APPENDIX 4.3A

→(DRN 00-1820, R10; 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

4.3A FUEL CYCLE 21

The following subsections discuss the fuel system design, nuclear design, thermal-hydraulic design and reactor protection and monitoring system changes for the subject fuel cycle at Waterford 3.
←(DRN 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308)
Operating conditions for this cycle were assumed to be consistent with those of previous cycles and are summarized as full power operation under base load conditions.
←(DRN 02-1477, R12)

Cycle 2 information was submitted to the NRC via References 1 and 2. The NRC's Safety Evaluation Report for Cycle 2 was provided in Reference 3.

4.3A.1 GENERAL DESCRIPTION

→(DRN 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308)
The Waterford-3 Cycle 21 core will consist entirely of assemblies of the Next Generation Fuel (NGF) design; specifically, Fresh Region EE assemblies, once burned Region DD, and twice burned Region CC and AA assemblies. See Sections 4.2.2.1 and 4.2.2.2 for details of the NGF fuel assembly and fuel rod designs.
←(DRN 00-1820, R10; 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308)

Control element assembly patterns and in-core instrument locations are shown in Figure 4.3A-4 and Figure 4.3A-5 respectively.

4.3A.2 FUEL SYSTEM DESIGN

4.3A.2.1 Mechanical Design

4.3A.2.1.1 Fuel Design

→(DRN 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308)
The Cycle 21 core consists of those assembly types and numbers listed in Table 4.3A-1. All fuel assemblies in the Cycle 21 core are of the NGF design.
←(DRN 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308)

→(DRN 02-1477, R12; 04-502, R13; 05-508, R14)
←(DRN 02-1477, R12; 04-502, R13; 05-508, R14)

→(DRN 04-502, R13)
←(DRN 04-502, R13)

→(DRN 00-1820, R10)
←(DRN 00-1820, R10)

4.3A.2.1.2 Clad Collapse

→(DRN 06-1059, R15; EC-9533, R302; EC-13881, R304)
The NGF fuel (UO_2) and IFBA rods in this cycle are initially pressurized with helium to the amount determined to be sufficient to prevent any gross clad deformation under the combined effect of external pressure and long term creep. The analyses of these rods credit the support of pellets and/or the holddown spring to prevent gross deformation (see also Sections 4.2.1.2.1 and 4.2.1.2.5).
←(DRN 06-1059, R15; EC-9533, R302; EC-13881, R304, LBDCR 15-035, R309)

4.3A.2.2 Mitigation of Guide Tube Wear

All fuel assemblies have stainless steel sleeves installed in the guide tubes to prevent guide tube wear.

4.3A.2.3 Thermal Design

→(DRN 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

The thermal performance of composite fuel rods that envelope the rods of fuel batches present in Cycle 21 have been evaluated using the NRC approved FATES3B version of the C-E fuel evaluation model (References 6, 7 and 32) and the Zirconium Diboride (ZrB_2) burnable absorber methodology described in Reference 35. The analysis was performed using a power history that enveloped the power and burnup levels representative of the peak pin at each burnup interval, from beginning of cycle to end of cycle burnups. The burnup range analyzed is in excess of that expected at the end of the Cycle.

←(EC-30663, R307, LBDCR 14-008, R308)

Reference 35 describes Westinghouse's 15 year fabrication and operational experience with ZrB_2 IFBA and the implementation and effect of using the coating on the C-E fuel assembly design and safety analyses. The neutronics effect, the helium production effect on internal gas pressure, and the mechanical and thermal effects of the coating thickness are all taken into account in the design and safety evaluations for C-E designed PWRs as described in that Reference.

←(DRN 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302)

→(EC-9533, R302)

The methodology for modeling the NGF design is described in the CE 16x16 Next Generation Fuel Topical Report, Reference 43.

←(EC-13881, R304)

4.3A.2.4 Chemical Design

→(EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308)

The metallurgical design specifications of the fuel cladding and other fuel assembly components for the NGF fuel used in Cycle 21 are essentially the same as those of the fuel regions included in Cycle 1. The NGF design of Region EE, Region DD, Region CC and Region AA include Optimized ZIRLO™ for the cladding and spacer grids (Reference 44) and ZIRLO™ for the CEA guide tubes (Reference 43). The introduction of these material changes does not impose any new water chemistry requirements relative to those employed for the standard fuel assembly.

