3.5 THERMAL AND HYDRAULIC DESIGN AND EVALUATION

3.5.1 GENERAL

This section presents the thermal and hydraulic characteristic data and design methodology. The objective of the thermal and hydraulic design of the reactor is to ensure that the core can meet steady state and transient performance requirements without violating the design bases. The principal thermal and hydraulic design bases are related to DNB, fuel/clad melting, and hydraulic loading. Instrument and control uncertainties, delays between parameter changes, RPS trip signals, and initiation of CEA movement are involved in the transient calculations. The RPS monitors and trips the reactor upon sensing an adverse condition. The Engineered Safety Feature Actuation Signal (ESFAS) provides automatic corrective action when operating parameters exceed their setpoints. The Safety Analysis determines the setpoints such that the design bases will not be exceeded during AOOs and most postulated accidents. The Safety Analysis (Chapter 14) discusses each of the DBEs in detail.

3.5.1.1 Cycle Summaries

The following cycle summaries provide a brief synopsis of major changes associated with each cycle.

a. Unit 1

1. Cycle 2

   The core power level was increased from 2560 MWt to 2700 MWt.

   Several design changes to the type D fuel improved the thermal performance and were included in the performance analysis. The parameters that affected the gap conductance, such as decreased pellet/cladding gap and the increased pellet density that decreased the effects of densification, were responsible for the improved thermal performance of the type D fuel.

2. Cycle 3

   Excessive CEA guide tube wear was identified. In order to reduce the wear on the guide tubes and CEA fingers, SS sleeves were inserted into the top end of some of the guide tubes. The sleeves protect the guide tubes against wear by CEA fingers.

   Part-length CEAs were also removed to reduce guide tube wear and dummy CEA plugs were inserted to minimize the increase in bypass flow.

   Minimum Departure from Nucleate Boiling Ratio was calculated using TORC/CE-1. TORC was used to generate the LCO in the Technical Specifications and was also used for all AOOs and postulated accidents. The TORC thermal hydraulics code replaces COSMO-INTHERMIC.

   TORC uses the CE-1 DNBR correlation, whereas previous cycles used COSMO-INTHERMIC, which uses the W-3 DNBR correlation (References 2, 3, 4, and 5).

   The detailed version of TORC (Reference 2) is a benchmarking code. The simplified version (Reference 3) runs considerably faster but is set to be more conservative when benchmarked against the detailed version.

Mechanical design and power distribution uncertainty factors used in the calculation of thermal margin were previously combined multiplicatively. In Cycle 3, these factors:

F_q^e = Engineering Factor,

F_q^n = Nuclear Factor,

F_q^f = Fuel Rod Bowing Factor and,

F_q^p = Poison Rod Factor

were combined using a root-sum-square (RSS) technique. The RSS technique is appropriate for combining random uncertainties. The multiplicative technique is appropriate for combining systematic/dependent uncertainties.

Augmentation factors were calculated using the FATES (Reference 1) fuel rod model.

3. Cycle 4

The guide tubes of reload fuel assemblies were either sleeved or their flow holes were changed. The sleeves protect the guide tubes against wear. The flow hole modification reduced coolant flow and thereby reduced the CEA vibration which caused the guide tubes to wear. Irradiated fuel assemblies previously resident in CEA locations, but not having sleeves, were sleeved in order to regain structural margin.

Fuel assemblies exceeding the 24,000 MWD/MTU rod bow penalty threshold were placed in core locations where their power density was sufficiently low to offset rod bow penalties on the MDNBR limit.

4. Cycle 5

The RPS was modified to include an asymmetric steam generator transient protection trip function. The trip function originates from the thermal margin/low pressure (TM/LP) logic and trips the reactor for those AOOs associated with secondary system malfunctions which would result in asymmetric primary loop temperatures. The limiting event is the loss of load to one steam generator caused by the closure of a single main steam isolation valve.

5. Cycle 6

The TORC/CE-1 thermal design code has been replaced by the CETOP/CE-1 code (Reference 12). The treatment of core system parameter uncertainties on the DNBR SAFDL has been changed from the deterministic approach to statistical combination of uncertainties (SCU) (References 13, 14, and 15). The calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) have been combined statistically with other uncertainty factors to define a new design limit on CE-1 MDNBR at the 95/95 confidence/probability level.

The performance of the fuel has been analyzed using FATES-3 (Reference 17), a fuel performance code.

The analysis with these methodology changes resulted in a MDNBR limit of 1.23.

6. Cycle 7

   The effects of fuel rod bowing on DNBR margin were evaluated using the methods described in Reference 11.

7. Cycle 8

   The PLCEA plugs were removed for Cycle 8 to facilitate the installation of the Reactor Vessel Level Monitoring System and to expedite refueling outage operations. An assessment concluded that the removal of the CEA plugs from all eight partial length rod locations has insignificant effect on the thermal and hydraulic design.

   The axial fuel densification factor is reduced from 1.01 to 1.002. A negative bias of 15% is added to the FTC data used in the safety analysis to establish consistency with the bias in the ROCS/DIT topical. This bias is used conservatively by selective application.

