4.3 Nuclear Design

This section describes the nuclear core design basis and the models used to analyze the fuel discussed in Section 4.2.

4.3.1 Design Basis

The design bases are those that are required for the plant to operate meeting all safety requirements. Safety design bases fall into two categories:

(1) The reactivity basis, which prevents an uncontrolled positive reactivity excursion

(2) The overpower bases, which prevent the core from operating beyond the fuel integrity limits

4.3.1.1 Reactivity Basis

The nuclear design shall meet the following basis: The core shall be capable of being made subcritical at any time or at any core condition with any control rod pair (with same HCU) fully withdrawn.

4.3.1.2 Overpower Bases

The Technical Specifications limits on Minimum Critical Power Ratio (MCPR) and the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) are determined such that the fuel will not exceed required licensing limits during abnormal operational occurrences or accidents.

4.3.2 Description

The ABWR core design consists of a light-water moderated reactor, fueled with slightly enriched uranium-dioxide. The use of water as a moderator produces a neutron energy spectrum in which fissions are caused principally by thermal neutrons. At normal operating conditions, the moderator boils, producing a spatially variable distribution of steam voids in the core. The ABWR design provides a system for which is reduced by an increase in the steam void content in the moderator. This void feedback effect is one of the inherent safety features of the ABWR system. Any system input which increases reactor power, either in a local or gross sense, produces additional steam voids which reduce reactivity and thereby reduce the power.

4.3.2.1 Nuclear Design Descriptions

The reference loading pattern is the basis for all fuel licensing. It is designed with the intent that it will represent, as closely as possible, the actual core loading pattern; however, there will be occurrences where the number and/or types of bundles in the reference design and the actual core loading do not agree exactly.

A reference core loading of 872 fuel bundles was used as the basis for the system dynamic response analyses in Section 6.3 and Chapter 15. [*This reference core loading pattern is provided in Figure 4.3-1.*]* Figure 4.3-2 shows an example of an initial core loading pattern.

4.3.2.2 Power Distribution

The core power distribution is a function of fuel bundle design, core loading, control rod pattern, core exposure distributions and core coolant flow rate. The thermal performance parameters, MAPLHGR and MCPR (defined in Table 4.3-1), limit unacceptable core power distribution.

4.3.2.2.1 Power Distribution Measurements

The techniques for measurement of the power distribution within the reactor core, together with instrumentation correlations and operation limits, are discussed in Reference 4.3-3.

4.3.2.2.2 Power Distribution Accuracy

The accuracy of the calculated power distribution is discussed in References 4.3-2, 4.3-3, and 4.3-5.

4.3.2.2.3 Power Distribution Anomalies

Stringent inspection procedures are utilized to ensure the correct arrangement of the core following fuel loading. A fuel loading error (a mislocated or a misoriented fuel bundle in the core) would be a very improbable event. The fuel loading error is discussed further in Chapter 15 and in References 4.3-2 and 4.3-5.

The inherent design characteristics of the BWR are well suited to limit gross power tilting. The stabilizing nature of the large moderator void coefficient effectively reduces the effect of perturbations on the power distribution. In addition, the incore instrumentation system, together with the online computer, provides the operator with prompt information on the power distribution so that he can readily use control rods or other means to limit the undesirable effects of power tilting. Because of these design characteristics, it is not necessary to allocate a specific margin in the peaking factor to account for power tilt. If, for some reason, the power distribution could not be maintained within normal limits using control rods and flow, then the total core power would have to be reduced.

4.3.2.3 Reactivity Coefficients

Reactivity coefficients, the differential changes in reactivity produced by differential changes in core conditions, are useful in calculating stability and evaluating the response of the core to external disturbances. The base initial condition of the system and the postulated initiating event determine which of the several defined coefficients are significant in evaluating the

* See Section 4.2.

response of the reactor. The coefficients of interest, relative to ABWR systems, are discussed here individually.

There are two primary reactivity coefficients that characterize the dynamic behavior of boiling water reactors: The Doppler reactivity coefficient and the moderator void reactivity coefficient. Also associated with the ABWR is a power reactivity coefficient and a temperature coefficient. The power coefficient is a combination of the Doppler and void reactivity coefficients in the power operating range, and the temperature coefficient is merely a combination of the Doppler and moderator temperature coefficients. Power coefficients are not specifically calculated for reload cores, while temperature coefficients normally are calculated.

4.3.2.3.1 Doppler Reactivity Coefficient

The Doppler coefficient is of prime importance in reactor safety. The Doppler coefficient is a measure of the reactivity change associated with an increase in the absorption of resonance-energy neutrons caused by a change in the temperature of the material in question. The Doppler reactivity coefficient provides instantaneous negative reactivity feedback to any rise in fuel temperature, on either a gross or local basis. The magnitude of the Doppler coefficient is inherent in the fuel design and does not vary significantly among BWR designs. For most structural and moderator materials, resonance absorption is not significant, but in U-238 and Pu-240 an increase in temperature produces a comparatively large increase in the effective absorption cross-section. The resulting parasitic absorption of neutrons causes a significant loss in reactivity. In ABWR fuel, in which approximately 97% of the uranium in UO_2 is U-238, the Doppler coefficient provides an immediate negative reactivity response that opposes increased fuel fission rate changes.

