ATTACHMENT 7 (105% OLTP)

Browns Ferry Nuclear Plant (BFN)
Unit 1

Technical Specifications (TS) Change 467-S

Revision of Technical Specifications to allow utilization of AREVA NP fuel and associated analysis methodologies

Mechanical Design Report

Attached is the non-proprietary version of the mechanical design report for 105% OLTP conditions.

An AREVA and Siemens company

ANP-2877NP
Revision 0

Mechanical Design Report for Browns Ferry Unit 1 Reload BFE1-9 ATRIUM-10 Fuel Assemblies (105% OLTP)

November 2009

AREVA

Mechanical Design Report for Browns Ferry Unit 1 Reload BFE1-9 ATRIUM-10 Fuel Assemblies (105% OLTP)

Nature of Changes

Item	Section(s) or Page(s)	Description of Change
1.	All	This is a new document.

Contents

Appendices

List of Tables

List of Figures

Total number of pages: 61

Nomenclature

Acronym	Definition
AOO	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BOL	Beginning of Life
BWR	Boiling Water Reactor
cal/g	Calorie per Gram
CFR	Code of Federal Regulations
CUF	Cumulative Usage Factor
EOL	End of Life
EPU	Extended Power Uprate
FDL	Fuel Design Limit
HALC	Harmonized Advanced Load Chain
ID	Inside Diameter
kW/ft	Kilowatt per Foot
LHGR	Linear Heat Generation Rate
LOCA	Loss-of-Coolant Accident
LTP	Lower Tie Plate
Mlbm/hr	Megapound-mass per Hour
MWd/kgU	Megawatt-Days per kilogram of Uranium
MWt	Megawatt-thermal
NAF	Neutron Absorber Fuel
NRC	U. S. Nuclear Regulatory Commission
OD	Outside Diameter
P	Primary stress intensity
PAPT	Protection Against Power Transients
P_b	Primary bending stress intensity
PCI	Pellet/Cladding Interaction
PLFR	Part-Length Fuel Rods
P_m	Primary membrane stress intensity
ppm	Parts per Million
psi	Pounds per Square Inch

Nomenclature (Continued)

Acronym	Definition
psia	Pounds per Square Inch Absolute
Q	Secondary stress intensity
S_m	Design stress intensity
SRP	Standard Review Plan
S_u	Ultimate stress
S_y	Yield stress
UCL	Upper Confidence Level
UTL	Upper Tolerance Limit
UTP	Upper Tie Plate

1.0 Introduction and Summary

This report provides a design description, mechanical design criteria, fuel mechanical analysis results, and test results for the fuel assembly and fuel channel design supplied by AREVA NP Inc. (AREVA) for Reload BFE1-9.

The AREVA ATRIUM-10 fuel assembly mechanical design is the same as that delivered for Browns Ferry Unit 3 Cycle 15.

ATRIUM-10 design features carried over from the Unit 3 cycle 15 reload include the use of the chamfered fuel pellet design, liner fuel rod cladding, Zircaloy-4 fuel channels, and the modified Harmonized Advanced Load Chain (HALC) upper tie plate. [

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Although Browns Ferry Unit 1 is not currently operating at Extended Power Uprate (EPU) condition, AREVA fuel supplied to Browns Ferry was previously analyzed and reported for EPU conditions. The EPU analyses are conservative for non-EPU conditions as well.

The fuel assembly design was evaluated according to the AREVA boiling water reactor (BWR) generic mechanical design criteria (Reference 1). The generic design criteria have been approved by the U.S. Nuclear Regulatory Commission (NRC) and the criteria are applicable to the subject design. The fuel channel design was evaluated to the criteria given in fuel channel topical report (Reference 2).