8. Cycle 9

   The fuel thermal performance calculations used FATES3B (Reference 18) which is an updated version of the FATES3 (Reference 17) fuel evaluation model. The statistically derived DNBR limit was reduced from 1.23 to 1.21. This reduction resulted from Nuclear Regulatory Commission (NRC) approval of a reduced CE-1 DNBR limit for CE's 14x14 fuel.

9. Cycle 10

   The DNBR SAFDL calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) have been combined statistically with other uncertainty factors using the extended statistical combination of uncertainties (ESCU) methodology of Reference 19. This combination is used to derive an overall uncertainty allowance which, when used with the CE-1 critical heat flux (CHF) correlation DNBR design limit of 1.15 for 14x14 fuel, provides a 95/95 probability/confidence level of assurance against DNB occurring during steady state operation and AOOs. The statistically derived ESCU uncertainty allowance includes a 0.006 DNBR rod bow penalty which accounts for the adverse effects of rod bowing on CHF for 14x14 fuel with burnup not exceeding 45 GWD/T.

10. Cycle 11

    The fresh assemblies (Batch N) in Cycle 11 employ the GUARDIAN™ debris-resistant fuel design and large envelope Zircaloy grids. This fuel design results in a greater hydraulic resistance than the debris-resistant LEF design of the Batch 2L fuel used in Unit 2 Cycle 9. This increase in hydraulic resistance results in a slight decrease in the inlet flow for the Batch N fuel. The TORC and CETOP models used in the DNB analysis account for this flow reduction.

11. Cycle 12

    The fresh assemblies (Batch P) in Cycle 12 employ the GUARDIAN™ design with an improved top grid design. The top laser-welded grid introduces a backup arch in each grid cell in addition to the existing backup arches in the peripheral cell locations. This results in a greater hydraulic resistance in the Batch P assemblies. The TORC and CETOP models used in the DNB analysis account for the hydraulic resistance of each assembly type.

12. Cycle 13

    The standard fresh assemblies (Batch 1R) in Unit 1 Cycle 13 employ the laser welded, straight strip GUARDIAN™ grid design. The Batch 1R fuel also utilizes laser welded wavy strip intermediate Zircaloy spacer grids. The Unit 1 Cycle 13 DNB analysis explicitly accounted for the resistance of each assembly type in the Unit 1 Cycle 13 core.

13. Cycle 14

    The standard fresh assemblies (Batch 1S) in Unit 1 Cycle 14 employ the laser welded, straight strip GUARDIAN™ grid design. The Batch 1S fuel also utilizes laser welded wavy strip intermediate Zircaloy spacer grids. The Unit 1 Cycle 14 DNB analysis explicitly accounted for the resistance of each assembly type in the Unit 1 Cycle 14 core.

14. Cycle 15

    The standard fresh assemblies (Batch 1T) in Unit 1 Cycle 15 employ the laser welded, straight strip GUARDIAN™ grid design. The Batch 1T fuel also utilizes laser welded wavy strip intermediate Zircaloy spacer grids. The Unit 1 Cycle 15 DNB analysis explicitly accounted for the resistance of each assembly type in the Unit 1 Cycle 15 core.

    Mid-cycle the ABB-NV CHF correlation was approved. Therefore, both the CE-1 and ABB-NV correlations are applicable for DNB analysis for Unit 1 starting in Cycle 15. Calculations done with the new correlation were implemented mid-cycle.

15. Cycle 16

    The 96 fresh Turbo assemblies (Batch 1V) both with ZIRLO and OPTIN cladding in Unit 1 Cycle 16 employ the same straight strip GUARDIAN™ grid design as earlier Batches S and T. Batch 1V Turbo fuel also utilizes two bottom and one top straight strip advanced no-vane Zircaloy spacer grids and five intermediate advanced Zircaloy spacer grids with mixing vanes. The Unit 1 Cycle 16 DNB analysis explicitly accounts for the resistance of each assembly type in Unit 1 Cycle 16 mixed core.

    ABB-NV and ABB-TV CHF correlations have been developed applicable to Westinghouse standard and Turbo types of fuel assemblies, respectively, in Reference 20. The ABB-TV correlation has a better CHF performance compared to the ABB-NV correlation due to the presence of mixing vane grids in Turbo type fuel assemblies.

    Because of the higher hydraulic resistance of the Turbo fuel assemblies compared to the standard fuel assemblies, Turbo assemblies lose flow to standard fuel assemblies along the height of the core. This loss of flow from Turbo fuel assemblies to the surrounding standard fuel assemblies in mixed core configuration such as in Unit 1 Cycle 16 is explicitly accounted for in the TORC DNB analysis.

Dual bundle tests have shown that the TORC prediction of this diversion cross-flow from a Turbo fuel assembly to a standard fuel assembly is fairly close with the test results. However, in order to conservatively compensate for any minor adverse effect on DNB margin assessment of Turbo type fuel assemblies due to small differences between the test results and TORC prediction of the diversion cross-flow from a Turbo fuel assembly to a standard fuel assembly, a margin neutral approach has been adopted for Unit 1 Cycle 16 DNB analysis. Based on this approach, Turbo fuel assemblies in Cycle 16 have been conservatively treated as standard fuel assemblies and the DNB margin assessment for Cycle 16 has been performed using ABB-NV CHF correlation documented in Reference 20. In other words, no margin credit has been taken for the ABB-TV CHF correlation that is applicable to Turbo fuel.

