3.4 NUCLEAR DESIGN AND EVALUATION

3.4.1 SUMMARY

This section summarizes the nuclear characteristics of the core and discusses the important design parameters which are of significance to the performance of the core during transient and steady state operation. A discussion of the nuclear design methods employed and comparisons with experiments which support the use of these methods is included. Summaries of nuclear parameters are presented in Table 3.4-1. Design limits for shutdown margin, reactivity coefficients, reactivity insertion rates, and power distribution are discussed in the appropriate sections.

Fuel enrichment and BPR distributions are shown in Table 3.3-1, Table 3.3-2, and Figure 3.3-4.

Physical features of the lattice, fuel assemblies, and CEAs are described in Section 3.3.2. The soluble boron insertion rates are sufficient to compensate for the maximum reactivity addition due to xenon burnout and normal plant cooldown.

3.4.2 REACTIVITY AND CONTROL REQUIREMENTS

At the beginning of each cycle, the core is loaded with sufficient fuel to generate essentially full power for the cycle length. This results in built-in excess reactivity (the reactivity present in the reactor with all control material withdrawn from the core) which must be sufficient to compensate for the reactivity lost during the cycle due to:

a. Fuel burnup;
b. Fission product buildup; and,
c. Negative reactivity feedbacks.

The excess reactivity must be stable and controlled through the cycle to permit power operations while maintaining the ability to rapidly shut down the reactor if necessary (Table 3.4-2).

Excess reactivity is controlled during the cycle by adjusting both the position of the CEAs and the concentration of boric acid dissolved in the RCS. The CEAs permit rapid changes in reactivity, as required for reactor trip, and may be used to compensate for changes in moderator temperature, fuel temperature, and moderator density associated with changes in power level.

Adjustment of the boric acid concentration is used to control the relatively slow reactivity changes associated with plant heatup and cooldown, fuel burnup, certain xenon variations, and slow power level changes. The use of boric acid dissolved in the reactor coolant makes it possible to maintain the CEAs in an essentially fully-withdrawn position during full power operation, thus minimizing distortions in power distribution. Although the boric acid system reduces reactivity relatively slowly, the rate of reduction is more than sufficient to maintain the shutdown margin against the effects of normal cooldown and xenon decay. Table 3.4-1 lists the predicted concentrations of natural boron required to maintain the first cycle critical under various conditions, assuming all CEAs to be fully withdrawn. The hot full power, equilibrium xenon, BOC boron concentration predictions for the present cycles are also given in Table 3.4-1.

Design criteria require reactivity control and stability. In a stable reactor, a reactivity perturbation during steady state operation leads to another steady state. This desirable, self-limiting characteristic is due to negative feedback of reactivity. The parameters used to quantify feedback are the reactivity coefficients which relate changes in core reactivity to variations in fuel and/or moderator conditions. Verifications of predicted nuclear parameters are conducted at the beginning of each cycle during startup testing (Chapter 13). Results of these tests verify prediction of the Moderator Temperature Coefficient. For each DBE in the Safety Analysis (Chapter 14), suitably conservative reactivity coefficient values are used. Values assumed in the transient analyses are listed in Chapter 14.

3.4.2.1 Fuel Temperature Coefficient

The FTC reflects the change of core reactivity per degree change in average fuel temperature. A change in fuel temperature affects the density of the fuel pellet and the nuclear density of the uranium in the pellet, thus changing the probability of interaction with a neutron. A fuel temperature change also modifies the reaction rates in uranium in both the thermal and epithermal neutron energy regimes.

The Doppler effect is the principal contributor to the change in reaction rate with fuel temperature in the epithermal range. This effect results from the increased (non-fissioning) neutron absorption in U-238 with increasing fuel temperature. This increase in neutron absorption rate with fuel temperature causes a negative FTC since the temperature increase decreases the fission rate. In the thermal energy regime, a change in reaction rate with fuel temperature arises from the effect of temperature dependent scattering properties of the fuel matrix on the thermal neutron spectrum. In typical PWR fuels containing strong resonance absorbers such as U-238 and Pu-240, the component of the FTC arising from the Doppler effect is more than a factor of ten larger than the thermal energy component.

The variation of FTC with temperature predicted for the first cycle is shown in Figure 3.4-1. The predicted first cycle hot, full power FTC was -1.06×10^{-5} $\Delta\rho$/°F which is approximately equivalent to -1.49×10^{-3} $\Delta\rho$/(kW/ft).

3.4.2.2 Moderator Temperature Coefficient

The MTC relates changes in reactivity to changes in the moderator average temperature and includes the effects of temperature on the moderator density.

Typically, an increase in the moderator temperature causes a decrease in moderator density and therefore less neutron thermalization which reduces core reactivity. When an increase in moderator temperature causes a decrease in reactivity, the core has a negative temperature coefficient. This adds to the stability since a temperature increase will reduce reactor power and vice versa.

