TENNESSEE VALLEY AUTHORITY
Division of Fuels
Nuclear Fuel Branch

TVA RELOAD CORE DESIGN AND ANALYSIS METHODOLOGY FOR THE BROWNS FERRY NUCLEAR PLANT

TVA-EG-047

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Chattanooga, Tennessee

January 1982

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FOR THE BROWNS FERRY NUCLEAR PLANT

1.0 Introduction

TVA has developed the capability to design and analyze BWR reload fuel cycles as necessary to prescribe safety and operational limits and to confirm that operation within those limits will be safe and efficient. This report contains a generic description of the TVA Reload Core Design and Analysis (RCDA) procedures and methods. This RCDA methodology will be used to generate data and information required for reload licensing amendments. It is intended that this report, together with various TVA topical reports referenced herein, will constitute a sufficiently comprehensive description of the TVA reload analysis process that detailed explanations of analytical techniques will not be necessary in specific reload licensing submittals.

TVA's RCDA program and this report are limited in scope to those analyses required to define an acceptable core configuration and confirm safe core operation throughout the fuel cycle. Fuel assembly design and LOCA analyses remain the responsibility of the fuel vendor, and TVA will obtain the fuel thermal, mechanical, and LOCA limits from the fuel vendor. The relationship between TVA and the fuel vendor with regard to RCDA is described more fully in section 3.0 of this report.

The TVA RCDA methodology will first be applied to the Browns Ferry units, starting with cycle 6 of unit 3. Table 1.1 shows all fuel cycles through 1986 for which reload licensing amendments will be based primarily on TVA analyses.

Table 1.1

FUEL CYCLES TO BE ANALYZED BY TVA THROUGH 1986

Unit	Cycle	Cycle Startup Date
Browns Ferry 3	6	Fall 1983
Browns Ferry 3	7	Spring 1985
Browns Ferry 2	7	Fall 1985
Browns Ferry 1	8	Spring 1986
Browns Ferry 3	8	Fall 1986

2.0 RCDA Process Overview

The TVA BWR RCDA process consists of several sequential phases composed of distinct analytical steps as illustrated in figure 2.1. Phase 1, a preliminary design phase, begins with the Cycle N Design Specifications (step 1.0). Cycle N energy requirements and plant design parameters affecting the reload analysis are specified in an Input Data Package that is verified by appropriate TVA organizations. Next, a Preliminary Fuel Cycle Design (step 1.1) is performed to determine the reload batch composition and to design a preliminary loading pattern. Input to the preliminary cycle design includes fuel assembly properties calculated in the Fuel Analyses (step 1.2). Physics and thermal-mechanical characteristics are calculated with TVA methods using vendor fuel assembly design specifications. Completion of the above three steps concludes the preliminary design phase.

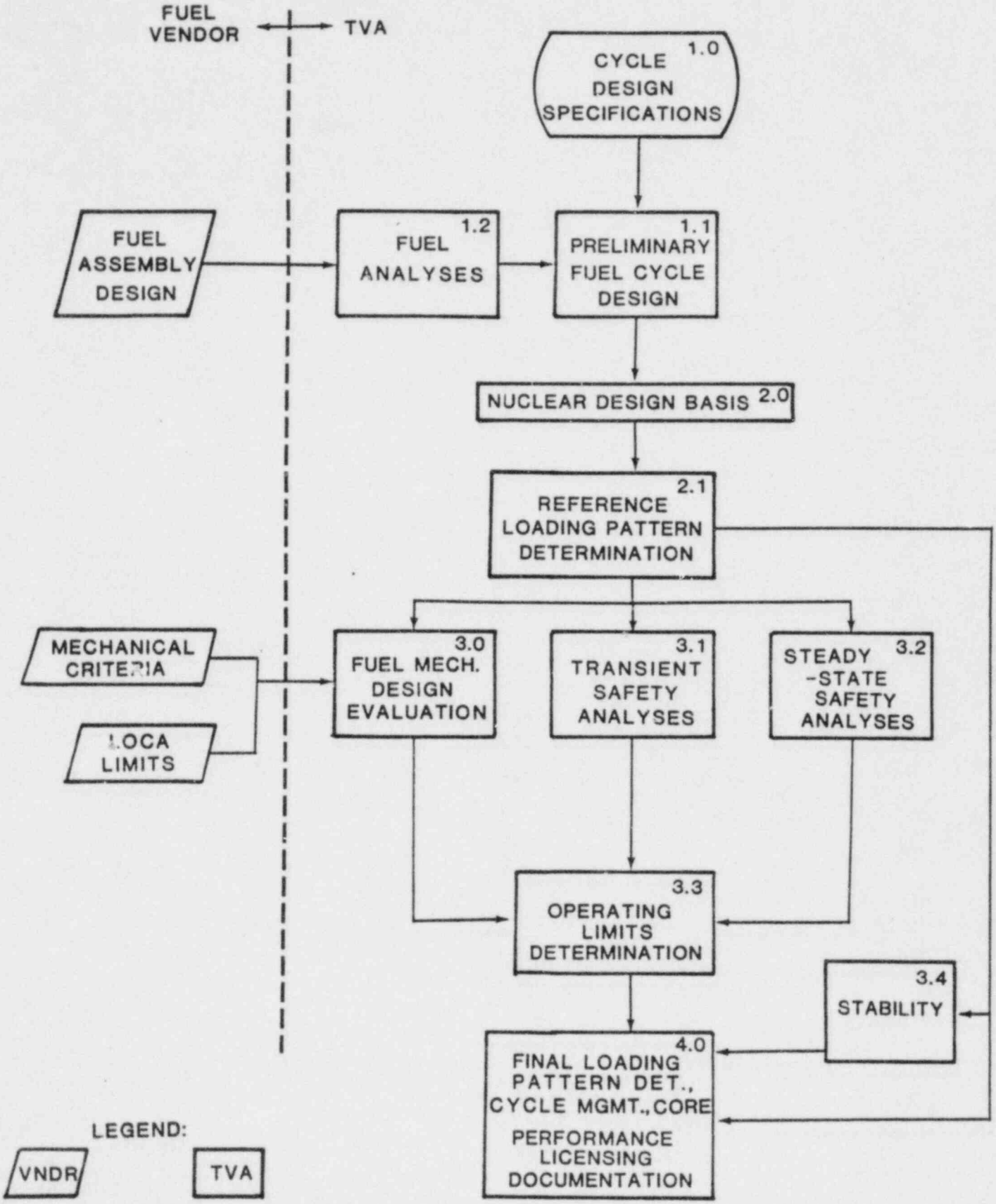
The reload design phase, Phase 2, begins with the determination of a Nuclear Design Basis (step 2.0). In this step, information from the preliminary design phase is compiled along with final reload design parameters and criteria. The Design Basis is then used as input to the Reference Loading Pattern Analyses (step 2.1). These analyses determine a core loading arrangement for licensing which meets energy and safety criteria specified in the design basis. The resulting reference loading pattern and supporting data are input to the safety analyses (Phase 3).

In Phase 3, the reference loading pattern is analyzed to verify that fuel failure criteria can be met and to determine the operating thermal limits. Step 3.0, Fuel Mechanical Design Evaluation, verifies that mechanical integrity criteria specified by the fuel vendor will not be violated. Also, MAPLHGR limits (specified by the fuel vendor) for conformance to LOCA criteria are incorporated into the evaluation. The Transient Safety Analyses (step 3.1) determine the reload core response to the rod drop accident and abnormal operational transients. The Steady-State Safety Analyses (step 3.2) determine the reload core response to loss of feedwater heating, rod withdrawal error, and fuel loading errors. Results from steps 3.0, 3.1, and 3.2 are evaluated in the Operating Limits Determination (step 3.3). In this evaluation the operating thermal limits for the reload cycle are determined. In addition to the safety analyses discussed above, the nuclear and hydrodynamic stability of the reload core must be analyzed (step 3.4). These stability analyses will be obtained from the fuel vendor until TVA develops its own stability analysis methodology.

