCA grid load margins.

Based on the grid load comparative study results between the OFA and VANTAGE 5 fuel assembly designs and the seismic/LOCA loads from WCAP-9401 encompassing the Byron/Braidwood Stations Units 1 and 2, it is concluded that the VANTAGE 5 fuel assembly design is structurally acceptable for all the Byron/Braidwood Stations Units 1 and 2. The same conclusion is also true for a transition core composed of both VANTAGE 5 and OFA assembly core configurations. The grids will not buckle due to combined impact loads of a seismic/LOCA event.

The coolable geometry is maintained. The stresses in the fuel assembly components resulting from seismic and LOCA induced deflections are well within acceptable limits. In accordance with Condition 2 of the VANTAGE 5 NRC SER for Reference 2, the VANTAGE 5 fuel assembly structural integrity is assured.

Core Components

The core components for the Byron/Braidwood Stations are designed to be compatible with both VANTAGE 5 and OFA assemblies. The OFA and VANTAGE 5 thimble tube provides sufficient clearance for insertion of control rods, WABA rods, source rods, and dually compatible thimble plugs to assure the proper operation of these core components.

The thimble plugs utilized by the plugging devices, source assemblies, and burnable absorber assemblies for the Byron/Braidwood Stations Units 1 and 2 Cycle 1 cores are a dually compatible design having an OD of 0.424 inch and a length of 8.077 inches. The thimble plug has been designed to be compatible with both the OFA and VANTAGE 5 designs from both a mechanical and thermal/hydraulic perspective.

Reduced length WABA rods will be introduced beginning with the Byron Station Unit 1 Cycle 4 reload VANTAGE 5 fuel. These assemblies incorporate a 120 inch pellet stack to accommodate the neutronic design and are described in Section 3.0. A description and evaluation of the WABA rods is presented in Reference 10.

3.0 NUCLEAR EVALUATION

The nuclear design portion of the Byron/Braidwood Stations Units 1 and 2 reload transition core analysis has two objectives. First, the impact on the key safety parameters must be determined for the transition to VANTAGE 5 fuel. These safety parameters are used as input to the FSAR Chapter 15 accident analyses. Second, the plant Technical Specifications that apply to nuclear design must be reviewed to determine if they remain appropriate or must be altered to accommodate the mixed OFA/VANTAGE 5 transition cores and a complete VANTAGE 5 core.

To satisfy these objectives, conceptual mixed OFA/VANTAGE 5 core and full VANTAGE 5 core models were constructed. Key safety parameters were then evaluated such that the expected ranges of variation of the parameters were determined for the transition to VANTAGE 5 fuel. The key safety parameters referred to here are those described in the standard reload design methodology, Reference 3. Since OFA and VANTAGE 5 fuel have the same pellet and fuel rod diameter, most reactivity parameters are insensitive to fuel type. The core peaking factors are primarily loading pattern dependent. The loading patterns developed take advantage of the design flexibility of the VANTAGE 5 features, which include IFBA loading, extreme low leakage, and higher discharge burnup. The observed variations in these loading pattern (LP) dependent parameters during the transition to VANTAGE 5 are typical of the normal cycle to cycle variations for non-transition fuel reloads. Therefore, most of the key safety parameters fall into this LP-dependent category.

Technical Specification modifications will be required as a result of increased peaking factors ($F_{\Delta H}^N$ and F_Q). The increased peaking factor limits allow for higher discharge burnup and increased low leakage which improves fuel economy and increases nuclear design flexibility.

In summary, the transition from the current all OFA core to VANTAGE 5 fuel will not cause changes to the current nuclear design bases given in the Byron/Braidwood Stations Units 1 and 2 updated FSAR. The evaluation of the transition and equilibrium cycle VANTAGE 5 cores presented in Reference 2, as well as the transition and equilibrium core evaluations for the Byron/Braidwood Stations Units 1 and 2, demonstrate that the impact of implementing VANTAGE 5 does not cause a significant change to the physics characteristics of the Byron/Braidwood Stations Units 1 and 2 cores beyond the normal range of variations seen from cycle to cycle.

4.0 THERMAL AND HYDRAULIC EVALUATION

The analysis of the OFA and VANTAGE 5 fuel will be based on the Improved Thermal Design Procedure (ITDP) described in Reference 11. The OFA fuel analysis will use the WRB-1 DNB correlation in Reference 13 while the VANTAGE 5 fuel will utilize the WRB-2 DNB correlation in Reference 2. These DNB correlations take credit for the significant improvement in the accuracy of the critical heat flux predictions over previous DNB correlations. The WRB-2 DNB correlation also takes credit for the VANTAGE 5 IFM grid. A DNBR limit of 1.17 is applicable for both the WRB-1 and WRB-2 correlations. Table 4.1 summarizes the pertinent thermal and hydraulic design parameters.

The design method employed to meet the DNB design basis is the Improved Thermal Design Procedure which has been approved by the NRC, Reference 12. Uncertainties in plant operating parameters, nuclear, and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least 95 percent probability at a 95 percent confidence level that the minimum DNBR will be greater than or equal to 1.17 for the limiting power rod. Plant parameter uncertainties are used to determine the plant DNBR uncertainties. These DNBR uncertainties, combined with the DNBR limit, establish a DNBR value which must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design DNBR value, the plant safety analyses are performed using values of input parameters without uncertainties. For this application, the minimum required DNBR values for the OFA fuel analysis are a 1.32 for thimble cold wall cells (three fuel rods and a thimble tube) and 1.34 for typical cell (four fuel rods). The design DNBR values for the VANTAGE 5 fuel are a 1.32 and a 1.33 for thimble and typical cells, respectively.

In addition to the above considerations, a plant-specific DNBR margin has been considered in the analyses. In particular, safety analysis DNBR limits of 1.47 for thimble and 1.49 for typical cells for OFA fuel, and 1.65 and 1.67 for thimble and typical cells respectively for the VANTAGE 5 fuel, were used

in the safety analyses. The DNB margin between the safety analysis DNBR values and the design DNBR values is used to accommodate appropriate fuel rod bow penalty, Reference 8. The remaining margin can be used for plant flexibility.

The OFA and VANTAGE 5 designs have been shown to be hydraulically compatible in Reference 2.

The phenomenon of fuel rod bowing, as described in Reference 8, must be accounted for in the DNBR safety analysis of Condition I and Condition II events for each plant application. Internal to the fuel rod, the IFBA and fuel pellet designs are not expected to increase the propensity for fuel rods to bow. External to the VANTAGE 5 fuel rod, the Inconel non-mixing vane and Zircaloy mixing vane grids provide fuel rod support. Additional restraint is provided with the Intermediate Flow Mixer (IFM) grids. The safety analysis for the Byron/Braidwood Stations Units 1 and 2 maintain sufficient margin between the safety analysis limits DNBRs and the design limit DNBRs to accommodate full-flow and low-flow DNBR penalties.