Although the reactivity change caused by the Doppler effect is small compared to other power-related reactivity changes during normal operation, it becomes very important during postulated rapid power excursions in which large fuel temperature changes occur. The most severe power excursions are those associated with rod drop accidents. A local Doppler feedback associated with a 1650°C to 2760°C temperature rise is available for terminating the initial excursion.

The Doppler coefficient is determined using the theory and methods described in Reference 4.3-3.

4.3.2.3.2 Moderator Void Coefficient

The moderator void coefficient should be large enough to prevent power oscillation due to spatial xenon changes yet small enough that pressurization transients do not unduly limit plant operation. In addition, the void coefficient in the ABWR has the ability to flatten the radial power distribution and to provide ease of reactor control due to the void feedback mechanism The overall void coefficient is always negative over the complete operating range since the ABWR design is under moderated.

A detailed discussion of the methods used to calculate void reactivity coefficients, their accuracy and their application to plant transient analyses, is presented in References 4.3-2, 4.3-3, and 4.3-5.

4.3.2.4 Control Requirements

The ABWR control rod system is designed to provide adequate control of the maximum excess reactivity anticipated during the plant operation. The shutdown capability is evaluated assuming a cold, xenon-free core.

4.3.2.4.1 Shutdown Reactivity

The core must be capable of being made subcritical, with margin, in the most reactive condition throughout the operating cycle with the most reactive control rod fully withdrawn and all other rods fully inserted. The shutdown margin is determined by using the BWR simulator code (see Section 4.3.3) to calculate the core multiplication at selected exposure points with the strongest rod fully withdrawn. The shutdown margin is calculated based on the carryover of the minimum expected exposure at the end of the previous cycle. The core is assumed to be in the cold, xenon-free condition in order to ensure that the calculated values are conservative. Further discussion of the uncertainty of these calculations is given in References 4.3-2, 4.3-3, and 4.3-5.

As exposure accumulates and burnable poison depletes in the lower exposure fuel bundles, an increase in core reactivity may occur. The nature of the increase depends on specifics of fuel loading and control state.

The cold k_{eff} is calculated with the strongest control rod out at various exposures through the cycle. A value R is defined as the difference between the strongest rod out k_{eff} at BOC and the maximum calculated strongest rod out k_{eff} at any exposure point. The strongest rod out k_{eff} at any exposure point in the cycle is equal to or less than:

$$k_{eff} = k_{eff}\,(\text{Strongest rod withdrawn})_{BOC} + R,$$

where: R is always greater than or equal to 0.

Sm is (as all other fission products) explicitly treated in the depletion calculations with the cross-section library.

The calculated values of k_{eff} with the strongest rod withdrawn at BOC and of R are reported in Table 4.3-2. For completeness, the uncontrolled k_{eff} and fully controlled k_{eff} values are also reported in Table 4.3-2.

4.3.2.4.2 Reactivity Variations

The excess reactivity designed into the core is controlled by the control rod system supplemented by gadolinia-urania fuel rods. Control rods are used during the cycle partly to compensate for burnup and partly to control the power distribution.

4.3.2.4.3 Standby Liquid Control System

The Standby Liquid Control System (SLCS) is designed to provide the capability of bringing the reactor, at any time in a cycle, from a full power and minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state. The requirements of this system are dependent primarily on the reactor power level and on the reactivity effects of voids and temperature between full-power and cold, xenon-free condition. The SLCS is discussed in Subsection 9.3.5.

4.3.2.5 Criticality of Reactor During Refueling

The core is subcritical at all times.

4.3.2.6 Stability

4.3.2.6.1 Xenon Transients

Boiling water reactors do not have instability problems due to xenon. This has been demonstrated by:

(1) Never having observed xenon instabilities in operating BWRs

(2) Special tests which have been conducted on operating BWRs in an attempt to force the reactor into xenon instability

(3) Calculations

Experience from a European BWR, operating with the SVEA-96 Optima2 fuel design, and with a considerably higher power density than in the ABWR design, has shown no instability problems due to xenon.

4.3.2.6.2 Thermal Hydraulic Stability

Thermal Hydraulic Stability is discussed in Section 15.9.

4.3.3 Analytical Methods

The nuclear evaluations of all cores are performed using the analytical tools and methods described in References 4.3-2 and 4.3-5.

The lattice analyses are performed during the bundle design process. The results of these single bundle calculations are reduced to “libraries” of lattice reactivities, relative rod powers, and few group cross-sections as functions of instantaneous void, exposure, exposure-void history, control state, and fuel and moderator temperature, for use in the core analysis. These analyses are dependent upon fuel lattice parameters only and are, therefore, valid for all plants and cycles to which they are applied.