The final phase in the RCDA process, Phase 4, involves determination of the final core loading pattern, analysis of the reload core steady-state performance, and specification of fuel cycle management recommendations (step 4.0). Near the end of cycle N-1, the cycle N final loading pattern is determined based on the reference licensing pattern and updated fuel assembly exposure data. Using the final pattern, cycle N performance is analyzed to evaluate compliance with thermal limits. Also, fuel management operating strategies are developed to meet cycle energy goals within the licensed operating limits. At EOC N-1, the assumed licensing basis for the reference loading pattern is verified to determine if additional licensing calculations are necessary. If warranted, additional analyses are performed and the reload licensing submittal is amended.

FIGURE 2.1

TVA RELOAD CORE DESIGN AND ANALYSIS PROCESS



3.0 Fuel Vendor - TVA Interface

The philosophy embodied in TVA reload fuel contracts involves a relatively clear division of responsibility between fuel assembly design and licensing and core design and licensing. TVA purchases from the fuel supplier fuel assemblies and related information sufficient to allow TVA to use and license the fuel assemblies. For example, a typical contract states that "the Contractor shall perform tests and analyses, and provide technical data and reports relating to its fuel assembly designs as may be required to enable TVA to obtain a license from the U.S. Nuclear Regulatory Commission in order to utilize one or more of Contractor's fuel assembly designs or types." TVA then is responsible for performing analyses required to (1) determine the manner of fuel utilization, (2) support day-to-day reactor operations, (3) license the manner of using Contractor's fuel assembly designs or types with the nuclear steam supply system, and (4) analyze, plan, and determine the properties of the reactor core with respect to neutronic characteristics, steady-state and transient minimum critical power ratio characteristics, thermal-hydraulic characteristics, and transient and accident analyses.

The vendor-TVA interface is illustrated in figure 2.1, which shows types of information received from the vendor and also where that information is used in the RCDA process. The vendor supplies a complete physical description of each fuel assembly which is used as input to lattice physics and fuel temperature calculations. The vendor also supplies criteria or limits which must be satisfied during the core design process in order to assure fuel assembly design calculations remain valid. These criteria and limits include linear heat generation rate (LHGR-kW/ft) versus exposure/residence time, LOCA limits, and the critical power ratio (CPR) safety limit. As described below, TVA also will receive transition boiling information needed for RCDA from the fuel supplier.

An essential part of any reload analysis methodology is the transition boiling correlation. TVA has obtained from the current Browns Ferry fuel supplier (General Electric Company) the right to use and reference the thermal analysis correlation "GEXL." In addition to descriptions of the GEXL correlation, its derivation, and limitations, TVA will receive from GE R-factor data for use in steady state and transient CPR calculations. These CPR calculation are described in subsequent sections of this report. It is anticipated that TVA's right to use GEXL will be retained for all of the reloads listed in table 1.1.

Finally, TVA has the option to purchase specific core analyses (such as a rod drop accident analysis) from the fuel supplier. Any reload licensing document which contains both TVA and vendor calculated results will clearly identify the source of the calculations. Also, TVA will promptly notify NRC of any significant changes in the vendor interfaces described in this report.

4.0 Fuel Mechanical Design

4.1 Analysis Objectives

The primary objective of the fuel assembly mechanical design analysis is to demonstrate that the fuel integrity is adequate to maintain the release of radioactivity within the limits of applicable regulations. Evaluations are made in conjunction with the core nuclear characteristics, core thermal-hydraulic characteristics, and safety evaluations to ensure that this objective is met for each reload core design.

TVA does not design fuel assemblies. As described in Section 3.0, the fuel assembly design and LOCA analyses will remain the responsibility of the fuel vendor. Under contractual agreements, the vendor supplies a complete physical description of each fuel assembly, LOCA limits, and the thermal and mechanical criteria which must be met by the final core design. For each reload core design performed by TVA, the direct analysis objective is to demonstrate compliance with the fuel mechanical design criteria and limits.

4.2 Design Criteria

The reload fuel mechanical design criteria are the operational and safety limits established for the fuel designs and the bases for these limits. For reload cores, the incoming fuel design is normally fixed and rarely a variable capable of being changed. Current practice is to establish operational and safety limits for the fuel designs comprising the reload core and to implement these limits as design criteria for safety evaluations. The mechanical design criteria and bases are provided by the fuel vendor. A detailed description of each mechanical design analysis and the associated criteria for current Browns Ferry fuel types can be found in the General Electric Generic Reload Fuel Application (reference 2).

4.3 Analysis Description

The power level, exposure range, and analysis assumptions considered by the fuel vendor in the mechanical design provide the basis for review of these same parameters for reload core designs. For each reload, the performance of the reload batch and other fuel remaining in the core is projected for the cycle length. The combination of power, exposure, and residence time from this projected performance are compared with the assumptions used in the generic analyses. The mechanical design is applicable only if the projected core operating conditions are bounded by the assumptions of the generic analyses.

One of the primary objectives of reload core design analyses is to demonstrate compliance with the established fuel safety limits for operational and abnormal plant transients. The fuel safety limits are established to preclude: (1) rupture of fuel rod cladding due to strain caused by relative expansion of the UO_2 pellets; and (2) severe overheating of the fuel cladding caused by inadequate cooling.

With the strain criteria, a value of 1 percent plastic strain of the zircaloy cladding has been established by the current fuel vendor (GE) as the safety limit. The 1 percent strain limits as a function of exposure

are documented in the GE Generic Reload Fuel Application (reference 2) for current Browns Ferry reload fuel types. The peak transient linear heat generation rate (LHGR) is compared to the established safety limits to demonstrate compliance.

Adequate cooling of the cladding during operational and abnormal transients is assured by demonstrating that transition boiling does not occur (CPR above safety limit). These evaluations are discussed in section 6.0 of this report.

Generic LOCA analyses have been performed by the fuel vendor for each fuel type currently used in Browns Ferry. LOCA analyses remain the responsibility of the fuel vendor. Reload core compliance with the LOCA analysis assumptions and criteria is assured by directly incorporating the appropriate MAPLHGR limits into the plant technical specifications.

For accident evaluations, certain other specific fuel criteria are applicable relative to peak fuel enthalpy, fuel machanical loadings, transition boiling, etc. The fuel criteria and safety limits applicable to accident events are discussed in section 6.

4.4 Analysis Output

The following specific items are documented as a result of the reload fuel mechanical design verification:

1. Internal documentation of reload core and fuel compliance with the vendor mechanical design criteria.

2. Verification of reload core and fuel compliance with the established fuel LHGR safety limits for operational and abnormal plant transients.

3. Documentation of the appropriate LOCA accident results and MAPLHGR limits for inclusion in the Reload Licensing Report and plant Technical Specifications.

5.0 Nuclear Design

5.1 Introduction

Reload core nuclear design analyses are essentially steady-state reactor physics calculations. In TVA's RCDA process, the nuclear design includes analytical steps in all of the design phases described in section 2.0. Specifically, the nuclear design analyses include: the Nuclear Design Basis (2.0)*; Reference Loading Pattern Analysis (2.1); SS Safety Analyses (3.2); and the Final Loading Pattern, Core Performance, and Cycle Management Analyses (4.0).

*Parenthetical numbers refer to the analysis steps of figure 2.1.

5.2 Nuclear Design Basis

5.2.1 Analysis Objective

The objective of this work is to compile and specify the reload cycle design parameters for RCDA. The Nuclear Design Basis consists of the design criteria, design goals, and input parameters for performing the steady-state reload core design. In the Nuclear Design Basis the following are specified:

- Projected end of previous cycle (N-1) conditions
- Fuel assemblies available for design reference loading
- Reload cycle (N) energy requirements
- Thermal design limits
- Reactivity and burnup limits
- Recommended design critical eigenvalues

5.2.2 Analysis Description

- Projected EOC N-1 Conditions

Nominal, minimum, and maximum cycle exposures for the end of cycle N-1 are specified in the nuclear design basis. A projection of the cycle N-1 core exposure distribution is made for the nominal cycle exposure to determine the basis for all cycle N reload design calculations. Also, a projection of the core exposure distribution is made for the minimum cycle N-1 exposure for licensing calculations. These exposure distribution calculations are performed using a 3-D reactor simulator model (references 1 and 6).