Mechanical analyses have been performed using NRC-approved design analysis methodology (References 3, 4, 5, 6, and 7). The methodology permits maximum licensed assembly and rod exposures of [ ] respectively. The analyses presented in this report evaluate the following maximum discharge exposures:

- [ ]
- [ ]

The analyses demonstrate that the mechanical criteria applicable to the design are satisfied when the fuel is operated at or below the linear heat generation rate (LHGR) limits presented in Figure 1.1 for normal operation and anticipated operational occurrences (AOOs).

The fuel assembly meets all mechanical compatibility requirements for use in Browns Ferry Unit 1. This includes compatibility with both co-resident fuel and the reactor core internals.

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Figure 1.1 LHGR Limits for Normal Operation and AOO

2.0 Design Description

The following sections describe the fuel assembly and fuel channel supplied by AREVA.

2.1 Fuel Assembly

The ATRIUM*-10 fuel assembly consists of a lower tie plate (LTP) and upper tie plate (UTP), 91 fuel rods, 8 spacer grids, a central water channel, and miscellaneous assembly hardware. Of the 91 fuel rods, 8 are part-length fuel rods (PLFRs). The structural members of the fuel assembly include the tie plates, spacer grids, water channel, and connecting hardware. The structural connection between the LTP and UTP is provided by the water channel. Seven spacers occupy the normal axial locations, while an eighth spacer is located just above the LTP to restrain the lower ends of the fuel rods.

The fuel assembly is accompanied by a fuel channel, as described later in this section.

Table 2.1 lists the main fuel assembly attributes and the appendix contains an illustration of the fuel bundle assembly.

2.1.1 Spacer Grid

The spacer grid is the ULTRAFLOW† design. It is a square grid of intersecting Zircaloy-4 strips with nickel alloy 718 springs. Within each cell, there are two springs and two opposing supports. A larger cell for the water channel is located one cell spacing off-center in a diagonal direction away from the control blade corner. Small swirl vanes are situated on the top edges of each cell and along the peripheral side strips of the grid. The upper edges of the four outer side strips also have lead-in tabs to reduce the possibility of interference during fuel channel installation.

Table 2.1 lists the main spacer grid attributes and the appendix provides an illustration of the spacer grid.

2.1.2 Water Channel

The water channel is made in the shape of a square duct with rounded corners from Zircaloy-4 sheet. Zircaloy-4 end fittings are welded to the upper and lower end. Inlet and outlet holes allow water to flow through the water channel.

*ATRIUM is a trademark of AREVA NP Inc.
†ULTRAFLOW is a trademark of AREVA NP Inc.

The structural tie between the LTP and UTP is provided by the water channel assembly. A connecting bolt extends from the water channel upper end fitting up to the UTP locking hardware. The LTP is secured to the water channel lower end fitting by a cap screw. Large cross-sectional threaded fasteners and connecting hardware ensure a strong connection between the two tie plates.

Table 2.1 lists the main water channel attributes and the appendix provides an illustration of a section of the water channel.

2.1.3 Lower Tie Plate

The diffuser box of the FUELGUARD‡ is cast of low carbon stainless steel and then machined. A support grid consisting of a geometrically captured matrix of curved blades and grid rods is brazed into the diffuser box prior to final machining. The FUELGUARD grid is designed to increase the debris filtering capability. Bushings are provided in the grid for the attachment of the water channel and PLFRs, but the LTP does not provide for lateral restraint of the full-length fuel rods. Instead, the full-length fuel rods rest on top of the grid rods and the lateral restraint is provided by the lowermost spacer grid.

Nickel alloy 718 seal springs are attached to each of the four sides to limit the bypass flow outside of the fuel channel.

Table 2.1 lists the main LTP attributes and the appendix provides an illustration of the LTP.

2.1.4 Upper Tie Plate and Connecting Hardware

The UTP is cast of low carbon stainless steel and then machined. Round bosses and connective webbing form a grid structure for lateral support of the fuel rod upper end caps. An integral cast bail handle connects with the grid at opposite corner posts to lift the assembly.