The core analysis is unique for each cycle. It is performed in the months preceding the cycle loading to demonstrate that the core meets all applicable safety limits. The principal tool used in the core analysis is the three-dimensional BWR simulator code, which computes power distributions, exposure, and reactor thermal-hydraulic characteristics, with spatially varying voids, control rods, burnable poisons and other variables.

4.3.4 Changes

Not applicable.

4.3.5 COL License Information

4.3.5.1 Thermal Hydraulic Stability

In the event the COL applicant chooses a fuel design different than that described herein, the methodology for demonstrating stability compliance will be that approved by the NRC .

4.3.6 References

4.3-1 "Supplemental Information for Toshiba ABWR DCD Renewal Amendment", (WCAP-17290-P Rev.0).

4.3-2 "Reference Safety Report for Boiling Water Reactor Reload Fuel and Core Analyses", (CENPD-300-P-A, July 1996).

4.3-3 "The Advanced PHOENIX and POLCA Codes for Nuclear Design of Boiling Water Reactors", (CENPD-390-P-A, December 2000).

4.3-4 Not Used

4.3-5 "Reference Safety Report for Boiling Water Reactor Fuel and Core Analyses Supplement 1 to CENPD-300-P-A," WCAP 17322-P, September 2010.

Table 4.3-1 Definition Of Fuel Design Limits

Maximum Average Planar Heat Generation Rate (MAPLHGR)

The MAPLHGR is the maximum average linear heat generation rate (expressed in kW/m) in any plane of a fuel bundle allowed by the Plant Technical Specifications for that fuel type. This parameter is obtained by averaging the linear heat generation rate over each fuel rod in the plane, and its limiting value is selected such that:

(a) The peak clad temperature during the design basis loss-of-coolant accident will not exceed 1204°C in the plane of interest, and

(b) All fuel design limits specified in Section 4.2 will be met.

Minimum Critical Power Ratio (MCPR)

The MCPR operating limit is the minimum CPR allowed by the Plant Technical Specifications for a given bundle type. The minimum CPR is a function of several parameters, the most important of which are bundle power, bundle flow and bundle R-factor. The R-factor is dependent upon the local power distribution and details of the bundle mechanical design including channel bow considerations. The limiting value of CPR is selected for each bundle type such that, during the most limiting event of moderate frequency, the calculated CPR in that bundle is not less than the safety limit CPR. The MCPR operating limit is attained when the bundle power, R-factor, flow, and other relevant parameters combine to yield the Technical Specification value.

Table 4.3-2 Calculated Core Effective Multiplication and Control System Worth—No Voids, 20°C

Beginning of Cycle, K-effective*	
Uncontrolled	1.12351
Fully Controlled	0.96284
Strongest Control Rod Pair Out	0.98936
R, Maximum Increase in Cold Core Reactivity with Exposure Cycle, Δk	0.0000

* For the core loading in Figure 4.3-1.

Equilibrium Core Loading Map

	17	16	15	14	13	12	11	10	9	8	7	6	5	4	3	2	1
1	3	3	3														
2	3	3	3	3	3	3											
3	3	3	3	3	3	3	3										
4	2	1C	2	2	3	3	3	3	3	3							
5	1C	2	1C	1C	1C	2	2	2	3	3	3						
6	2	1A	2	1C	2	1C	1C	1C	2	3	3	3					
7	1A	2	1A	2	1A	2	1C	2	1C	1C	2	3	3				
8	2	1A	2	1B	2	1C	2	1C	2	1C	1C	3	3	3			
9	1A	2	1B	2	1B	2	1A	2	1A	2	1C	2	3	3			
10	2	1A	2	1A	2	1B	2	1B	2	1A	2	1C	2	3			
11	1A	2	1A	2	1A	2	1A	2	1A	2	1C	1C	3	3	3		
12	2	1B	2	1B	2	1B	2	1B	2	1A	2	1C	2	3	3	3	
13	1B	2	1A	2	1B	2	1A	2	1C	2	1C	2	1C	3	3	3	
14	2	1A	3	2	2	1B	2	1B	2	1C	2	1C	1C	2	3	3	
15	1A	2	1A	2	1C	2	1B	2	1A	2	1A	2	1C	2	3	3	3
16	2	1A	2	1B	2	1B	2	1A	2	1A	2	1C	2	1C	3	3	3
17	3	2	1A	2	1B	2	1B	2	1A	2	1B	2	1C	2	3	3	3

Bundle Type	Enrichment		Number of Bundles
1A	Fresh bundle type A	4.23 wt %	116
1B	Fresh bundle type B	4.22 wt %	76
1C	Fresh bundle type C	4.22 wt %	128
2	One-cycle-burnt fuel		320
3	Two-cycle-burnt fuel		232

[Figure 4.3-1 Core Loading Map Used for Response Analyses]*

* See Section 4.2 for restriction to change this figure.

Figure 4.3-2 [Information not included in DCD (Refer to Reference 4.3-1)]