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shall be capable of being made subcritical at any time or at any core condition with the highest worth control fully withdrawn and all other rods fully inserted.

4.3.1.2 Overpower Bases

The Technical Specifications limits on Safety Limit Minimum Critical Power Ratio (SLMCPR) 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 in which the fission rate 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

For the purpose of this Safety Analysis Report, a reference core loading of 872 fuel bundles was used as the basis for the system dynamic response analyses in Chapter 15. This reference core loading pattern (an equilibrium core with a 25% batch fraction) is provided in Figure 4.3-1. See also Section 4.3 of Reference 4.3-4.

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), prevent unacceptable core power distributions.

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 4.3-1.

4.3.2.2.2 Power Distribution Accuracy

The accuracy of the calculated power distribution is discussed in Reference 4.3-2.

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, but calculations have been performed to determine the effects of such events on CPR. The fuel loading error is discussed further in Chapter 15 and Subsection 6.3.2.2.7 of Reference 4.3-4.

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 the operators 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 reactivity coefficients are discussed in Sections 4.2 and 4.3 of Reference 4.3-4.

4.3.2.4 Control Requirements

The Lungmen NPS control rod system is designed to provide adequate control of the maximum excess reactivity anticipated during the plant operation (see Appendix 4A). 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 highest worth control rod or any control rod pair with same Hydraulic Control Unit (HCU), fully drawn 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 Reference 4.3-3.

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 Beginning of Cycle (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:

$$k_{eff} = k_{eff}\,(\text{Strongest rod pair withdrawn})_{BOC} + R$$

where:

R is always greater than or equal to 0. The value of R includes equilibrium Samarium (Sm).

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 (SLC) 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 SLC is discussed in Subsection 9.3.5 and Subsection 4.3.2.1.3 of Reference 4.3-4.

4.3.2.5 Criticality of Reactor During Refueling

During refueling operation, safety system interlocks provide assurance that inadvertent criticality does not occur because a control rod has been removed or is withdrawn in coincidence with another control rod.

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

All of these indicators have proven that xenon transients are highly damped in a BWR due to the large negative power coefficient.

Analysis and experiments conducted in this area are reported in References 4.3-2 and 4.3-4.

4.3.2.6.2 Thermal Hydraulic Stability

The compliance of GE fuel designs to the criteria set forth in General Design Criterion 12 is demonstrated provided that the stability compliance criteria are satisfied using approved methods as discussed in Section 9.1 of Reference 4.3-4.

4.3.3 Analytical Methods

The nuclear evaluations of all General Electric cores are performed using the analytical tools and methods described in Reference 4.3-4, Section 4.0.

4.3.4 Changes

Not applicable.

4.3.5 References

4.3-1 J.F. Carew, *Process Computer Performance Evaluation Accuracy*, NEDO-20340, June 1974.

4.3-2 R.L. Crowther, *Xenon Considerations in Design of Boiling Water Reactor*, APED-5640, June 1968.

4.3-3 *BWR/4,5,6 Standard Safety Analysis Report,* Revision 2, Chapter 4, June 1977.

4.3-4 *GESTAR III Republic of China, General Electric Standard Application for Reactor Fuel*, NEDE-24011-P-A-7 RC.

4.3-5 *R-Factor Calculation Method for GE11, GE12, and GE13 Fuel*, NEDC-32505P, November 1995.

4.3-6 GE12 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDE-32417P, December 1994.

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 Reference 4.3-6, Section 2.2 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 (Reference 4.3-5). 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 MCPR. 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	1.0973
Fully Controlled	0.9416
Strongest Control Rod Out	0.9800
R, Maximum Increase in Cold Core Reactivity with Exposure Cycle, Δk	0.0000

* For the core loading in Figure 4.3-1.

F1 = GE12 3.48% Enriched High Gadolina Bundles
F2 = GE12 3.48% Enriched Low Gadolina Bundles
1 = GE12 3.48% One Cycle Fuel (220 Bundles)
2 = GE12 3.48% Two Cycle Fuel (216 Bundles)
3 = GE12 3.48% Three Cycle Fuel (216 Bundles)

Note: Loading pattern is shown for quarter core only. Rotational symmetry applies.

I \ J	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1															3	3	3
2												3	3	3	2	1	1
3											3	2	1	1	1	1	2
4								3	3	3	2	F2	2	F2	3	F1	2
5							3	1	1	1	F2	1	2	1	F2	3	2
6						3	1	1	2	F2	1	F1	2	2	1	F1	2
7					3	1	1	3	F1	3	F2	3	F2	3	F2	3	1
8				3	1	1	3	F1	1	F1	3	F1	2	1	3	F1	2
9				3	1	2	F1	1	2	2	F2	3	2	2	F2	1	2
10				3	1	F2	3	F1	2	2	1	F1	1	2	2	F2	1
11			3	2	F2	1	F2	3	F2	1	2	3	F2	1	2	3	1
12		3	2	F2	1	F1	3	F1	3	F1	3	F2	2	F2	3	F1	2
13		3	1	2	2	2	F2	2	2	1	F2	2	2	2	F2	1	2
14		3	1	F2	1	2	3	1	2	2	1	F2	2	2	1	F1	1
15	3	2	1	3	F2	1	F2	3	F2	2	2	3	F2	1	2	3	2
16	3	1	1	F1	3	F1	3	F1	1	F2	3	F1	1	F1	3	F1	1
17	3	1	F2	3	1	2	F1	1	2	2	F1	1	2	2	F1	3	3

Figure 4.3-1 Equilibrium Core Loading Map