The Westinghouse transition core DNB methodology is given in References 1 and 14 and has been approved by the NRC via Reference 15. A change to the VANTAGE 5 transition core penalty is discussed in Reference 16 and the recent NRC generic approval of this change is given in Reference 17. This methodology has been extended further in Reference 18. Using this methodology, transition cores are analyzed as if they are full cores of one assembly type (full OFA or full VANTAGE 5) applying the applicable core penalties.

The fuel temperatures for use in safety analysis calculations for the VANTAGE 5 fuel are evaluated using the same methods as those used to evaluate the current OFA fuel. Westinghouse uses the PAD performance code described in Reference 5 to perform both design and licensing calculations.

TABLE 4.1

Byron/Braidwood Thermal and Hydraulic Design Parameters

Thermal and Hydraulic Design Parameters		Design Parameters
(Using ITDP)		
Reactor Core Heat Output, MWt		3,411
Reactor Core Heat Output, 10^6 BTU/Hr		11,639
Heat Generated in Fuel, %		97.4
Core Pressure, Nominal, psia		2,280
Radial Power Distribution (OFA)		$1.49[1+0.3(1-F)]$
(VANTAGE 5)		$1.59[1+0.3(1-P)]$
Minimum DNBR at Nominal Conditions [#]		
Typical Flow Channel	(OFA)	2.43
	(VANTAGE 5)	2.49
Thimble (Cold Wall) Flow Channel	(OFA)	2.29
	(VANTAGE 5)	2.39
Limit DNBR for Design Transients		
Typical Flow Channel	(OFA)	1.49
	(VANTAGE 5)	1.67
Thimble (Cold Wall) Flow Channel	(OFA)	1.47
	(VANTAGE 5)	1.65
DNB Correlation	(OFA)	WRB-1
	(VANTAGE 5)	WRB-2

TABLE 4.1 (continued)

Byron/Braidwood Thermal and Hydraulic Design Parameters

HFP Nominal Coolant Conditions [#]	Design Parameters
Vessel Minimum Measured Flow ⁺ Rate (including Bypass), 10^6 lbm/hr GPM	144.8 390,390
Vessel Thermal Design Flow ⁺ Rate (including Bypass), 10^6 lbm/hr GPM	140.0 377,600
Core Flow Rate ⁺ (excluding Bypass, based on TDF) 10^6 lbm/hr GPM	131.2 353,811
Fuel Assembly Flow Area for Heat Transfer, ft ²	54.13
Core Inlet Mass Velocity, 10^6 lbm/hr-ft ² (Based on TDF)	2.42
Pressure Drop across Core, psi (Based on Best Estimate Flow)	(OFA) 26.0 (VANTAGE 5) 29.0

TABLE 4.1 (continued)

Byron/Braidwood Thermal and Hydraulic Design Parameters

Thermal and Hydraulic Design Parameters	Design Parameters
(Based on Thermal Design Flow)	
Nominal Vessel/Core Inlet Temperature, °F	558.4*
Vessel Average Temperature, °F	588.4
Core Average Temperature, °F	592.0
Vessel Outlet Temperature, °F	618.4
Average Temperature Rise in Vessel, °F	60.0
Average Temperature Rise in Core, °F	63.6
 Heat Transfer	
Active Heat Transfer Surface Area, ft ²	57,505
Average Heat Flux, BTU/hr-ft ²	197,180
Average Linear Power, kw/ft	5.45
Peak Linear Power for Normal Operation, ** kw/ft	13.61
Temperature at Peak Linear Power for Prevention of Centerline Melt, °F	4,700

Based on Safety Analysis $T_{in} = 559.9$ °F and Pressure = 2280 psia

* Safety Analysis $T_{in} = 559.9$ °F

+ Includes 10% steam generator tube plugging

** Based on 2.50 F_D loading factor

5.0 ACCIDENT EVALUATION

5.1 Non-LOCA Accidents

This section addresses the impact on non-LOCA accident analyses of the following proposed changes and design safety analysis assumptions for the Byron/Braidwood Stations Units 1 and 2. The non-LOCA safety evaluation will consider both the OFA-VANTAGE 5 transition cores as well as full VANTAGE 5 cores.

Proposed Change to the Licensing Basis:

- VANTAGE 5 Fuel design

Design Safety Analysis Assumptions:

- Increased Design Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^N$) and F_Q

VANTAGE 5

The design features of VANTAGE 5 fuel considered in the non-LOCA analysis are:

- VANTAGE 5 Fuel Rod Dimensions
- Axial Blankets
- Integral Fuel Burnable Absorbers (IFBAs)
- Intermediate Flow Mixer Grids (IFMs)
- Debris Filter Bottom Nozzle (DFBN)
- Increased Fuel Enrichment
- Extended Burnup Capability

Fuel Rod Dimensions

The VANTAGE 5 fuel rod dimensions which determine the safety analysis temperature versus linear power density relationship are identical to the OFA rod design. These dimensions include rod diameter, pellet diameter, initial pellet-to-clad gap size, and stack height. Therefore, the non-LOCA safety

analysis fuel temperature and rod geometry assumptions consider this geometry change and bound both OFA and VANTAGE 5 fuel.

The VANTAGE 5 fuel DNB analysis uses the Improved Thermal Design Procedure (Reference 11) and the WRB-2 correlation which is described in Appendix A of Reference 2. The OFA fuel DNB analysis for this evaluation uses the Improved Thermal Design Procedure (Reference 11) and the WRB-1 correlation (Reference 13).

Axial Blankets and IFBAs

Axial blankets reduce power at the ends of the rod which increases axial peaking at the interior of the rod. Used alone, axial blankets reduce DNB margin, but the effect may be offset by the presence of reduced length absorbers (Integral Fuel Burnable Absorbers - IFBA and Wet Annular Burnable Absorbers - WABA) which flatten the power distribution. The net effect on the axial shape is a function of the number and configuration of IFBAs in the core and time in life. The effects of axial blankets and IFBAs on the reload safety analysis parameters are taken into account in the reload design process. The axial power distribution assumption in the safety analyses kinetics calculations have been determined to be applicable for evaluating the introduction of axial blankets in the Byron/Braidwood Stations Units 1 and 2.