←(EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308)

4.3A.2.5 Shoulder Gap Adequacy

→(DRN 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308)

Adequate shoulder gap is predicted for all NGF Regions of fuel in Cycle 21. This conclusion is based upon the fuel rod growth models of Reference 34 for Zircaloy, Reference 45 for ZIRLO™, and Reference 43 for Optimized ZIRLO™. The shoulder gap evaluation for Regions with the NGF design demonstrates that the initial shoulder gap reduction of approximately 0.5 inches relative to the non-NGF design is accommodated by the improved dimensional stability of the NGF cladding and CEA guide tube materials (Optimized ZIRLO™ and ZIRLO™, respectively).

←(DRN 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308)

4.3A.3 NUCLEAR DESIGN

4.3A.3.1 Physics Characteristics

4.3A.3.1.1 Fuel Management

→(DRN 00-1820, R10; 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308)

The Cycle 21 core consists of those assembly types and numbers listed in 4.3A-1. Twenty-one (21) Region BB and seventy-six (76) Region CC assemblies irradiated during Cycle 20 will be removed from the core and replaced with ninety-six (96) fresh Region EE assemblies and one (1) twice-burned Region AA assembly that was not loaded in the Cycle 20 core. One hundred (100) Region DD and twenty (20) Region CC assemblies in the core during Cycle 20 will be retained for Cycle 21.

←(DRN (00-1820, R10; 02-1477, R12; 04-502, R13; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

→(DRN 00-1820, R10; 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

The Cycle 21 core makes use of a low-leakage fuel management scheme in which eight (8) previously burned Sub-Region CA, CC, DA, DD, DF, and DG assemblies are each placed on the core periphery. One (1) previously burned Sub-Region AD assembly is placed in the center location. The ninety-six (96) fresh Region EE (Sub-Regions EA, EB, EC, ED, and EE) assemblies are located throughout the interior of the core, where they are arranged with other previously burned Region DD and CC fuel assemblies in a pattern that minimizes power peaking, and reduces both core leakage and the total neutron fluence to the reactor vessel.

The Cycle 21 center assembly is a twice-burned Region AA assembly that was held in the Spent Fuel Pool during the Cycle 19 and Cycle 20 operations. It had previously been loaded in the core for the Cycle 17 and Cycle 18 operation.

Fuel rod enrichment and Zirc Diboride configurations for the Region EE fuel are presented in Figure 4.3A-1.

The Cycle 21 reload fuel enrichment and region size will provide a nominal best estimate cycle length of 485 EFPD (498 EFPD with coastdown) based on operation at 3716 MWth and a Cycle 20 nominal endpoint of 507 EFPD. Depending on the actual Cycle 20 endpoint, the Cycle 21 core could deliver as much as 498 EFPD (511 EFPD with coastdown) or as little as 477 EFPD (490 EFPD with coastdown) on a best estimate basis. The Cycle 20 termination burnup has been assumed to be between 482 and 522 EFPD.

Figures 4.3A-3a and 4.3A-3b display the beginning of Cycle 21 and the end of Cycle 21 (502 EFPD) assembly average burnup distributions. These burnup distributions are based on Cycle 20 endpoints of 482 and 522 EFPD, respectively.

Table 4.3A-2 provides a comparison of characteristic physics parameters for Cycle 21 to the same parameters for Cycle 20, the Reference Cycle. The values in this table are intended to represent nominal core parameters. Those values used in the safety analyses (see Chapter 15) contain appropriate uncertainties, or incorporate values to bound future operating cycles, and in all cases are conservative with respect to the values calculated for Cycle 21.

Table 4.3A-3 presents a summary of CEA reactivity worths and allowances for the end of Cycle 21 full power steam line break transient. The full power steam line break was chosen as a reasonable illustration of the CEA reactivity worth.

