ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATIONS CHANGES

NORTH ANNA UNITS 1 AND 2

REVISED PAGES FOR CURRENT

NORTH ANNA

TECHNICAL SPECIFICATIONS

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be ≤ 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. $T_{avg} \geq 500°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 2 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to:

1. ≤ 66% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are open, or

2. ≤ 71% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above the design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 324°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification, and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the cold leg stop valve provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its cold leg stop valve ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

a. T_{avg} greater than or equal to 500°F, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any full length rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the rod drop times within limits but determined with 2 reactor coolant pumps operating, operation may proceed provided THERMAL POWER is restricted to:

   1. Less than or equal to 66% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are open, or

   2. Less than or equal to 71% of RATED THERMAL POWER when the reactor coolant stop valves in the nonoperating loop are closed.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of full length rods shall be demonstrated through measurement prior to reactor criticality:

a. For all rods following each removal of the reactor vessel head,

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once per 18 months.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with all reactor coolant loops in operation and maintain the DNBR above the design limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two RHR loops to be OPERABLE.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 340°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into or (2) by restricting starting from the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

The requirement to maintain the boron concentration of an isolated loop greater than or equal to the boron concentration of the operating loops ensures that no reactivity addition to the core could occur during startup of an isolated loop. Verification of the boron concentration in an idle loop prior to opening the cold leg stop valve provides a reassurance of the adequacy of the boron concentration in the isolated loop. Operating the isolated loop on recirculating flow for at least 90 minutes prior to opening its cold leg stop valve ensures adequate mixing of the coolant in this loop and prevents any reactivity effects due to boron concentration stratifications.

Startup of an idle loop will inject cool water from the loop into the core. The reactivity transient resulting from this cool water injection is minimized by delaying isolated loop startup until its temperature is within 20°F of the operating loops. Making the reactor subcritical prior to loop startup prevents any power spike which could result from this cool water induced reactivity transient.

CHANGE PAGES FOR

NORTH ANNA

MERITS TECHNICAL SPECIFICATIONS

MERITS FORMAT

THIS SPECIFICATION WILL BE RELOCATED TO CHAPTER 16.2 (TECHNICAL REQUIREMENTS) OF THE UFSAR PER THE MERITS PROGRAM. THE SURVEILLANCE REQUIREMENTS HAVE BEEN RETAINED IN MERITS SPECIFICATION 3.1.4, "ROD GROUP ALIGNMENT LIMITS". THIS HAS BEEN NOTED AND WILL BE INCORPORATED AS PART OF THE RELOCATED SPECIFICATION.

ATTACHMENT 2

DISCUSSION AND SAFETY EVALUATION

1.0 INTRODUCTION AND SUMMARY

North Anna Units 1 and 2 have been operating with cores fueled by Westinghouse 17x17 low-parasitic (LOPAR) fuel assemblies. For North Anna Unit 1 Cycle 9, Unit 2 Cycle 8 and subsequent cycles, it is planned to refuel both units with new 17x17 fuel assemblies with Westinghouse 17x17 VANTAGE 5H fuel assembly design features which were approved for use in Reference 1 (Westinghouse Reference Core Report - VANTAGE 5H Fuel Assembly, February 1989). The new assembly design is designated North Anna Improved Fuel (NAIF). As a result, future core loadings would range from approximately a 1/3 NAIF - 2/3 LOPAR mixed core to eventually a 100% NAIF fueled core. The NAIF design includes the following VANTAGE 5H design features:

1) Low Pressure Drop Intermediate Zircaloy Grids
2) Reconstitutable Top Nozzle (RTN)
3) Extended Burnup Capability

Another assembly design feature, which will be included in NAIF assembly design, is the reconstitutable Debris Filter Bottom Nozzle (DFBN). This nozzle will replace the standard bottom nozzle used on the LOPAR fuel.

The RTN, DFBN and Extended Burnup Capability were previously evaluated as a part of the reload safety evaluation process and implemented in North Anna 1 Cycle 8 and North Anna 2 Cycle 7. The addition of the Zircaloy grids to the VANTAGE 5H and DFBN fuel features currently being used at North Anna will decrease neutron parasitic capture and thereby permit more efficient fuel usage. This safety evaluation addresses the impact of Zircaloy intermediate grids on plant design and operation, although the other changes associated with the NAIF assembly are discussed, also.