A large round boss is located in the central part of the UTP grid for attaching the UTP to the assembly. Through this boss, the UTP is secured to the upper end of the water channel with a latching mechanism at the top of the connecting bolt.

The Harmonized Advanced Load Chain (HALC) modifications have improved the UTP connection by making it simpler and more robust. A compression nut on the end of the connecting bolt is used to retain the locking lug, locking ring, and locking spring. To remove the UTP, the compression spring only needs to be depressed enough to unseat the locking lug. The locking ring and locking spring must then be compressed to rotate the locking lug to align and

‡FUELGUARD is a trademark of AREVA NP Inc.

engage with the locking ring. The UTP can then be removed. Installation is accomplished in the reverse manner.

Table 2.1 lists the main UTP attributes and the appendix provides an illustration of the UTP and locking components.

2.1.5 Fuel Rods

The fuel rods are made with Zircaloy-2 [ ]. The cladding has an inner liner [ ]. The rods contain fuel pellets composed of sintered UO_2 or UO_2-Gd_2O_3. A stainless steel plenum spring on the upper end of the fuel column helps in maintaining a compact fuel column during shipment and initial reactor operation.

Zircaloy-2 end caps are welded on both ends. [ ].

The PLFRs have a special lower end fitting that engages in bushings in the LTP grid. This is done to keep the PLFRs in the proper axial location in the fuel assembly with the lower ends engaged in the lowermost spacer grid. A lower plenum is included in the rod to minimize the amount of PLFR upper plenum volume in the active region of the fuel. A small Zircaloy open-ended tube is positioned at the lower end of the column to support the fuel pellets and provide a space for the lower plenum.

Features are included on the upper and lower ends of the PLFRs to allow for remote removal and installation in the case of fuel surveillance or repair. To remove a PLFR, it is necessary to grapple the upper end and pull on the rod with a moderate amount of force to disengage the lower end fitting from the LTP grid.

Table 2.1 lists the main fuel rod attributes. The appendix provides illustrations of the full-length fuel rod and the PLFR.

2.2 Fuel Channel and Components

The fuel channel is a square duct with rounded corners, and it is open at both ends. It encloses the sides of each fuel assembly for the main purpose of providing a flow boundary between the active coolant flow and the core bypass flow. The fuel channel also lends considerable stiffness to the channeled fuel assembly and provides a bearing surface for the guidance of the control blade during movement. Gussets are welded at two opposite corners of the top end of the fuel channel for support and attachment to the fuel assembly. The fuel channel outer geometry is designed for compatibility with the control blade.

The fuel channel assembly also includes channel spacers and channel fasteners. The channel spacers and fasteners are designed to maintain proper spacing of the assemblies in the core.

Table 2.2 lists the fuel channel component attributes. The appendix provides illustrations of the fuel channel with the channel spacers installed and of the fuel channel fastener.

Table 2.1 Fuel Assembly and Component Description

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Table 2.1 Fuel Assembly and Component Description (continued)

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Table 2.2 Fuel Channel and Fastener Description

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3.0 Fuel Design Evaluation

A summary of the mechanical methodology and results from the design evaluations is provided in this section. Results from the mechanical design evaluation demonstrate that the design satisfies the mechanical criteria to the analyzed exposure and LHGR limits.

3.1 Reactor Conditions

The reactor operating conditions and duty cycles covered by the mechanical evaluations are provided in Table 3.1 and Table 3.2.

Design power histories are used as input to RODEX2A for the fuel rod analyses. The UO_2 fuel rod power history is shown in Figure 3.1. This power history was derived from the normal operating LHGR limit (Figure 1.1) following the methods described in References 1 and 5.

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Note that the fuel rod analyses are limited by the rod average exposure. The LHGR limit can be linearly extrapolated beyond the highest point shown in the curve in Figure 1.1 provided the rod average exposure limit is observed.