16. Cycle 17

    Eighty-eight fresh assemblies were installed for Unit 1 Cycle 17 (batch designation 1W).

    Batch 1W is the third batch of VAP for Unit 1. Erbia remains the burnable absorber, the Erbia fuel pins have cutback regions of 10.5 inches at the top of the rod and 12.0 inches at the bottom.

    Batch 1W is the second batch of the Turbo fuel assembly design for Unit 1. All but four assemblies utilized the same design and the same grid cage design as the U2C15 Westinghouse LFAs, see Section 3.7.3.13. The four different assemblies do not have the increased backup-arch length and have been given a unique sub-batch identifier. All of the fuel was manufactured by Westinghouse at their Columbia, SC facility. The cladding material for all fresh fuel is ZIRLO™.

    As Unit 1 contains approximately 85% of the Turbo fuel assembly design, transient analyses were updated to utilize the ABB-TV CHF correlation. Since the Turbo fuel has a non-mixing vane lower axial section and an upper section with mixing grids, both the ABB-NV and the ABB-TV CHF correlations are applied in the safety analysis.

17. Cycle 18

    The fresh assemblies (Batch 1X) for Unit 1 Cycle 18 employ the laser-welded, straight strip GUARDIAN grid design. Cycle 18 also employs the third full batch of the Turbo advanced grid design for Unit 1 and is the first Unit 1 core to contain all Turbo fuel assemblies.

18. Cycle 19

    Ninety-six fresh assemblies (Batch 1Z) were loaded for Unit 1 Cycle 19. All assemblies utilized in Cycle 19 employ the laser-welded, straight strip GUARDIAN grid design and the TURBO advanced grid design.

19. Cycle 20

    The rated thermal power was increased from 2700 MWt to 2737 MWt.

    Ninety-two fresh assemblies (eighty-eight Batch AA and four Batch 2X7) were loaded for Unit 1 Cycle 20. All assemblies utilized in Cycle 20 employ the laser-welded, straight strip GUARDIAN grid design and the TURBO advanced grid design.

20. Cycle 21

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. A thermal hydraulic compatibility analysis was performed to assess the impact on DNB performance of the core with addition of the AREVA fuel assemblies and the co-resident fuel assemblies.

    AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

21. Cycle 22

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. A thermal hydraulic compatibility analysis was performed to assess the impact on DNB performance of the core with addition of the AREVA fuel assemblies and the co-resident fuel assemblies.

    AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

22. Cycle 23

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

23. Cycle 24

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA/Framatome. These assemblies used the HTP™ correlation to determine DNBR. AREVA/Framatome design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

24. Cycle 25

    One hundred one fresh assemblies were loaded, which were manufactured by Framatome. These assemblies used the HTP™ correlation to determine DNBR. Four of the fresh assemblies contained 8 armoring inert rods per bundle. The Framatome design uses HTP™ / HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

. Unit 2

1. Cycle 2

   The core power level was increased from 2560 MWt to 2700 MWt.

   The following modifications made to Unit 1 Cycles 3 and 4 were also made to the design for Unit 2 Cycle 2:

   a) SS sleeves were installed in CEA guide tubes of selected reload fuel assemblies;
   b) The size and number of CEA guide tube flow holes were modified on other selected fuel assemblies;
   c) Part-length CEAs were removed and replaced by CEA plugs;
   d) Minimum Departure from Nucleate Boiling Ratio was calculated using TORC/CE-1; and
   e) Mechanical design and power distribution uncertainty factors, used in the calculation of thermal margin, which were previously combined multiplicatively were combined using an RSS technique.

2. Cycle 3

   All fuel assemblies placed in CEA locations had SS sleeves installed in the guide tubes in order to prevent guide tube wear.

   Augmentation factors were calculated using the FATES (Reference 1) fuel rod model.

   Fuel assemblies exceeding the 24,000 MWD/MTU rod bow penalty threshold were placed in core locations where their power density was sufficiently low to offset rod bow penalties on the MDNBR limit.

3. Cycle 4

   The RPS was modified to include the asymmetric steam generator transient protection trip function. The trip function originates from the TM/LP logic and trips the reactor for those AOOs associated with secondary system malfunctions which would result in asymmetric primary loop temperatures. The most limiting event is the loss of load to one steam generator caused by the closure of a single main steam isolation valve.

4. Cycle 5

   The TORC/CE-1 thermal design code has been replaced by the CETOP/CE-1 Code (Reference 12). The treatment of core system parameter uncertainties has been changed from the deterministic approach to SCU (References 13, 14, and 15). The DNBR SAFDL calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) have been combined statistically with other uncertainty factors to define a new design limit on CE-1 MDNBR at the 95/95 confidence/probability level.

   The analysis with these methodology changes resulted in a MDNBR limit of 1.23.

5. Cycle 6

   The effects of fuel rod bowing on DNBR margin were evaluated using the methods described in Reference 11.