Significant operating experience on Zircaloy clad fuel with Zircaloy grids has been obtained from a number of commercial reactors operating with regions of VANTAGE 5 and optimized fuel assemblies (OFAs) with 14x14, 15x15 and 17x17 arrays (Reference 2, "Operational Experience with Westinghouse Cores," August 1988). No Zircaloy grid abnormalities or fuel

rod abnormalities related to the Zircaloy grids have been observed. The NAIF assembly uses the VANTAGE 5H low pressure drop Zircaloy grid design which was developed for use with the 0.374 inch diameter rod rather than the 0.360 inch diameter OFA rod. The VANTAGE 5H grid design is based on the OFA Zircaloy grid design and operating experience. A comparison of the VANTAGE 5H and OFA (and VANTAGE 5) grid designs is given in Reference 1. The VANTAGE 5H Zircaloy grid pressure drop is significantly less than the OFA Zircaloy grid and nearly identical to the LOPAR Inconel grid.

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 a third cycle of irradiation in September 1988 and were discharged with an average burnup of 46,050 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. A full region of 17x17 VANTAGE 5 fuel is in its first cycle of operation in the Callaway plant and in the V. C. Summer plant. In addition a full region of VANTAGE 5H fuel is in its first cycle of operation in the Salem Unit 1 plant.

In addition to the fuel assembly modifications, thimble plug assemblies will be removed from North Anna 1 Cycle 9, North Anna 2 Cycle 8 and subsequent cycles.

2.0 SUMMARY AND CONCLUSIONS

Consistent with the Westinghouse and Virginia Electric and Power Company standard reload methodologies for analyzing cycle specific reloads (Reference 3, "Reload Nuclear Design Methodology" September 1986, and Reference 4, "Westinghouse Reload Safety Evaluation Methodology" March 1978), parameters are chosen to maximize the applicability of the transition evaluations for future cycles and to facilitate subsequent determinations of the applicability of 10 CFR 50.59. The mechanical, thermal/hydraulic, nuclear and accident evaluations considered the impact of thimble plug removal.

The VANTAGE 5H hydraulic test program discussed in Reference 1, showed that the VANTAGE 5H grid and LOPAR fuel Inconel mixing vane grid had the same hydraulic characteristics. Therefore, no transition core DNB penalty or LOCA PCT transition core penalty needed to be applied to the evaluation for a NAIF/LOPAR assembly mixed core.

The results of evaluation/analyses and tests described herein lead to the following conclusions:

1. The North Anna Improved Fuel (NAIF) assemblies are mechanically and hydraulically compatible with the current LOPAR fuel assemblies, control rods and reactor internals interfaces.

2. Commercial operating and/or demonstration experience with 14x14, 15x15 and 17x17 OFAs and 17x17 VANTAGE 5 assemblies containing Zircaloy grids provides evidence of satisfactory operation of the VANTAGE 5H Zircaloy grids used on the NAIF assemblies.

3. Demonstration experience with 17x17 VANTAGE 5 assemblies containing removable nozzles provides evidence of satisfactory operation of 17x17 NAIF removable nozzle assemblies.

4. Thimble plug removal will increase the design core bypass flow slightly from 4.5% to 6.5% and the best estimate bypass flowrate by approximately 1.5%. These revised values will be used in the transient and thermal/hydraulic analyses.

5. Changes in the nuclear characteristics due to the transition from LOPAR to NAIF will be within the range normally seen from cycle-to-cycle due to fuel management effects.

6. Plant operating limitations given in the Technical Specifications will be satisfied with the proposed changes noted in Section 7.0 of this report.

3.0 MECHANICAL EVALUATION

3.1 Mechanical

The NAIF assemblies have been designed to be compatible with the LOPAR assemblies, reactor internals interfaces, and fuel handling and refueling equipment. Figure 1 gives a side-by-side comparison of the fuel assemblies. The grid elevations for the two assembly designs are nearly the same such that any integral contact between the assemblies will be grid-to-grid. By matching grid elevations, any crossflow maldistribution between the LOPAR and NAIF fuel assembly is minimized. The assembly envelopes and fuel rod diameters are the same. In addition the top and bottom grids in both designs are Inconel. Changes from the LOPAR to the NAIF assembly design include: decreases in guide thimble and instrumentation tube diametral dimensions, the change from use of the six intermediate LOPAR Inconel grids to a VANTAGE 5H Zircaloy grid design, the change to the VANTAGE 5H removable top nozzle (RTN) with modified holddown spring, the change to a low profile debris filtering bottom nozzle (DFBN), and the change to the longer VANTAGE 5H fuel rod and fuel assembly structure.