IFM Grids

The IFM grid feature of the VANTAGE 5 fuel design increases DNB margin. The fuel safety analysis limit DNBR values contain significant DNB margin (see Section 4.0). This DNB margin was set to ensure that the core thermal safety limits for the VANTAGE 5 fuel with an $F_{\Delta H}^N$ of 1.65 are acceptable. The OFA fuel core limits are more restrictive than the VANTAGE 5 fuel core limits. Thus, the most restrictive core limits correspond to the OFA fuel design. Any transition core penalty is accounted for with the available DNBR margin.

The IFM grid feature of the VANTAGE 5 fuel design increases the core pressure drop. The control rod scram time to the dashpot is increased from 2.4 to 2.7

seconds. The increased drop time primarily affects the fast reactivity transients. These accidents have been reanalyzed for this report. The revised safety analysis assumption was incorporated in all the reanalyzed events requiring this parameter and the remaining transients have been evaluated.

Debris Filter Bottom Nozzle

Core flow areas and loss coefficients were preserved in the design of the debris filter bottom nozzle. As such, no parameters important to non-LOCA safety analyses are impacted.

Fuel Enrichment

The VANTAGE 5 fuel design increased fuel enrichment is conservatively bounded by the maximum safety analysis assumption of 5.0 w/o.

Extended Burnup Fuel Assembly Design

WCAP-10125-P-A, "Extended Burnup Evaluation of Westinghouse Fuel," (Reference 6) evaluates the impact of extended burnup on the design and operation of Westinghouse fuel. The major effect of the extended burnup rod design is on power sharing between fresh and burned assemblies.

5.1.1 Increased Design Enthalpy Hot Channel Factor ($F_{\Delta H}^N$)

The $F_{\Delta H}^N$ for the OFA fuel during the transition cycles is 1.55. The $F_{\Delta H}^N$ for VANTAGE 5 fuel is 1.65. The non-LOCA calculations, applicable for the VANTAGE 5 core, have assumed a full power $F_{\Delta H}^N$ of 1.65. This is a conservative safety analysis assumption in this report.

The design core limits for this report incorporate the increased $F_{\Delta H}^N$ for VANTAGE 5 fuel.

5.1.2 Increase in LOCA F_Q

The increase in the Technical Specification maximum LOCA F_Q from 2.32 to 2.50 is conservatively accounted for in the non-LOCA transients.

5.1.3 Non-LOCA Safety Evaluation Methodology

The non-LOCA safety evaluation process is described in References 1 and 2. The process determines if a core configuration is bounded by existing safety analyses in order to confirm that applicable safety criteria are satisfied. The methodology systematically identifies parameter changes on a cycle-by-cycle basis which may invalidate existing safety analysis assumptions and identifies the transients which require re-evaluation. This methodology is applicable to the evaluation of VANTAGE 5 transition and full cores.

Any required re-evaluation identified by the reload methodology is one of two types. If the identified parameter is only slightly out of bounds, or the transient is relatively insensitive to that parameter, a simple evaluation may be made which conservatively evaluates the magnitude of the effect and explains why the actual analysis of the event does not have to be repeated. Alternatively, should the deviation be large and/or expected to have a significantly or not easily quantifiable effect on the transients, new analyses are required. The analysis approach will typically utilize the analytical methods which have been used in previous submittals to the NRC. These methods are those which have been presented in FSARs, subsequent submittals to the NRC for a specific plant, reference SARs, or report submittals for NRC approval.

The key safety parameters are documented in Reference 3. Values of these safety parameters which bound both fuel types (OFA and VANTAGE 5) were assumed in the safety analyses. For subsequent fuel reloads, the key safety parameters will be evaluated to determine if violations of these bounding values exist. Re-evaluation of the affected transients would take place and would be documented for the cycle specific reload design, as per Reference 3.

5.1.4 Reduced Temperature Operation

The T_{hot} Reduction Program (Reference 20) was originally performed for the Byron and Braidwood Stations to permit operation at lower primary temperatures in order to reduce the propensity of primary water stress corrosion cracking in the steam generator tubes. This program permitted operation with a RCS average temperature between 569.1°F and 588.4°F. The potential impact of the VANTAGE 5 fuel has been evaluated with respect to the results and conclusions of the T_{hot} Reduction Program. As long as the plants are in compliance with the requirements defined by Reference 20, the T_{hot} Reduction Program can be used in conjunction with VANTAGE 5 fuel and the conclusions presented in the UFSAR and References 19 and 20 for the non-LOCA transients remain valid.

5.1.5 Conclusions

Descriptions of the transients evaluated and analyzed for this report, method of analysis, results, and conclusions are contained in Attachment 3. The analytical procedures and computer codes used are identified in Sections 15.3 and 15.4. For those events explicitly analyzed, Attachment 3 has been prepared consistent with the format of the Byron/Braidwood Stations Units 1 and 2 UFSAR.

For each of the accidents analyzed, it was found that the appropriate safety criteria are met. For each of the accidents not analyzed, evaluations were performed which determined that the existing conclusions remain applicable for the proposed changes to the plant.

5.2 LOCA Accidents

5.2.1 Large Break LOCA

5.2.1.1 Description of Analysis/Assumptions for 17X17 VANTAGE 5 Fuel

The large break loss-of-coolant accident (LOCA) analysis for the Byron/Braidwood Stations Units 1 and 2, applicable to a full core of VANTAGE 5 fuel assemblies, was performed to develop Byron/Braidwood specific peaking factor limits. This is consistent with the methodology employed in the Reference Core Report for 17X17 VANTAGE 5, Reference 2. The Westinghouse 1981 Evaluation Model + BASH, References 21 and 22, was utilized and a spectrum of cold leg breaks were analyzed for the Byron/Braidwood Stations Units 1 and 2 that bound both the nominal operating conditions and the reduced temperature operation previously analyzed in Reference 20. Other pertinent analysis assumptions include: a core thermal power of 3411 MWt, 15% steam generator tubes plugged in each of four steam generators (i.e. uniform among the loops), an $F_{\Delta H}^N$ of 1.65, and fuel data based on the new fuel thermal model, Reference 5. The most limiting break determined from the reduced temperature analysis was reanalyzed at the nominal operating temperatures. The analysis results, tables and figures are presented in Attachment 4.

VANTAGE 5 fuel features, as applied at the Byron/Braidwood Stations Units 1 and 2, result in a fuel assembly that is more limiting than the OFA fuel assembly with respect to large break LOCA ECCS performance, Reference 2. As such, VANTAGE 5 fuel has been analyzed herein.

5.2.1.2 Method of Analysis

The methods used in analyzing the Byron/Braidwood Stations Units 1 & 2

for VANTAGE 5 fuel, including computer codes used and assumptions are described in detail in Attachment 4, Section 15.6.5.