6. Cycle 7

   The statistically derived DNBR limit was reduced from the value of 1.23 to a value of 1.21. The reduction results from NRC approval of a reduced CE-1 DNBR limit for CE's 14x14 fuel. At the time the SCU analysis was approved for the Calvert Cliffs units, NRC review of the applicability of the CE-1 CHF correlation to rods with nonuniform APDs was incomplete. An interim CE-1 DNBR limit of 1.19 was thus used in the original SCU analysis. In the review of CE's nonuniform APD topical report, the NRC reduced the CE-1 DNBR limit from 1.19 to 1.15 for 14x14 fuel. The SCU DNBR limit was correspondingly reduced from 1.23 to 1.21. The 1.21 SCU DNBR limit includes the following penalties imposed by the NRC in their review of the SCU analysis.

   - Critical heat flux correlation cross validation penalty (5% increase in standard deviation of CHF correlation uncertainty distribution).
   - T-H code uncertainty penalty (5%, equal to two standard deviations).

   The 1.21 SCU DNBR limit also includes a 0.006 DNBR rod bow penalty which accounts for the adverse effects of rod bowing on CHF for 14x14 fuel with burnup not exceeding 45 GWD/T.

   The axial fuel densification factor was reduced from 1.01 to 1.002 to make it consistent with existing calculations.

   The PLCEA plugs were removed for Cycle 7 to facilitate the installation of the Reactor Vessel Level Monitoring System and to expedite refueling outage operations.

7. Cycle 8

   The thermal performance of the fuel was evaluated using the FATES3B (Reference 18) fuel evaluation model.

8. Cycle 9

   The fresh assemblies (Batch L) in Cycle 9 have small flow hole LEF plates, which result in greater hydraulic resistance than the LEF plates of the irradiated fuel. This increase in hydraulic resistance will result in a slight decrease in the inlet flow for these Batch L assemblies. The cycle specific TORC and CETOP models used in the Cycle 9 DNB analyses accounted for this flow reduction.

   The calculational factors (engineering heat flux factor, engineering factor on hot channel heat input, rod pitch and clad diameter factor) have been combined statistically with other uncertainty factors using the ESCU methods (Reference 19). This combination is used to derive an overall uncertainty allowance which, when used with the CE-1 CHF correlation design limit of 1.15 for 14x14 fuel, provides a 95/95 probability/confidence level of assurance against DNB occurring during steady state operation or AOOs.

9. Cycle 10

   The fresh assemblies (Batch 2M) in Cycle 10 employ the GUARDIAN™ debris resistant fuel design. This fuel design results in a greater hydraulic resistance than the Batch 2L fuel which trapped debris by employing small flow holes in the lower end fitting.

10. Cycle 11

   The standard fresh assemblies (Batch 2N) in Unit 2 Cycle 11 are the second Unit 2 batch to employ the GUARDIAN™ design. For Batch 2N, a straight strip GUARDIAN™ grid design was introduced, replacing the previously employed wavy GUARDIAN™ grid design. The new straight strip GUARDIAN™ grid design has lower hydraulic resistance than the Batch 2M wavy strip GUARDIAN™ grid design.

   Laser welded wavy strip intermediate Zircaloy spacer grids were also introduced for the Batch 2N fuel, replacing the previously employed TIG welded wavy strip intermediate Zircaloy spacer grids. The laser welded intermediate grids have a slightly lower hydraulic resistance than the TIG welded grids. The Cycle 11 DNB analysis explicitly accounted for the resistance of each assembly type in the Unit 2 Cycle 11 core.

11. Cycle 12

   The standard fresh assemblies (Batch 2P) in Unit 2 Cycle 12 employ the laser welded, straight strip GUARDIAN™ grid design. The Batch 2P fuel also utilizes laser welded wavy strip intermediate Zircaloy spacer grids. The perimeter strips have small guide holes to match pins on the grid assembly weld fixture. This is to ensure more consistent alignment of the grid strips within the fixture during welding. The Unit 2 Cycle 12 DNB analysis explicitly accounted for the resistance of each assembly type in the Unit 2 Cycle 12 core.

12. Cycle 13

   The standard fresh assemblies (Batch 2R) in Unit 2 Cycle 13 employ the laser welded, straight strip GUARDIAN™ grid design. The Batch 2R fuel also utilizes laser welded wavy strip intermediate Zircaloy spacer grids. The DNB analysis for Unit 2 Cycle 13 explicity accounted for the resistance of each assembly type.

13. Cycle 14

   The fresh assemblies (Batch 2S) in Unit 2 Cycle 14 employ the laser-welded, straight-strip GUARDIAN™ grid design. The Batch 2S fuel also utilizes laser-welded wavy-strip intermediate Zircaloy spacer grids. The DNB analysis for Unit 2 Cycle 14 explicitly accounted for the resistance of each assembly type. The ABB-NV DNB correlation was used instead of the CE-1 correlation.

14. Cycle 15

    The fresh assemblies (Batch 2T) manufactured by Westinghouse in Unit 2 Cycle 15 employ the laser-welded, straight strip GUARDIAN™ grid design. Unit 2 Cycle 15 also contains the first full batch of the Turbo advanced grid design for Unit 2. Unit 1 received the first full batch of the Turbo advanced grid design at Calvert Cliffs in Unit 1 Cycle 16. The Turbo grid features include mixing vanes (at five of the eight spacer grid locations) and new rod retention device known as I-springs (at all eight spacer grid locations). The Unit 2 Cycle 15 DNB analysis explicitly accounts for the resistance of each assembly type in the Unit 2 Cycle 15 mixed core.