The 17x17 NAIF (VANTAGE 5H) assembly guide thimbles are similar in design to their counterparts in the LOPAR fuel assemblies except for a 8 mil ID and OD reduction above the dashpot and an decreased length due to the removable top nozzle. The diameter reduction is due to a reduced grid cell. This results from use of thicker grid strap material for the Zircaloy grids. Below the dashpot the NAIF and LOPAR assembly guide thimble dimensions are identical. The VANTAGE 5H guide thimble tube ID continues to provide an adequate nominal diametral clearance for the control rods and other core components. However, due to the reduced clearance, the time to the dashpot for accident analyses has conservatively been determined to increase from 2.2 seconds (to dashpot) for the LOPAR assemblies to 2.7 seconds for the NAIF assemblies. The increase in rod drop time required accident reanalyses as described in Section 6.0 of this report.

The six intermediate (mixing vane) grids on the NAIF assembly are made of Zircaloy material rather than Inconel which is currently used in the LOPAR design. These VANTAGE 5H grids are designed for the same pressure drop as the Inconel grid. Relative to the Inconel grid, the VANTAGE 5H Zircaloy grid strap thickness and strap height are increased for structural performance. The VANTAGE 5H 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 VANTAGE 5H Zircaloy grid has superior dynamic structural performance relative to the Inconel grid. Structural testing was performed and analyses have shown the VANTAGE 5H Zircaloy grid seismic/LOCA grid load margin is superior to that of the Inconel grid.

The VANTAGE 5H Zircaloy grid assembly design bases and evaluation are given in Section 2.2 in Reference 1.

The fuel assembly top nozzle for the VANTAGE 5H assembly differs from the current design in two ways: a groove is provided in each thimble thru-hole in the nozzle plate to facilitate removal; and the nozzle adapter plate is reduced to provide additional space for fuel rod growth. In the LOPAR design the Zircaloy thimbles are mechanically joined through expanded bulges to the uppermost grid stainless steel sleeve which is welded to the top nozzle adapter plate. The VANTAGE 5H assembly top nozzle design and methods for removal/reattachment are the same as those which received NRC generic approval, as shown in Reference 1.

The 17x17 debris filter bottom nozzle (DFBN) differs from the standard 17x17 bottom nozzle as follows:

1) The DFBN uses a modified pattern with a smaller flow hole size. The holes are sized to minimize the passage of debris through the nozzle.

2) The DFBN uses a decreased nozzle height and thinner plate to accommodate the extended burnup fuel rod.

3) Skirts are added to the bottom periphery of the plate between the legs to strengthen the plate and legs (see Figure 1).

The DFBN retains the reconstitutable feature found on the LOPAR design, which uses a locking cup to lock the thimble screws on the guide thimble assembly in place.

Both NAIF and LOPAR fuel rod designs retain the nominal pellet stack height of 144 inches; however, the NAIF fuel rod (standard VANTAGE 5H rod) length increases by 0.640 inches due to an increased gas plenum (for extended burnup) and to provide a longer lead-in chamfer on the bottom end plug for the removable top nozzle reconstitution feature. Fuel rod performance is shown to satisfy the UFSAR fuel rod design basis on a region by region basis. These same bases are applicable to the Westinghouse LOPAR fuel design and the NAIF design. Fuel performance evaluations are completed for each fuel region to demonstrate that the design criteria will be satisfied for the fuel rod types in the core under the planned operating conditions.

The rod bow behavior of the NAIF assemblies is predicted to be within the bounds of existing LOPAR assembly rod bow data. The most probable causes of significant rod bow are rod-grid and pellet-clad interaction forces and wall thickness variation (WTV). The NAIF assembly will have reduced grid forces (due to Zircaloy grid) and the same fuel tube thickness-to-diameter ratio (t/d) as the LOPAR assembly, which should tend to decrease NAIF rod bow compared to LOPAR fuel. For a given burnup, the magnitude of rod bow gap closure for the NAIF assembly is conservatively taken to be the same as that applied to the LOPAR fuel assembly. Additional evaluations related to DNBR penalties are given in Section 5.0.