When sufficient boron is present in the moderator, a reduction in moderator density also causes a reduction in the boron density in the core, thus producing a positive contribution to the MTC. One core design objective is to limit the MTC to values near zero, if positive, or to slightly negative values (the MTC positive limits are listed in the Technical Specifications). In order to limit the dissolved boron concentration and its positive contribution to the MTC, BPRs (shims) may be provided in the cycle design (Tables 3.3-1 and 3.3-2). The reactivity control provided by the shims makes possible a reduction in the dissolved boron concentration and therefore a reduction in the MTC.

Moderator Temperature Coefficient values for various core conditions during the first cycle and for the current cycle are given in Table 3.4-1. As shown in the table, the least negative value at full power conditions occurs in the unrodded core when the dissolved boron content is at its maximum. The MTC becomes more negative at the EOC due mainly to the reduction in the dissolved boron content with burnup. CEA insertion provides a negative contribution to the coefficient since a corresponding reduction in the dissolved boron content is required to maintain the reactor critical.

The effects of plutonium and fission products on the MTC are small when compared to the effects of dissolved boron changes. The buildup of fission product xenon supplies a positive contribution to the MTC for a constant boron concentration. However, when the dissolved boron concentration is reduced by the reactivity equivalent of the xenon, MTC becomes more negative.

The change in MTC as a function of boron concentration is almost linear and was about $+0.16 \times 10^{-4}$ $\Delta\rho$/F per 100 ppm soluble boron for the first cycle.

3.4.2.3 Moderator Pressure Coefficient

The Moderator Pressure Coefficient is the change in reactivity per unit change in RCS pressure. Since an increase in pressure increases the water density, the pressure coefficient is opposite in sign to the temperature coefficient. The reactivity effect of increasing the pressure is reduced in the presence of dissolved boron because an increase in coolant density also increases the boron density. The Moderator Pressure Coefficient decreases as the RCS boron concentration increases. The calculated pressure coefficients for the beginning and end of the first cycle at full power were $+0.3 \times 10^{-6}$ $\Delta\rho$/psi and $+2.6 \times 10^{-6}$ $\Delta\rho$/psi, respectively. The Moderator Pressure Coefficient was measured to be $-.5 \times 10^{-6}$ $\Delta\rho$/psi at the beginning of Unit 1 Cycle 1 with a nominal RCS temperature of 450°F by changing pressure in the 1100 psia to 2250 psia range. The pressure coefficient of reactivity is relatively insignificant and is several orders of magnitude smaller than the MTC.

3.4.2.4 Moderator Void Coefficient

The occurrence of small amounts of local subcooled boiling in the reactor during full power operation may result in small steam bubbles (voids). These are called voids because they contain almost no moderating nuclei. The average void fraction is the fraction by volume of the moderator that is in void form and it is substantially less than 1% at normal operating conditions. The change in reactivity associated with these voids in the moderator is the Void Coefficient of Reactivity. An increase in voids reduces the moderator density and decreases core reactivity. The presence of soluble boron tends to add a positive contribution to the void coefficient because an increase in voids results in a reduction in boron density in the core. The calculated values at the beginning and at the end of the first cycle were -0.1×10^{-3} $\Delta\rho$/% void and -1.3×10^{-3} $\Delta\rho$/% void, respectively.

3.4.2.5 Power Coefficient

The Power Coefficient is the change in core reactivity per percent change incore power level. Although all of the previously mentioned coefficients (FTC, MTC, Moderator Pressure Coefficient and the Moderator Void Coefficient) contribute to the Power Coefficient, only the MTC and the FTC are significant. To determine the change in reactivity with power, it is necessary to know the change in the average moderator temperature and effective fuel temperature with power.

The average moderator temperature is a linear function of power. The effective fuel temperature is dependent on both power level and burnup. Due to fuel pellet cracking and fission gas release with irradiation, this functional relationship changes during the cycle.

The Power Coefficient can be obtained from the following equation:

$$\frac{dRho}{dP} = \left(\frac{dRho}{dT_f}\ x\ \frac{dT_f}{dP}\right) + \left(\frac{dRho}{dT_m}\ x\ \frac{dT_m}{dP}\right)$$

The first term of the equation provides the fuel temperature contribution to the Power Coefficient. The first term is the product of FTC of reactivity and the effective change of fuel temperature with respect to power. The second term in the equation provides the moderator contribution to the Power Coefficient. The first factor is the MTC and the second factor is a constant since the moderator temperature is a linear function of power.

Since the factors $dRho/dT_f$ and $dRho/dT_m$ are functions of one or more independent variables (e.g., burnup, temperature, soluble boron content, xenon worth, and CEA insertion), the total Power Coefficient, dRho/dP, also depends on these variables.

Plots of the calculated FTC and a plot of the predicted Power Coefficient for the beginning of the first cycle are shown in Figures 3.4-1 and 3.4-2, respectively. The full power value of the Power Coefficient for the unrodded first cycle is $-1.49\times10^{-3}\ \Delta\rho/(kW/ft)$. The Power Coefficient becomes more negative with burnup due to the increasing negative FTC and MTC.

3.4.3 SHUTDOWN REACTIVITY CONTROL

The reactivity worth requirement of all the CEAs is determined by:

a. Shutdown reactivity margin;
b. Power defect (including moderator voids); and,
c. CEA bite.

