3.6 NUCLEAR DESIGN

This section describes the nuclear core design basis and the models used to analyze the fuel discussed in Subsection 3.2, "Fuel Mechanical Design." Detailed documentation of the nuclear design of General Electric (GE) fuel product lines is contained in GESTAR II (General Electric Standard Application For Reactor Fuel) (Reference 1). The nuclear design criteria of AREVA fuel is described in Reference 15 with the detailed design provided in a reload specific fuel nuclear design report.

3.6.1 Power Generation Objective

The objectives of the fuel nuclear design are as follows:

a. To attain rated power generation from the nuclear fuel for a given period of time,

b. To attain reactor nuclear stability throughout core life,

c. To allow normal power operation of the nuclear fuel without sustaining fuel damage.

3.6.2 Power Generation Design Basis

1. Fuel nuclear design shall provide sufficient excess reactivity during power operation to achieve the core design burnup.

2. Fuel nuclear design shall provide sufficient negative reactivity feedback to facilitate normal maneuvering and control.

3. Fuel nuclear design shall, in combination with reactivity control systems, allow continuous, stable regulation of core excess reactivity.

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

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be capable of being made subcritical at any time or at any core condition with the highest worth control rod fully withdrawn.

3.6.3.2 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.

3.6.4 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. Void feedback effects are one inherent safety feature of the BWR. 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.

3.6.4.1 Nuclear Design Description

The reference loading pattern for each cycle is documented in the GE Supplemental Reload Licensing Report (SRLR) or the AREVA Reload Licensing Analysis Report. The current licensing report for each BFN unit is included in Appendix N of the FSAR.

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 GESTAR II for GE analyzed reload cores and in Reference 12 for AREVA analyzed reload cores. 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. If the final loading plan does not meet the necessary criteria, a re-examination of the parameters that determine the operating limits is performed. 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.

3.6.4.2 Power Distribution

The core power distribution is a function of fuel bundle design, core loading, control rod pattern, core exposure distribution, and core coolant flow rate. The thermal performance parameters MAPLHGR, MLHGR, and MCPR (defined in Table 3.6-1) limit unacceptable core power distributions.

3.6.4.2.1 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 2 for GE analyzed cores and in Reference 11 for AREVA analyzed cores.

3.6.4.2.2 Power Distribution Accuracy

The accuracy of the calculated power distribution is discussed in References 3, 4, 18, 19, and 20 for GE analyzed reload cores and in References 11 and 13 for AREVA analyzed reload cores.

3.6.4.2.3 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 and LHGR. The fuel loading error is discussed further in GESTAR II (Reference 1) and in References 12 and 17.

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.

3.6.4.3 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 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 and temperature coefficients are not specifically calculated for reload cores.

3.6.4.3.1 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.

For GE analyses, the Doppler coefficient is determined using the theory and methods described in Reference 5. The application of the Doppler coefficient to the AREVA analysis of the rod drop accident is discussed in Reference 12.

3.6.4.3.2 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 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 Reference 5 for GE analyzed reload cores and in References 11 and 12 for AREVA analyzed cores.

3.6.4.4 Control Requirements

The 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.

3.6.4.4.1 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 a BWR simulator code (see Subsection 3.6.5, “Analytical Methods”) 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 6 and 7 for GE analyses and in References 11 and 12 for AREVA analyses.

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:

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

where,

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

3.6.4.4.2 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.

3.6.4.4.3 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 conditions. The shutdown capability of the SLCS is reported in the AREVA Reload Licensing Analysis Report.

3.6.4.5 Criticality of Reactor During Refueling

The core is subcritical at all times during refueling. This is ensured by a combination of refueling interlocks and analytical verification of shutdown margin. Shutdown margin is determined by using a BWR simulator code (see Subsection 3.6.5, “Analytical Methods”) to calculate the core multiplication with the strongest rod fully withdrawn for the final reload core configuration and for limiting interim core configurations in the case of an incore shuffle.

3.6.4.6 Stability

3.6.4.6.1 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.
Analyses and experiments conducted in this area are reported in Reference 8.

3.6.4.6.2 Thermal Hydraulic Stability

The compliance of GE fuel designs to the criteria set forth in General Design Criterion 12 is demonstrated provided that the following stability compliance criteria are satisfied using approved methods:

(1) Neutron flux limit cycles, which oscillate up to 120% APRM high neutron flux scram setpoint or up to the LPRM upscale alarm trip (without initiating scram) prior to operator mitigating action shall not result in exceeding specified acceptable fuel design limits.
(2) The individual channels shall be designed and operated to be hydrodynamically stable or more stable than the reactor core for all expected operating conditions (analytically demonstrated).

The GE methodology for demonstrating the above has been reviewed and approved by the NRC in Reference 9. The AREVA methodology for demonstrating the above has been reviewed and approved by the NRC in Reference 14. The stability compliance of the fuel designs described in Subsection 3.2, “Fuel Mechanical Design,” has been demonstrated on a generic basis for GE reloads and approved by the NRC in Reference 9. See Subsection 3.7.6.2, “Thermal Hydraulic Stability Performance,” for additional information regarding core thermal-hydraulic stability.