- Fuel Assemblies Available for Design Reference Loading

The number of fuel assemblies, fuel assembly types, and assembly identification numbers are specified in the Nuclear Design Basis. The specification includes fresh and exposed fuel assemblies which will be available for loading in cycle N. Known or suspected damaged fuel assemblies are excluded from the list of possible assemblies for reload design.

. Reload Cycle (N) Energy Requirements

Cycle N target energy (MWD) and an acceptable variation about the target are specified in the Design Basis. Also, planned startup and shutdown dates, cycle operating capacity factor, and operational plans for coastdown or load following are specified. These items are determined by the TVA operating divisions and documented in a Design Input Data Package which is incorporated into the Design Basis. Preliminary fuel cycle design calculations with the 3-D simulator model verify that the cycle energy requirements can be met within expected operating thermal limits and maximum fuel burnup limits. This verification is repeated in the reference loading pattern calculations.

. Thermal Limits

The thermal limits which must be addressed during the nuclear design process are maximum linear heat generation rate (MLHGR), minimum critical power ratio (MCPR), and maximum average planar linear heat generation rate (MAPLHGR). The nuclear design criterion with regard to thermal limits is that the reload core design must be capable of normal steady-state operation without violating the thermal limits. In order to verify this criterion, values are established in the Design Basis for the actual or expected cycle N operating thermal limits. Actual limits are available for MLHGR and MAPLHGR at the start of the nuclear design process. The MCPR limit is not known until both nuclear and safety analyses are completed. Consequently, a target MCPR is selected for the initial nuclear design work. Compliance with the actual limit is confirmed upon completion of the MCPR operating limit analyses. The MLHGR operating limit is defined by the fuel vendor to prevent steady-state operation which would result in exceeding the cladding 1 percent plastic strain limit in the event of an overpower transient. GE has defined the operating limit MLHGR for 7 x 7 fuel as 18.5 kW/ft and for 8 x 8 fuel as 13.4 kW/ft. These MLHGR values represent the TVA design criteria for GE fuel assemblies. The MAPLHGR operating limit is also defined by the fuel vendor for each fuel assembly type to prevent steady-state operation which would result in exceeding the Appendix K criteria in the event of a LOCA. The MAPLHGR limits defined by GE will be the TVA design criteria.

The MCPR operating limit is determined each cycle from nuclear design and safety analyses. This limit is established to prevent steady-state operation which could result in going below the critical power safety limit (1.07) assuming the most severe anticipated operational transient. Since the actual cycle N operating limit is a result of the reload analyses, the cycle N-1 operating limit MCPR is usually selected as a "target" MCPR for the nuclear design analyses. As noted above, compliance with the actual MCPR limit is confirmed once the limit is established.

. Reactivity and Burnup Limits

Design limits are specified for the Doppler coefficient, void coefficient, cold shutdown margin, SLCS shutdown margin, and maximum fuel pellet exposures as shown in table 5.2.1. Compliance with the design limits are normally demonstrated in the reference loading pattern calculations and ensure compliance with the indicated technical specification limits.

. Recommended Design Critical Eigenvalues

Target hot and cold critical eigenvalues for use in the 3-D simulator model are specified in the design basis. The target critical eigenvalues are determined by statistical analysis of calculated critical eigenvalues from core follow calculations.

These determinations are made for both cold configurations and hot operating conditions. As the data base of critical comparison points changes, new statistical analyses are performed to adjust the target eigenvalues appropriately with greater emphasis placed on comparisons with the most recent plant data.

TABLE 5.2.1

REACTIVITY AND BURNUP LIMITS

Doppler coefficient - negative
Void coefficient - negative

Cold shutdown margin

Design max. cold k_{eff}, SRO* <0.99
Technical specification limit $k_{eff} \leq 0.9962$

SLCS shutdown margin

Design max. k_{eff} at 600 ppm <0.99
Technical specification limit - k_{eff} <1.0

Burnup Limit

Max. pellet burnup <50 GWD/MT (P8x8R)
<44 GWD/MT (7x7, 8x8, 8x8R)

*Strongest rod out.

5.2.3 Analysis Output

The output of the Nuclear Design Basis determination is a data book containing the discussed design criteria and design input parameters. This data book will also reference pertinent information such as data libraries, computer codes, previous analyses, regulatory information, and plant data which should be considered in the reload design. When completed, the Nuclear Design Basis data book will serve as the input source of information for the reference loading pattern calculations and other reload core design and licensing analyses.

5.3 Reference Loading Pattern Analysis

5.3.1 Analysis Objectives

Determination of the reference loading pattern is the primary function of the reload core nuclear design phase. The objectives of the reference

loading analysis are: (1) to determine the reload core loading arrangement for cycle N and (2) to verify that the core loading meets the design criteria. The reference loading pattern is used in the reload licensing safety calculations and as the basis for generating the actual loading pattern.

5.3.2 Design Criteria

Nuclear design criteria to be applied in the reference loading pattern determination are shown in table 5.3.1. These design criteria have a basis defined in the Browns Ferry Technical Specifications or the FSAR and generally define limits necessary to ensure safe reactor operation. These criteria are translated into "Limiting Conditions for Operation" (LCOs) in plant technical specifications. TVA designs the reference loading pattern to maintain appropriate margins to the listed criteria. These design margins are specified to account for uncertainties in the nuclear design models relative to the plant process computer so that the margin between LCOs and monitored results will be sufficient to allow flexibility in plant operations.

TABLE 5.3.1

REFERENCE LOADING DESIGN CRITERIA

1. 0.38 % ΔK cold shutdown margin (strongest rod out)

2. K_{eff} <1.0 for standby liquid control system shutdown at 600 ppm, no control rod movement

3. Maximum in-sequence control rod worth results in < 280 cal/gm enthalpy rise due to a rod drop accident (RDA analyses are discussed in section 6.0)

4. MLHGR $\leq$ operating limit MLHGR (13.4 kW/ft - 8 x 8)
   (18.5 kW/ft - 7 x 7)

5. MAPLHGR $\leq$ operating limit MAPLHGR (as defined by fuel vendor for each fuel type)

6. MCPR $\geq$ operating limit MCPR

5.3.3 Analysis Description

The reference loading pattern determination includes fuel shuffling based on experience and shuffling algorithms, and cycle analyses to verify that the loading pattern meets the various design criteria. These cycle analyses include the following:

1. Power distribution-depletion calculations to the end of cycle N

2. Cold shutdown margin analysis (strongest rod out, 68°F, no xenon) throughout cycle N

3. Hot excess reactivity calculations throughout cycle N

4. Standby liquid control system analysis (68°F, all rods out, 600 ppm boron, no xenon, most reactive exposure)

Additionally, the following analyses may be included in the reference loading calculations:

1. Control rod worth determinations

2. Stepwise power distribution-depletion calculations with expected full power control rod patterns.

3. Cycle N exposure capability and hot excess reactivity starting from various EOC N-1 burnups.

The 3-D simulator code is used to perform the reference loading calculations. Design input is obtained from the Nuclear Design Basis, thermal-hydraulic and nuclear physics data libraries, and exposure history data from core follow analyses.

5.3.4 Analysis Output

Primary outputs from the reference loading analysis include the reference loading pattern, calculated safety parameters (CSDM and SLCS capability), predicted operating thermal performance (MLHGR, MAPLHGR, MCPR), and burnup-dependent reactor core conditions throughout cycle N for input to the safety analysis. These parameters will be documented in the reload licensing report submitted to obtain an operating license. Also, the reference loading pattern will be the basis for determining the final loading pattern in the fuel cycle management/core performance phase of the reload design.