5.2.1.3 Results

The results of this analysis, including tabular and plotted results of the break spectrum analyzed are provided in Attachment 4, Section 15.6.5, which has been prepared using the NRC Standard Format and Content Guide, Regulatory Guide 1.70, Revision 2 for accidents applicable to the Byron/Braidwood Stations Units 1 and 2.

Reference 22 stated three restrictions related to the use of the 1981 EM + BASH calculational model. The application of these restrictions to the plant specific large break LOCA analysis was addressed with the following conclusions:

Byron/Braidwood Stations Units 1 and 2 are neither an Upper Head Injection (UHI) or Upper Plenum Injection (UPI) plant so restriction 1 does not apply.

The Byron/Braidwood Stations Units 1 and 2 plant specific LOCA analysis analyzed both minimum and maximum ECCS cases to address restriction 2. The $C_D=0.6$ Double Ended Cold Leg Guillotine (DECLG) with minimum ECCS flows was found to result in the most limiting consequences.

Generic sensitivity studies were performed by Westinghouse for a typical 4-loop plant using different power shapes. This sensitivity study demonstrated that the chopped cosine was the most limiting power shape, Reference 22. A chopped cosine power shape was used in the large break LOCA analysis for the Byron/Braidwood Stations Units 1 and 2 thus satisfying restriction 3.

5.2.1.4 Conclusions

The large break LOCA analysis performed for the Byron/Braidwood Stations Units 1 and 2 has demonstrated that for breaks up to a double-ended severance of the reactor coolant piping, the Emergency Core Cooling System (ECCS) will meet the acceptance criteria of Title 10 CFR Part 50 Section 46. That is:

1. The calculated peak cladding temperature will remain below the required 2200°F.
2. The amount of fuel cladding that reacts chemically with the water or steam does not exceed 1% of the hypothetical amount that would be generated if all the zirconium metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.
3. The localized cladding oxidation limit of 17 percent is not exceeded during or after quenching.
4. The core remains amenable to cooling during and after the LOCA.
5. The core temperature is reduced and decay heat is removed for an extended period of time. This is required to remove the heat produced by the long-lived radioactivity remaining in the core.

The time sequence of events for all breaks analyzed is shown in Table 15.6-1 of Attachment 4, Section 15.6.5.

The large break LOCA analysis for the Byron/Braidwood Stations Units 1 and 2 assuming a full core of VANTAGE 5 fuel; utilizing the 1981 EM + BASH calculational model, resulted in a peak cladding temperature of 1883.1°F for the limiting DECLG break at a total peaking factor of 2.50. The maximum local metal-water reaction was 3.25%, and the total core wide metal-water reaction was less than 0.3% for all cases analyzed. The clad temperature transients turn around at a time when the core geometry was still amenable to cooling.

The effect of the transition core cycles are conservatively evaluated to be at most 50.F higher in calculated peak cladding temperature which would yield a transition core PCT of 1933.1°F. The transition core penalty can be accommodated by the margin to the 10 CFR 50.46, 2200°F limit.

It can be determined from the results contained in Attachment 4, Section 15.6.5 that the ECCS analysis for the Byron/Braidwood Stations Units 1 and 2 remains in compliance with the requirements of 10CFR50.46 including consideration for transition core configurations.

5.2.2 Small Break LOCA

5.2.2.1 Description of Analysis and Assumptions for 17X17 VANTAGE-5

The small break loss-of-coolant accident (LOCA) was analyzed assuming a full core of VANTAGE 5 fuel to determine the peak cladding temperature. This is consistent with the methodology employed in WCAP-10444-P-A, Reference 2, for 17x17 VANTAGE 5 transition. The currently approved NOTRUMP Model Small Break ECCS Evaluation Model, Reference 23, was utilized for a spectrum of cold leg breaks. Attachment 4, Section 15.6.5, includes a full description of the analysis and assumptions utilized for the Westinghouse VANTAGE 5 ECCS LOCA analysis. Pertinent assumptions include an $F_{\Delta H}^N$ of 1.65 for both the Byron and Braidwood Stations, a total peaking factors corresponding to 2.5 at the core mid-plane, 15% steam generator tube plugging, and a core thermal power level of 3411 MWt. The most limiting break determined from the nominal operating temperature small break LOCA analysis was reanalyzed at the reduced operating temperatures.

Sensitivity studies performed using the NOTRUMP small break evaluation model have demonstrated that VANTAGE 5 fuel is more limiting than OFA fuel

in calculated ECCS performance. Similar studies using the WFLASH evaluation model, have previously shown the OFA fuel is more limiting than LOPAR fuel. For the small break LOCA, the effect of the fuel difference is most pronounced during core uncover periods and, therefore, shows up predominantly in the LOCTA-IV calculation in the evaluation model analysis. Consequently, the previous conclusion drawn from the WFLASH studies, regarding the fuel difference, may be extended to the NOTRUMP evaluation model analysis. On this basis, only VANTAGE 5 fuel was analyzed, since it is the most limiting of the two types of fuel (OFA and VANTAGE 5) that would reside in the core at the Byron/Braidwood Stations Units 1 and 2.

5.2.2.2 Method of Analysis

The methods of analysis, including codes used and assumptions, are described in detail in Attachment 4, Section 15.6.5.

5.2.2.3 Results

The results of this analysis, including tabular and plotted results of the break spectrum analyzed, are provided in Attachment 4, Section 15.6.5.

5.2.2.4 Conclusions

The small break VANTAGE 5 LOCA analysis for the Byron/Braidwood Stations Units 1 and 2, utilizing the currently approved NOTRUMP Evaluation Model resulted in a peak cladding temperature (PCT) of 1453.1°F for the 3-inch diameter cold leg break at the nominal operating temperatures. The 3-inch break size was used in a similar analysis at the reduced operating temperatures which resulted in a PCT of 1424.5°F. The analysis assumed a limiting small break power shape consistent with a LOCA $F_Q(z)$ envelope of 2.50 at the core midplane elevation and 2.31 at the top of the core. The maximum local-water reaction is 0.48 percent, and the total core metal-water reaction is less than 0.3 percent for all cases analyzed. The clad temperature transients turn around at a time when the core geometry is still amenable to cooling.

Analyses presented in Attachment 4, Section 15.6.5 show that one centrifugal pump and one high head pump, together with the accumulators, provide sufficient core flooding to keep the calculated peak clad temperature well below the required limits of 10 CFR 50.46 (for all units). It can also be seen that the ECCS analysis remains in compliance with all other requirements of 10 CFR 50.46 and the peak cladding temperature results are well below the peak cladding temperatures calculated for the large break LOCA. Adequate protection is therefore afforded by the ECCS in the event of a small break LOCA.