←(DRN (00-1820, R10; 02-1477, R12; 04-502, R13; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

→(DRN 02-1477, R12; 04-502, R13)

The CEA core locations and group identifications are shown in Figure 4.3A-4. At the end of Cycle 11, the eight (8) Part-Length CEAs comprising Bank P were replaced with full-length, full strength CEAs and reassigned to Bank A. Four (4) full-length CEAs in Shutdown Bank A were reassigned to Bank P. Additionally, the four 4 Element CEAs in Shutdown Bank A, that span two fuel assemblies at the core periphery's major axes, were removed from the core. The Waterford 3 CEA Bank configurations are shown in Figure 4.3A-4. Commencing with Cycle 12, the Waterford 3 core has a total of 87 CEAs, all of the standard five element design. The assumed power dependent insertion limits (PDIL) for regulating groups and CEA Group P are shown in Figures 4.3A-6 and 4.3A-7 respectively. Table 4.3A-4 shows the reactivity worths of various CEA groups calculated at full power conditions for this cycle and the Reference Cycle.

←(DRN 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15)

4.3A.3.1.2 Power Distribution

Figures 4.3A-8 through 4.3A-10 illustrate the calculated All Rods Out (ARO) planar radial power distributions during this cycle. The one-pin planar radial power peaks presented in these figures represent the middle region of the core. Time points at the beginning, middle, and end of cycle were chosen to display the variation in maximum planar radial peak as a function of burnup.

The calculated radial power distributions described in this section do not include any uncertainties or allowances. The calculations performed to determine these radial power peaks explicitly account for augmented power peaking which is characteristic of fuel rods adjacent to the water holes.

→(DRN 02-1477, R12; 04-502, R13)

The following endpoints apply to Figures 4.3A-8 through 4.3A-10:

→(DRN 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

BOC21 values based on EOC20 = 482 EFPD, BOC21 = 0 EFPD
MOC21 values based on EOC20 = 522 EFPD, MOC21 = 240 EFPD
EOC21 values based on EOC20 = 522 EFPD, EOC21 = 502 EFPD

←(DRN 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304, LBDCR 14-008, R308, LBDCR 15-035, R309)

4.3A.3.1.3 Maximum Fuel Rod Burnup

→(DRN 00-1820, R10; 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304, LBDCR 14-008, R308, LBDCR 15-035, R309)

The Cycle 21 length will be limited to assure the maximum projected fuel rod burnup is less than the 60,000 MWD/T limit presented in Reference 34. The physics data which are input to cycle safety and fuel performance analyses are developed from explicit fine mesh calculations of fuel rod power and exposure. Burnup dependent physics data (e.g., maximum fuel rod fluence and fuel rod power histories) conservatively envelope core and fuel rod behavior at maximum burnups as well as lower burnups.

←(DRN 00-1820, R10; 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

4.3A.3.2 Safety Related Data

4.3A.3.2.1 Augmentation Factors

As indicated in Reference 5, the increased power peaking associated with the small interpellet gaps found in modern fuel rods (non-densifying fuel in pre-pressurized tubes) is insignificant compared to the uncertainties in the safety analyses. The report concluded that augmentation factors can be eliminated from the reload analyses of any reactor loaded exclusively with this type of fuel. Therefore, augmentation factors have been eliminated for Waterford 3.

4.3A.3.3 Physics Analysis Methods

4.3A.3.3.1 Analytical Input To In-Core Measurements

→(EC-9533, R302)

In-core detector measurement constants to be used in evaluating the reload cycle power distributions were calculated in accordance with Reference 42.

←(EC-9533, R302)

4.3A.3.3.2 Uncertainties In Measured Power Distribution

→(EC-9533, R302)

The planar radial power distribution measurement uncertainty based upon Reference 42 is applied to the COLSS and CPC on-line calculations which use planar radial power peaks. The axial and three dimensional power distribution measurement uncertainties were determined using the values in Reference 42 in conjunction with other monitoring and protection system measurement uncertainties.