5.4 Steady-State Safety Analyses

5.4.1 Analysis Objective

The primary objective of the steady-state analyses is to calculate the core response to the loss of feedwater heating (LFWH) transient, rod withdrawal error (RWE) transient, and fuel loading errors (FLE). The LFWH and RWE events are relatively slow transients which can be analyzed in a quasi-static manner using a steady-state 3-D simulator model.

5.3.3 Analysis Description

The reference loading pattern determination includes fuel shuffling based on experience and shuffling algorithms, and cycle analyses to verify that the loading pattern meets the various design criteria. These cycle analyses include the following:

1. Power distribution-depletion calculations to the end of cycle N

2. Cold shutdown margin analysis (strongest rod out, 68°F, no xenon) throughout cycle N

3. Hot excess reactivity calculations throughout cycle N

4. Standby liquid control system analysis (68°F, all rods out, 600 ppm boron, no xenon, most reactive exposure)

Additionally, the following analyses may be included in the reference loading calculations:

1. Control rod worth determinations

2. Stepwise power distribution-depletion calculations with expected full power control rod patterns.

3. Cycle N exposure capability and hot excess reactivity starting from various EOC N-1 burnups.

The 3-D simulator code is used to perform the reference loading calculations. Design input is obtained from the Nuclear Design Basis, thermal-hydraulic and nuclear physics data libraries, and exposure history data from core follow analyses.

5.3.4 Analysis Output

Primary outputs from the reference loading analysis include the reference loading pattern, calculated safety parameters (CSDM and SLCS capability), predicted operating thermal performance (MLHGR, MAPLHGR, MCPR), and burnup-dependent reactor core conditions throughout cycle N for input to the safety analysis. These parameters will be documented in the reload licensing report submitted to obtain an operating license. Also, the reference loading pattern will be the basis for determining the final loading pattern in the fuel cycle management/core performance phase of the reload design.

5.4 Steady-State Safety Analyses

5.4.1 Analysis Objective

The primary objective of the steady-state analyses is to calculate the core response to the loss of feedwater heating (LFWH) transient, rod withdrawal error (RWE) transient, and fuel loading errors (FLE). The LFWH and RWE events are relatively slow transients which can be analyzed in a quasi-static manner using a steady-state 3-D simulator model.

5.4.2 Acceptance Criteria

The acceptance criteria applied to the steady-state safety analyses are the fuel cladding integrity safety limits (MCPR and LHGR) as currently defined for the applicable vendor fuel types. These criteria for GE fuel types currently expected to be used at Browns Ferry are defined in reference 2.

5.4.3 Analysis Description

5.4.3.1 Loss of Feedwater Heating Transient

Feedwater heating can be lost by closure of the heater extraction line or feedwater bypassing around the heaters. The maximum number of feedwater heaters which can be tripped or bypassed by a single event represents the most severe transient. The instantaneous loss of feedwater heating results in an increase in core inlet subcooling and a corresponding power increase. Even for the worst case, the power increase is at a very moderate rate. Therefore, analysis of the event with a static neutronics model with coupled thermal-hydraulic feedback is acceptable. It is also preferable because of accurate modeling of the power increase using the 3-dimensional neutronics model. The steady-state analysis is also conservative since the heat flux increase does not lag behind the nuclear power increase as in a transient model.

The LFWH analytical procedure assumes that 100°F feedwater heating can be lost by a single event. The method of analysis is an iterative technique to determine the power increase due to the cooler feedwater. Starting from the nominal safety analysis conditions corresponding to 105 percent steam flow (104.5 percent thermal power, 100 percent flow), a power search/power distribution calculation is performed with the 3-D simulator for the 100°F LFWH. The results of the simulator calculation are used to determine inlet subcooling and pressure for the calculated higher power level. These new core conditions are then used to perform another power search calculation. This procedure is repeated until a converged 3-D power search calculation is obtained for the core conditions corresponding to the loss of 100°F feedwater heating. The results of the converged solution are then input to a hot channel analysis to determine the CPR for a fuel assembly initially operating on the expected MCPR limit.

5.4.3.2 Rod Withdrawal Error

The rod withdrawal error is assumed to occur while the reactor is operating at 100 percent of rated power. It is assumed that the operator continuously withdraws the maximum worth control rod until the withdrawal is stopped by the rod block monitor (RBM) system. For analysis purposes, a limiting operating control rod pattern is selected which results in the reactor being placed on thermal design limits near the error rod. Also, for conservatism, no xenon is assumed present and the analysis is performed at the most reactive point in the cycle. These are highly abnormal conditions and could only be achieved by deliberate operator action.

Since the power change due to the withdrawn control rod is relatively slow, the RWE is analyzed using the static 3-D simulator model. First, at the core conditions described above, the highest worth control rod is determined. Then, a limiting control rod pattern is searched for in which

the highest worth rod is inserted and some neighboring fuel assemblies are operating at or near the expected thermal limits. Note that this condition would result in a highly unusual control rod pattern. With this rod pattern, a series of simulator calculations are run withdrawing the highest worth rod in several increments from full-in to full-out. The change in reactor power level and power distribution along with simulated LPRM readings are calculated for each rod position. These results are input to a RBM response calculation. The limiting MLHGR and MCPR for RBM setpoints from 104 percent to 110 percent are determined and reported in the licensing submittal. To add additional conservatism, the RBM response is determined for the worst allowable combination of failed LPRM strings in the RBM channels.

5.4.3.3 Fuel Loading Error

A fuel loading error can result from the loading of a fuel assembly in an incorrect location or by rotating a fuel assembly in the correct location with respect to the correct loading orientation. Both of these events are assumed to be undetected in subsequent core verifications and the core is assumed to be operated with the loading error. The FLE safety analysis determines if a loading error would result in the limiting MCPR or would result in violation of the cladding 1 percent plastic strain limit (LHGR).

The bundle mislocation error assumes a fresh reload bundle is erroneously loaded in an unmonitored bundle location which, at or near the end of cycle N, results in the lowest MCPR and highest MLHGR (assuming the highest power monitored bundle in a symmetric location is operating at the LHGR limit). The MCPR and MLHGR calculated for this abnormal condition must not violate the fuel cladding integrity limits. These values of MCPR and MLHGR are calculated with the 3-D simulator model.

The bundle misorientation error assumes a bundle is rotated 180 degrees from its correct orientation. The resulting change in power due to the rotation is calculated. Rotated bundle local peaking factors are analyzed with the lattice physics data generation code (references 3 and 6) to determine the MLHGR. Rotated bundle R-factors (supplied by the fuel vendor) and the change in power are used to determine the MCPR in a hot channel analysis. The worst case MLHGR and MCPR due to a misorientation error are reported in the licensing submittal.

5.4.4 Analysis Output

The result of the steady-state safety analyses are input to the operating limits determination to be used in setting the cycle N MCPR operating limit. Also, the calculated MLHGR and MCPR due to these safety analyses are reported in the reload licensing report.

5.5 Core Performance, Cycle Management, Final Loading Pattern Analyses

5.5.1 Analysis Objectives

The core performance, cycle management, and final loading pattern analyses are last in the sequence of calculations performed in support of reload cycle N. The objectives of these analyses are threefold:

1. Calculate the core performance throughout cycle N for full power, rodded core configurations.

2. Determine the optimum fuel cycle operating strategy to meet cycle N goals.

3. Determine the final core loading pattern and verify that the licensing design basis is valid at the actual EOC N-1 conditions.

5.5.2 Design Criteria

The design criteria for determination of the final loading pattern are the same as those applied to the reference loading pattern analyses. In addition, conditions assumed for EOC N-1 in the licensing analyses must be verified. In performing the reference loading pattern calculations and subsequent safety analyses, conditions at the EOC N-1 are projected forward, resulting in an assumed core average exposure and axial exposure distribution. At the EOC N-1, the actual core conditions will be compared to the assumed conditions for the cycle N licensing analyses to verify that the results are still valid.