5.2.3 Transition Core Effects on LOCA

When assessing the effect of transition cores on the large break LOCA analysis, it must be determined whether the transition core can have a greater calculated peak cladding temperature (PCT) than either a complete core of the OFA assembly design or a complete core of the VANTAGE 5 design. For a given peaking factor, the only mechanism available to cause a transition core to have a greater calculated PCT than a full core of either fuel is the possibility of flow redistribution due to fuel assembly hydraulic resistance mismatch. Hydraulic resistance mismatch will exist only for a transition core and is the only unique difference between a complete core of either fuel type and the transition core.

5.2.3.1 Large Break LOCA

The large break LOCA analysis was performed with a full core of VANTAGE 5 and conservatively applies the blowdown results to transition cores. The VANTAGE 5 differs hydraulically from the OFA assembly design it replaces.

The difference in the total assembly hydraulic resistance between the two designs is approximately 10% higher for VANTAGE 5.

An evaluation of hydraulic mismatch of approximately 10% showed an insignificant effect on blowdown cooling during a LOCA. The SATAN-VI computer code models the crossflows between the average core flow channel (N-1 fuel assemblies) and the hot assembly flow channel (one flow assembly) during blowdown. To better understand the transition core large break LOCA blowdown transient phenomena, conservative blowdown fuel clad heatup calculations have been performed to determine the clad temperature effect on the new fuel design for mixed core configurations. The effect was determined by reducing the axial flow in the hot assembly at the appropriate elevations to simulate the effects of the transition core hydraulic resistance mismatch. In addition, the Westinghouse blowdown evaluation model was modified to account for grid heat transfer enhancement during blowdown for this evaluation. The results of this evaluation have shown that no peak cladding temperature penalty is observed during blowdown for the mixed core. Therefore, it is not necessary to perform a blowdown calculation for the VANTAGE 5 transition core configuration because the evaluation model blowdown calculation performed for the full VANTAGE 5 core is conservative and bounding.

Since the overall resistance of the two types of fuel is essentially the same, only the crossflows during core reflood due to Intermediate Flow Mixing grid need be evaluated. The LOCA analysis uses the BASH computer code to calculate the reflood transient, Reference 21, which utilizes the BART code, Reference 24. A detailed description of the BASH code is given in Attachment 4. Fuel assembly design specific analyses have been performed with a version of the BART computer code, which accurately models mixed core configurations during reflood. Westinghouse transition core designs, including specific 17X17 OFA to VANTAGE 5 transition core cases, were analyzed. For this case, BART modeled both fuel assembly types and predicted the reduction in axial flow at the appropriate elevations. As expected, the increase in hydraulic resistance for the VANTAGE 5 assembly was shown to produce a reduction in reflood steam flow rate for the VANTAGE 5 fuel at mixing vane grid elevations for transition core configurations. This reduction in steam flow rate is partially offset by the fuel grid heat

transfer enhancement predicted by the BART code during reflood. The various fuel assembly specific transition core analyses performed resulted in peak cladding temperature increases of up to 50.F for core axial elevations that bound the location of the PCT. Therefore, the maximum PCT penalty possible for VANTAGE 5 fuel residing in a transition core is 50.F, Reference 2. Once a full core of VANTAGE 5 fuel is achieved the large break LOCA analysis will apply without the transition core penalty.

5.2.3.2 Small Break LOCA

The NOTRUMP computer code, Reference 25, is used to model the core hydraulics during a small break event. Only one core flow channel is modeled in the NOTRUMP code, Reference 23, since the core flow during a small break is relatively slow, providing enough time to maintain flow equilibrium between fuel assemblies (i.e., no crossflow). Therefore, hydraulic resistance mismatch is not a factor for small break. Thus, it is not necessary to perform a small break evaluation for transition cores, and it is sufficient to reference the small break LOCA for the complete core of the VANTAGE 5 fuel design, as bounding for all transition cycles.

5.2.4 Containment Integrity Mass and Energy Releases

The effect that design changes to the reactor fuel can have on Containment Mass and Energy releases used to determine Containment Peak Pressure are dependent upon:

- 1) The change in core fluid volume as a result of the new fuel design.
- 2) Increase or Decrease in core stored energy.
- 3) Effect of the new fuel design on reflood flooding rates as a result of core flow area or hydraulic resistance changes.

The VANTAGE 5 fuel design utilizes a fuel rod identical in diameter to the 17X17 OFA fuel presently installed in the Byron/Braidwood Stations Units 1 and 2. Therefore, the core stored energy remains unchanged from that assumed in the current Updated FSAR analyses for mass and energy releases calculated for a hypothetical LOCA. The VANTAGE 5 fuel assembly will not change the core fluid volume. However, the use of Intermediate Flow Mixing grids will increase hydraulic resistance which would reduce the rate of mass and energy releases to the containment. Thus the implementation of VANTAGE 5 fuel at the Byron/Braidwood Stations Units 1 and 2 will not result in an increase in the containment peak pressure reported in the Byron/Braidwood Stations Updated FSAR or increase the offsite radiological consequences associated with high containment pressures resulting from a hypothetical LOCA. Based on this evaluation a reanalysis of Containment Integrity Mass and Energy releases was deemed unnecessary for the implementation of VANTAGE 5 fuel at the Byron/Braidwood Stations Units 1 and 2.

5.2.5 Steam Generator Tube Rupture

The analysis for a steam generator tube rupture accident (SGTR) presented in the Byron/Braidwood Stations Updated FSAR was performed to ensure that the offsite radiation doses remain below the 10 CFR 100 limits. The primary thermal hydraulic parameters affecting this conclusion are the extent of fuel failure, the primary to secondary break flow through the ruptured tube, and the mass released to the atmosphere from the steam generator with the ruptured tube. The FSAR SGTR analysis is based on an assumption of 1% defective fuel, and the initial primary and secondary coolant activities are assumed to correspond to the specific activity limits in the Technical Specifications. These assumptions will not be affected by the change to VANTAGE 5 fuel. The primary to secondary break flow and the mass release to the atmosphere are dependent upon the initial reactor and steam generator conditions of power, pressure and temperature. The implementation of VANTAGE 5 fuel will not change the initial operating conditions at the Byron/Braidwood Stations

Units 1 & 2, and therefore the consequences of a SGTR will not be increased by the implementation of VANTAGE 5 fuel. Thus, a reanalysis of the FSAR SGTR analysis was determined to be unnecessary for the implementation of VANTAGE 5 fuel and the SGTR analysis in the Byron/Braidwood Stations Updated FSAR is considered to be bounding.