The wear of a fuel rod is dependent on both the support provided by the assembly skeleton and the flow environment to which it is subjected. Since the VANTAGE 5H Zircaloy grid, which is used on the NAIF assembly, has the same pressure drop as the LOPAR Inconel grid, any crossflow between adjoining assemblies is minimized. A very conservative test was conducted consisting of 1000 hours of hot test time in a two assembly test loop. For the test, a VANTAGE 5H type assembly, with grid fuel cells modified to end-of-life conditions, was placed adjacent to an assembly designed to maximize adverse crossflow conditions. The test results showed no evidence of fretting wear; extrapolation of the data shows that fuel rod wear would be less than 10 percent of the cladding thickness for a reactor residence time of 48 months at full flow conditions.

The above tests on rod integrity have also been supported by analytical results. The analysis accounted for rod vibrations caused by both axial and cross flows and the effect of potential fuel rod to grid gaps.

In addition to grid differences, crossflows can be affected by the change in the distribution of core outlet loss coefficients resulting from the removal of thimble plugging devices. The core outlet loss coefficient (PFO) distribution shows an increase in PFO mismatch after thimble plug removal. Westinghouse has performed fuel rod vibration tests with a much larger PFO mismatch between two 17x17 fuel assemblies than would exist in the North Anna core after removal of all or any combination of thimble plugs. The results showed that there was no significant difference in fuel rod response between the tests performed with and without this large PFO mismatch. Therefore it is concluded that thimble plug removal will not have a detrimental effect on fuel rod vibration and wear.

Westinghouse studies on control rod wear have shown that most of the wear tends to be in the upper internals region. When thimble plugs are removed, the hydraulic resistance at the outlet for these

assemblies is reduced. This in turn causes the flow through the RCCA guide tubes to be reduced, because more flow is now going through the outlet of the assemblies which were previously fitted with thimble plugs. This reduction of flow through the RCCA guide tubes is in the direction that would tend to reduce control rod wear. In addition, it was concluded that the maximum core outlet loss coefficient (PFO) mismatch between an RCC location and an adjacent assembly does not increase with thimble plug removal for the North Anna reactor upper internals configuration. As a result, the magnitude of the crossflow seen by the control rods through the gap between the top nozzle and the upper core plate and the vibration of the rods caused by this crossflow will not be increased. Therefore, thimble plug removal will not have an adverse impact on control rod wear for the North Anna reactors.

4.0 NUCLEAR EVALUATION

The transition from 17x17 LOPAR fuel assemblies to 17x17 NAIF assemblies will not result in changes from the current nuclear design bases as given in the UFSAR and applied to subsequent North Anna Units 1 and 2 Reload Safety Evaluations. Although the physics characteristics are slightly different for a core of NAIF fuel when compared to 17x17 LOPAR fuel, evaluations show that the differences are well within the normal range of variations seen from cycle to cycle.

Thus any significant changes in nuclear characteristics found in the LOPAR/NAIF transition cores or eventually in a 100% NAIF core will be due to fuel management variables (number of feed assemblies, feed enrichment, cycle length, etc.) and not due to the change in fuel assembly design. As in current practice, each reload core design will be evaluated to assure that design and safety limits are satisfied according to the reload methodology (Reference 3).

5.0 THERMAL AND HYDRAULIC DESIGN

The 17x17 LOPAR and 17x17 VANTAGE 5H fuel assembly designs have been shown to be hydraulically compatible in the VANTAGE 5H Fuel Assembly Report (Reference 1). Therefore, no transition core penalty is applied to either the LOPAR or NAIF assemblies during the transition cycles.

The same calculational methods currently used on the 17x17 LOPAR fuel and described in the UFSAR and Reference 5 ("Statistical DNBR Evaluation Methodology," June 1987) are applicable to the evaluation of a core containing both 17x17 LOPAR and 17x17 NAIF assemblies. The WRB-1 correlation, which is applicable to the 17x17 LOPAR and VANTAGE 5H assemblies (Reference 6, Letter, NRC to Virginia Electric and Power Company, "Qualification of the WRB-1 Correlation in the Virginia Power COBRA Code," July 25, 1989), was used in the DNB safety analyses performed to support the implementation of VANTAGE 5H fuel.