5.5.3 Analysis Description

Approximately two months prior to shutdown of cycle N-1, a preliminary loading pattern for cycle N is developed and core performance calculations are made to establish cycle management recommendations. The latest core follow results and best estimate of projected EOC N-1 burnup and operating conditions are utilized. A preliminary loading pattern is obtained by arranging the fuel assemblies available for reloading as closely as possible to the design reference loading pattern used for licensing analyses.

After a preliminary pattern is developed, stepwise power distribution-depletion calculations are made with target, full power control rod patterns to predict core performance. The cycle N predicted core performance is expressed in terms of the calculated thermal limits (MCPR, MLHGR, and MAPLHGR) at rated core conditions. The following calculations are also performed:

1. Cycle N target power and exposure distribution

2. Cold shutdown margin throughout cycle N

3. SLCS analysis at the most reactive exposure point

4. Hot excess reactivity throughout cycle N

The design criteria for the preliminary loading pattern are the same as those for the reference loading pattern. Input is obtained from the Nuclear Design Basis, Reference Loading Pattern determination, and the Operating Thermal Limits determination. All of the calculations are performed using the 3-D simulator model.

After shutdown of cycle N-1, the final cycle N loading pattern is determined and the licensing basis verified. The preliminary loading pattern, actual EOC N-1 exposure, and exposure distribution are used to develop the final loading pattern. Normally, if no damaged fuel assemblies exist among the exposed assemblies to be reloaded, the preliminary loading pattern will become the final loading pattern. However, results of the licensing basis verification may require further fuel shuffling to ensure that the design operating limits are met.

Verification of the licensing basis is performed by comparing the actual EOC N-1 core average exposure and exposure distribution against the assumed EOC conditions. If the actual EOC N-1 conditions differ from the licensing basis, it will be necessary to consider reanalyzing certain safety parameters. These parameters include the CSDM, SLCS worth, RDA, and the limiting abnormal operational transient event. If the reanalysis results are less conservative than the reference loading licensing results, an amendment to the reload submittal will be made. Another function to be performed after the final loading pattern is determined is the Shutdown Margin Demonstration sequence. Control rod withdrawal sequences are developed such that the CSDM can be determined during the startup test program. After the shutdown margin demonstration test is performed, the results are analyzed to verify the actual CSDM available. Determination of the control rod withdrawal sequence and analysis of the test results are performed with the 3-D simulator.

5.5.4 Analysis Output

The results of the core performance and cycle management analyses are documented in TVA internal fuel cycle design reports. Additional safety analyses performed to support determination of a final loading pattern will be reported to the NRC in amendments to the reload licensing report if less conservative than the reference licensing report or to take credit for undue conservatisms in the reference licensing analyses.

6.0 Safety and Transient Analysis

6.1 Analysis Objectives

The objective of the plant safety analyses is to evaluate the ability of the plant to operate without undue hazard to the health and safety of the public. The cycle 1 safety evaluation is contained in the plant Final Safety Analysis Report (FSAR). For each subsequent refueling, reload core specific safety evaluations are performed and documented in the reload licensing report.

The objective of the reload core safety evaluations is to demonstrate the continued applicability of analyses contained in previous licensing submittals or, if required, to perform new safety analyses and document the results in the reload licensing report.

6.2 Design Criteria

The design criteria applicable to the evaluation of a particular safety-related event depend upon the category of event under consideration. The distinctions between event categories are primarily related to the frequency of anticipated occurrence, number of equipment failures or operator errors required to initiate the event, and applicable regulatory requirements. For the purposes of safety evaluations, it is convenient to place all potential occurrences with safety implications into two general categories: (1) Normal and Abnormal Operational Events, and (2) Accident or Design Basis Events. Although other schemes for categorizing safety related incidents can and have been used by licensing applicants, the Browns Ferry FSAR uses only two categories and this remains the basis upon which the reloads are licensed.

The design criteria applicable to Normal and Abnormal Operational Events are as follows:

1. No fuel failures can be calculated as a result of a plant transient.

2. The release of radioactive material to the environment must not exceed the limits of 10 CFR 20.

3. The nuclear steam system stress must not be in excess of that allowed for transients by applicable industry codes.

4. The reactor core, core flow channels, and total system must be inherently stable for disturbances of any critical variable.

In practice, the critical power ratio (CPR) is the primary means of demonstrating compliance with design criteria 1 and 2 (no fuel failures). As discussed in Section 3.0, TVA has obtained from the current Browns Ferry fuel supplier the right to use and reference the General Electric Thermal Analysis Basis "GETAB," including R-factor data for use in steady-state and transient CPR calculations of GE supplied fuel. The minimum allowable transient CPR is such that, considering uncertainties in the CPR correlation, in fuel manufacturing, and in monitoring the core operating state, more than 99.9 percent of the fuel rods would be expected to avoid boiling transition. Additionally, during normal and abnormal operational events,

the fuel can not exceed the 1-percent plastic strain limit imposed on the cladding regardless of CPR.

The allowable overpressure for the reactor vessel is defined by the ASME Boiler and Pressure Vessel Code, Section III, class 1. Because the ASME code permits pressure transients up to 10 percent over design pressure, the design pressure portion of the reactor coolant pressure boundary meets the design requirement if peak nuclear system pressure remains below 1375 psig (110 percent x 1250 psig) for Browns Ferry.

Three types of stability are considered in the design of boiling water reactors: (1) reactor core stability, (2) channel hydrodynamic stability, and (3) total system stability. A stable system is demonstrated if no divergent oscillation develops within the system as a result of disturbances of any critical variable such as steam flow, pressure, neutron flux, or recirculation flow.

The design criteria applicable to Accident or Design Basis Events are as follows:

1. Calculated radioactive material release will not exceed the limits of 10 CFR 100.

2. Catastrophic failure of fuel cladding will not occur, including fragmentation of fuel cladding and excessive fuel enthalpy is not predicted.

3. Nuclear system or containment stresses will not exceed those allowed for accidents by applicable codes.

4. Radiation overexposure of plant personnel in the control room will not occur.

In practice, safety evaluations of accidents and design basis events use one or more of the following criteria to predict fuel failures for radioactive release calculations:

1. Transient CPR below the established safety design limit.

2. Cladding strain above the 1-percent plastic strain safety limit.

3. Excessive cladding temperature.

4. Radially averaged fuel enthalpy above the established licensing safety limit. (Currently 170 cal/gm)

5. Gross compression, distortion and bending energy above the fuel vendor established material capabilities.

6.3 Abnormal Operational Events

Abnormal operational transients are moderate frequency (once per year to once in 20 years) transients caused by a single operator error or equipment malfunction. Acceptable CPR margin and margin to the fuel linear heat rate (LHR) safety limit (1-percent plastic strain) must be demonstrated for each

event. The following is a list of the Browns Ferry FSAR abnormal operational transients grouped according to the effects on key core parameters:

Events Resulting in a Nuclear System Pressure Increase:

1. Generator Trip - Turbine Control Valve Fast Closure
2. Loss of Condenser Vacuum
3. Turbine Trip
4. Turbine Trip @ High Power without Bypass
5. Turbine Trip @ Low Power without Bypass
6. MSIV Closure - All valves
7. MSIV Closure - One valve
8. Pressure Regulator Failure

Events Resulting in a Reactor Vessel Water Temperature Decrease:

1. Loss of Feedwater Heaters
2. Operation of the RHRS Heat Exchangers - Decay Heat System
3. Inadvertent Pump Start
   A. Control rod drive and makeup water - fall high-flow
   B. Startup of RCIC system
   C. Startup of HPCI System - 5000 gpm

Events Resulting in a Positive Reactivity Insertion:

1. Continuous rod withdrawal during power range operation
2. Continuous rod withdrawal during reactor startup
3. Control rod removal error during refueling
4. Fuel assembly insertion error during refueling

Events Resulting in a Reactor Vessel Coolant Inventory Decrease:

1. Pressure regulator failure
2. Inadvertent opening of a relief or safety valve
3. Loss of feedwater flow
4. Loss of auxiliary power

Events Resulting in a Core Coolant Flow Decrease:

1. Recirculation Flow Control Failure - Decreasing Flow

2. Trip of one recirculation pump

3. Trip of two recirculation pump MG set drive motors

Events Resulting in a Core Coolant Flow Increase:

1. Recirculation Flow Control Failure - Increasing Flow

2. Startup of Idle Recirculation Pump

Event Resulting in Excess of Coolant Inventory:

1. Feedwater Controller Failure - Maximum Demand

Abnormal operating transients are evaluated to determine the plant operating MCPR limit and to demonstrate that a plant operating within limits will sustain essentially no fuel damage from anticipated transients. In addition to these analyses, evaluations of less frequent postulated events are made to ensure an even greater depth of safety.