    ABB-NV and ABB-TV CHF correlations have been developed applicable to Westinghouse standard and Turbo types of fuel assemblies, respectively, in Reference 20. The ABB-TV correlation has a better CHF performance compared to the ABB-NV correlation due to the presence of mixing vane grids in Turbo type fuel assemblies.

    Because of the higher hydraulic resistance of the Turbo fuel assemblies compared to the standard fuel assemblies, Turbo assemblies lose flow to standard fuel assemblies along the height of the core. This loss of flow from Turbo fuel assemblies to the surrounding standard fuel assemblies in mixed core configuration such as in Unit 2 Cycle 15 is explicitly accounted for in the TORC DNB analysis.

    Dual bundle tests have shown that the TORC prediction of this diversion cross-flow from a Turbo fuel assembly to a standard fuel assembly is fairly close with the test results. However, in order to conservatively compensate for any minor adverse effect on DNB margin assessment of Turbo type fuel assemblies due to small differences between the test results and TORC prediction of the diversion cross-flow from a Turbo fuel assembly to a standard fuel assembly, a margin neutral approach has been adopted for Unit 2 Cycle 15 DNB analysis. Based on this approach, Turbo fuel assemblies in Cycle 15 have been conservatively treated as standard fuel assemblies and the DNB margin assessment for Cycle 15 has been performed using ABB-NV CHF correlation documented in Reference 20. In other words, no margin credit has been taken for the ABB-TV CHF correlation that is applicable to Turbo fuel.

    Batch 2T also contains LFAs from Westinghouse and FANP/AREVA. See Section 3.7.3.13 for detailed discussion on the Unit 2 Cycle 15 LFA.

    In order to accommodate growth of the guide tube thimbles, the center guide tube recess hole in the lower end fitting of the fuel assembly was made approximately 1-7/8 inches deeper. This was accomplished via a process known as electric discharge machining and was performed or the fresh fuel after delivery.

15. Cycle 16

    The fresh fuel assemblies (Batch 2V) for Unit 2 Cycle 16 employ the laser-welded, straight strip GUARDIAN grid design. Cycle 16 also contains the second full batch of the Turbo advanced grid design for Unit 2. The TORC DNB analysis explicitly accounts for the difference in the hydraulic resistance between the Turbo and standard fuel assemblies. This is the second cycle for the Westinghouse and AREVA LFAs.

16. Cycle 17

    The fresh assemblies (Batch 2W) for Unit 2 Cycle 17 employ the laser-welded, straight strip GUARDIAN grid design. Cycle 17 also employs the third full batch of the Turbo advanced grid design for Unit 2 and is the first Unit 2 core to contain all Turbo fuel assemblies.

17. Cycle 18

    The rated thermal power was increased from 2700 MWt to 2737 MWt.

    Ninety-six fresh assemblies (Batch 2X) were loaded for Unit 2 Cycle 18. All assemblies utilized in Cycle 18 employ the laser-welded, straight strip GUARDIAN grid design and the TURBO advanced grid design.

18. Cycle 19

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. A thermal-hydraulic compatibility analysis was performed to assess the impact on DNB performance of the core with the addition of the AREVA fuel assemblies and the co-resident fuel assemblies.

    AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

19. Cycle 20

    One-hundred fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. A thermal-hydraulic compatibility analysis was performed to assess the impact on DNB performance of the core with the addition of the AREVA fuel assemblies and the co-resident fuel assemblies.

    AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

20. Cycle 21

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR.

    AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

21. Cycle 22

    Ninety-six fresh assemblies were loaded, which were manufactured by AREVA. These assemblies used the HTP correlation to determine DNBR. AREVA design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods.

22. Cycle 23

    Ninety-seven fresh assemblies were loaded, which were manufactured by Framatome. These assemblies used the HTP™ correlation to determine DNBR.

    The Framatome design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and MS® clad fuel rods"

23. Cycle 24

    Ninety-six fresh assemblies, including one Lead Test Assembly (LTA), were loaded, which were manufactured by Framatome. These assemblies used the HTP™ correlation to determine DNBR. The Framatome design uses HTP™/HMP™ spacer grids, FUELGUARD™ lower end fitting, reconstitutable upper end fitting, and M5® clad fuel rods. An alternate reduced free length plenum spring was approved for use starting with U2C24.

3.5.2 THERMAL AND HYDRAULIC DESIGN BASES

Avoidance of thermally- or hydraulically-induced fuel damage during normal steady state operation and during AOOs is the principal thermal and hydraulic design basis. In order to satisfy the design basis for reactor operation, the following design limits are established, but exceeding these limits will not necessarily result in fuel damage. The RPS provides for automatic reactor trip and the ESFAS provides other corrective action before these design limits are violated for AOOs. However, there is a small probability of limited fuel damage for certain other DBEs discussed in Chapter 14.

3.5.2.1 Minimum Departure from Nucleate Boiling Ratio

The minimum allowed DNBR provides at least a 95% probability with a 95% confidence that DNB does not occur on a fuel rod having the calculated MDNBR during steady state operation and AOOs. The ABB-NV correlation coupled with the CETOP code provides at least this probability and confidence. The DNBR limit may be modified to account for the possibility of fuel rod bow at burnups in excess of 45,000 MWD/MTU . Starting with Unit 1 Cycle 17, the ABB-TV correlation was used in conjunction with the ABB-NV correlation to make DNB determinations.