The total worth of all CEAs provides adequate shutdown at full power even if the CEA with the most worth is stuck in the fully withdrawn position. Table 3.4-2 compares available CEA reactivity with the various required reactivity components at BOC and EOC for the first cycle. The table also lists the current cycle's most limiting values of reactivity worths and allowances. Individual components are discussed below.

3.4.3.1 Shutdown Reactivity Margin

The shutdown margin requirement is based on the reactivity requirements for the most limiting postulated accident. This requirement varies throughout core life as a function of fuel depletion, boron concentration, and RCS temperature.

Sufficient CEA worth must be available for rapid insertion to ensure that:

a. The reactor can be made subcritical from all operating conditions.
b. Reactivity transients associated with postulated accident conditions are controllable.

A steam line break or excess load event (Chapter 14) at EOC requires the maximum CEA shutdown margin due to the large, rapid cooldown and the large, negative MTC. These accidents provide the basis for the shutdown margin Technical Specification with RCS temperature above 200°F.

Allowances of 2.0% and 2.4% $\Delta\rho$ at the beginning of first cycle and at the end of first cycle, respectively, were made for the predicted shutdown margin and safety feature allowances at hot, zero power conditions. The current cycle values of CEA worths and allowances for the limiting event are listed in Table 3.4-2.

3.4.3.2 Power Defect

The power defect is the reactivity change in the core from hot zero power to a higher power. During a reactor trip, this increase in reactivity must be compensated for by the CEAs.

The power defect increases as the power level is increased and results from the following changes:

a. Fuel Temperature Variation
b. Moderator Temperature Variation
c. Moderator Voids Variation

These three reactivity variations are described below.

a. Fuel Temperature Variation

The reactivity increase that occurs when the fuel temperature decreases from its full (or other) power value to its zero power value is primarily due to changes in epithermal absorption resonance's (the Doppler effect) in U-238. As exposure accumulates, concentrations of plutonium increase making the FTC more negative. However, pellet swelling and clad creepdown associated with increasing exposure improves the heat transfer which decreases the fuel temperature. The competing effects of a more negative FTC and decreasing fuel temperature tends to minimize the burnup dependence of the fuel temperature defect. The Unit 1 Cycle 1 reactivity increase associated with a fuel temperature decrease from full power to zero power was 1.7% $\Delta\rho$ at BOC and 1.8% $\Delta\rho$ at EOC.

b. Moderator Temperature Variation

The average reactor coolant temperature increases with increasing power. This decreases the moderator density and causes a reactivity change which is usually negative (except at very high boron concentrations). The moderator temperature variation allowance is large enough to compensate for any reactivity increase that may occur when the moderator temperature decreases from full power to zero power. This reactivity increase, which is primarily due to the negative MTC, is largest at the EOC when the soluble boron concentration is near zero and the moderator coefficient is strongly negative. At BOC, when the MTC is less negative, the reactivity change associated with the moderator temperature change is smaller.

c. Moderator Voids Variation

Increasing the power level causes a decrease in reactivity resulting from formation of small steam bubbles (voids) due to local boiling. The average void content in the core is very small and is estimated to be less than 1% at full power. As with the moderator temperature effect, the maximum increase in reactivity from full to zero power occurs at EOC when the least amount of dissolved boron is present. At BOC the void coefficient is essentially zero.

3.4.3.3 Control Element Assembly Bite and Power Dependent Insertion Limits

Control element assembly bite is the CEA reactivity worth permitted to be inserted in the core when critical for power shaping and to compensate for minor variations in moderator temperature, boron concentration, xenon concentration, and power level.

The substantially smaller power defects for shutdown initiated at lower power levels allow a reduction in the CEA worth required to be available for shutdown. The corresponding increase in allowed CEA insertion is reflected by the transient Power Dependent Insertion Limits (PDILs) of the Technical Specifications. The transient PDIL restricts the amount of CEA insertion into the core while at power. The amount of CEA insertion permitted by PDIL provides sufficient reactivity for control while ensuring the minimum shutdown requirement is maintained in the withdrawn CEAs.

The allowance for first cycle CEA bite was 0.1% $\Delta\rho$. In addition, 0.1% $\Delta\rho$ was allowed for the first cycle for compensating fuel depletion effects between adjustments of the dissolved boron concentration. The current allowances for CEA bite are reflected in the Technical Specification transient PDIL curves.

3.4.3.4 Shutdown Conditions

Boric acid is used to provide a large margin for shutdown and refueling. After a normal shutdown or reactor trip, boric acid is injected into the RCS to compensate for reactivity increases caused by normal cooldown and xenon decay. Although the boric acid system reduces reactivity slowly, compared to CEAs, the rate of reduction is more than sufficient to maintain the shutdown margin against the effects of cooldown and xenon decay. The boron concentration established for refueling is listed in the Technical Specifications. This boron concentration provides more than adequate negative reactivity to maintain the shutdown condition.