6.4 Accidents and Design Basis Events

Accidents are events which have a projected frequency of occurrence of less than once in every one hundred years for an operating BWR. Some fuel damage is allowed for accident evaluations within the limits of the previously described criteria related to radiological releases, containment stresses, and plant personnel safety. The broad spectrum of postulated accident events is covered by five categories of design basis events. The five categories and the associated Browns Ferry FSAR design basis events are as follows:

1. Accidents that result in radioactive release from the fuel with the nuclear system barrier, primary containment, and secondary containment initially intact.

   Design Basis Accident - Single control rod drop accident

2. Accidents that result in radioactive release directly to the primary containment.

   Design Basis Accident - Loss-of-coolant accident (rupture of one recirculation loop)

3. Accidents that result in radioactive release directly to the secondary containment with the primary containment initially intact.

   Design Basis Accident--(a) steamline break inside reactor building, (b) dropping a fuel assembly into fuel pool. These are bounded by categories 4 and 5 below.

4. Accidents that result in radioactive releases directly to the secondary containment with the primary containment not intact.

   Design Basis Accident - Refueling accident (fuel assembly drops on core during refueling)

5. Accidents that result in radioactive material releases outside the secondary containment.

   Design Basis Accident - Steamline break accident (main steamline breaks outside of secondary containment)

6.5 Reload Licensing Basis

6.5.1 Limiting Abnormal Operational Events

A plant and fuel cycle specific MCPR operating limit is determined for each reload to ensure that the CPR safety limit will not be violated during all normal and abnormal operational events. The MCPR operating limit is obtained by the addition of the ΔCPR calculated for the most severe abornmal operational event to the CPR safety limit (a detailed description of the ΔCPR analysis method is included in Section 6.6). The limiting abnormal operational events are evaluated for each reload to demonstrate that the core design meets the design criteria of essentially no fuel damage, system pressure less than 1375 psig, and core system stability requirements. The following abnormal events have been identified as potentially limiting for GE BWR-4 plants:

1. Generator Load Rejection without Bypass (GLRWOBP)

2. Turbine Trip without Bypass (TTWOBP)

3. Loss of Feedwater Heaters (LFWH)

4. Inadvertent HPCI Startup (IHPCI)

5. Feedwater Controller Failure (FWCF)

6. Main Steamline Isolation Valve Closure (MSIVC)

7. Rod Withdrawal Error (RWE)

8. Stability Analysis for Core, Core Channels, and System

A brief description of each transient is as follows:

GLRWOBP - Fast closure of the turbine control valves is initiated whenever electrical grid disturbances occur which result in significant loss of load on the generator. The turbine control valves are required to close as rapidly as possible to prevent overspeed of the turbine generator shaft. The closing causes a sudden reduction in steam flow which results in a pressure increase in the reactor vessel. The pressure increase collapses voids and produces a power increase. In normal operation, the bypass valves would open and reduce the severity of the transient.

TTWOBP - A variety of NSSS and BOP disturbances will initiate a turbine trip. Some examples are high reactor water level, loss of control fluid pressure, low condenser vacuum, moisture separator high level, and large turbine vibrations. Turbine trips generally occur to protect the turbine from damage. In normal operation, the bypass valves would open to reduce the severity of the transient.

Without modifications to existing plant equipment or changes in the historic analysis assumptions, the TTWOBP will not be limiting for reload analyses performed specifically for the Browns Ferry Nuclear Plants (BFNP). For BFNP, the GLRWOBP transient will always be more severe than the TTWOBP transient because (1) the control valve closure time initiated on generator load rejections is less than the stop valve closure time initiated on turbine trips (greater pressure and power increase for GLR) and (2) the turbine overspeed during a GLR may result in a slightly increased core flow and void collapse.

LFWH - Feedwater heating can be lost in at least two ways: (1) the steam extraction line to a heater is closed, or (2) feedwater is bypassed around the heater. Either case produces a gradual cooling of the feedwater supplied to the reactor. This increases core power due to the negative void coefficient. The maximum number of feedwater heaters which can be tripped or bypassed by a single event (equipment failure or operator error) represents the most severe LFWH transient.

IHPCI - The IHPCI transient is initiated when the HPCI pumps are inadvertently started (either by equipment failure or operator error) and cold water is injected into the feedwater line. This produces an increase in inlet subcooling and a subsequent power increase.

Without modifications to existing plant equipment or changes in the historic analysis assumptions, the IHPCI transient will not be limiting for reload analyses performed specifically for the Browns Ferry Nuclear Plants (BFNP). For BFNP, the LFWH transient is more severe than the IHPCI transient. The minimum CPR for both transients occurs when the reactor reaches a new steady-state power (or at scram if final power is greater than the scram setpoint) determined by the final inlet enthalpy. A larger enthalpy reduction results in a more severe (higher final power and larger CPR) transient. Based on the maximum HPCI flowrate and minimum HPCI enthalpy, the IHPCI transient results in a smaller reduction in core inlet enthalpy and consequently is less severe than the LFWH transient.

FWCF - Feedwater control can be lost due to the failure of a control device which can increase the feedwater flow. The failure results in an increase in coolant inventory. The excess inventory produces an increase in core inlet subcooling and reactor water level. Reactor power will increase until a turbine trip occurs due to a high reactor water level.

MSIVC - Various steamline and nuclear system malfunctions or operator actions can initiate a MSIV closure. Examples are: low steamline pressure, high steamline flow, high steamline radiation, and low reactor water level. Most MSIVCs (except closures due to isolation valve failure or operator error) occur to prevent core uncovery or high release of radioactive steam from containment. A MSIVC transient produces a system pressure increase and a corresponding power increase. The MSIVC

transient, as analyzed for the SAR, take credit for reactor trips initiated by the valve closures and these transients are not limiting for CPR considerations.

The MSIVC event analyzed for reload applications assumes the closure of all main steamline isolation valves with reactor trip initiated on high flux. Reliance on the flux trip assumes multiple failures of the position switch of each MSIV. The transient, as analyzed, is technically not an operational transient since it assumes multiple failures in the reactor protection system. The results of this analysis are used only to demonstrate that the peak allowable pressure of 110 percent of the vessel design pressure is not exceeded for the most severe isolation event.

RWE - All system equipment is assumed to be operating normally; hence, the only transient phenomena of interest are local core power perturbations. While operating in the power range, the operator is assumed to make a procedural error and withdraw the maximum worth control rod. It is assumed that the operator ignores the peak linear power limit alarms and the rod block monitor alarms and continues the rod withdrawal. The core average power and local power in the vicinity of the withdrawn control rod will increase and could potentially cause fuel damage. The core behavior during the event is calculated using a steady-state 3-dimensional coupled nuclear-thermal hydraulic BWR core simulator. The approach assumes the core neutronic and heat transfer parameters are in equilibrium. Core conditions are calculated as a functon of rod block setpoints and a rod block setpoint is selected for the reload core design such that CPR and fuel cladding safety limits can not be exceeded, i.e., the event will be terminated before fuel damage occurs because the rod block system will physically not allow further rod withdrawal.