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the HTP™ correlation was used to determine DNBR for the AREVA/Framatome fuel assemblies. The burnup limit for fuel rod bow impact on DNBR is greater than fuel assembly burnups for all anticipated core designs. Therefore, it is not necessary to modify the HTP™ correlation limit due to the effects of fuel rod bow.

3.5.2.2 Fuel Design Basis

a. Fuel Melt

   The UO_2 melting point will not be reached during steady state operation and AOOs. For Westinghouse fuel assemblies, the UO_2 melting point is 5080°F unirradiated, and reduced by 58°F per 10,000 MWD/MTU burnup and reduced by 10.4°F for each weight percent of erbia of the maximum core erbia loading. For AREVA/Framatome fuel assemblies, a bounding value of 4595°F was used for the UO_2 melting point. This value bounds all anticipated limiting fuel burnup distributions and Gadolinia concentrations. The thermal and hydraulic parameters which influence the fuel centerline temperature include maximum linear heat rate (LHR), coolant velocity, pressure, temperature, clad temperature, fuel-to-gap conductance, fuel burnup, and UO_2 temperature.

b. Fuel Cladding Integrity

The fuel design bases for fuel clad integrity and fuel assembly integrity are given in Section 3.2.3.5. Thermal and hydraulic parameters that influence the fuel integrity include maximum LHR, core coolant velocity, coolant temperature, clad temperature, fuel-to-clad gap conductance, fuel burnup, and UO_2 temperature.

The cladding minimizes deformation from external hydraulic pressure or internal gas/pellet pressure. Excessive contraction of the clad may lead to power spiking and excessive expansion may significantly decrease the flow channel area.

Conformance with the design limits and conformance with the design bases are sufficient to ensure fuel clad integrity, fuel assembly integrity, and the avoidance of thermally- or hydraulically-induced fuel damage for steady-state operation and AOOs.

3.5.2.3 Hydraulic Stability

Reactor internal flow passages and fuel coolant channels are designed to prevent hydraulic instabilities. Permissible flow maldistributions are limited by design to be compatible with the specified thermal design criteria.

3.5.3 STATISTICAL COMBINATION OF UNCERTAINTIES

The input data required for a detailed thermal-hydraulic analysis can be defined by type: (1) system parameters which describe the physical system and are not monitored during reactor operation; and (2) state parameters, which describe the operational state of the reactor and are monitored during operation. There is a degree of uncertainty in the value used for each of the input parameters used in the design safety analyses. This uncertainty has been handled in the past by assuming that each variable affecting DNB is at the extreme most adverse limit of its uncertainty range. The assumption that all factors are simultaneously at their most adverse values leads to conservative restrictions in reactor operation.

Beginning with Unit 1 Cycle 6, a new methodology was applied to statistically combine uncertainties in the calculation of new limits for Calvert Cliffs. These limits will ensure that neither the DNB nor fuel centerline melt design bases will be violated. The methodology is presented in three parts (References 13, 14, and 15). Part 1 (Reference 13) describes the application of the SCU to the development of the local power density (LPD) and TM/LP LSSSs. These are used in the analog RPS to protect against fuel centerline melt and DNB, respectively. Part 2 (Reference 14) uses SCU methods to develop a new DNBR limit which accommodates system parameter uncertainties. Part 3 (Reference 15) uses SCU methods to define LCOs.

For Unit 1 Cycle 10, an improved method was used for statistically combining uncertainties for the CE calculated TM/LP LSSS and DNB LCO. The extended combination of uncertainties (ESCU) methodology (Reference 19) is a modification of the SCU methodology (References 13, 14, and 15).

With the introduction of AREVA fuel assemblies for Unit 2 Cycle 19 and Unit 1 Cycle 21, a new method for the statistical combination of uncertainties was used to verify the TM/LP LSSS, the LPD LSSS, the DNB LCO, and LPD LCO. This methodology is described in Section 14.1.4.1 and Reference 23.

3.5.4 REACTOR HYDRAULICS

3.5.4.1 Coolant Flow

Coolant enters the four inlet nozzles and flows down through the annular plenum between the reactor vessel and the core support barrel. Coolant continues through the flow skirt to the plenum below the core lower support structure. Pressure losses in the skirt and lower support structure help to even out the inlet flow distribution to the core. The coolant passes through the openings in the lower core plate and flows axially through the fuel assemblies. After passing through the core, the coolant flows past the fuel alignment plate and into the region outside the CEA shrouds. From this region the coolant flows across the CEA shrouds and passes out through the outlet sleeves on the core barrel to the two outlet nozzles.

The principal core bypass routes (coolant flow paths other than through the fuel assemblies and next to the fuel rods) are direct inlet-to-outlet coolant flow at the joint between the core support barrel sleeve and the outlet nozzle and the flow in the radial reflector region between the core shroud and core support barrel. A small portion flows into the guide tubes in the fuel assemblies. The flow through the guide tubes has been modified by a reduction in size of the flow holes in the bottom of the guide tubes in other fuel assemblies. The coolant required to cool the CEAs flows in the annulus between the CEA and the guide tube and into the region outside the CEA shrouds. A similar but smaller leakage will occur around the restriction to flow at the upper end of those guide tubes without CEAs. The design limits the total guide tube flow and core bypass flow to a maximum of 3.9% of total reactor vessel flow as compared to the calculated bypass flow shown in Table 3.5-2.