The limiting DNB transients that had to be reanalyzed to support the implementation of the NAIF fuel product were the Loss-of-Flow and Locked Rotor events (see Section 6). The Loss-of-Flow transient was evaluated using the Virginia Power statistical DNB methodology (Reference 5) and the results compared to a 1.46 design limit DNBR (Reference 7, Letter, NRC to Virginia Electric and Power Company, "North Anna Units 1 and 2 - Approval of Continued Use of Negative Coefficient for NA-1 and Issuance of Amendment for NA-2," June 25, 1989). The Locked Rotor event was analyzed deterministically with the WRB-1 correlation.

The main impact of thimble plug removal is the increase in core bypass flow. Hydraulic calculations have shown that the design value of core bypass flow assumed in the deterministic DNB evaluations increases from 4.5% to 6.5%. The best estimate bypass flowrate used in the statistical DNB evaluations increases by 1.5%. These increases in bypass flow and their impact on DNB was directly accounted for in the DNB transients which were reanalyzed, Locked Rotor and Loss-of-Flow. For those transients which were not reanalyzed (see Section 6) the loss of margin resulting from the increased bypass flow and the associated reduction of core flow

is accommodated by retained DNBR margin. The loss of retained DNBR margin was approximately 2% and 2.5% for the statistical DNBR and deterministic DNBR transients, respectively.

Thimble plug removal also results in a reduction to the fuel assembly hydraulic loss coefficient. Westinghouse has performed hydraulic tests to quantify the magnitude of this effect. Based on these tests, it is estimated that there will be a slight increase in the primary system flow rate (less than 1 percent) due to thimble plug removal from the North Anna cores. No fuel assembly mechanical design criteria are impacted by this slight increase in flow rate.

Sensitivity studies were performed to demonstrate the insensitivity of as calculated DNBRs to non-uniform outlet pressure distributions and to variations in outlet loss coefficients. The variations in outlet loss coefficient due to thimble plug removal for the North Anna cores are bounded by these sensitivity studies. It is concluded, therefore, that removal of all or any combination of thimble plugs will not result in the reduction of DNBR margin due to mismatches in core outlet pressure gradients and loss coefficients.

The current Technical Spacification core safety limits (T.S. Figure 2.1-1) and the associated set points remain bounding for the transition cores and the VANTAGE 5H cores.

6.0 ACCIDENT EVALUATION

Those accidents analyzed and reported in the UFSAR which could potentially be affected by the NAIF reload have been reviewed. As discussed in Section 3.0 of this report, rod drop time is increased from 2.2 to 2.7 seconds (to dashpot) for control/shutdown rods in NAIF guide thimbles. Accident transients significantly affected are "fast" transients for which the protection system responds by tripping the reactor within a few seconds after the transient begins. The transients that fall into this category are Loss of Flow, Locked Rotor, and Rod Ejection. In these reanalyses, the impact of the increased drop time was considered for all

transition core configurations, including an all-LOPAR fueled core. Other non-LOCA accidents analyzed in the UFSAR were individually reviewed and were evaluated to be minimally affected by the increased rod drop time.

The simultaneous Loss of Flow from the three coolant pumps, Locked Rotor, and Rod Ejection accidents were reanalyzed with RETRAN (Reference 8, "Reactor System Transient Analysis Using the RETRAN Computer Code," May 1985 and Reference 9, "VEPCO Evaluation of the Control Rod Ejection Transient," December 1984) to account for the increased rod drop time. Results for these accident reanalyses (Tables 1 - 3) showed that the safety limits and criteria are satisfied for the increased rod drop time. These reanalyses will be incorporated in the next UFSAR update. As discussed in Section 5.0, the increase in bypass flow resulting from thimble plug removal was explicitly accounted for in these reanalyzed accidents.

The existing Large Break Loss-of-Coolant (LOCA) analysis for North Anna Units 1 and 2 was performed with the 1981 Evaluation Model using BART and is applicable to a full core of 17x17 LOPAR fuel. Sensitivity studies (Reference 1) show that BART Analysis results for a full core of 17x17 VANTAGE 5H fuel result in a 30°F to 100°F peak clad temperature (PCT) benefit when compared to a full core of 17x17 LOPAR fuel. For a given peaking factor, the only mechanism available to cause a greater calculated PCT for the transition core than a full core of either design is the possibility of flow redistribution due to a mismatch of fuel assembly hydraulic resistances. As discussed previously, full assembly testing of the VANTAGE 5H fuel (Reference 1) has demonstrated that the 17x17 VANTAGE 5H and 17x17 LOPAR fuel designs are hydraulically equivalent and therefore, flow redistribution does not occur. Based on this information and realizing that local flow redistribution induced by local hydraulic mismatches is the sole contributor in determining transition core penalties, it is concluded that there is no LOCA PCT transition core penalty associated with the 17x17 NAIF and 17x17 LOPAR fuel mixed core.

