ATTACHMENT 3

PROPOSED CHANGES TO APPENDIX A,
TECHNICAL SPECIFICATIONS, OF FACILITY
OPERATING LICENSES NPF-76 AND NPF-80,
SOUTH TEXAS PROJECT UNITS 1 & 2

Revision to: 5.3.1 and 6.9.1.6

5.3 REACTOR CORE

FUEL ASSEMBLIES

or ZIRLO ,ZIRLO

5.3.1 The reactor core shall contain 193 fuel assemblies. Each fuel assembly shall consist of a matrix of zircaloy clad fuel rods with an initial composition of natural or slightly enriched uranium dioxide as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with NRC-approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core regions.

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 57 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 158.9 inches of absorber material. The absorber material within each assembly shall be silver-indium-cadmium or hafnium. Mixtures of hafnium and silver-indium-cadmium are not permitted within a bank. 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 13,814 ± 100 cubic feet at a nominal T_{avg} of 561°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological towers shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

MONTHLY OPERATING REPORTS

6.9.1.5 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.6.a Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle, or any part of a reload cycle for the following:

1. Moderator Temperature Coefficient BOL and EOL limits, and 300 ppm surveillance limit for Specification 3/4.1.1.3,

2. Shutdown Bank Insertion Limit for Specification 3/4.1.3.5,

3. Control Bank Insertion Limits for Specification 3/4.1.3.6,

4. Axial Flux Difference limits and target band for Specification 3/4.2.1,

5. Heat Flux Hot Channel Factor, K(Z), Power Factor Multiplier, and (F_{xy}^{RTP}) for Specification 3/4.2.2, and

6. Nuclear Enthalpy Rise Hot Channel Factor, and Power Factor Multiplier for Specification 3/4.2.3.

The CORE OPERATING LIMITS REPORT shall be maintained available in the Control Room.

6.9.1.6.b The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in:

1. WCAP 9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July, 1985 (W Proprietary).

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limit, 3.1.3.6 - Control Bank Insertion Limits, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - Nuclear Enthalpy Rise Hot Channel Factor.)

~~1~~ 1.A. WCAP 12942-P-A, "SAFETY EVALUATION SUPPORTING A MORE NEGATIVE EOL MODERATOR TEMPERATURE COEFFICIENT TECHNICAL SPECIFICATION FOR THE SOUTH TEXAS PROJECT ELECTRIC GENERATING STATION UNITS 1 AND 2."

CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient)

2. WCAP 8385, "POWER DISTRIBUTION AND LOAD FOLLOWING PROCEDURES TOPICAL REPORT", September, 1974 (W Proprietary).

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

3. Westinghouse letter NS-TMA-2198, T.M. Anderson (Westinghouse) to K. Kniel (Chief of Core Performance Branch, NRC) January 31, 1980 - Attachment: Operation and Safety Analysis Aspects of an Improved Load Follow Package.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control). Approved by NRC Supplement No. 4 to NUREG-0422, January, 1981 Docket Nos. 50-369 and 50-370.)

4. NUREG-0800, Standard Review Plan, U.S. Nuclear Regulatory Commission, Section 4.3, Nuclear Design, July, 1981. Branch Technical Position CPB 4.3-1, Westinghouse Constant Axial Offset Control (CAOC), Rev. 2, July 1981.

(Methodology for Specification 3.2.1 - Axial Flux Difference (Constant Axial Offset Control).)

5a 5. WCAP-10266-P-A, Rev. 2, WCAP-11524-NP-A, Rev. 2, "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code", Kabadi, J.N., et al., March 1987; including Addendum 1-A, "Power Shape Sensitivity Studies," December 1987 and Addendum 2-A, "BASH Methodology Improvements and Reliability Enhancements" May 1988.

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

5b. (see insert on attached page)

6.9.1.6.c The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

6.9.1.6.d The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector.

Insert for Technical Specification page 6-22

5b. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," April, 1995 (W Proprietary) for Loss of Coolant Accident (LOCA) Evaluation models with ZIRLO clad fuel for rod heatup calculation.

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