3.5.4.2 Pressure Losses

The irrecoverable pressure losses from the inlet to outlet nozzles are calculated using standard loss coefficient methods and information from flow model tests. The nominal design pressure losses are listed in Table 3.5-3.

3.5.4.3 Partial Flow Operation

The plant operates with all four Reactor Coolant Pumps functioning. Partial pump operation will only occur during transients prior to trip and is discussed in Chapter 14. The most limiting partial pump operation is during a Seized Rotor Event which is more limiting than a four-pump Loss of Coolant Flow Event.

Partial pump operation creates an unbalanced inlet and outlet nozzle flow rate and a core inlet flow maldistribution. Furthermore, unbalanced steam generator flow rates create a greater possibility of temperature nonuniformities at the core inlet due to incomplete mixing of the incoming coolant.

3.5.5 MAXIMUM CORE TEMPERATURE

The maximum core temperature occurs at the center of the hottest pellet. The temperature drops radially across the pellet, gap, clad and coolant film. Heat transfer correlations relate the physical properties, heat flux, and temperature drops. The different physical geometry and properties necessitate separate correlations. The Jens-Lottes/Dittus-Boelter equation, clad conductivity, gap conductance, and pellet conductance relate the temperature drops across the coolant film, cladding, gap, and pellet, respectively.

For AREVA/Framatome fuel, the maximum fuel centerline temperature reached during the event is explicitly calculated using heat structure in the S-RELAP5 model which represents the hot node in the core for either a UO_2 rod or Gadolinia rod, whichever is limiting for the fuel centerline melt.

3.5.6 DEPARTURE FROM NUCLEATE BOILING

3.5.6.1 Design Approach to Departure from Nucleate Boiling

The margin to DNB at any point in the core is expressed in terms of the DNBR. The DNBR is defined as the ratio of the heat flux predicted to produce DNB at specific local conditions to the calculated local heat flux at the same local conditions. At some point in the core the DNBR is a minimum and, at this point, the margin to DNB for the core is evaluated. The following items are important in determining the core margin to DNB:

a. the coolant inlet conditions (e.g., pressure, temperature, and velocity distribution),
b. the geometry,
c. the power level,
d. the nuclear power distribution,
e. the analytical methods used to predict local coolant conditions, and
f. the correlation used to predict DNB heat flux.

Correlations of DNB are derived from experimental data and reduced to key parameters. Correlations for DNB are intended only to predict actual DNB and, therefore, the concept of DNB ratio can be misleading if one attempts to associate a physical meaning rather than a statistical meaning. Because of the uncertainties associated with predicting DNB there is a finite probability that if a channel is operated at a specified DNB ratio greater than unity based on a particular correlation, it will be at or above its DNB heat flux. Therefore, the proper interpretation of DNB ratio is that it is a measure of the probability that DNB would occur in the particular design situation to which the DNB correlation is applied. This interpretation assumes that all operating parameters are known precisely and that the probability being evaluated is only that associated with the correlation. The approach used in design is to select core operating conditions and analytical methods in such a way that there is a very small probability that the actual hot subchannel coolant conditions are more severe than the calculated conditions used as input to the DNB correlation. Starting with Unit 1 Cycle 17, the ABB-TV correlation was used in conjunction with the ABB-NV correlation to make DNB determinations.

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the HTP correlation (Reference 24) was used to determine DNBR for the AREVA/Framatome fuel assemblies. The DNBR limit accounts for state and system parameter uncertainties.

3.5.6.2 Evaluation of Margin to DNB

DNBR analysis is performed over a wide range of coolant parameters to determine the envelope in which the DNBR is at least greater than the SAFDL. The inlet coolant flow distribution used was empirically developed from scale models.

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the XCOBRA-IIIC code was used to calculate the core thermal-hydraulic conditions for the AREVA/Framatome assemblies. These conditions are then used with the HTP correlation to determine DNBR for the AREVA/Framatome assemblies.

3.5.7 VAPOR FRACTION

At steady state the reactor is operated at a negative coolant quality (subcooled temperature). Therefore, vapor formation is minimized and, for the most adverse steady state conditions core vapor fraction is less than 0.1%. To avoid the possibility of DNB resulting from local flow oscillations, a conservative vapor fraction limit prevents flow instabilities. The limits to assure stable flow are based on avoiding flow regime changes in the hot channel that could affect the flow pressure drop characteristics and cause an instability. A thermal margin trip will occur before the flow instability limit is reached; thus DNB resulting from flow oscillations is prevented.

AREVA/Framatome fuel was added to the core starting from Unit 2 Cycle 19 and Unit 1 Cycle 21. Flow instability is not part of the AREVA evaluation and no changes to the flow stability occurred with the transition to AREVA/Framatome fuel.

3.5.8 THERMAL AND HYDRAULIC EVALUATION

The margin to CHF or DNB is expressed in terms of DNBR.

