3. Nuclear Design

This section describes the nuclear core design basis and the models used to analyze the fuel detailed in References 3–2 and 3–3. All fuel designs either meet the criteria of Subsection 1.1.3 or are separately approved by the NRC.

3.1 Design Bases

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

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be capable of being made subcritical at any time or at any core condition with the highest worth control rod fully withdrawn.

3.1.2 Overpower Bases

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

3.2 Description

The BWR 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 BWR design provides a system for which reactivity 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 BWR system. Any system input that increases reactor power, either in a local or gross sense, produces additional steam voids that reduce reactivity and thereby reduce the power.

3.2.1 Nuclear Design Description

The reference loading pattern for each cycle is documented in the FSAR or in the Supplemental Reload Licensing Report.

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.

Any differences between the reference loading pattern and the actual loading pattern are evaluated as described in Section 3.4.

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, MLHGR and MCPR (defined in Table 3–1), limit unacceptable core power distributions.

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 3–1.

3.2.2.2 Power Distribution Accuracy

The accuracy of the calculated power distributions is discussed in References 3–4, 3–5, 3–16, 3–17 and 3–18.

3.2.2.3 Power Distribution Anomalies

The power distribution anomaly resulting from a fuel loading error does not generally result in the limiting delta-CPR compared to the other events analyzed for each reload cycle. As such, the event has a very remote likelihood of resulting in fuel failures.~~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 analyzed as an Infrequent Incident when appropriate core verification procedures are utilized to ensure the correct arrangement of the core following fuel loading.** Fuel loading error is discussed further in the country–specific supplement to this document.

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 in–core instrumentation system, together with the on–line 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.

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

There are two primary reactivity coefficients that characterize the dynamic behavior of boiling water reactors; these are the Doppler reactivity coefficient and the moderator void reactivity coefficient. Also associated with the BWR are 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 and temperature coefficients are not specifically calculated for reload cores.

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 BWR 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 3000°F to 5000°F temperature rise is available for terminating the initial excursion.

The Doppler coefficient is determined using the theory and methods described in Reference 3–6.

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 a BWR 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 BWR design is undermoderated.

A detailed discussion of the methods used to calculate void reactivity coefficients, their accuracy and their application to plant transient analyses, is presented in Reference 3–6.

3.2.4 Control Requirements

The General Electric BWR 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.

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 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 3–7 and 3–8.

As exposure accumulates and burnable poison depletes in the lower exposure fuel bundles, an increase in core reactivity may occur. The nature of this 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:

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

where

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

The calculated values of k_{eff} with the strongest rod withdrawn at BOC and of R are reported in the FSAR or in the supplemental reload licensing report. For completeness, the uncontrolled k_{eff} and fully controlled k_{eff} values are also reported in the FSAR or in the supplemental reload licensing report.

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.

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, to a subcritical condition with the reactor in the most reactive xenon–free state with all of the control rods in the full–out condition. 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 shutdown capability of the SLCS is given in the FSAR or the supplemental reload licensing report.

3.2.5 Criticality of Reactor During Refueling

The core is subcritical at all times.

3.2.6 Stability

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, and (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 Reference 3–9.

3.2.6.2 Thermal Hydraulic Stability

This subject is covered in the country–specific supplement to this document.

3.3 Analytical Methods

The nuclear evaluations of all General Electric BWR cores are performed using the analytical tools and methods described in this section. There are two sets of procedures available for fuel design and licensing analysis: GENESIS and GEMINI. The nuclear physics methods described in References 3–4, 3–7, 3–10 and 3–11 are utilized as part of the GENESIS group. The advanced physics methods described in References 3–5 and 3–16 are utilized as part of the GEMINI group. The particular procedure that can be utilized is optional. In either case, the nuclear evaluation procedure is best addressed as two parts: lattice analysis and core analysis.

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, exposure–control 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 Boiling Water Reactor Simulator code, which computes power distributions, exposure, and reactor thermal–hydraulic characteristics, with spatially varying voids, control rods, burnable poisons and other variables.

3.4 Final Loading Pattern Comparison

(Reload Cores)

3.4.1 Introduction and Bases

Because the reload licensing process requires an assumption as to the condition of the core at the end of the previous cycle, it is possible that the as–loaded core may not be identical to the reference core. To assure that licensing calculations performed on the reference core are applicable to the as–loaded core, certain key parameters, which affect the licensing calculations, are examined to assure that there is no adverse impact; only when this examination has been completed and it has been established that the as–loaded core satisfies the licensing basis will the core be operated.

3.4.2 Acceptable Deviation from Reference Core Design

The parameters that measure the deviation between the reference core and the actual core have been identified and are discussed in this section. Sensitivity studies have been conducted to accurately determine how these parameters may be allowed to vary without adversely affecting the licensing analysis.

The parameters discussed in the following sections are routinely checked for every reload.