3.2 Fuel Rod Evaluation Summary

The results from the analyses are listed in Table 3.3. Summaries of the methods and codes used in the evaluation are provided in the following paragraphs. The design criteria are also listed, along with references to the appropriate sections of the design criteria topical reports (References 1 and 2). Details of the methodology can be found by consulting the referenced documents.

The fuel rod mechanical design criteria are summarized below:

- **Internal Hydriding.** The fabrication limit for total hydrogen in the fuel pellets is less than or equal to 2.0 ppm to preclude cladding failure caused by internal sources of hydrogen.
- **Cladding Collapse.** Creep collapse of the cladding and subsequent potential failure is avoided by eliminating the formation of axial gaps. The pellet/clad gap is evaluated [ ] to ensure the cladding does not [ ].
- **Overheating of Cladding.** The design basis requires that 99.9% of the fuel rods do not experience transition boiling. Prevention of potential fuel failure from overheating of the cladding is accomplished by minimizing the probability of exceeding thermal margin limits on limiting fuel rods during normal operation and AOO.
- **Overheating of Fuel Pellets.** The fuel centerline temperature must remain below melting during normal operation and AOO. The melting point of the fuel includes adjustments for burnup and gadolinia content.
- **Stress and Strain Limits.** The uniform cladding strain during a transient must be less than 1%. For pellet exposures greater than [ ], the transient strain limit is reduced to [ ]. As a related criterion, fuel melting is not allowed during normal operation and AOO. In addition, the steady-state cladding creep strain shall not exceed 1%. Cladding stresses are restricted to satisfy the limits established from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code.
- **Cladding Rupture.** Fuel failures from cladding rupture must not be underestimated when analyzing a loss-of-coolant accident (LOCA).
- **Fuel Rod Mechanical Fracturing.** Fuel rod cladding stresses are limited to satisfy the ASME B&PV Code, Section III, Division 1, Appendix F criteria for faulted conditions.
- **Fuel Densification and Swelling.** There are no specific limits on combined fuel densification and swelling. Instead, the densification and swelling are shown to be acceptable in the thermal-mechanical analysis of fuel temperature, cladding strain, and rod internal pressure.

3.2.1 Internal Hydriding

Internal hydriding is prevented by careful control of moisture and other hydrogenous impurities during fuel fabrication. A fabrication limit of less than or equal to 2.0 ppm hydrogen is specified for fuel pellet fabrication. The fuel pellets are tested on a routine basis during fabrication to ensure acceptable hydrogen levels in the fuel.

3.2.2 *Cladding Collapse*

Cladding collapse is evaluated using the RODEX2A and COLAPX codes (References 6 and 3). The analysis demonstrates that [

]. RODEX2A is used to calculate the uniform creepdown of the cladding and provide initial conditions to COLAPX. The COLAPX code calculates the ovalization of the cladding under the influence of external pressure, fast neutron flux, and temperature. The gap conditions are evaluated after the first [
]. The methodology for the analysis is described in References 3 and 8.

The results show positive gap in compliance with the design criteria.

3.2.3 *Overheating of Cladding*

This evaluation is covered separate from this report.

3.2.4 *Overheating of Fuel Pellets*

Fuel centerline temperature is evaluated using the RODEX2A code (Reference 6) for both normal operating conditions and AOOs. The design power history is used as input for calculating the normal operating temperatures (see Section 3.1 for the power history description). For AOOs, the fuel temperatures are calculated using the same design power history, except that additional calculations are performed at elevated power levels as a function of exposure corresponding with the Protection Against Power Transients (PAPT) LHGR limit (see Figure 1.1). Adjustments to the fuel melt temperature are made for exposure and gadolinia content.

Results of the fuel temperature analysis are provided in Figure 3.2, Figure 3.3, and Figure 3.4 for the urania, gadolinia, and PLFR, respectively. The calculated maximum fuel temperature remains below the fuel melt temperature.