3.4.4 CONTROL ELEMENT ASSEMBLY PATTERN, OPERATIONS, AND WORTHS

The CEA is described in Section 3.3.2.4 and shown in Figures 3.3-8, 3.3-9A and 3.3-9B. The CEAs are designated as Regulating CEAs or Shutdown CEAs. The Regulating CEAs are divided into five groups (1 through 5). The Shutdown CEAs are divided into three groups (A, B, and C). The locations of all the CEAs in one of four symmetrical core quadrants are shown on Figure 3.3-10.

All CEAs within a particular group are designated to be withdrawn or inserted nearly simultaneously. For startups the Shutdown CEA groups are withdrawn without overlap; then the Regulating groups are withdrawn with overlap.

The PDILs specify the maximum permitted CEA insertion as discussed in Section 3.4.3.3. The PDIL curve for the current cycle of each Unit is in the Technical Specifications. The PDIL curve illustrates the Regulating CEA Groups insertion order (5-4-3-2-1) with overlap between successive groups. The typical CEA withdrawal procedure is as follows:

a. With the reactor subcritical, Shutdown Group A is fully withdrawn, followed by Shutdown Group B and then Shutdown Group C.
b. Withdrawal of regulating CEAs commences starting with Regulating Group No. 1. CEA withdrawal continues with the prescribed overlap of Regulating Groups 2 through 5.

All CEAs are inserted for cold shutdown conditions and are essentially fully withdrawn during full power steady state operation. Reactivity insertion rates are discussed in Section 3.4.5. The allowable CEA misalignment within any CEA group is specified in Technical Specifications and intentional misalignment of CEAs within a group for the purpose of power shaping is not allowed.

The accidents involving CEAs are analyzed with conservative assumptions as discussed in Chapter 14. The CEA withdrawal accident is analyzed with the maximum calculated differential reactivity insertion rate resulting from a sequential CEA bank withdrawal with overlap. The CEA drop accident is analyzed by selecting the dropped CEA that maximizes the increase in the radial peaking factor. The typical reactivity insertion during a reactor scram is presented in Chapter 14. This reactivity insertion is computed by static or space time axial models and is used for all accidents which are terminated by a scram.

3.4.5 REACTIVITY INSERTION RATES

Normal operating practices require reactivity changes to accommodate power level changes, fuel depletion, temperature control, etc. The principal reactivity control mechanisms are CEA Regulating Groups and boration/dilution. The analysis of CEA withdrawal events (Chapter 14) shows that maximum CEDM speed results in a differential reactivity per inch and consequence which will not exceed the Specified Acceptable Fuel Design Limits (SAFDLs). The analysis of the boron dilution event (Chapter 14) for the worst case of initial refueling in the drained-down mode shows that adequate time exists to take corrective measures.

Reactivity addition rates due to control rods vary with the CEA group, CEA group position, RCS temperature, dissolved boron concentration, fuel depletion, power level, and power distribution. Reactivity addition rates due to changes in dissolved boron vary with CEA insertion, temperature, fuel depletion, and power level due primarily to their effect on the dissolved boron concentration. Both spectral and spatial self-shielding effects are involved.

3.4.6 POWER DISTRIBUTION

3.4.6.1 General

The core is designed and the reactor is operated to maintain a relatively uniform power distribution. Significant deviations from the expected power distribution are restricted by the Limiting Safety System Settings (LSSSs) [e.g., Axial Shape Index (ASI)] and the Limiting Conditions for Operation (LCOs) [e.g., F_r^T, Peak Linear Heat Rate (PLHR) and T_q]. These operating limits are bounded by sufficient thermal margin to prevent DNB and fuel/clad melting during Anticipated Operational Occurrences (AOOs).

3.4.6.2 Objective

A stable uniform power distribution increases thermal margin by minimizing peak heat flux and enthalpy rise. The relative power distribution is approximately the ratio of the maximum local power to the core average power. The power peaking factors (F_r^T, ASI, and T_q) are minimized by design to allow operational flexibility to deviate from the nominal core power distribution without encroaching upon the LSSSs and LCOs.

3.4.6.3 Fuel Management and Operations

The peaking factors are most strongly influenced by the core loading which is therefore designed to reduce the inherent power peaking. To accomplish this goal in reload core designs, the arrangement of fresh and depleted assemblies together increases the power sharing. Reload batches may have multiple enrichments. Fuel assemblies located in a low neutron leakage or high power density region may have BPRs. Figures 3.4-3 through 3.4-5 show the assembly location for the first and current cycles. The expected power distributions at selected burnups and CEA insertions for the first and current cycles are shown in Figures 3.4-6 through 3.4-22. Figures 3.4-23 through 3.4-26 show the effect of CEA insertion on power peaking and distribution for the first cycle.

The radial power distribution and fuel depletion are almost insensitive to reactor operations due to the high order of radial symmetry maintained and the negative radial and azimuthal stability factor. The APD, however, is subject to nonuniform axial temperatures, burnups, control rods, and xenon oscillations. Prudent use of CEAs will control undesirable axial power oscillations and the PDILs (first cycle shown in Figure 3.4-27) on CEA position prevent the expected burnup distribution from being negated by excessive CEA use.