←(EC-9533, R302)

4.3A.3.3.3 Nuclear Design Methodology

→(DRN 06-1059, R15)

Beginning with Cycle 15, the Advanced Nodal Code (ANC) (References 37, 38, and 39) was implemented in the reload design analysis. ANC is an advanced nodal analysis theory code capable of two- or three-dimensional calculations. Also, beginning with Cycle 15, PARAGON (Reference 40) computer code was implemented in the reload design analysis. PARAGON is a two-dimensional transport theory based code that calculates lattice physics constants. These are the same methods and models that have been used in other Westinghouse reload cycle designs. These codes are replacements for the ROCS/DIT computer codes.

The primary purpose of PARAGON is to provide input data for use in three dimensional core simulator codes. This includes macroscopic cross sections, microscopic cross sections for feedback adjustments to the macroscopic cross sections, pin factors for pin power reconstruction calculations, and discontinuity factors for a nodal method solution. PARAGON can be used as a standalone or as a direct replacement for all the previously licensed Westinghouse Pressurized Water Reactor ("PWR") lattice codes, such as PHOENIX-P, as approved by the NRC in Reference 40.

→(EC-9533, R302)

PARAGON is a two-dimensional multi-group neutron (and gamma) transport code. The PARAGON flux solution calculation uses Collision Probability theory within the interface current method to solve the integral transport equation. Throughout the whole calculation, PARAGON uses the exact heterogeneous geometry of the assembly and the same energy groups as in the cross-section library to compute the multi-group fluxes for each micro-region location of the assembly.

In order to generate the multi-group data that will be used by a core simulator code, PARAGON goes through four steps of calculations: resonance self-shielding, flux solution, homogenization, and burnup calculation.

ANC (for Advanced Nodal Code) is the three-dimensional core simulator code in the Westinghouse nuclear design code system. The ANC nodal flux solution is based on a set of two-group diffusion theory nodal balance equations that are solved using a solution method based on the nodal expansion method (NEM). This method and the specific approximations made in the ANC implementation provide an accurate representation of the core nodal neutronics. ANC is used to calculate core reactivity, reactivity coefficients, critical boron, rod worths, and core, assembly, and rod power distributions for normal and off-normal conditions for use in design and safety analyses. The ANC computer code is also used in the COLSS/CPC uncertainty analysis, as a replacement for the ROCS code, which in turn was a replacement for the FLAIR computer code.

←(DRN 06-1059, R15; EC-9533, R302)

4.3A.4 THERMAL-HYDRAULIC DESIGN

4.3A.4.1 <u>DNBR Analysis</u>

→(DRN 02-523, R12; 03-2058, R14; EC-9533, R302; EC-13881, R304; EC-30663, R307)

Steady state DNBR analyses at the rated power level of 3716 MWT have been performed using the TORC computer code described in Reference 11, the WSSV-T and ABB-NV critical heat flux correlations applicable to NGF assemblies described in Reference 41 and 46, respectively, the TORC modeling methods described in References 11 and 13, and the CETOP code described in Reference 14.

←(DRN 02-523, R12; 03-2058, R14; EC-9533, R302; EC-30663, R307)

Table 4.3A-5 contains a list of pertinent thermal-hydraulic design parameters. The Modified Statistical Combination of Uncertainties (MSCU) methodology presented in Reference 15 was applied with Waterford 3 specific data using the calculational factors listed in Table 4.3A-5 and other uncertainty factors at the 95/95 confidence/probability level to define a design limit of 1.24 over a DNBR range of 1.0 to 1.24, applicable to both the ABB-NV and WSSV-T correlations.

←(EC-13881, R304)

The DNBR limit includes the following allowances:

→(EC-13881, R304)

1. NRC specified allowances for TORC code uncertainty.

2. Rod bow penalty as discussed in Section 4.3A.4.2 below.

→(EC-9533, R302)

←(EC-9533, R302; EC-13881, R304)

4.3A.4.2 Effects Of Fuel Rod Bowing on DNBR Margin

→(DRN 03-2058, R14; EC-9533, R302; EC-13881, R304)

Effects of fuel rod bowing on DNBR margin have been incorporated in the safety and setpoint analyses in the manner discussed in Reference 19. The penalty used for this analysis is valid for bundle burnups up to 33,000 MWD/T. This penalty is included in the 1.24 DNBR limit, applicable to both the ABB-NV and WSSV-T correlations.