3.5.8.1 Statistical Analysis of Hot Channel Factors

Random variations from nominal values in enrichment, pellet density, pellet diameter and clad diameter, will affect the engineering heat flux factor on heat flux. Similar random variations in heat flux, as well as rod diameter, pitch, and bow contribute to the enthalpy rise factor. The calculation of these factors uses randomly-collected inspection data on "as-manufactured" fuel assemblies. Statistically random and independent constituent uncertainties are combined statistically.

3.5.8.2 Fuel Temperature Conditions

An assessment of the fuel centerline temperature has been made. The results demonstrate that a significant margin to centerline fuel melting will exist over the normal range of plant operation.

3.5.8.3 Flow Stability

Flow oscillations of significant amplitude may be sustained in some channels when heat is added to two-phase flow in parallel channels. This possibility is mitigated by the low vapor fraction during steady state and AOOs.

The two-phase flow regimes may be classed as separated or homogeneous. Separated flow is annular or slug. Homogeneous flow is bubbly or froth flow. For homogeneous flow, the channel pressure drop continuously increases with increasing flow rate or increasing vapor fraction. A change in the flow regime to separated flow results in a change in the flow characteristics and flow oscillations in the parallel channels are then possible. Cross flow tends to damp the oscillations and tends to make the open channel array stable when parallel closed channels would not be stable.

3.5.8.4 AREVA/Framatome Fuel Assemblies

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the HTP correlation was used to determine DNBR for the AREVA/Framatome fuel assemblies. The HTP correlation is described in Reference 24 and is applicable to the following operating conditions and nominal range of fuel design parameters:

Parameter	**Minimum Value**	**Maximum Value**
Pressure, psia	1385	2425
Local Mass Flux, mlb/hr/ft^2	0.498	3.573
Inlet Enthalpy (Btu/lb)	382.3	649.9
Local Quality	---	0.515
Fuel Rod Diameter (in)	0.360	0.440
Fuel Rod Pitch (in)	0.496	0.580
Axial Spacer Span (in)	10.5	26.2
Hydraulic Diameter (in)	0.4571	0.5334
Heated Length (ft)	8.0	14.0

Based on the overall core conditions calculated at selected times during a transient evaluation, the XCOBRA-IIIC fuel assembly thermal-hydraulic code is used to calculate the flow and enthalpy distributions for the entire core and the DNB performance for the DNB-limiting assembly. The XCOBRA-IIIC model consists of a thermal-hydraulic model of the core (representing each assembly by a single “channel”) linked to a detailed thermal-hydraulic model of the limiting assembly (representing each subchannel by a single “channel”). The limiting assembly DNBR calculations are performed using an approved DNB correlation.

The use of XCOBRA-IIIC is limited to the “snapshot” mode when used for transients and is restricted from use in LOCAs and other calculations with flow reversal and recirculation as per the NRC licensing restrictions (Reference 25). This mode is based on a series of steady-state calculations for input over a series of time steps.

3.5.9 REFERENCES

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2. CENPD-161-P-A, "TORC Code: A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986
3. CENPD-206-P-A, "TORC Code Verification and Simplified Modeling Methods," June 1981
4. 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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23. EMF-1961(P)(A), Revision 0, “Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors,” Siemens Power Corporation, July 2000

24. EMF-92-153(P)(A), Revision 1, “HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel,” Siemens Power Corporation, March 2005

25. Letter from Mr. D. V. Pickett (NRC) to Mr. G. H. Gellrich (CCNPP), dated February 18, 2011, Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 - Amendment Re: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel (TAC Nos. ME2831 and ME2832)

TABLE 3.5-2

REACTOR COOLANT FLOWS IN BYPASS CHANNELS

BYPASS ROUTE	UNIT 1 PERCENT OF TOTAL REACTOR FLOW	UNIT 2 PERCENT OF TOTAL REACTOR FLOW
Outlet nozzle clearances	0.663	0.929
Alignment keyways	0.099	0.099
Core shroud annulus	0.624	0.594
Guide tubes	1.85	1.85
Total bypass	3.24	3.47

NOTE: The Unit 1 and Unit 2 bypass flows are based on a full core consisting of Framatome HTP^{Tm} assemblies. Units 1 and 2 bypass flow accounts for 35 ICI thimbles. The core shroud annulus bypass flow accounts for the appropriate number of top flange holes. The outlet nozzle clearance on Unit 2 is longer than that on Unit 1, accounting for an increase in the Unit 2 outlet nozzle bypass flow. The total design bypass flow rate used for safety analysis is typically 3.9% which accounts for uncertainties and the increase in hydraulic resistance due to postulated crud buildup.

TABLE 3.5-3

DESIGN REACTOR PRESSURE LOSSES

	VELOCITY, ft/sec	UNIT 1[(a)] PRESSURE LOSS, psid	UNIT 2[(a)] PRESSURE LOSS, psid
Inlet Nozzle and 90° Turn	42.0	5.7	5.7
Lower Plenum	7.5	9.6	9.6
Core	15.5	16.2	16.2
Core Outlet to Outlet Nozzle	46.8	6.7	6.7
TOTAL		38.2	38.2

(a) The values presented are valid for 2737 MWt, 2250 psia, 396,125 gpm, 548°F inlet temperature, and a full core of AREVA HTP™ fuel. Bounding analyses are also performed for the limiting conditions on power, pressure, flow, and temperature with mixed cores using both HTP™ and Turbo fuel.