5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircaloy-4 or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with 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 ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 87 control element assemblies.

5.4 NOT USED

5.5 METEOROLOGICAL TOWERS LOCATION

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

5.6 FUEL STORAGE

CRITICALITY

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

a. A normal k_{eff} of less that or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties.

b. A nominal 10.185 inch center-to-center distance between fuel assemblies placed in Region 1 (cask storage pit) spent fuel storage racks.

c. A nominal 8.692 inch center-to-center distance between fuel assemblies in the Region 2 (spent fuel pool and refuelling canal) racks, except for the four southern-most racks in the spent fuel pool which have an increased N-S center-to-center nominal distance of 8.892 inches.

d. New or partially spent fuel assemblies may be allowed unrestricted storage in Region 1 racks.

e. New fuel assemblies may be stored in the Region 2 racks provided that they are stored in a "checkerboard pattern" as illustrated in Figure 5.6-1.

f. Partially spent fuel assemblies with a discharge burnup in the "accceptable range" of Figure 5.6-2 may be allowed unrestricted storage in the Region 2 racks.

g. Partially spent fuel assemblies with a discharge burnup in the "unacceptable range" of Figure 5.6-2 may be stored in theRegion 2 racks provided that they are stored in a "checkerboard pattern", as illustrated in Figure 5.6-1, with spent fuel in the "acceptable range" of Figure 5.6-3.

h. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent.

5.6.2 The k_{eff} for new fuel stored in the new fuel storage racks shall be less than or equal to 0.95 when flooded with unborated water and shall not exceed 0.98 when aqueous foam moderation is assumed.

DRAINAGE

5.6.3 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation +40.0 MSL. When fuel is being stored in the cask storage pit and/or the refueling canal, these areas will also be maintained at +40.0 MSL.

CAPACITY

5.6.4 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1849 fuel assemblies in the main pool, 255 fuel assemblies in the cask storage pit and after permanent plant shutdown 294 fuel assemblies in the refueling canal. The heat load from spent fuel stored in the refueling canal racks shall not exceed 1.72x10E6 BTU/Hr. Fuel shall not be stored in the spent fuel racks in the cask storage pit or the refueling canal unless all of the racks are installed in each respective area per the design.

5.7 NOT USED

CORE OPERATING LIMITS REPORT COLR (Continued)

6) "CESEC - Digital Simulation for a Combustion Engineering Nuclear Steam Supply System," (CE letter LD-82-001 and NRC SE to CE dated April 3, 1984). (Methodology for Specification 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.1 for Movable Control Assemblies - CEA Position, 3.1.3.6 for Regulating and group P CEA Insertion Limits, and 3.2.3 for Azimuthal Power Tilt).

7) "Qualification of Reactor Physics Methods for the Pressurized Water Reactors of the Entergy System," ENEAD-01-P. (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).

8) "Fuel Rod Maximum Allowable Gas Pressure," CEN-372-P-A. (Methodology for Specification 3.2.1, Linear Heat Rate).

9) "Technical Description Manual for the CENTS Code," WCAP-15996-P-A. (Methodology for Specification 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.1 for Movable Control Assemblies - CEA Position, 3.1.3.6 for Regulating and group P CEA Insertion Limits, and 3.2.3 for Azimuthal Power Tilt).

10) "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model," CENPD-132, Supplement 4-P-A. (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt and 3.2.7 for ASI).

11) "Implementation of ZIRLO Material Cladding in CE Nuclear Power Fuel Assembly Designs," CENPD-404-P-A (Methodology for Specification 3.1.1.3 for MTC, 3.2.1 for Linear Heat Rate, 3.2.3 for Azimuthal Power Tilt, and 3.2.7 for ASI).

12) "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," WCAP-11596-P-A; "ANC: A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A; and "ANC: A Westinghouse Advanced Nodal Computer Code: Enhancements to ANC Rod Power Recovery," WCAP-10965-P-A Addendum 1. (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).

13) "Qualification of the Two-Dimensional Transport Code PARAGON," WCAP-16045-P-A (Methodology for Specifications 3.1.1.1 and 3.1.1.2 for Shutdown Margins, 3.1.1.3 for MTC, 3.1.3.6 for Regulating and group P CEA Insertion Limits, 3.1.2.9 Boron Dilution (Calculation of CBC & IBW), and 3.9.1 Boron Concentration).

6.9.1.11.2 The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal 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.

6.9.1.11.3 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.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted in accordance with 10 CFR 50.4 within the time period specified for each report.

6.10 Not Used