←(EC-9533, R302; EC-13881, R304)

For assemblies with burnup greater than 33,000 MWD/T sufficient available margin exists to offset rod bow penalties due to the lower radial power peaks in these higher burnup batches. Hence the rod bow penalty based upon Reference 19 for 33,000 MWD/T is applicable for all assembly burnups expected.

←DRN 03-2058, R14)

4.3A.5 REACTOR PROTECTION AND MONITORING

4.3A.5.1 Introduction

The Core Protection Calculator (CPC) System is designed to provide the low DNBR and high Local Power Density (LPD) trips to (1) ensure that the specified acceptable fuel design limits on departure from nucleate boiling and centerline fuel melting are not exceeded during Anticipated Operational Occurrences (AOOs) and (2) assist the Engineered Safety Features System in limiting the consequences of certain postulated accidents. The CPCS is further described in subsection 7.2.1.1.2.5.

The CPC/CEAC in conjunction with the balance of the Reactor Protection System (RPS) must be capable of providing protection for certain specified design basis events, provided that at the initiation of these occurrences the Nuclear Steam Supply System, its sub-systems, components and parameters are maintained within operating limits and Limiting Conditions for Operation (LCOs).

4.3A.5.2 CPCS Software Modifications

The CPC/CEAC software for Waterford 3 was modified prior to Cycle 2. This modification implemented the CPC Improvement Program, including algorithms and plant specific data base changes, changes to the list of addressable constants and implementation of the Reload Data Block (RDB).

The Waterford 3 CPC/CEAC algorithms are the same as those implemented at SONGS-2 and -3 (Cycle 3) and at ANO-2 (Cycle 6) and described in References 21 and 22. The revised list of addressable constants are defined in Reference 23. The software modifications are described in References 23, 24, 25, and 29. All changes were implemented per the established software change procedures, References 26 and 27.

4.3A.5.3 Addressable Constants

Certain CPC constants are addressable so that they can be changed as required during operation. Addressable constants include (1) constants that are measured during startup (e.g., shape annealing matrix, boundary point power correlation coefficients, and adjustments for CEA shadowing and planar radial peaking factors), (2) uncertainty factors to account for processing and measurement uncertainties in DNBR and LPD calculations (BERRO through BERR4), (3) trip setpoints and (4) miscellaneous items (e.g., penalty factor multipliers, CEAC penalty factor time delay, pre-trip setpoints, CEAC inoperable flag, calibration constants, etc.).

Trip setpoints, uncertainty factors and other addressable constants have been determined consistent with the software and methodology established in the CPC Improvement Program (Reference 23, 24 and 25) and the cycle design, performance, and safety analyses.

4.3A.5.4 Digital Monitoring System (COLSS)

The Core Operating Limit Supervisory System (COLSS) is a monitoring system that initiates alarms if the LCO on DNBR, peak linear heat rate, core power, axial shape index, or core azimuthal tilt are exceeded. The COLSS is further described in subsection 7.7.1.5. The COLSS data base and uncertainties have been updated to reflect the current core design.

4.3A.6 REFERENCES TO APPENDIX 4.3A

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2. W3P86-3328 dated October 1, 1986.

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4. C.O. Thomas (NRC), to A.E. Scherer (C-E), "Acceptance for Referencing of Licensing Special Report LD-84-043, CEA Guide Tube Wear Sleeve Modification," September 7, 1984.

5. EPRI NP-3966-CCM, "CEPAN Method of Analyzing Creep Collapse of Oval Cladding Volume 5: Evaluation of Interpellet Gap Formation and Clad Collapse in Modern PWR Fuel Rods," April 1985.

6. CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report," July 1974.

7. CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989.

8. R.A. Clark (NRC) to A.E. Lundvall, Jr., (BG&E), "Safety Evaluation of CEN-161 (FATES3)," March 31, 1983.

9. ENEAD-02-NP, “Verification of CECOR Coefficient Methodology for Application to Pressurized Water Reactors of the Entergy System,” September 1994.