3.2.5 *Stress and Strain*

3.2.5.1 Pellet-Cladding Interaction (PCI)

Cladding strain caused by transient-induced deformations of the cladding is calculated using the RODEX2 and RAMPEX codes. The design power history (see Section 3.1) is analyzed using RODEX2 to provide initial ramping conditions for the RAMPEX code. Ramps are prescribed in

RAMPEX up to the PAPT LHGR overpower limit to evaluate cladding strain. Conservative design inputs are selected according to the methodology described in Reference 8.

In addition to the transient strain analysis, a steady-state creep strain analysis is performed using RODEX2A. The design power history is used along with conservative inputs for fuel parameters.

The results are summarized in Table 3.3 and Figure 3.5. In both analyses, the cladding strain satisfies the strain limit of 1%. In addition, for pellet exposures [ ], the cladding transient strain is less than [ ] (Reference 7). The transient strain results satisfy the criteria.

3.2.5.2 Cladding Stress

Cladding stresses are calculated using solid mechanics elasticity solutions and finite element methods. The stresses are conservatively calculated for the individual loadings and are categorized as follows:

Cladding Steady-State Stresses - Loads and Categories

Category	Membrane	Bending
Primary	Differential pressure (uniform hoop, axial, and radial stresses)	Differential pressure and ovality Flow-induced vibration
Secondary	Differential thermal expansion Steady-state PCI	Restraint against mechanical and thermal bow Steady-state PCI Spacer contact stresses

Stresses are calculated at beginning of life (BOL) and at end of life (EOL), at the cladding outer and inner diameter in the three principal directions. At EOL, the stresses due to mechanical bow and contact stress are assumed to decrease to lower levels due to irradiation relaxation. The separate stress components are then combined, and the stress intensities for each category are compared to their respective limits.

The end cap stresses are evaluated for loadings from differential pressure, differential thermal expansion, rod weight, and plenum spring force.

The design limits are based on the ASME B&PV Code (Reference 9) and the minimum specified material tensile strength properties.

Table 3.3 contains the results for comparison with the design criteria.

3.2.6 *Cladding Rupture*

This evaluation is covered separate from this report.

3.2.7 *Fuel Rod Mechanical Fracturing*

See Section 3.4.4 for the evaluation of fuel rod mechanical fracturing.

3.2.8 *Fuel Densification and Swelling*

Fuel densification and swelling are limited by the design criteria for fuel temperature, cladding strain, cladding collapse, and internal rod pressure criteria. Although there are no explicit criteria for fuel densification and swelling, the effect of these phenomena are included in the RODEX2A and RODEX fuel rod performance codes. See the other applicable sections of this report for the design evaluation.

3.3 Fuel Assembly Evaluation Summary

The fuel system mechanical design criteria are summarized below (Reference 1):

- **Stress, Strain, or Loading Limits on Assembly Components.** The structural integrity of fuel assembly components is assured by establishing limits on stresses and deformations due to handling, operational, and accident loads. Load and stress limits, as applicable, are derived from the ASME B&PV Code Section III. In addition, the loadings on components are evaluated for compliance with fuel handling and structural deformation (postulated accident) criteria.

- **Fatigue.** The criteria limit the cladding cyclic fatigue for significant cyclic loads to be less than **[ ]**. The fatigue life includes a factor of safety of two on the stress level or a factor of safety of 20 on the number of cycles, whichever is more conservative.

- **Fretting Wear.** The design basis for fretting wear is that fuel rod failures due to fretting shall not occur. There is no specific wear limit. The acceptability of fretting resistance is verified **[ ]**.

- **Oxidation, Hydriding, and Crud Buildup.** There is no specific limit on oxide thickness. AREVA BWR poolside measurement data indicate that the amount of oxidation is low. The effect of oxidation is included in the fuel rod thermal analyses and the cladding stress analysis.