3.4.6.4 Power Peaking Limits

Specified Acceptable Fuel Design Limits require that the core power distribution does not result in either fuel/clad melt or a DNB. Assurance that SAFDLs are not exceeded is obtained through the RPS and the Technical Specifications which enforce LCO. Limiting Conditions for Operation are established such that the initial conditions assumed in the analysis of AOOs and postulated accidents are conservative with respect to allowed reactor conditions. However, during certain AOOs the margin to fuel design limits deteriorates either in a manner that is undetected by the RPS or in such a manner that the RPS would not act before some margin loss has occurred. Therefore, the reactor must be operated such that losses in margin do not result in exceeding SAFDLs before the RPS restores the reactor to a safe condition. Limiting Conditions for Operation assure that sufficient initial margin to overcome margin losses exists.

3.4.6.5 Power Distribution Monitoring Capability

Neutron flux detectors are provided both within the active core (incore) and outside (excore) the reactor vessel. The RPS continuously monitors the excore detectors and reactor coolant variables to determine whether the plant is being operated within the LSSSs. Indicators are provided to solicit operator action prior to reaching an LSSS. In the event that conditions reach an LSSS, the RPS will automatically initiate a reactor trip.

Incore detectors provide the detailed power distributions necessary for Technical Specification surveillance of power peaks and core data trends.

The 35 incore self-powered rhodium detector strings are placed in the center CEA guide tube of selected assemblies. Each detector string has four, 40 cm long, rhodium detectors located at approximately 20, 40, 60, and 80% core height.

3.4.7 REACTOR STABILITY

3.4.7.1 General

Pressurized water reactors with negative overall power coefficients are inherently stable with respect to power oscillations. Therefore, this discussion will be limited to xenon-induced power distribution oscillations.

Xenon-induced oscillations occur as a result of rapid perturbations to the power distribution which cause the xenon and iodine fission product distributions to be out of phase with the perturbed power distribution. This results in a shift in the iodine and xenon distribution that causes the power distribution to change in an opposite direction from the initial perturbation and, thus, initiate oscillations. The magnitude of the power distribution oscillation can either increase or decrease with time. Thus, the core can be considered to be either unstable or stable with respect to these oscillations. Xenon stability analyses on previous Calvert Cliffs cores indicate that any radial and azimuthal xenon oscillations induced in the core would be damped. Axial xenon oscillations, however, could exhibit instabilities during later portions of the cycle in the absence of appropriate control action. Before discussing the methods of analysis and control, it is appropriate to reiterate several important aspects of the xenon oscillation phenomenon.

a. The time scale for the oscillations is long and any induced oscillation typically exhibits a period of about one day.

b. Xenon oscillations are readily detectable as discussed below.

c. As long as the initial power peak associated with the perturbation initiating the oscillation is acceptable, the operator has time, in the order of from hours to days, to take appropriate remedial action before the allowable peaking factors are exceeded.

3.4.7.2 Method of Analysis

A xenon oscillation may be described by the following equation:

$$\phi(\bar{r},t) = \phi_o(\bar{r}) + delta\, \phi_o(\bar{r}) e^{bt} \sin(\omega t + \sigma)$$

where:

$\phi(\bar{r},t)$	is the space-time solution of the neutron flux
$\phi_o(\bar{r})$	is the initial fundamental flux
delta $\phi_o(\bar{r})$	is the perturbed flux mode
b	is the stability index
ω	is the frequency of the oscillation
σ	is a phase shift

The stability of a reactor can be characterized by a stability index or a damping factor which is defined as the natural exponent which describes the growing or decaying amplitude of the oscillation.

A positive stability index (b) indicates an unstable core. A zero or a negative value indicates stability for the oscillatory mode being investigated. The stability index is generally expressed in units of inverse hours, so that a value of -0.01/hr would mean that the amplitude of each subsequent oscillation cycle decreases by about 25% for a period of about 30 hours for each cycle. Xenon oscillation modes can be classified into three general types: radial, azimuthal, and axial.

3.4.7.3 Radial Stability

A radial xenon oscillation consists of a power shift inward and outward from the center of the core to the periphery. This oscillatory mode is generally more stable than an azimuthal mode.

To confirm that the radial mode is extremely stable, for the first cycle a space-time calculation was run for a reflected, zoned core 11' in diameter without including the damping effects of the negative power coefficient. The initial perturbation was a poison worth of 0.4% in reactivity placed in the central 20% of the core for 1 hr. Following removal of the perturbation, the resulting oscillation was followed in 4-hr time steps for a period of 80 hours. The resulting oscillation died out very rapidly with a damping factor of about -0.06/hr. If this damping coefficient is corrected for a finite time mesh, it would become even more strongly convergent. On this basis, it is concluded that radial oscillations are highly unlikely.

This conclusion is of particular significance because it means that there is no type of oscillation where the inner portions of the core act independently of the peripheral portions of the core, whose behavior is more closely followed by the excore detectors. Primary reliance is placed on these detectors for the detection of any xenon oscillations.