3.4.2.1 Core Average EOC Exposure

The reference core is designed and licensed on the assumption of a specific value for the core average exposure at the end of the previous cycle. Significant deviation from the assumed value requires that the impact on all licensing calculations be determined.

3.4.2.2 Core Average EOC Axial Exposure Distribution

An evaluation is made between the previous cycle EOC axial exposure distribution assumed for the reference core and the final EOC axial exposure distribution of the previous cycle core.

3.4.2.3 Number of Reload Bundles

The number of new bundles actually loaded cannot be greater than the corresponding number in the reference core, without specific evaluations of the impact on licensing results.

3.4.2.4 Type and Number of Exposed Bundles

The most reactive available bundles of the types and numbers specified in the reference core are used. If the number of available bundles of a given type is less than specified in the reference core, bundles of a different type but of lower reactivity may be substituted without re–analysis. The core is then reviewed to ascertain that the new core nuclear parameters are equal to or conservative relative to the reference core values.

3.4.2.5 Locations of Reload Bundles

A fresh bundle may be loaded only into a location that has been designated in the reference core to receive a fresh bundle or a new analysis is required. When reload batch size is decreased, deletions may be made only of fresh bundles scheduled to be loaded in peripheral, control–rod–centered four–bundle cells. The number of fresh bundles deleted shall not exceed the smallest of either 10% of the reload batch or 2% of the total core without reanalysis.

3.4.2.6 Locations of Exposed Bundles

Bundles remaining in the core should preferentially be loaded into locations designated for that bundle type in the reference core, except for changes necessitated by changes in available inventory. Such changes are made in the regions of least importance. Individual bundle locations are assigned by matching individual bundle exposures and burn histories as closely as possible to those designated in the reference core.

3.4.2.7 Shuffling of Edge Bundles

The reflector distorts the flux within those bundles that are located on the core edge. The effect of this distortion is to introduce a small–added uncertainty in the bundle nuclear characteristics. To avoid concentrating these bundles, the following principle is used: A given control cell should, if practical, contain no more than one bundle which saw duty in a location on the core edge during the previous cycle.

3.4.2.8 Symmetry

Calculation of the Fuel Cladding Integrity Safety Limit MCPR by the GETAB analysis assumes core quadrant fuel bundle type symmetry. No such assumption is necessary in the other areas of the safety analysis. It should be noted that the Fuel Cladding Integrity Safety Limit MCPR was derived for a reasonably bounding power distribution and should also apply for the case of asymmetric reactor power. This is discussed further in Reference 3–12.

When the reactor core is being operated with a mirror or rotationally symmetric control rod pattern, the neutron flux at similarly symmetric narrow–narrow gap locations in the four quadrants is considered to be equal. This fact is used to reflect the readings of the real strings into their symmetric counterpart locations where no real strings exist. This reflection is done prior to the commencement of the power distribution calculations.

In the few instances where fuel bundles near the edge are quadrant–loaded asymmetrically, the error induced by reflecting real readings is partially negated by the fuel type dependent correlations. Any remaining error is considered to be of negligible second order. Further, because such bundles are in low power regions, it is highly unlikely that one of them is a limiting bundle.

In the rare case of the reactor being operated with an asymmetric control rod pattern, the reflection of real string readings is not utilized. In this instance, readings at locations without strings are inferred by interpolation of the real string values in the immediate vicinity.

3.4.2.9 Shutdown Margin

The cold shutdown margin is always recalculated for the final core loading. Adequate shutdown margin is verified experimentally during the startup.

3.4.3 Re–Examination of Bases

If the final loading plan does not meet the criteria of Subsection 3.4.2, a re–examination of the parameters that determine the operating limits is performed. Based on results of the sensitivity studies of the operating limits to these parameters, conservative bounds have been set on the allowable change from the reference. These parameters are:

1. Scram reactivity insertion.
2. Dynamic void coefficient.
3. Peak fuel enthalpy during rod drop accident.
4. Cold shutdown margin.
5. Standby liquid control system shutdown margin.
6. Change in critical power ratio due to a misloaded fuel assembly. **(When analyzed as an AOO.)**
7. Rod block monitor response to a rod withdrawal error.
8. Safety Limit MCPR.

These parameters were chosen by one of the following two criteria:

(1) It is a parameter whose magnitude or behavior is explicitly reported in the supplemental reload licensing report.

Examples:

Cold shutdown margin, peak fuel enthalpy in Rod Drop Accident, change in CPR due to a misloaded assembly, and Rod Block Monitor response.

(2) It is a parameter important to the quantification of an operating limit.

Examples:

Scram reactivity insertion and dynamic void coefficient affect the operating limit MCPR.

The Doppler coefficient and delayed neutron fraction were excluded because these are slowly varying functions of exposure that do not change significantly over the expected range of exposure deviations.