Stability Analysis - Reload criteria limit core designs to stable configurations; thus, unstable operational transients are precluded.

For reload cores, two types of stability are examined. First is the hydrodynamic channel stability of one or more types of channels operating in parallel with other channels in the cores. This is considered because flow oscillations may impede heat transfer to the moderator and drive the reactor into power oscillations. Second is the reactivity feedback stability of the entire reactor core which also involves power oscillations. Total system dynamics are comprised of the dynamics of the control system combined with those of the basic process and is, therefore, not reanalyzed for reload cores. The assurance that the total plant is stable and, therefore, has significant design margin is demonstrated analytically when the acceptable performance limit of a decay ratio less than 1.0 or a damping coefficient greater than 0.0 is met for each type of stability discussed.

6.5.2 Limiting Accident Events

From the broad spectrum of postulated accidents, six categories of design basis events have been identified which may require analysis each refueling. These events are the control rod drop, main steamline break, loss-of-coolant, refueling, recirculation pump seizure, and fuel assembly loading accidents.

Control Rod Drop - This accident assumes the breakage or disconnection of an inserted control rod drive from the cruciform control blade. The blade sticks in the inserted position as the rod drive is withdrawn. The blade then falls to the position of the rod drive. The analysis of this accident is performed at various reactor operating states; the key reactivity feedback affecting the shutdown of the initial power burst is the Doppler coefficient. Final shutdown is achieved by scramming all but the dropped rod.

The number of failed fuel rods assumed for radiological consequences is based upon CPR, fuel mechanical limits, or fuel peak enthalpy. The most limiting rod drop situations are from low power initial conditions and the fuel peak enthalpy limit is the most common failure mode. Currently fuel is assumed failed above 170 cal/gm (radially averaged) and no fuel can be above 280 cal/gm.

The rod drop accident is addressed for each reload core design. Results of the analysis are supplied in the cycle specific reload licensing submittal. The current fuel vendor (General Electric) has performed generic sensitivity analyses for group notch plants. The cycle licensing submittal will usually be based on comparisons to the generic analysis to demonstrate that the Doppler reactivity coefficient, accident reactivity functions, and scram reactivity functions are within the bounding values which ensure the 280 cal/gm design limit will not be exceeded. A reload core specific analysis will be performed in cases where the bounding conditions are not met.

Main Steam Line Break - Results for this accident depend on the thermal-hydraulic parameters of the overall reactor and primary coolant activity. The FSAR analysis of this event demonstrated no transition boiling concern. Hence, unless reactor or systems equipment changes are made which adversely affect this transient, no reload specific analysis is required.

Loss-of-Coolant Accident - The current fuel vendor performs the emergency core cooling system (ECCS) evaluations and develops the models which are employed to determine the effects of the loss-of-coolant accident (LOCA) in accordance with 10 CFR 50.46 and Appendix K. The current ECCS models and computer codes are identified as LAMB, SCAT, SAFE, REFLOOD, and CHASTE. The LAMB code calculates the short-term blowdown response and core flow, which are input into the SCAT code to calculate blowdown heat transfer coefficients. The SAFE code is used to determine longer term system response and flows from the various ECCS systems. Where appropriate, the output of SAFE is used in the REFLOOD code to calculate liquid levels. The results of these codes are used in the CHASTE code to calculate fuel clad temperatures and maximum average planar linear heat generation rates (MAPLHGR) for each fuel type.

GE separates plants into groups for the purpose of LOCA analysis. Within each plant group there will be a single lead plant analysis which provides the basis for the selection of the most limiting break size yielding the highest peak cladding temperature (PCT). Also, the lead plant analysis provides an expanded documentation base to provide added insight into evaluation of the details of particular phenomena. The remainder of the plants in that group will have nonlead plant analyses referenced to the lead plant analysis.

New MAPLHGR values are provided for each reload in which a new fuel type is introduced. These analyses are bounding in the sense that they address all reload core configurations containing the new fuel type in combination with previously licensed fuel. LOCA licensing analyses have previously been submitted which address all current fuel designs for Browns Ferry. If changes are made in the current fuel designs, reactor internals, or system equipment which require new LOCA evaluations, these analyses will be provided by the fuel vendor. Reload core compliance with the LOCA analysis assumptions and criteria is assured by directly incorporating the appropriate MAPLHGR limits supplied by the fuel vendor into the plant technical specifications as discussed in section 4.3.

Refueling Accident - The analysis of this event assumes that a fuel assembly is dropped onto the top of the core during refueling with the drywell head and reactor vessel head removed. The radioactive material released is available for transport directly to the containment. The FSAR analysis of this event was bounding for the 7 x 7 fuel design. The Generic Reload Fuel Application (Reference 2) presented the results of analyses for the 8 x 8 and 8 x 8R fuel designs and determined that 8 x 8 fuel designs result in radiological releases less than 96 percent of those presented in the FSAR. Thus, the FSAR results are bounding for all current reload fuel types. Unless a new fuel design is introduced to the core, no new reload specific refueling accident analysis is required.

Recirculation Pump Seizure - This accident is assumed to occur as a consequence of an unspecified, instantaneous stoppage of a recirculation pump while the reactor is operating at full power. This accident is bounded by the LOCA analysis which assumes a severance of the recirculation line. No reload specific analysis is required.

Fuel Assembly Loading - The analysis of this event assumes that a reload bundle is incorrectly rotated or inserted in an improper location and the error is not discovered in the subsequent core verification. The event is classified as an accident and not an abnormal condition because two independent errors are assumed to occur. Multiple potential core configurations which could result from fuel loading errors are analyzed for each reload core design using a steady-state 3-D coupled nuclear-thermal-hydraulic BWR core simulator. The ΔCPRs resulting from the limiting rotated and improperly located fuel bundles are reported in the reload licensing submittal. The analysis of this event is discussed in detail in Section 5 of this report.

6.6 Analysis Description

6.6.1 MCPR Operating Limit Calculational Procedures

A plant unique MCPR operating limit is evaluated for each reload cycle to ensure that the fuel cladding integrity safety limit is not exceeded during reactor operation. The operating limit CPR (OLCPR) is determined by the addition of the ΔCPR (including any adjustment factors) for the limiting abnormal operational transient to the fuel safety limit CPR (SLCPR). The SLCPR is determined by a statistical combination of uncertainties to ensure that 99.9 percent of the fuel rods are expected to avoid boiling transition.

For each reload cycle, the ΔCPR is calculated for a group of transients that have been identified as bounding for all postulated abnormal operational transients. Rapid core-wide events are analyzed using the RETRAN (reference 4) computer code. Localized and/or slower events are analyzed with a series of steady state coupled nuclear-thermal-hydraulic calculations using the 3-D simulator code (reference 1).

Section 6.5.1 identified the events which are bounding for abnormal operational transients. The loss of feedwater heaters (LFWH) and rod withdrawal error (RWE) transients are analyzed using the 3-D simulator code as described in section 5. With the exception of the stability analysis, all other abnormal operational transients are evaluated using RETRAN.

RETRAN is a computer code developed to describe the behavior of light-water reactor systems during postulated accidents and operational transients. Reference 4 provides a description of the theory, coding, input, and applications of the code. RETRAN includes proven thermal-hydraulic models obtained from previous codes, as well as improved models from current codes and RETRAN development work. Of particular importance in the analyses of BWR pressurization transients are the models added for 1-dimensional kinetics and nonhomogeneous (slip) flow. The code allows the user to develop models with varying levels of geometric detail depending on the intent of the analysis.