- **Rod Bow.** The maximum rod closure, as calculated by the approved AREVA bow model, shall not impact thermal margins.
- **Axial Irradiation Growth.** Fuel assembly components, including the fuel channel, shall maintain clearances and engagements, as appropriate, throughout the design life.
- **Rod Internal Pressure.** The rod internal pressure is limited to [ ].
- **Assembly Liftoff.** The fuel shall not levitate under normal operating or AOO conditions. Under postulated accident conditions, the fuel shall not become disengaged from the fuel support. These criteria assure control blade insertion is not impaired.
- **Fuel Assembly Handling.** The fuel assembly shall withstand, without permanent deformation, all normal axial loads from shipping and fuel handling operations. [ ].

  During handling, the plenum spring shall maintain a force against the fuel column stack to prevent column movement.
- **Miscellaneous Component Criteria.** The compression spring must support the weight of the UTP and fuel channel throughout the design life.

  The LTP seal spring shall limit the bypass coolant leakage rate between the LTP and fuel channel. The seal spring shall accommodate expected channel deformation while remaining in contact with the fuel channel. Also, the seal spring shall have adequate corrosion resistance and be able to withstand the operating stresses without yielding.

3.3.1 Stress, Strain or Loads on Assembly Components

The fuel assembly components are evaluated for structural integrity by the evaluation of significant loads experienced by the components during normal operation, AOOs, and under faulted conditions. Those components that are subjected to significant loads during normal operation include the fuel rod cladding, water channel, LTP, UTP, and tie plate connecting hardware. For faulted conditions, the major structural components of the fuel assembly (spacer grids, water channel, tie plates) and the fuel rods undergo additional loading.

3.3.1.1 Normal Operation and AOOs

During normal operation (and AOOs), the fuel rod cladding experiences the greatest relative stress and strain. The water channel is subjected to differential pressure loads. For the structural components, fuel handling produces the highest loads for the water channel, tie plates, and tie plate connecting hardware. The fuel rod cladding stress and strain are addressed in Section 3.2.5. Fuel handling loads are covered under Section 3.3.9.

Water channel stresses during normal operation are calculated using either conventional elasticity theory or the finite element method. The primary loading comes from the differential pressure. A secondary load occurs as a result of differential thermal expansion between the water channel and fuel rods. Stresses are calculated in the cross-section of the channel and at the welded connections to the end fittings. Stress limits are derived from the ASME B&PV Code (Reference 9) using the minimum-specified tensile properties of the water channel material.

See Table 3.3 for results from the component strength evaluations.

3.3.1.2 Loads During Postulated Accidents

Component integrity during faulted conditions is described in Section 3.4.4.

3.3.2 Fatigue

Fuel rod cladding fatigue is calculated using the RODEX2 and RAMPEX codes. The design power history (see Section 3.1) is analyzed using RODEX2 to provide initial, steady-state ramping conditions for the RAMPEX code. Conservative design inputs are selected according to the methodology described in Reference 8. Ramps are prescribed in RAMPEX corresponding to the power changes listed in Table 3.2. For each duty cycle (i.e., type of power change), cladding cyclic stresses are obtained from RAMPEX output. Corresponding to each duty cycle, the allowable number of cycles (N_{allow}) is computed using a fatigue S-N curve for Zircaloy (Reference 10). The fatigue design curve includes a factor of safety of 2 on stress or 20 on the number of cycles, whichever is more conservative. The fatigue curve also includes a correction for the effect of the maximum mean stress. Table 3.2 lists the number of analyzed power changes n_{design} for the various design duty cycles. For each duty cycle, the resulting fatigue is calculated as n_{design} divided by N_{allow}. The total cumulative fatigue usage is calculated by summing the individual contributions of each duty cycle using Miner's rule:

$$\mathrm{CUF} = \sum \frac{n_{design}}{N_{allow}}$$

The fuel rod cladding cumulative usage results are reported in Table 3.3.