10. CENPD-266-P-A, "The ROCS and DIT Computer Codes for Nuclear Design," April 1983.

10B. CENPD-275-P-A, "C-E Methodology for Core Designs Containing Gadelinia-Urania Burnable Absorbers," May 1988.

11. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986.

12. CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids, Part 1, Uniform Axial Power Distribution," September 1976.

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→(EC-13881, R304)

16. Deleted

17. Deleted

←(EC-13881, R304)

18. NUREG-0787, Supplement 1, "Safety Evaluation Report Related to the Operation of Waterford Steam Electric Station, Unit No. 3," Docket No. 50-382, October 1981.

19. CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983.

20. Robert A. Clark (NRC) to William Cavanaugh III, (AP&L), "Operation of ANO-2 During Cycle 2," July 21 1981 (Safety Evaluation Report and License Amendment No. 26 for ANO-2).

21. CEN-304-P, Rev. 01-P, "Functional Requirements for a Control Element Assembly Calculator," May 1986.

22. CEN-305-P, Rev. 01-P, "Functional Requirements for a Core Protection Calculator," May 1986.

23. CEN-308-P-A, "CPC/CEAC Software Modifications for the CPC Improvement Program," April 1986.

24. CEN-310-P-A, "CPC and Methodology Changes for the CPC Improvement Program," April 1986.

25. CEN-330-P-A, "Rev. 00-P, "CPC/CEAC Software Modifications for the CPC Improvement Program Reload Data Block," October 1987.

26. CEN-39(A)-P, Rev. 03, "CPC Protection Algorithm Software Change Procedure," January 1986.

27. CEN-39(A)-P, Supplement 1-P, Rev. 03-P, "CPC Protection Algorithm Software Change Procedure Supplement 1," April 1986.

28. CEN-323-P-A, "Reload Data Block Constant Installation Guidelines," September 1986.

29. CEN-281(S)-P, "CPC/CEAC Software Modifications for San Onofre Nuclear Steam Generating Station Units No. 2 and 3," July 1984.

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31. Deleted.

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33. CEN-372-P-A, “Fuel Rod Maximum Allowable Gas Pressure,” May 1990

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→(EC-13881, R304)

35. WCAP-16072-P-A, “Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs,” August 2004.

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→(DRN 03-270, R12-B)

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←(DRN 03-270, R12-B)

→(EC-9533, R302)

37. WCAP-11596-P-A, “Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,” June 1988

38. WCAP-10965-P-A, “ANC: A Westinghouse Advanced Nodal Computer Code,” September 1986

39. WCAP-10965-P-A Addendum 1, “ANC: A Westinghouse Advanced Nodal Computer Code: Enhancements to ANC Rod Power Recovery,” April 1989

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41. WCAP-16523-P-A, Rev. 0. “Westinghouse Correlations WSSV and WSSV-T for Predicting Critical Heat Flux in Rod Bundles with Side-Supported Mixing Vanes”, August 2007.

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→(EC-13881, R304, LBDCR 14-008, R038)

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45. CENPD-404-P-A, “Implementation of ZIRLO™ Material Cladding in CE Nuclear Power Fuel Assembly Designs,” November 2001.

46. CENPD-387-P-A, “ABB Critical Heat Flux Correlations for PWR Fuel,” May 2000.

←(EC-13881, R304, LBDCR 14-008, R308)