3.4.7.4 Azimuthal Stability

An azimuthal oscillation consists of an X-Y power shift from one side of the reactor core to the other. Azimuthally-symmetric operation and design practices and a negative stability index ensure proper azimuthal power distribution.

3.4.7.5 Axial Stability

Axial xenon oscillations consist of a power shift between the top and bottom of the reactor core. This type of oscillation may be unstable toward the EOC.

3.4.7.6 Detection and Control of Oscillations

Primary reliance for the detection of any xenon oscillations is placed on the excore flux monitoring instrumentation. As indicated earlier, oscillations in modes such as radial, which would allow the center of the core to behave independently from the peripheral portions of the core, are highly unlikely and this lends support to reliance on the excore detectors for this purpose.

Although the primary response of these detectors will be to the power in the peripheral fuel assemblies, the lower modes of any induced oscillations will affect the power shapes in these peripheral assemblies. Therefore, azimuthal or axial flux tilts can be observed and identified with the use of incore or excore instrumentation and appropriate remedial action can be taken.

In addition, the incore flux monitoring instrumentation is used to verify the correlation between indications from the excore detectors and the space-dependent flux distribution within the core.

The reactor is operated in such a manner as to avoid inducing sizable spatial perturbations. As was discussed previously, radial and azimuthal xenon oscillations are expected to be damped in the Calvert Cliffs cores. Axial oscillations, however, may be undamped in the latter stages of core life. These unstable xenon oscillations require control action to prevent them from building in magnitude; however, they are very slow acting and thus leave time for appropriate control strategies to be determined. The part-length CEAs initially installed to control axial oscillations were removed since full-length CEAs have proved to be effective in controlling all xenon oscillations.

3.4.8 NEUTRON FLUX AT PRESSURE VESSEL

The original design of the reactor vessel considers the fast neutron fluence (neutron energy greater than 1 MeV) to the inner wall of the vessel. The fluence was determined by combining the results of the computer code P3MG1 (Reference 7) and SHADRAC (Reference 2). A detailed neutron transport analysis using the discrete ordinate computer code DOT-4 (Reference 10) is periodically performed to determine the fast neutron fluence on the vessel. The analysis considers the neutron flux from previous fuel cycles, and projected low fluence core design. The method is verified by examination of surveillance capsules. Beginning with Unit 1 Cycle 11 and Unit 2 Cycle 10, low fluence fuel management is used to reduce the fluence at the vessels' beltline region welds.

3.4.9 ANALYTICAL METHODS

3.4.9.1 General

Calvert Cliffs reactor cores are designed using a series of calculations to determine the energy and spatial dependent neutron flux, the integral or differential core reactivity, and the power distribution. The computer code DIT (Reference 4) calculates the energy dependent flux, the spatial dependent flux, and the flux weighted assembly-wide cross-sections necessary to perform core-wide calculations. Originally, the computer code PDQ (Reference 3) was used to calculate a few group pin power distribution. Starting with Unit 2 Cycle 8 and Unit 1 Cycle 10 the fine-mesh code MC (Reference 4) replaced PDQ. The computer code ROCS (Reference 4) was used to calculate a coarse-mesh two- or three-dimensional power distribution and the core averaged reactivity coefficients.

Starting with Unit 2 Cycle 16 and Unit 1 Cycle 18, the PARAGON and ANC computer codes are used for nuclear design analysis. PARAGON (Reference 11) calculates the energy dependent flux, the spatial dependent flux, and the flux weighted assembly-wide cross-sections, and the pin peak reconstruction factors necessary to perform core-wide calculations. The computer code ANC (References 12, 13, and 14) calculates a coarse and fine mesh two- or three-dimensional power distribution and the core averaged reactivity coefficients. Previously the computer codes DIT and ROCS (Reference 4) were used for these purposes. Both PARAGON and ANC have been extensively benchmarked to a wide variety of measurements including those taken on several past cycles of the Calvert Cliffs units.

Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, the MICBURN-3/CASMO-3G and PRISM computer codes are used for nuclear design analysis. MICBURN-3/CASMO-3G (Reference 15) calculates the cross-sections, discontinuity factors, and heterogeneous form functions for the core simulator code, PRISM. The computer code PRISM (Reference 15) calculates the core-wide power distribution in three dimensions. MICBURN-3/CASMO-3G and PRISM have been extensively benchmarked to a wide variety of measurements including those taken on several past cycles of the Calvert Cliffs units.

MICBURN-3 is a multigroup one-dimensional transmission probability code which calculates the microscopic burnup in Gadolinium-loaded fuel containing initially homogeneously distributed poison. These cross-sections, as a function of absorber number density, are input to CASMO-3.

CASMO-3G is a multi-group, two-dimensional transport theory code for burnup calculations on assemblies. CASMO-3G is capable of modeling the geometry of the Calvert Cliffs cores including non-symmetric fuel bundles. The microscopic depletion is calculated in each fuel rod and burnable absorber rod. The output consists of cross-sections, discontinuity factors, and heterogeneous form functions for the core simulator code, PRISM.

