Attachment 2

Technical Specifications Changes

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 45 psig and a temperature of 280°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4 or ZIRLO. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1780 grams uranium. The initial core loading shall have a maximum enrichment of 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.3 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

2a. WCAP-9220-P-A. Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL - 1981 VERSION", February 1982 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2b. WCAP-9561-P-A, ADD. 3, Rev. 1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS - SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL", JULY, 1986, (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2c. WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March 1987 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2d. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2e. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code", August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2f. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY REPORT," June 1990 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor.)

3a. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology", June 1987.

(Methodology for LCO 3.2.3, Nuclear Enthalpy Rise Hot Channel Factor).

3b. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code", July 1990.

(Methodology for LCO 3.2.3 Nuclear Enthalpy Rise Hot Channel Factor).

4. VEP-NE-1-A, "Vepco Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," March 1986.

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor and LCO 3.2.1 - Axial Flux Difference.)

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy-4 or ZIRLO. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1780 grams uranium. The initial core loading shall have a maximum enrichment of 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 4.3 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is approximately 10,000 cubic feet at nominal operating conditions.

2a. WCAP-9220-P-A. Rev. 1, "WESTINGHOUSE ECCS EVALUATION MODEL - 1981 VERSION", February 1982 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2b. WCAP-9561-P-A, ADD. 3, Rev. 1, "BART A-1: A COMPUTER CODE FOR THE BEST ESTIMATE ANALYSIS OF REFLOOD TRANSIENTS - SPECIAL REPORT: THIMBLE MODELING IN W ECCS EVALUATION MODEL", JULY, 1986, (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2c. WCAP-10266-P-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", March 1987 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2d. WCAP-10054-P-A, "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2e. WCAP-10079-P-A, "NOTRUMP, A Nodal Transient Small Break and General Network Code", August 1985 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor).

2f. WCAP-12610, "VANTAGE+ FUEL ASSEMBLY REPORT," June 1990 (W Proprietary).

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor.)

3a. VEP-NE-2-A, "Statistical DNBR Evaluation Methodology", June 1987.

(Methodology for LCO 3.2.3, Nuclear Enthalpy Rise Hot Channel Factor).

3b. VEP-NE-3-A, "Qualification of the WRB-1 CHF Correlation in the Virginia Power COBRA Code", July 1990.

(Methodology for LCO 3.2.3 Nuclear Enthalpy Rise Hot Channel Factor).

4. VEP-NE-1-A, "Vepco Relaxed Power Distribution Control Methodology and Associated FQ Surveillance Technical Specifications," March 1986.

(Methodology for LCO 3.2.2 - Heat Flux Hot Channel Factor and LCO 3.2.1 - Axial Flux Difference.)

Attachment 3

Significant Hazards Consideration

SIGNIFICANT HAZARDS CONSIDERATION

Virginia Electric and Power Company plans to insert fuel assemblies containing fuel rods, guide thimble tubes, instrumentation tubes, and mid-span grids fabricated with Westinghouse Electric Corporation's (Westinghouse's) advanced zirconium alloy material, ZIRLO, into the North Anna Units 1 and 2 reactors, beginning with Cycle 11 at each unit. In the current fuel design, these components are fabricated from Zircaloy-4. Changing the material of these components from Zircaloy-4 to the ZIRLO alloy will provide operational benefit relative to the current fuel design due to the ZIRLO alloy's improved corrosion resistance and dimensional stability under irradiation.

Because the Technical Specifications define the fuel rod cladding material as Zircaloy-4, implementation of this material change requires changes to the Technical Specifications. Technical Specification 5.3.1 is being modified to allow the use of either Zircaloy-4 or ZIRLO fuel rod cladding, and an additional reference for the calculation of the heat flux hot channel factor for LOCA evaluations of fuel with ZIRLO cladding is being defined in Technical Specification 6.9.1.7.e. The use of the ZIRLO fabricated guide thimble tubes, instrumentation tubes, and mid-span grids does not require changes to the Technical Specifications.

Virginia Electric and Power Company has reviewed the Technical Specifications changes against the criteria of 10 CFR 50.92, and has concluded that the changes do not pose a significant hazards consideration. Specifically, operation of North Anna Power Station in accordance with the Technical Specifications changes will not:

1. Involve a significant increase in the probability or consequence of an accident previously evaluated. The North Anna fuel assemblies containing fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy meet the same fuel assembly and fuel rod design bases as the current fuel assemblies fabricated with Zircaloy-4 components. In addition, the 10 CFR 50.46 criteria will be applied to the fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy. The use of these fuel assemblies will not result in a change to the North Anna Units 1 and 2 reload design and safety analysis limits. The ZIRLO alloy is similar in chemical composition to Zircaloy-4, and also has physical and mechanical properties similar to those of Zircaloy-4. Thus the cladding integrity is maintained and the structural integrity of the fuel assembly is not affected. The ZIRLO clad fuel rods improve corrosion resistance and dimensional stability. Since the dose predictions in the safety analyses are not sensitive to the fuel rod cladding material changes as specified in this report, the radiological consequences of accidents previously evaluated in the safety analyses remain valid. Therefore, neither the probability of occurrence nor the consequences of any accident previously evaluated is significantly increased.

2. Create the possibility of a new or different kind of accident from any accident previously identified, since the North Anna Units 1 and 2 fuel assemblies containing fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy will satisfy the same design bases used for previous fuel regions containing Zircaloy-4 components. Since the original design criteria

are being met, the fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy will not be initiators for any new accident. All design and performance criteria will continue to be met and no single failure mechanisms have been created. In addition, the use of these fuel assemblies does not involve any alteration to plant equipment or procedures which would introduce any new or unique operational modes or accident precursors. Therefore, the possibility for a new or different kind of accident from any accident previously evaluated is not created.

3. Involve a significant reduction in a margin of safety. The North Anna Units 1 and 2 fuel assemblies containing fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy do not change the North Anna Units 1 and 2 reload design and safety analysis limits. The use of fuel assemblies containing fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy will take into consideration the normal core operating conditions allowed in the Technical Specifications. For each cycle reload core these fuel assemblies will be specifically evaluated using approved reload design methods and approved fuel rod design models and methods. This will include consideration of the core physics analysis peaking factors and core average linear heat rate effects. In addition, the 10 CFR 50.46 criteria will be applied each cycle to the fuel rods, guide thimble tubes, instrumentation tubes and mid-span grids fabricated with ZIRLO alloy. Analyses or evaluations will be performed each cycle to confirm that 10 CFR 50.46 will be met. Therefore, the margin of safety as defined in the Bases

to the North Anna Unit 1 Technical Specifications and the North Anna Unit 2 Technical Specifications is not significantly reduced.

Virginia Electric and Power Company concludes that the activities associated with the proposed Technical Specifications changes satisfy the no significant hazards consideration criteria of 10 CFR 50.92 and, accordingly, a no significant hazards consideration finding is justified.