Because the existing analysis for a full core of 17x17 LOPAR fuel bounds a full core of 17x17 NAIF fuel and since there is no transition core penalty, the current LOCA analysis remains bounding.

7.0 TECHNICAL SPECIFICATION CHANGES

Based on the preceding evaluations the following technical specification changes for North Anna Units 1 and 2 are required to support the transition to NAIF fuel:

1. Modifications to Specification 3.1.3.4 to permit an increase in the control rod drop time to 2.7 seconds.

2. Modifications to Specification Bases 3/4.4.1 to reflect the use of the WRB-1 correlation and to make the section consistent with Section B 3/4.2.4 and the Bases for Section 2.0.

These changes are given in the proposed Technical Specification page changes (Attachment 1).

8.0 10 CFR 50.59 EVALUATION

The proposed change to the fuel assembly design has been determined not to result in an unreviewed safety question as defined in 10 CFR 50.59. The basis for this determination is as follows:

1. Neither the probability of occurrence nor the consequence of an accident or malfunction of equipment important to safety previously evaluated in the safety analyses is increased. The evaluations of the Nuclear, Thermal/Hydraulic and Mechanical design effects of the NAIF design support the conclusion that the assembly changes are within established design criteria. Consequently, no new mechanisms have been introduced to increase the probability of an accident.

   The Loss of Flow, Locked Rotor and Rod Ejection were reanalyzed to the same design limits used in previous submittals and documented in the UFSAR. The other non-LOCA and LOCA accidents discussed in the UFSAR were evaluated and determined to be unaffected by the use of 17x17 NAIF fuel. Since none of the design limits were violated, the

evaluation of the UFSAR transients indicate that operation under the proposed Technical Specification changes would not result in an increase in accident consequences.

2. The possibility for an accident or malfunction of a different type than evaluated in the safety analyses. The NAIF design satisfies the current fuel assembly design criteria. In addition, the impact of the removal of the thimble plugs was evaluated and determined not to increase fuel rod fretting or control rod wear. The impact of thimble plug removal upon core by-pass flow was addressed in evaluation of the non-LOCA and LOCA accidents. These accidents continue to meet their design limits. Therefore, the assembly design change and thimble plug removal do not involve any alteration to plant equipment or procedures which would introduce any new or unique accident precursors.

3. The margin of safety is not reduced. The reanalyses of the Loss of Flow, Locked Rotor and Rod Ejection transients meet the current design limits. Therefore, there is no reduction in the margin of safety.

9.0 REFERENCES

1. Davidson, S. L. (Ed.) "Westinghouse Reference Core Report - VANTAGE 5H Fuel Assembly," Addendum 2-A to WCAP-10444-P-A, February 1989.

2. Foley, T. and Skaritka, J., "Operational Experience with Westinghouse Cores," (through December 31, 1987), WCAP-8183-Rev. 16, August 1988.

3. "Reload Nuclear Design Methodology," VEP-FRD-42 Revision 1-A, September 1986.

4. Bordelon, F. M., et al, "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272 (Prop.) and WCAP-9273 (Non-Prop.), March 1978.

5. Anderson, R. C., "Statistical DNBR Evaluation Methodology," VEP-NE-2-A, June 1987.

6. Letter from G. C. Lainas (NRC) to W. R. Cartwright (Virginia Electric and Power Company), "Qualification of the WRB-1 Correlation in the Virginia Power COBRA Code," Serial No. 89-571, July 25, 1989.

7. Letter from L. B. Engle (NRC) to W. R. Cartwright (Virginia Electric and Power Company), "North Anna Units 1 and 2 - Approval of Continued Use of Negative Moderator Coefficients for NA-1 and Issuance of Amendment for NA-2," NRC Letter No. 89-498, dated June 30, 1989.