5.2.6 Blowdown Reactor Vessel and Loop Forces

The forces created by a hypothesized break in the RCS piping are principally caused by the motion of the decompression wave through the RCS. The strength of the decompression wave is primarily a result of the assumed break opening time, break area and RCS operating conditions of power, temperature and pressure. These parameters will not be effected by a change in fuel at Byron/Braidwood Stations Units 1 and 2 from 17X17 OFA to VANTAGE 5. The forces in the vicinity of the core are effected by the core flow area/volume. Since there will be no change in the core flow area/volume for VANTAGE 5 fuel, there will be no change in the forces calculated for a hypothesized LOCA. Forces acting on the RCS loop piping as a result of a hypothesized LOCA are not influenced by changes in fuel assembly design. Thus the implementation of VANTAGE 5 fuel at the Byron/Braidwood Stations Units 1 and 2 will not result in an increase of the calculated consequences of a hypothesized LOCA on the reactor vessel internals or RCS loop piping. The LOCA hydraulic forces used in Reference 9 and the evaluation for the LOCA hydraulic forces performed in Reference 20, for the T_{hot} reduction program, along with the current Byron/Braidwood Stations Updated FSAR analysis for forces on the reactor internals and RCS piping resulting from a hypothesized LOCA are considered to be bounding to the application of VANTAGE 5 fuel at the Byron/Braidwood Stations Units 1 and 2.

5.2.7 Post-LOCA Long-Term Core Cooling - ECCS flows, core subcriticality and switchover of the ECCS to hot leg recirculation

The implementation of VANTAGE 5 fuel at the Byron/Braidwood Stations Units 1 and 2 does not affect the assumptions for decay heat, core reactivity or boron concentration for sources of water residing in the containment sump Post LOCA. Thus, these licensing requirements associated with LOCA are not significantly affected by the implementation of VANTAGE 5 fuel. Additionally, Westinghouse/Commonwealth Edison Company perform an independent check on core subcriticality for each fuel cycle operated at the Byron/Braidwood Stations Units 1 and 2.

ATTACHMENT 2
TECHNICAL SPECIFICATION CHANGES
FOR BRAIDWOOD STATION UNITS 1 AND 2
TRANSITION TO WESTINGHOUSE 17x17 VANTAGE 5 FUEL ASSEMBLIES

ATTACHMENT 2

TECHNICAL SPECIFICATION CHANGES

FOR BRAIDWOOD STATION UNITS 1 AND 2

TRANSITION TO WESTINGHOUSE 17X17 VANTAGE 5 FUEL ASSEMBLIES

Record of Request for Offsite Review

Station Braidwood On-Site Review No. 89-091
Submitted by Lee Bush Date 7-22-89

- Test or experiment not involving an unreviewed safety question.
- Proposed test or experiment involving an unreviewed safety question.
- Proposed change to procedure, equipment or system involving an unreviewed safety question.
- Proposed change to Tech. Spec. or license.
- Unanticipated deficiency of design or operation of safety related structures, systems, or components.
- Proposed change to GSEP.
- Referral by Technical Staff Supervisor, Station Superintendent, Division Manager Nuclear Stations, or Manager of Quality Assurance.

Additional subject description: _____

Supporting documents attached: _____

Date required for Offsite Review completion: _____

Received by _____ Date _____
Senior Participant

Offsite Review No. _____

(Final)

APPROVED
OCT 06 1983

Grandwood On-site Review and Investigation Report

Review Number: 89-091

Date: 9-7-89

Subject Review: Vantage 5 Transition Licensing Amendment

- Disciplines Required:
- A Nuclear Power Plant Technology
 - B Reactor Operations
 - C Reactor Engineering
 - D Chemistry
 - E Radiation Protection
 - F Instrumentation and Control
 - G Mechanical and Electrical Systems

APPROV

JUL 29 1989

ON-SITE REVIEW

Participants: OP Eng _____
TSS _____
C. Wiegand _____

OSR Membership Approved Donald P. Paine 9/7/89
Technical Staff Supervisor / Date

10CFR50.59 Safety Evaluation is Required - - - - - Y/N Yes
 If yes, attach completed documentation in accordance with SWAP 1205-6.

10CFR50.59 Safety Evaluation Requires Action - - - - - X/N Yes

Submittal to Offsite Review is Required - - - - - Y/N Yes

Findings and Recommendations:

On-site Review has reviewed the proposed Westinghouse changes and has determined it is acceptable to be sent to offsite review and the NRC.
(see attached sheets)

 On-Site Review Committee: Signature indicates concurrence with Findings and Recommendations and 10CFR50.59 Safety Evaluation.

Signatures	Discipline(s)	Date
<u>[Signature]</u>	<u>C</u>	<u>9-22-89</u>
<u>D. Paine</u>	<u>A, C, F, G</u>	<u>9/25/89</u>
<u>Om Wiegand</u>	<u>SNE</u>	<u>9-27-89</u>
<u>[Signature]</u>	<u>A, B, C, G</u>	<u>9-28-89</u>

Approved by: [Signature] 9/29/89
 STATION MANAGER (Final) DATE

Procedure/Test/Modification Number: OSR No. 59-091 Rev. NA

Title: VANTAGE 5 TRANSITION LICENSING AMENDMENT

1.a. Description of the change(s) to the procedure or test or description of the Mod:
To USE VANTAGE 5 FUEL IN TRANSITION CYCLES

1.b. Purpose of the change(s) to the procedure or test or purpose of the Mod:
To USE VANTAGE 5 FUEL IN TRANSITION CYCLES

1.c. List systems and components affected:

REACTOR CORE

2. List reference documents reviewed which describe the components or administrative controls applicable to the procedure or test. Proposed tests must include review of UFSAR chapter 14.

REVIEW DOCUMENTS REVIEWED

- | | |
|---|--|
| a. UFSAR Section(s) <u>3, 4.2, 4.3, 4.4, 11.0, 15.0</u> | f. Fire Protection Report Section(s) <u>NONE</u> |
| b. SER Section(s) <u>4</u> | g. JIO's <u>NONE</u> |
| c. Tech Specs <u>Sections 2.0, 3/4.1, 3/4.2, 5.3 and bases.</u> | h. CFR's <u>10CFR50 App.A Criteria 10, 11, 60, 61, 62.</u> |
| d. Previous Safety Evaluations <u>Byron</u> | i. Reg. Guides <u>NONE</u> |
| e. Unit Operating License <u>NPF-72, NPF-77</u> | j. Other <u>NCAP 10444-P-A</u> |

3. State the effects of the proposed change on the following functions:

- | | |
|---------------------------|--|
| a) Site or Security | <u>NONE</u> |
| b) Mechanical | <u>Vantage 5 is compatible with OFA</u> |
| c) Structural | <u>Vantage 5 is seismic qualified</u> |
| d) Electrical | <u>NONE</u> |
| e) Instrument and Control | <u>NONE</u> |
| f) Fire Protection | <u>NONE</u> |
| g) Radiological | <u>VANTAGE 5 is similar or better than OFA</u> |
| h) Flood Protection | <u>NONE</u> |
| i) Administrative Control | <u>NONE</u> |

NOTE: FOR MODIFICATIONS, SETPOINT CHANGES, COMPONENT REPLACEMENTS, QUESTIONS 4, 5, AND 6 ARE NOT APPLICABLE.