3.3.3 Fretting Wear

Fretting wear is evaluated by testing, as described in Section 4.4. [

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3.3.4 Oxidation, Hydriding and Crud Buildup

RODEX2A is used to calculate fuel rod cladding external oxidation. AREVA BWR fuel rod cladding oxidation data show the amount of oxidation to be low (Reference 7). Because of the low amount of corrosion, there is [ ]. The maximum amount of wall thinning is taken into account in the cladding steady-state stress analysis for EOL conditions.

The effect of corrosion is required to be included in the fuel temperature analysis (Section 3.2.4) through the use of an enhancement to the corrosion model. The additional enhancement is an input to the RODEX2A corrosion model such that the EOL oxidation is nearly equivalent to the maximum amount of oxidation observed on AREVA BWR fuel rod cladding.

The steady-state stress and fuel temperature results are reported in Table 3.3.

3.3.5 Rod Bow

Rod bow is calculated using the approved model described in Reference 11. Based on cladding geometry, span length and rod support conditions, the model applies equally well to the ATRIUM-10 design as to the AREVA 9x9 and 8x8 designs. Rod closure measurement data taken on lead fuel assemblies validate the application of the rod bow model to the ATRIUM-10 design. The rod closure due to rod bow is assessed for impact on thermal margins in a separate report.

3.3.6 Axial Irradiation Growth

Three growth calculations are considered for the ATRIUM-10 design: (1) minimum fuel rod clearance between the LTP and UTP, (2) minimum engagement of the fuel channel with the LTP seal spring, and (3) external channel engagement (e.g., channel fastener springs). Rod growth, assembly growth, and fuel channel growth are calculated using correlations derived from growth data. The evaluation of initial engagements and clearances accounts for the stackup of fabrication tolerances on individual component dimensions.

The rod growth correlation was established from 7x7, 8x8, and 9x9 data. Additional comparisons have been made to the 10x10 data to demonstrate the applicability of the growth correlation to the ATRIUM-10 design. The maximum rod growth is predicted using the 95/95 UTL of the data at the EOL exposure and fluence level.

Assembly growth is dictated by the water channel growth. The growth of the water channel and the fuel channel is based on fuel channel growth data. These data and the correlation of growth are described in Reference 7. The minimum and maximum 95/95 tolerance limits of the growth, as appropriate, are used to obtain EOL growth values.

To calculate the minimum fuel rod clearance at EOL, the initial minimum clearance between the fuel rod and tie plates is calculated from the stackup of fabrication dimensions and tolerances. The maximum differential growth is then calculated as the maximum fuel rod growth minus the minimum fuel assembly (water channel) growth. The EOL minimum clearance is obtained by subtracting the maximum differential growth from the initial minimum clearance.

Fuel channel and LTP seal spring engagement is calculated in a similar manner as for the fuel rod clearance. Minimum overlap is based on a stackup of component fabrication dimensions and tolerances. The EOL engagement is calculated by subtracting the differential growth between the fuel assembly (water channel) and the fuel channel from the initial overlap.

The channel fastener springs must engage with the springs on adjacent fuel assemblies through EOL. This includes the situation of placing fresh fuel adjacent to co-resident fuel in its last cycle of operation. Again, manufacturing tolerances and maximum growth variations are considered in the evaluation.

The minimum EOL rod growth clearance and EOL fuel channel engagement with the seal spring are listed in Table 3.3. The channel fastener spring axial compatibility is reported in Table 3.5.

3.3.7 Rod Internal Pressure

Fuel rod internal pressure is calculated using the RODEX2A code. The design power history is used as input (see Section 3.1) along with the fuel rod input parameter methodology as described in References 3 and 8. An additional factor is applied to the power inputs to account for power uncertainty. The maximum [ ] is included in the analysis.

In addition to evaluating the maximum rod pressure, [

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The results are listed in Table 3.3. Figure 3.6 shows the calculated rod internal pressure as a function of rod exposure.