The PRISM code performs core-wide two-group calculations. It uses pin power reconstruction to establish the individual rod histories and reactivities. The reactor core for Calvert Cliffs is modeled as 4 radial nodes and 32 axial nodes per assembly. With this reactor model, axial effects, including predicted values of LHR, F_r^T, and F_z can be studied. Thermal hydraulic feedback and axial exposure distribution effects on power shapes, rod worths, and cycle lifetime are explicitly included in the PRISM analysis.

The computer code INCA (Reference 5) was used to perform the on-line incore calculations. In Unit 2 Cycle 9 and Unit 1 Cycle 11, INCA is replaced by CECOR 3.3 for power distribution surveillance. Starting with Unit 2 Cycle 19 and Unit 1 Cycle 21, CECOR is replaced by POWERTRAX for power distribution surveillance. Satisfactory comparisons between these measured data and the predictions validate the design procedures.

3.4.9.2 Coarse-Mesh Diffusion Calculations - Westinghouse only

Coarse-mesh calculations are performed by ANC. This code contains a nodal solution to the diffusion equation to obtain a high degree of accuracy with a low number of spatial mesh points. The reactor core is typically modelled in ANC as four radial nodes and 20 to 30 axial nodes per fuel assembly. The axial and radial reflector regions are explicitly represented as additional nodes that are attached to the core boundary. ANC also contains equilibrium thermal models required to correctly determine the neutronic impact of changes in the spatial distributions of fuel temperatures and moderator density.

3.4.9.3 Power Distribution Monitoring

During normal operating conditions, signals proportional to the rhodium activation rates are obtained from the incore detectors. These signals are related to local power by use of calculated signal-to-power conversion factors for the appropriate core conditions. The measured signals are corrected for background, calibration, and depletion.

The POWERTRAX system replaces CECOR starting with Unit 2 Cycle 19 and Unit 1 Cycle 21. POWERTRAX provides a method of synthesizing detailed three dimensional assembly and peak-pin power distributions. This method is described below.

a. Calculated 3-D Nodal Power and Signal Distributions

The calculated nodal power and detector signals are provided by the POWERTRAX system nodal simulator.

b. Measured Powers at Operable Detector Locations

Measured powers at the operable detector locations are generated by multiplying the calculated nodal power by the corresponding ratio of the measured to calculate detector signals. The calculation is performed at all axial detector levels.

c. Measured Powers at other Locations

Other locations include those detector axial levels in un-instrumented assembly locations and those locations with failed detectors. The measured nodal powers at other locations are computed using the relation between the calculated nodal powers at these locations and at the locations with operable detectors.

d. Radial Power Distributions

A radial power distribution is a combination of the measured powers at operable detector locations and measured power at other locations.

e. Axial Power Profile

The axial power profiles are derived from the axial profiles from the 3-D nodal simulator calculation and adjusted by the differences in the measured and calculated radial power distribution described above. These adjustments are made by modulating the calculated nodal power distribution with the ratio of inferred segment powers to the calculated nodal powers.

f. Final Normalization

The resulting 3-D power distribution is then re-normalized to the core thermal power to produce the final inferred 3-D nodal power distribution.

g. Peak Pin Power Distributions

Using the calculated nodal power distribution provided in step a, F_r (F_r-total is the maximum average pin power integrated over the entire core height; F_r-unrodded is the maximum average pin power integrated over the unrodded portion of the core) and pin-to box factor (PF-total is calculated for each assembly as the F_r-total value divided by the measured assembly average power; PF-unrodded is calculated for each assembly as the F_r-unrodded value divided by the average of the measured nodal powers averaged over the unrodded planes) can be obtained.

h. Azimuthal Power Tilt

The azimuthal power tilt is computed by determining the maximum values and locations of the maximum values from quadrant power tilt calculations for the upper and lower halves of the reactor core at each degree of rotation angle for 360° rotationally.

3.4.10 REFERENCES

1. Deleted
2. SHADRAC, "Shield Heating and Dose Rate Attenuation Calculation," G30-1365, March 25, 1966
3. W.R. Cadwell, "PDQ-7 Reference Manual," WAPD-TM-678, January 1968
4. CENPD-266-P-A, "The ROCS and DIT Computer Codes for Nuclear Design," April 1983
5. CENPD-153-P, Rev. 1-P-A, "Evaluation of Uncertainties in the Nuclear Power Peaking Measured by the Self-Powered Fixed In-core Detector System," May 1980
6. Deleted
7. CENPD 302, "Fast Attenuation by the P3MG, C-17 Shielding Method," July 1967
8. Deleted
9. Deleted
10. ORNL-5851, "An Updated Version of the DOT-4 One- and Two-Dimensional Neutron/Photon Transport Code," W.A. Rhoades, R.L. Childs, Oak Ridge National Laboratory, Oak Ridge, TN, July 1982
11. WCAP-16045-P-A, “Qualification of the Two Dimensional Transport Code PARAGON,” August 2004
12. WCAP-10965-P-A, “ANC: A Westinghouse Advanced Nodal Computer Code,” September 1986
13. WCAP-10965-P-A Addendum 1, “ANC: A Westinghouse Advanced Nodal Computer Code: Enhancements to ANC Rod Power Recovery,” April 1989