8. VEP-FRD-41A, "Reactor System Transient Analyses Using the RETRAN Computer Code," May 1985.

9. VEP-NFE-2-A, "Vepco Evaluation of the Control Rod Ejection Transient," December 1984.

TABLE 1

LOSS OF FLOW RESULTS

	Calculated Minimum Value	Limit
DNBR	1.48	1.46

TABLE 2

LOCKED ROTOR RESULTS

	Calculated Value	Limit
System Pressure (psia)	2713	2750
% of Rods Below DNBR Limit	< 13	13*

*Current UFSAR design basis

TABLE 3

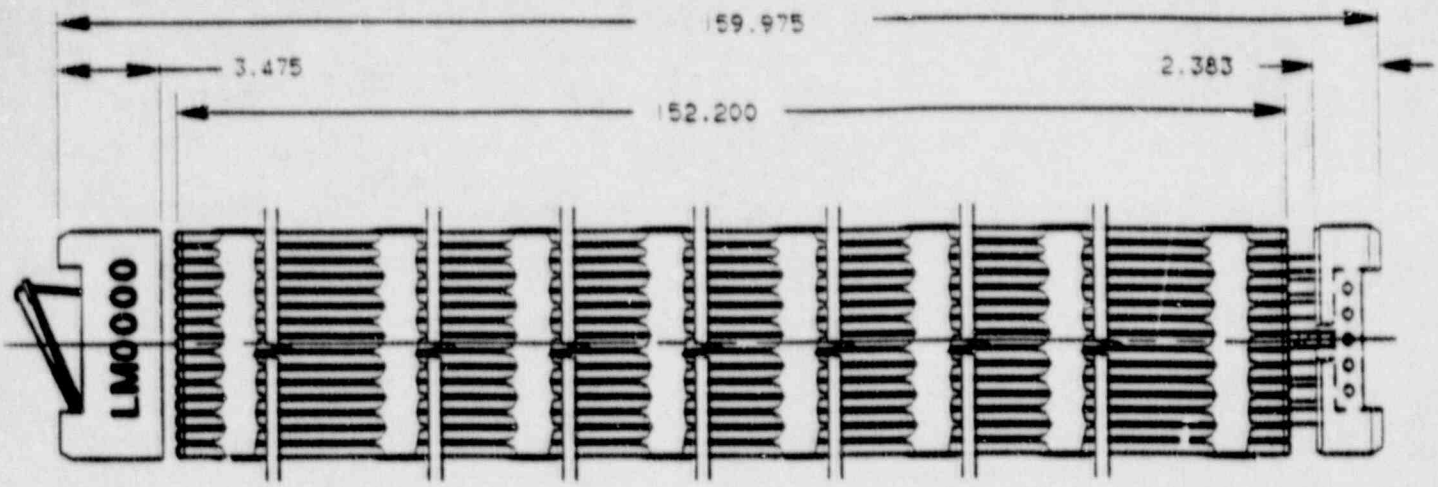
SUMMARY OF ROD EJECTION ANALYSIS PARAMETERS AND RESULTS

Case	BOL-HZP***	BOL-HFP	EOL-HZP	EOL-HFP	LIMIT
Percent Fuel Melt Temperatures (°F)	0	0	0	10**	10% Melt
Max. Clad Inner Temperature (°F)	2564	2514	2575	2447	2700
Average Fuel Enthalpy (cal/gm)	152	184	154	178	200****

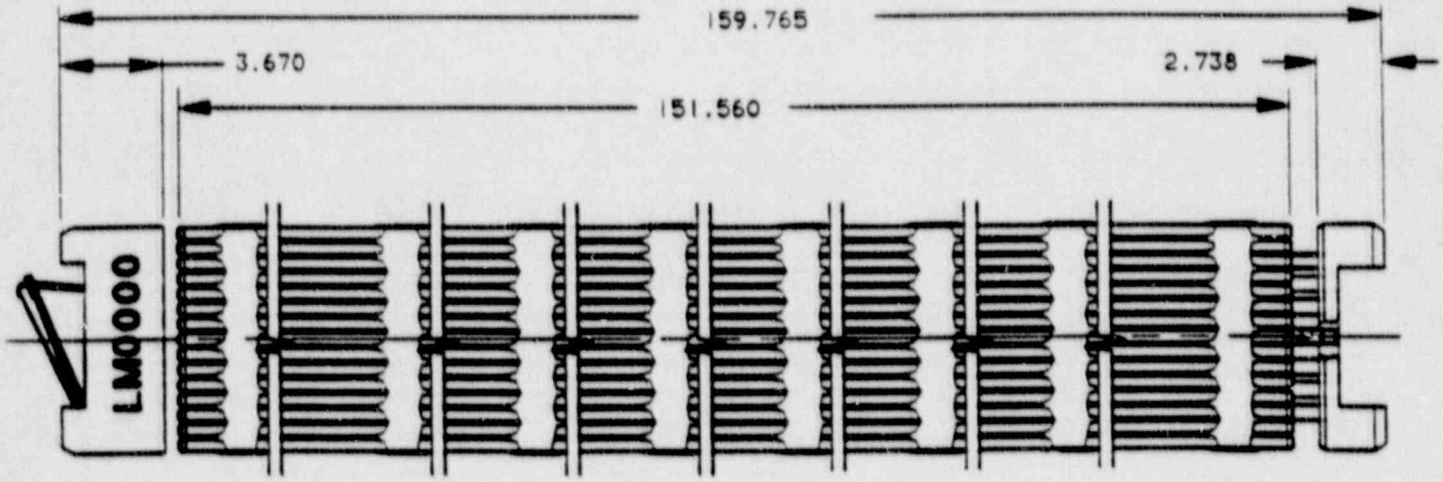
**Less than 10% of fuel melt at fuel rod hot spot