3.6.5 Analytical Methods

Nuclear evaluations are performed by GE using the analytical tools and methods described in Reference 4. AREVA nuclear evaluations are performed using the analytical tools and methods described in References 11 and 12.

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 a 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.

3.6.6 Reactivity of Fuel in Storage

The NRC amended its regulations in December 1998 to give licensees the option of either meeting the criticality accident requirements of 10 CFR 70.24 paragraphs (a) through (c) in handling and storage areas for Special Nuclear Material (SNM), or electing to comply with certain requirements in a new section, 50.68, in 10 CFR part 50. Browns Ferry has chosen to comply with the new requirements of 10 CFR 50.68(b).

To meet these new requirements the quantity of SNM, other than nuclear fuel stored onsite, shall be less than the quantity necessary for a critical mass. The quantity of SNM specified to be enough for a critical mass in Section 1.1 of Regulatory Guide 10.3, "Guide for the Preparation of Applications for Special Nuclear Material Licenses of Less than Critical Mass Quantities" is 350 gram of U-235, 200 grams of U-233, and 200 grams of Pu-239. The combined total of non-fuel SNM maintained at BFN is far less than this quantity. The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five percent by weight. Requirements for new fuel storage are described in Section 10.2 of the FSAR. Requirements for spent fuel storage are described in FSAR, Section 10.3.5. Requirements for the reactor building ventilation radiation monitors are described in FSAR, Section 7.12.5. The existing radiation monitors on the refuel zone are considered to meet 10 CFR 50.68(b) requirements. The remaining 50.68 (b) requirements for fuel handling outside of approved storage areas are contained in plant procedures.

The basic criterion associated with the storage of irradiated (spent) and new fuel is that the effective multiplication factor of fuel stored under normal conditions will be ≤0.95 for high density racks. For storage of new fuel in the new fuel storage vaults, the effective multiplication factor will be ≤0.90 for dry conditions and ≤0.95 for flooded conditions. [Note: Placement of fuel in the new fuel storage vaults is currently prohibited at Browns Ferry. This restriction is administratively controlled by BFN Site Procedures.]

The current and legacy fuel products have been assessed and shown to meet the criteria per Reference 10.

3.6.7 References

1. "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, (See Appendix N for applicable revision).

2. J. F. Carew, "Process Computer Performance Evaluation Accuracy," NEDO-20340, June 1974.

3. C. L. Martin, "Lattice Physics Methods Verification," NEDO-20939-A, January 1977.

4. "Steady-State Nuclear Methods," NEDE-30130-P-A (Proprietary) and NEDO-30130-A, April 1985.

5. R. C. Stirn, “Generation of Void and Doppler Reactivity Feedback for Application to BWR Design,” NEDO-20964, December 1975.

6. G. R. Parkos, “BWR Simulator Methods Verification,” NEDO-20946-A, January 1977.

7. “BWR/4,5,6 Standard Safety Analysis Report, Revision 2,” Chapter 4, June 1977.

8. R. L. Crowther, “Xenon Considerations in Design of Boiling Water Reactors,” APED-5640, June 1968.

9. G. A. Watford, “Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria,” December 1982 (NEDE-22277-P-1).

10. “Browns Ferry Nuclear Plant Units 1, 2, and 3 Spent Fuel Storage Pool Criticality Safety Analysis for ATRIUM™ 10XM (ANP-3160(P) Revision 0), October 2012.

11. EMF-2158(P)(A) Revision 0, “Siemens Power Corporation Methodology for Boiling Water Reactors: Evaluation and Validation of CASMO-4/MICROBURN-B2,” Siemens Power Corporation, October 1999.

12. XN-NF-80-19(P)(A) Volume 1 and Supplements 1 and 2, “Exxon Nuclear Methodology for Boiling Water Reactors - Neutronic Methods for Design and Analysis,” Exxon Nuclear Company, March 1983.

13. EMF-2493(P) Revision 0, “MICROBURN-B2 Based Impact of Failed/Bypassed LPRMs and TIPs, Extended LPRM Calibration Interval, and Single Loop Operation on Measured Radial Bundle Power Uncertainty,” Siemens Power Corporation, December 2000.

14. EMF-CC-074(P)(A) Volume 4 Revision 0, “BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2,” Siemens Power Corporation, August 2000.

15. ANF-89-98(P)(A) Revision 1 and Supplement 1, “Generic Mechanical Design Criteria for BWR Fuel Designs,” Advanced Nuclear Fuels Corporation, May 1995.

16. ANF-524(P)(A) Revision 2 and Supplements 1 and 2, “ANF Critical Power Methodology for Boiling Water Reactors,” Advanced Nuclear Fuels Corporation, November 1990.

17. XN-NF-80-19(P)(A) Volume 4 Revision 1, “Exxon Nuclear Methodology for Boiling Water Reactors Application of the ENC Methodology to BWR Reloads,” Exxon Nuclear Company, April 1986.

18. 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.

19. Methodology and Uncertainties for Safety Limit MCPR Evaluation, NEDC-32601P-A, August 1999.

20. Power Distribution Uncertainties for Safety Limit MCPR Evaluations, NEDC-32694P-A, August 1999.