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

15. EMF-96-029(P)(A), Volumes 1 and 2, “Reactor Analysis System for PWRs Volume 1 - Methodology Description, Volume 2 - Benchmarking Results,” Siemens Power Corporation, January 1997

TABLE 3.4-1

NUCLEAR PARAMETERS

I. FIRST CYCLE NUCLEAR PARAMETERS

<u>Control Characteristics</u>	
k_{eff}, BOL, No Control Element Assemblies or Dissolved Boron with BPRs in	
Cold (68°F)	1.194
Hot (532°F), Zero Power	1.152
Hot (572°F), Full Power	1.128
Hot, Equilibrium Xe, Full Power	1.094
<u>Total CEA Worth, %</u>	
BOL	
Hot (572°F)	9.8
Cold (68°F)	5.7
EOC	
Hot (572°F)	9.7
Cold (68°F)	5.6
<u>Dissolved Boron</u>	
Dissolved Boron Content for Criticality, ppm, (CEAs withdrawn, BOL)	
Cold (68°F)	1120
Hot (532°F), Zero Power, Clean	1095
Hot (572°F), Full Power	960
Hot (572°F), Equilibrium Xe, Full Power	725
Dissolved Boron Content for Refueling, ppm	1720
Boron Worth, ppm/%	
Hot (572°F)	86
Cold (68°F)	69
<u>Reactivity Coefficients (CEAs Withdrawn Unless Otherwise Indicated)</u>	
MTC, $\Delta\rho$/°F	
Hot (572°F)	
BOC, 960 ppmb	-0.20×10^{-4}
BOC, 847 ppmb (1% CEAs In)	-0.49×10^{-4}
EOC, Zero ppmb	-1.96×10^{-4}
EOC, Zero ppmb (1% CEAs In)	-2.20×10^{-4}
Cold (68°F) Zero Power	
BOC, 1120 ppmb	-0.06×10^{-4}
FTC, $\Delta\rho$/°F	
Hot, Zero Power	-1.46×10^{-5}
Full Power	-1.06×10^{-5}
Moderator Void Coefficient, $\Delta\rho$/% Void	
Hot, Operating, BOL	-0.1×10^{-3}
EOC	-1.3×10^{-3}
Moderator Pressure Coefficient, $\Delta\rho$/psi	
Hot, Operating, BOL	$+0.3 \times 10^{-6}$
EOC	$+2.6 \times 10^{-6}$

TABLE 3.4-1

NUCLEAR PARAMETERS

II. CURRENT CYCLE NOMINAL NUCLEAR CHARACTERISTICS

	UNIT 1 CYCLE 25	UNIT 2 CYCLE 24
Dissolved Boron, ppm		
Dissolved Boron Content for Criticality, CEAs Withdrawn, Hot Full Power, Equilibrium Xenon, BOC	1364	1227
Boron Worth, ppm/% $\Delta\rho$		
Hot Full Power, BOC	146	144
Hot Full Power, EOC	106	106
Moderator Temperature Coefficient (CEAs Withdrawn), 10^{-4} $\Delta\rho$/°F Hot Full Power, Equilibrium Xenon		
BOC	-0.39	-0.64
EOC	-2.84	-2.84

TABLE 3.4-2

CEA REACTIVITY WORTH AND ALLOWANCES, (% Δρ)

I. UNIT 1 FIRST CYCLE VALUES

	BOC	**EOC**
Fuel Temperature Variation	1.7	1.8
Moderator Temperature Variation	0.7	1.4
Moderator Voids	0.0	0.1
CEA Bite and Boron Deadband	0.2	0.2
Shutdown Margin and Safety Features Allowance	2.0	2.4
Total Reactivity Allowances	4.6	5.9
Stuck CEA Allowance	1.9	2.2
Calculated CEA Worth at 572°F (77 Full-Length CEAs)	9.8	9.7
Uncertainty Allowance and Margin	3.3	1.6

II. CURRENT CYCLE LIMITING VALUES FOR EXCESS SHUTDOWN MARGIN

	UNIT 1 CYCLE 25	**UNIT 2 CYCLE 24**
Limiting Condition	EOC HFP	EOC HFP
Control Rod Worth		
ARI (All rods inserted)	8.471	8.405
MRR (Most reactive rod)	1.558	1.577
PDIL	0.184	0.176
(ARI-MRR-PDIL)*0.9	6.056	5.987
Positive Reactivity Insertion		
Power Defect	2.215	2.223
Axial Flux Redistribution	0.185	0.171
Coolant Void Effects	0.050	0.050
Total Positive Reactivity Insertion	2.450	2.444
Shutdown Margin		
(ARI-MRR-PDIL)*0.9 - Total Positive Reactivity Insertion	3.606	3.543
Required Shutdown Margin	3.500	3.500
Excess Shutdown Margin	0.106	0.043