4. Does the procedure or procedure revision constitute a change to a procedure as described in the UFSAR? YES () NO ()
5. Does the test involve operating methods or configurations contrary to those as described in the UFSAR? YES () NO ()
6. Is a Tech. Spec. change required? YES () NO ()

- If ALL of the answers in 4, 5, or 6 are NO, this evaluation is complete.
- If ANY of the answers in 4, 5, or 6 are YES, answer the 10 CFR 50.59 questions on BWAP 1205-6T1.

APPROVED

APR 13 1989

BRADWOOD

(Final)

 9-11-89
Prepared By

10CFR50.59 FORMAT FOR SAFETY EVALUATION

Station Braidwood Unit 1+2 System HOUSE FUEL OSR No. 89-091

Modification No. _____ Procedure No. _____ Rev. _____

Test No. _____ Rev. _____ Experiment No. _____ Rev. _____

Procedure/Test/Experiment Title _____

Equipment Name VANTAGE 5 FUEL

Equipment Number _____

Description _____

VANTAGE 5 Transition Licensing Amendment

10 CFR 50.59 Questions

1) Is a change to the Technical Specifications needed? YES NO _____

2) Is the probability of an occurrence, the consequence of an accident, or malfunction of equipment important to safety as previously evaluated in the PSAR, increased? YES _____ NO , because:

See attached

3) Is the possibility for an accident or malfunction of a different type than any previously evaluated in the PSAR created? YES _____ NO , because:

See attached

4) Is the margin of safety, as defined in the basis for any Technical Specification, reduced? YES _____ NO , because:

See attached

APPROVED

JUL 29 1986

BRAIDWOOD
ON-SITE REVIEW

Performed By: [Signature] Date 9-11-89

Approved By: [Signature] Date 9/22/89

7128P(072586)
0973A

VANTAGE 5 FUEL 50.59 CONTINUATION

General Note: The answers to the three questions for determining if an unreviewed safety question exists are based on the current safety analysis and all additional analyses performed in support of the Vantage 5 upgrade as documented in the Reload Transition Safety Report (RTSR) and Westinghouse's Safety Evaluation. These additional analyses will be documented in the UFSAR through its normal amendment process. Detailed support of the conclusion drawn in the answers to the questions below can be found in the Safety Evaluation provided by Westinghouse which is included in the licensing package.

All the proposed changes to the Technical Specifications except one are necessitated to support or are consistent with Vantage 5 design characteristics. The one exception is to Specification 3.1.1.3, Moderator Temperature Coefficient. One of the proposed changes to this specification is to add a paragraph a.4 to the Action Statements allowing for the provisions of Specification 3.0.4 to be not applicable. This proposal is administrative in nature to clarify the intent of Action "a" to allow power operations to continue and mode changes, if necessary, provided Actions a.1, a.2, and a.3 are complied with.

Question 2: The Westinghouse Vantage 5 reload fuel assemblies for the Braidwood Station are mechanically compatible with the current OFA fuel assemblies, control rods, and reactor internals interfaces. The Vantage 5 fuel assembly responses under seismic and LOCA excitations were determined using the analytical model representation of the Braidwood reactor core. Analysis of the 17 x 17 Vantage 5 fuel assembly component stresses and grid impact forces due to postulated faulted condition accidents verified that the Vantage 5 fuel assembly design is structurally acceptable. The Vantage 5 and OFA fuel assemblies satisfy the safety criteria which form the current design basis for Braidwood Station. Further, the reload Vantage 5 fuel assemblies are hydraulically compatible with the OFA fuel assemblies from the previous core. Changes in the nuclear characteristics due to the transition from OFA to Vantage 5 fuel will be within the range normally seen from cycle to cycle due to fuel management effects.

Question 3: This is based on the fact that the Vantage 5 fuel is compatible to the OFA fuel in form, fit, and function, and the method and manner of plant operation is unchanged.

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Question 4: Although some margins have been affected, the core design and safety analyses results show the core's capability for operating safely for the rated Braidwood Station Units 1 and 2 design thermal power with $F_{\Delta H}$ of 1.65 (Vantage 5), and 1.55 (OFA), $F_Q = 2.50$, and steam generator tube plugging levels up to $Q_{10\%}$ (15% for LOCA).

Previously reviewed licensed safety limits continue to be met when the Braidwood Station Units 1 and 2 are reloaded with Vantage 5 fuel. Plant operating limitations given in the Technical Specifications will be satisfied with the proposed amendment. A reference is established upon which to base Westinghouse reload safety evaluations for future reloads with Vantage 5 fuel.

In a transition core of Vantage 5 and OFA assemblies, the IFM grids in the Vantage 5 assemblies result in a localized flow redistribution between adjacent Vantage 5 and OFA assemblies. The effect of this localized flow redistribution is bounded by applying penalties to the transition core DNBR and Large Break LOCA Peak Clad Temperature results. In addition, the core hydraulic resistance due to the IFM grids results in an increase in the control rod scram time to the dashpot from 2.4 and 2.7 seconds. This increase, as well as the other effects of the change in design, have been incorporated into the non-LOCA and LOCA analyses, indicate that the ANS Condition II, III and IV acceptance criteria endorsed by NRC NUREG-0800, are still met.