The RETRAN system model used to analyze abnormal operational transients for reload submittals is described in reference 5. Verification of the model is also presented in the referenced report. This model evolved over several years of BWR transient analysis and is capable of simulating a wide range of NSSS disturbances. Major features of the model are:

1. One-dimensional core kinetics model with 26 axial regions.

2. A multi-volume steam line model capable of predicting wave phenomena.

3. Nonhomogenous flow (slip) is assumed.

4. The average water density in the core is calculated in 14 axial volumes.

5. The effects of subcooled boiling on the core neutronics are modeled.

6. The core thermal-hydraulic response is matched to the results from more detailed thermal-hydraulic codes.

7. The core bypass region and bypass flow paths are modeled.

8. Heat transfer to the coolant and fuel temperature are calculated at each core control volume.

9. Non-equilibrium effects in the reactor downcomer region are taken into account.

10. Recirculation loops are modeled. Recirculation and jet pump performance are matched to vendor data.

11. Enthalpy transport delays in the feedwater lines, recirculation lines, and downcomer are taken into account.

12. The recirculation system (drive motor, coupler, generator) is modeled.

13. All major control systems are modeled: turbine speed and pressure control, feedwater and level controls, and recirculation speed controls.

14. The ability to simulate automatic trips (reactor protection system, safety/relief valves, pump trips, etc.) is included.

Because RETRAN is a "best estimate" computer code, sufficient conservatism must be applied to the input data supplied to the code or to its results. To accomplish this, trip setpoints, circuit delays, and component performance are assumed at their adverse tolerances. The major assumptions used in the transient analyses are identified below. These assumptions form the basis from which reactor behavior is analyzed for a licensing submittal. The major assumptions are:

1. The reactor is initially operating at 104.5-percent rated power (105-percent steam flow).

2. All trip set points (for flux, pressure, level, etc.) are equal to the most adverse values specified in the Technical Specifications.

3. Neutronics data is generated at the limiting core exposure for each transient (generally EOC unless specific exposure dependent evaluations are performed).

4. The average speed of all the control rod drives following a scram is assumed to be at the plant technical specification value.

5. Circuitry delays in the reactor protection system as well as other equipment circuit delays (S/R valve opening, RPT, etc.) are assumed at the maximum specified values.

6. The set points of the safety/relief (S/R) valves are assumed to be 1-percent above the nominal set points to account for initial set point errors and any set point drift.

7. It is assumed that one S/R valve at the lowest pressure set point fails to open.

8. Equipment performance such as S/R valve opening characteristics, recirculation pump drive train inertia, MSIV closure times, etc., are all assumed to be at their adverse tolerances.

9. All plant control systems (unless otherwise specified) continue normal operation.

10. The reactor is operating in the manual flow-control mode. In automatic control, some reduction in core power results from runback of the recirculation pump.

The ΔCPR for each transient is calculated using a RETRAN "hot-channel" model and the overall reactor response obtained from the system model. The hot-channel model represents a single fuel assembly located between the upper and lower plenums. A more detailed description is available in reference 5. The results from the system model are used to provide time dependent boundary conditions (upper and lower plenum conditions and normalized power) to the hot-channel model. This model is then used to evaluate the local fluid conditions in a single fuel assembly during a transient.

The results from the RETRAN hot-channel model are used to calculate the CPR during a transient. The GEXL correlation and assembly flow, power, enthalpy distribution, and pressure time histories (from hot-channel results) are used to calculate the transient CPR. The CPR for each transient is determined by subtracting the minimum CPR during the transient from the initial CPR.

The initial step in determining the operating limit CPR (OLCPR) is to evaluate the ΔCPR for the transients that are normally limiting. The calculation of the ΔCPR involves an iterative process. First, a bundle power is assumed and a hot-channel analysis is performed. The hot-channel results are then used to predict the CPR during the transient. If the minimum CPR predicted is equal to the safety limit CPR (SLCPR), the ΔCPR is calculated using the results of this iteration. If the minimum CPR calculated is greater than the SLCPR, the bundle power assumed is increased and the hot-channel analyses is repeated. The reverse is true if the minimum CPR predicted is less than the SLCPR. Because a change in initial CPR of 0.05 results in a change of about .01 in ΔCPR, it is not necessary to continue the iteration if the minimum CPR calculated is within .02 of the SLCPR.

The Technical Specification OLCPR is determined by adjusting the deterministic ΔCPR value resulting from RETRAN evaluations of core-wide transients such that the desired protection probability and confidence level are attained relative to the safety limit. The basis for the adjustment factors and the method of application are discussed in chapter 7 (to be submitted to NRC in July 1982) of reference 5. The OLCPR is obtained by adding the maximum adjusted ΔCPR to the SLCPR.

The operating limit MCPR must be increased at low flow conditions. This is because, in the BWR, power increases as core flow increases, which results in a corresponding lower MCPR. If the MCPR at a reduced flow condition were at the 100 percent power and flow MCPR operating limit, a sufficiently large inadvertent flow increase could cause the MCPR to decrease below the Safety Limit MCPR. Therefore, the required operating limit MCPR is increased at reduced core flow rates by a flow factor, K_f, such that:

$$\text{Required MCPR Operating Limit} = K_f * \text{MCPR Operating Limit @ 100\% core flow}$$

The appropriate K_f factors are documented in the plant Technical Specifications and will be amended for reload cores if required.

6.6.2 Rod Drop Analysis Procedure

TVA is currently developing rod drop analysis codes, methods, and procedures to be submitted for NRC review. Until these analyses are developed and approved for licensing applications, TVA will purchase any required cycle specific evaluations from the fuel vendor.

6.6.3 Stability Analysis Procedure

TVA is currently developing BWR stability analysis codes, methods, and procedures to be submitted for NRC review. Until these analyses are developed and approved for licensing applications, TVA will purchase cycle specific stability evaluations from the fuel vendor.

6.7 Analysis Output

The following table indicates the major output items resulting from the Safety Analysis and Transient Evaluations of BWR reload cores.

References

1. S. L. Forkner, G. H. Meriwether, and T. D. Beu, "Three-Dimensional LWR Core Simulation Methods," TVA-TR78-03A, 1978.

2. NEDE-24011-P-A-1, "Generic Reload Fuel Application," General Electric Company Licensing Topical Report, July 1979.

3. B. L. Darnell, T. D. Beu, and G. W. Perry, "Methods for the Lattice Physics Analysis of LWR's," TVA-TR78-02A, 1978.

4. McFadden, et al, "RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems," EPRI NP-1850-CCM, May 1981

5. Forkner, et al, "BWR Transient Analysis Model Utilizing the RETRAN Program," TVA-TR81-01, December 1981.

6. TVA-TR79-01A, Verification of TVA Steady-State BWR Physics Methods, January 1979.

Analysis Output Item	Primary Documentation
1. Reload Unique Transient Initial Conditions	Reload Licensing Submittal
2. Transient Analysis Results - Limiting Events	Reload Licensing Submittal-Operating Limit Determination
3. MCPR Operating Limit	Reload Licensing Submittal-Technical Specifications
4. Over pressurization Summary	Reload Licensing Submittal
5. Stability Results	Reload Licensing Submittal
6. Control Rod Drop Analysis Results	Reload Licensing Submittal

Glossary of Terms

BOC - beginning of cycle
Cycle N - reload fuel cycle being designed
CSDM - cold shutdown margin
EOC - end of cycle
FLE - fuel loading error
FWCF - feedwater controller failure
GLRWOBP - generator load rejection without bypass
IHPCI - inadvertent high pressure coolant injection
LCO - limiting condition for operation
LFWH - loss of feedwater heaters
LOCA - loss of coolant accident
MLHGR - maximum linear heat generation rate (kW/ft)
MCPR - minimum critical power ratio
MAPLHGR - maximum average planar linear heat generation rate (kW/ft)
MSIVC - main steamline isolation valve closure
OLCPR - operating limit critical power ratio
RCDA - reload core design and analysis
RDA - rod drop accident
RWE - rod withdrawal error
RBM - rod block monitor
SLCS - standby liquid control system
S/R - safety/relief
SLCPR - safety limit critical power ratio