→(DRN 00-1820, R10; 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

Waterford - 3 Cycle 21 Core Loading Description

Sub-Batch ID	Number of Assemblies	Pattern ID	UO_2 Rods per Assembly	Nominal Enrichment (wt. %)	ZrB_2 Rods per Assembly	Shim Loading (ZrB_2)	Number of Fuel Rods (Including ZrB_2 Rods)	Number of ZrB_2 Rods
EA	20	PAT1633IFB	164	4.38	20	2.0 X	3680	400
		(60 IFBA)	12	3.98	40	2.0 X	1040	800
EB	8	PAT1649IFB	116	4.38	68	2.0 X	1472	544
		(112 IFBA)	8	3.98	44	2.0 X	416	352
EC	4	PAT1632IFB	176	3.98	8	2.0 X	736	32
		(48 IFBA)	12	3.58	40	2.0 X	208	160
ED	48	PAT1649IFB	116	3.98	68	2.0 X	8832	3264
		(112 IFBA)	8	3.58	44	2.0 X	2496	2112
EE	16	PAT1636IFB	112	3.98	72	2.0 X	2944	1152
		(124 IFBA)	0	3.58	52	2.0 X	832	832
Total	96						22656	9648
DA	16	PAT1632IFB	176	4.53	8	2.0 X	2944	128
		(48 IFBA)	12	4.23	40	2.0 X	832	640
DB	4	PAT1648IFB	124	4.53	60	2.0 X	736	240
		(88 IFBA)	24	4.23	28	2.0 X	208	112
DC	12	PAT1649IFB	116	4.53	68	2.0 X	2208	816
		(112 IFBA)	8	4.23	44	2.0 X	624	528
DD	8	PAT1650IFB	92	4.53	98	2.0 X	1472	736
		(136 IFBA)	8	4.23	44	2.0 X	416	352
DE	8	PAT1649IFB	116	3.83	68	2.0 X	1472	544
		(112 IFBA)	8	3.53	44	2.0 X	416	352
DF	20	PAT1636IFB	112	3.83	72	2.0 X	3680	1440
		(124 IFBA)	0	3.53	52	2.0 X	1040	1040
DG	32	PAT1650IFB	92	3.83	92	2.0 X	5888	2944
		(136 IFBA)	8	3.53	44	2.0 X	1664	1408
Total	100						23600	11280
CA	8	PAT1643IFB	160	4.23	24	2.0 X	1472	192
		(68 IFBA)	8	3.92	44	2.0 X	416	352
CC	8	PAT1648IFB	124	3.81	60	2.0 X	1472	480
		(88 IFBA)	24	3.51	28	2.0 X	416	224
CE	4	PAT1649IFB	116	3.81	68	2.0 X	736	272
		(112 IFBA)	8	3.51	44	2.0 X	208	176
Total	20						4720	1696
AD	1	PAT1635IFB	136	3.90	48	2.0 X	184	48
		(100 IFBA)	0	3.50	52	2.0 X	52	52
Total	1						236	100
Grand Total	217						51212	22724

←(DRN 00-1820, R10; 02-1477, R12; 04-502, R13; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

TABLE 4.3A-2 Revision 309 (06/16)

→(DRN 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

NOMINAL PHYSICS CHARACTERISTICS

	Units	Reference Cycle**	Cycle 21*
Dissolved Boron			
Dissolved Boron Concentration for Criticality, CEAs Withdrawn, Hot Full Power, Equilibrium Xenon	PPM	527	583
Inverse Boron Worth			
Hot Full Power, Equilibrium Xenon			
BOC	PPM/%Δρ	130	130
EOC	PPM/%Δρ	103	104
Moderator Temperature Coefficients			
Hot Full Power, Equilibrium Xenon			
BOC	10^{-4} Δρ/°F	-1.5	-1.4
EOC	10^{-4} Δρ/°F	-2.8	-2.9
Doppler Coefficient			
Hot Zero Power, BOC	10^{-5} Δρ/°F	-1.7	-1.7
Hot Full Power, Equilibrium Xenon			
BOC	10^{-5} Δρ/°F	-1.6	-1.6
EOC	10^{-5} Δρ/°F	-1.8	-1.8
Total Delayed Neutron Fraction βeff			
BOC	-----------	0.0061	0.0061
EOC	-----------	0.0050	0.0050
Neutron Generation Time, τ*			
BOC	10^{-6} sec	17.0	18.0
EOC	10^{-6} sec	27.9	27.9

* values vary with cycle
** Reference cycle is Cycle 20

←(DRN 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

TABLE 4.3A-3 Revision 309 (06/16)

LIMITING VALUES OF
REACTIVITY WORTHS AND ALLOWANCES FOR HOT
FULL POWER STEAM LINE BREAK, %$\Delta\rho$, END-OF-CYCLE (EOC)