ONSITE REVIEW 89-091

The Onsite review committee find the proposed amendment to the Technical Specifications addressing Vantage 5 fuel acceptable, and recommends submittal of these changes to Offsite review and the NRC. The following is a summary of the proposed changes and the actions necessary to implement these changes upon receipt:

- 1) Table 2.2-1, will be changed to revised the Delta I offset wings for the OTDT trip setpoint. These changes will require the OTDT loops to be recalibrated in accordance with the revised limits. These revised limits will require revision of several Instrument Maintenance procedures. These changes will be tracked by item numbers 456-130-89-0.2-0101, for Unit 1 and 456-130-89-0.2-0102 for Unit 2. This change will also require revision of the PLS manual. This change will be tracked by item number 456-130-89-0.2-0109.
- 2) 2.1.1 bases will be revised to address the new DNB correlations, Safety Analysis DNBR limits, and add new FNDH values associated with Vantage 5 fuel. These bases changes do not require any specific actions to implement.
- 3) 3.1.1.3 and associated bases will be revised to address the changes in the Moderator Temperature Coefficient required due to Vantage 5 fuel. These changes are necessary because moderator temperature coefficient (MTC) will be initially increasing (becoming more positive) with core exposure. As such the least negative value for MTC may not necessarily be at the beginning of core life. The provisions of 3.0.4 are not applicable is being added to allow transition into modes 2 and 1 from mode 3 as long as rod withdrawal limits are established per action statement A. In addition, the surveillance requirements and bases associated with MTC have also been revised to address the concerns above. These above changes will require the following actions to implement: LCDAR procedures - BwOS 1.1.3-1a will require revision to address 3.0.4 not being applicable. The statement of applicability associated with BwVS 1.1.3.a-1 must be revision to address the most restrictive point in core life for MTC. These actions will be tracked by item numbers 456-130-89-0.2-0103 and 456-130-89-0.2-0104 respectively.
- 4) 3.1.3.4, will be revised to address an increase in rod drop times associated with the Vantage 5 fuel. This change will require BwVS 1.3.4-1 to address this change in rod drop times. Tracking item 456-130-89-0.2-0105 will be assigned to track the completion of this item.

5) 3.2.2 and associated bases, will be revised to address the increased FQ(Z) limits associated with Vantage 5 fuel. This change will require revision of BwVS 10.2.2-1 to address these revised limits. The surveillance requirement exempting the area surrounding the grid straps from peaking factor limits will be changed. The grid straps associated with the Vantage 5 fuel do not significantly alter the flux distribution. No surveillance changes are required to address this change. The current surveillance procedure and program does measure the the peaking factors in the grid plane, and the revised Fxy limits can be applied to these planes. Several UFSAR changes will be necessary to address the FQ(Z) limits. The revision of BwVS 10.2.2-1 will be tracked by item number 456-130-89-0.2-0106. Item number 456-130-89-0.2-0108 will track the completion of the UFSAR revisions.

6) 3.2.3 and its bases, will be revised to address the new FNDH limits associated with Vantage 5 fuel. This change will require the following station procedures to be revised: 1BwVS 2.2.2-1, 2BwVS 2.2.2-1, 1BwVS 2.3.2-1, 2BwVS 2.3.2-1, and BwVS 10.2.2-1.

In addition to these procedure changes the UFSAR must be revised to address these new limits. Item number 456-130-89-0.2-0107 will track the completion of the above procedure revisions. Item number 456-130-89-0.2-0108 will track the completion of the UFSAR revisions.

7) 3.2.1 bases, has been revised to reflect the change made to the FQ(Z) limits, and the use of rod bow penalties due to extended burnup. no actions are required to implement these changes.

With the change in fuel types, a Startup Report must be submitted in accordance with specification 6.9.1. Submittal of this report will be tracked by item number 456-130-89-0.2-0110.

Technical Specification Changes for VANTAGE 5 Fuel

<u>PAGE</u>	<u>SECTION</u>	<u>DESCRIPTION OF CHANGE</u>	<u>JUSTIFICATION</u>
2-8	Table 2.2-1	Revised the $F(\Delta I)$ offset wings and gains with cycle specific identification.	These changes are due to the VANTAGE 5 fuel design.
B 2-1	2.1.1	Added DNB correlations and design and Safety Analysis DNBR limits for the VANTAGE 5 fuel. Added new $F_{\Delta H}^N$ values.	These changes reflect the DNB correlations and the values for $F_{\Delta H}^N$ for the VANTAGE 5 and OFA fuel.
B 2-2	Basis		
3/4 1-4	3.1.1.3	BOL deleted from MTC LCD and Surveillance	These changes reflect increasing MTC with burnup before decreasing toward EOL for VANTAGE 5 core and to allow entry into Modes 1 and 2, if the requirements of the Action Statements are met.
3/4 1-5		4.1.1.3 modified to compare BOL MTC with predicted MTC with burnup and develop rod withdrawal limits to keep MTC negative. Added "Provisions of Specification 3.0.4 are not applicable." to the Action Statement.	
3/4 1-19	3.1.3.4	Revised the rod drop time to ≤ 2.7 seconds and added cycle specific identification.	This change is the result of an increase in the core hydraulic resistance due to the VANTAGE 5 fuel design.

<u>PAGE</u>	<u>SECTION</u>	<u>DESCRIPTION OF CHANGE</u>	<u>JUSTIFICATION</u>
3/4 2-4	3.2.2	Added new F_Q limit and cycle specific identification.	This change reflects the value for F_Q assumed in the safety analysis for the VANTAGE 5 fuel design.
3/4 2-5	3.2.2	Replaced Figure 3.2-2 with 2 segment curve.	This curve is consistent with the VANTAGE 5 analysis.
3/4 2-7	3.2.2	In 4.2.2.2.f.3 add "(except VANTAGE 5 fuel assembly IFM grids)".	The VANTAGE 5 fuel assembly IFM grids will not significantly distort the indicated flux during the F_{xy} surveillance.
3/4 2-8	3.2.3	Revised the $F_{\Delta H}^N$ limits.	These changes reflect the values for $F_{\Delta H}^N$ assumed in the safety analyses for VANTAGE 5 and OFA fuel.
B 3/4 1-2	3/4 1.1.3 Basis	Reworded Surveillance justification paragraph.	This change reflects increasing MTC with burnup before decreasing toward EOL for VANTAGE 5 core.
B 3/4 2-1	3/4.2 Basis	Revised basis discussion of DNB.	These changes reflect the new DNB correlations used for the VANTAGE 5 and OFA fuel.

<u>PAGE</u>	<u>SECTION</u>	<u>DESCRIPTION OF CHANGE</u>	<u>JUSTIFICATION</u>
B 3/4 2-1	3/4.2.1 Basis	Changed the axial peaking factor multiplier to F_Q limit.	This change reflects the value for F_Q assumed in the safety analyses for either OFA or VANTAGE 5 fuel design.
B 3/4 2-4	3/4.2.2 3/4.2.3	Revised basis discussion for rod bow penalty.	This change reflects the new DNB correlations used for VANTAGE 5 and OFA fuel.
B 3/4 2-5	3/4.2.2 3/4.2.3	Revised basis discussion of $F_{\Delta H}^N$ limits.	Revised $F_{\Delta H}^N$ limits to include VANTAGE 5 fuel design.