***BOL-HZP Beginning of Life - Hot Zero Power

****200 cal/gm irradiated and 225 cal/gm unirradiated



17 X 17 NAIF FUEL ASSEMBLY



17 X 17 RECONSTITUTABLE LOPAR FUEL ASSEMBLY

NORTH ANNA UNITS 1 & 2

FIGURE 1
17 X 17 NAIF / LOPAR
FUEL ASSEMBLY COMPARISON

ATTACHMENT 3

10 CFR 50.92 EVALUATION

10 CFR 50.92 NO SIGNIFICANT HAZARDS DETERMINATION

The proposed changes do not involve a significant hazards consideration because operation of North Anna Units 1 and 2 in accordance with these changes would not:

(1) involve a significant increase in the probability or consequence of an accident previously evaluated. Neither the probability of occurrence nor the consequence of an accident or malfunction of equipment important to safety previously evaluated in the safety analyses is increased. The evaluations of the Nuclear, Thermal/Hydraulic and Mechanical design effects of the NAIF design support the conclusion that the assembly changes are within established design criteria. Consequently, no new mechanisms have been introduced to increase the probability of an accident.

The Loss of Flow, Locked Rotor and Rod Ejection were reanalyzed to the same design limits used in previous submittals and documented in the UFSAR. The other non-LOCA and LOCA accidents discussed in the UFSAR were evaluated and determined to be unaffected by the use of 17x17 NAIF. Since none of the design limits were violated, the evaluation of the UFSAR transients indicate that operation under the proposed Technical Specification changes would not result in an increase in accident consequences.

(2) create the possibility of a new or different kind of accident from any accident previously identified. The possibility for an accident or malfunction of a different type than evaluated in the safety analyses. The NAIF design satisfies the current fuel assembly design criteria. In addition, the impact of the removal of the thimble plugs was evaluated and determined not to increase fuel rod fretting or control rod wear. The impact of thimble plug removal upon core by-pass flow was addressed in evaluation of the non-LOCA and LOCA accidents. These accidents continue to meet their design limits. Therefore, the assembly design change and thimble plug removal do not involve any alteration to plant equipment or procedures which would introduce any new or unique accident precursors.

(3) involve a significant reduction in the margin of safety. The margin of safety is not reduced. The reanalyses of the Loss of Flow, Locked Rotor and Rod Ejection transients meet the current design limits. Therefore, there is no reduction in the margin of safety.

We have also reviewed the examples of types of amendments which the NRC considers not likely to involve significant hazards considerations (51 FR 7744, March 6, 1986) and found that Example (iii) was directly applicable to the proposed change. Example (iii) states: "For a nuclear power reactor, a change resulting from a nuclear reactor core reloading, if no fuel assemblies significantly different from those found previously acceptable to the NRC for a previous core at the facility in question are involved. This assumes that no significant changes are made to the acceptance criteria for the technical specifications, that the analytical methods used to demonstrate conformance with the technical specifications and regulations are not significantly changed and that the NRC has previously found such methods acceptable." The analyses show that the 17x17 VANTAGE 5H assemblies meet the design criteria applicable to the 17x17 LOPAR fuel assemblies. The analytical methods used will remain unchanged.