→(DRN 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

	Reference Cycle**	Cycle 21*
Net Available Scram Worth (No LOAC)	8.1	7.9

* values vary with cycle
** Reference cycle is Cycle 20

←(DRN 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

TABLE 4.3A-4 Revision 309 (06/16)

REACTIVITY WORTH OF CEA REGULATING GROUPS
AT HOT FULL POWER, %$\Delta\rho$

→(DRN 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

	Beginning of Cycle		End Of Cycle	
	Reference Cycle**	Cycle 21*	Reference Cycle**	Cycle 21*
←(EC-13881, R304)				
Group P @ 0"	0.4	0.4	0.4	0.4
Group 6 @ 0"	0.4	0.4	0.4	0.4
Group 5 @ 0"	0.4	0.3	0.4	0.4

Note: Values shown assume sequential group insertion

* Values vary with cycle

→(EC-13881, R304)

** Reference cycle is Cycle 20

←(DRN 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, 309)

→(DRN 00-1820, R10; 02-523, R12; 02-1477, R12; 04-502, R13 ,03-2058, R14; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)

Cycle 21
Thermal-Hydraulic Parameters at Full Power

General Characteristics	Units	Cycle 20	Cycle 21
Total Heat Output (Core Only)	(MW_{th})	3716	3716
	(10^6 Btu/hr)	12680	12680
Fraction of Heat Generated in Fuel Rod	---	0.975	0.975
Primary System Pressure (Nominal)	(psia)	2250	2250
Inlet Temperature (Maximum Indicated)	(°F)	543	543
Total Reactor Coolant Flow (Minimum Steady State)	(gpm)	390,220	390,220
	(10^6 lbm/hr)	148.0	148.0
Coolant Flow Through Core (Minimum)	(10^6 lbm/hr)	144.2	144.2
Hydraulic Diameter (Nominal Channel)	(ft)	0.041	0.041
Core Average Mass Velocity	(10^6 lbm/hr-ft^2)	2.55	2.55
Pressure Drop Across Core (at Minimum Steady State Core Flow Rate)	(psi)	20.7	20.7
Total Pressure Drop Across Vessel (Based on Nominal Dimensions and Minimum Steady State Flow)	(psi)	46.6	46.6
Core Average Heat Flux (Accounts for Fraction of Heat Generated in Fuel Rod and Axial Densification Factor)	(Btu/hr-ft^2)	198,016(1)	198,016(1)
Total Heat Transfer Area (Accounts for Axial Densification Factor)	(ft^2)	62,432(1)	62,432(1)
Film Coefficient at Average Conditions	(Btu/hr-ft^2-°F)	6092	6092
Average Film Temperature Difference	(°F)	32.50(1)	32.50(1)
Average Linear Heat Rate of Undensified Fuel Rod (Accounts for Fraction of Heat Generated In Fuel Rod)	(kw/ft)	5.67(1)	5.67(1)
Average Core Enthalpy Rise	(Btu/lbm)	88.0	88.0
Maximum Clad Surface Temperature	(°F)	656.76(1)	656.76(1)
Engineering Heat Flux Factor	---	1.03(2),(3)	1.03(2),(3)
Engineering Factor on Hot Channel Heat Input	---	1.03(2),(3)	1.03(2),(3)
Rod Pitch, Bowing and Clad Diameter Factor	---	1.05(2),(3)	1.05(2),(3)
Fuel Densification Factor (Axial)	---	1.002	1.002

(1) Based on 100 shims (non fuel rods) in the core and 217 NGF assemblies.
(2) These factors have been combined statistically with other uncertainty factors at 95/95 confidence/probability level and included in the design limit on ABB-NV minimum DNBR and WSSV-T minimum DNBR.
(3) These values are generic based on fuel design drawing tolerances and are also applicable to NGF.

←(DRN 00-1820, R10; 02-523, R12; 02-1477, R12; 04-502, R13 ,03-2058, R14; 05-508, R14; 06-1059, R15; EC-9533, R302; EC-13881, R304; EC-30663, R307, LBDCR 14-008, R308, LBDCR 15-035, R309)