3.5 Reactivity of Fuel in Storage

The basic criterion associated with the storage of both irradiated (spent) and new fuel is that the effective multiplication factor of fuel stored under normal conditions will be ≤ 0.90 for the regular density rack and ≤ 0.95 for the high–density racks. Abnormal storage conditions are limited to a $k_{eff} \leq 0.95$ for both high and regular density designs. A list of normal and abnormal storage conditions is presented in Chapter 9 of Reference 3–13. These storage criteria will be satisfied if the uncontrolled lattice k_∞ calculated in the normal reactor core configuration meets the following condition for General Electric designed fuel storage racks.

(a) $k_\infty \leq 1.31$ for 20°C to 100°C for regular spent fuel storage racks with an interrack spacing ≥ 11.875 inches.

(b) $k_\infty \leq 1.30$ for 20°C to 100°C for regular spent fuel storage racks with an interrack spacing ≥ 11.71 inches.

(c) $k_\infty \leq 1.33$ for 20°C to 100°C for high density fuel storage racks.

(d) $k_\infty \leq 1.31$ for 20°C to 100°C for regular new fuel vault storage racks with an interrack spacing ≥ 10.50 inches.

These criteria apply to the storage racks designed by General Electric at all plants.

The peak uncontrolled k_∞ values show that the fuel storage criteria will be satisfied for the Type a and Type b rack spacing and for the Type c high density fuel storage rack (Reference 3–14) designed by the General Electric Company. They also show that the storage criteria will be satisfied for the new fuel vault storage racks (Type d).

3.6 References

3–1 J. F. Carew, *Process Computer Performance Evaluation Accuracy*, NEDO–20340–1, December 1984.

3–2 *General Electric Fuel Bundle Designs*, NEDE–31152–P, Revision 8, April 2001.

3–3 *General Electric Fuel Bundle Designs Evaluated with TEXICO/CLAM Analyses Bases*, latest version, NEDE–31151–P.

3–4 C. L. Martin, *Lattice Physics Methods Verification*, NEDO–20939–A, January 1977.

3–5 *Steady–State Nuclear Methods*, NEDE–30130–P–A (Proprietary) and NEDO–30130–A, April 1985.

3–6 R. C. Stirn, *Generation of Void and Doppler Reactivity Feedback for Application to BWR Design*, NEDO–20964–A, December 1, 1986.

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3–9 R. L. Crowther, *Xenon Considerations in Design of Boiling Water Reactors*, APED–5640, June 1968.

3–10 C. L. Martin, *Lattice Physics Methods*, NEDE–20913–P–A (Proprietary) and NEDO–20913–A, February 1977.

3–11 J. A. Woolley, *Three–Dimensional BWR Core Simulator*, NEDO–20953–A, January 1977.

3–12 *Process Computer Performance Evaluation Accuracy Amendment 1*, NEDO–20340–1, December 1984.

3–13 *General Electric Standard Safety Analysis Report*, General Electric Company, 22A7007, Revision 14.

3–14 *Design Report and Safety Evaluation for High Density Fuel Storage System*, NEDE–24076–1–P, May 1979.

3–15 *R–Factor Calculation Method for GE11, GE12 and GE13 Fuel*, NEDC–32505P–A, Revision 1, July 1999.

3–16 Letter from Ralph J. Reda to R. C. Jones, Jr., “Implementation of Improved GE Steady–State Nuclear Methods,” Letter No. MFN–098–96, July 2, 1996.

3–17 *Methodology and Uncertainties for Safety Limit MCPR Evaluation*, NEDC–32601P–A, August 1999.

3–18 *Power Distribution Uncertainties for Safety Limit MCPR Evaluations*, NEDC–32694P–A, August 1999.

Table 3–1

Definition of Fuel Design Limits

Maximum Linear Heat Generation Rate (MLHGR) The MLHGR is the maximum linear heat generation rate expressed in kW/ft for the fuel rod with the highest surface heat flux at a given nodal plane in the bundle. The LHGR operating limit is bundle type dependent. The MLHGR can be monitored to assure that all mechanical design requirements will be met.
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) The MAPLHGR is the maximum average linear heat generation rate (expressed in kW/ft) 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 2200°F in the plane of interest, and (b) all fuel rod thermal–mechanical design limits specified in Section 2 will be met if the exposure–dependent MLHGR is not monitored for that purpose.
Minimum Critical Power Ratio (MCPR) The critical power ratio is defined as the ratio of the critical power (bundle power at which some point within the assembly experiences onset of boiling transition) to the operating bundle power. The critical power is determined at the same mass flux, inlet temperature, and pressure that exists at the specified reactor condition. Thermal margin is stated in terms of the minimum value of the critical power ratio, MCPR, which corresponds to the most limiting fuel assembly in the core.
Operating Limit 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 (Reference 3–15). 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.