3.3.8 Assembly Liftoff

Fuel assembly liftoff is calculated under both normal operating conditions (including AOOs) and under faulted conditions. For normal operating conditions, the net axial force acting on the fuel assembly is calculated by summing the loads from gravity, hydraulic resistance from coolant flow, difference in fluid flow entrance and exit momentum, and buoyancy. The calculated net force is confirmed to be in the downward direction, indicating no liftoff. Maximum hot channel conditions are used in the calculation because the greater two-phase flow losses produce a higher uplift force.

Mixed core conditions for liftoff are considered on a specific basis as determined by the plant and the other fuel types. Analyses to date indicate a large margin to liftoff under normal operating conditions. Fuel liftoff in BWRs under normal operating conditions is, therefore, considered to be a small concern.

Liftoff under faulted conditions is described in Section 3.4.4.

3.3.9 Fuel Assembly Handling

The fuel assembly structural components are assessed for axial fuel handling loads by testing. To demonstrate compliance with the criteria, the test is performed by loading a test assembly **[**

]. The testing is described further in Section 4.1.

Also, the plenum spring must not allow the fuel column to shift as a result of the maximum axial handling load. This spring force requirement is demonstrated through a combination of design calculations and testing.

3.3.10 Miscellaneous Components

3.3.10.1 Compression Spring Forces

The ATRIUM-10 has a single large compression spring mounted on the central water channel. The compression spring serves the same function as for previous designs by providing support for the UTP and fuel channel. The spring force is calculated based on the deflection and specified spring force requirements. Irradiation-induced relaxation is taken into account for EOL conditions. The minimum compression spring force at EOL is shown to be greater than the combined weight of the UTP and fuel channel (including channel fastener hardware). Since the compression spring does not interact with the fuel rods, no additional consideration is required for fuel rod buckling loads.

3.3.10.2 LTP Seal Spring

The LTP seal spring is the same design as used on previous AREVA designs. Flow testing is used to confirm acceptable bypass flow characteristics. The seal spring is designed with adequate deflection to accommodate the maximum expected channel bulge while maintaining acceptable bypass flow. Nickel alloy 718 is selected as the material because of its high strength at elevated temperature and its excellent corrosion resistance. Seal spring stresses are analyzed using a finite element method.

3.4 Fuel Coolability

For accidents in which severe fuel damage might occur, core coolability and the capability to insert control blades are essential. Normal operation and AOO must remain within the thermal margin criteria. Chapter 4.2 of the Standard Review Plan (SRP) provides several specific areas important to fuel coolability: Embrittlement, Violent Expulsion of Fuel, Fuel Ballooning, and Structural Deformations. The topics other than structural deformations are addressed separate from this report.

The fuel coolability design criteria are summarized below (Reference 1):

- **Cladding Embrittlement.** The requirements on cladding embrittlement are contained with the LOCA requirements in 10CFR50.46.
- **Violent Expulsion of Fuel.** For a severe reactivity-initiated accident, the radially averaged energy deposition at the highest axial location is restricted according to the guidelines contained in Regulatory Guideline 1.77.
- **Fuel Ballooning.** The effect of potential cladding ballooning on flow blockage and cladding rupture is considered in the LOCA analysis according to 10CFR50 Appendix K requirements.
- **Structural Deformations.** Deformations or stresses from postulated accidents are limited according to requirements contained in the ASME B&PV Code, Section III, Division 1, Appendix F, and SRP 4.2 Appendix A.

3.4.1 Cladding Embrittlement

This evaluation is covered separate from this report.

3.4.2 Violent Expulsion of Fuel

This evaluation is covered separate from this report.

3.4.3 Fuel Ballooning

This evaluation is covered separate from this report.

3.4.4 Structural Deformations

The methodology for analyzing the fuel under the influence of seismic/LOCA analysis loads is described in References 2, 12, and 13. Evaluations performed for the fuel under combined seismic/LOCA loadings include mechanical fracturing of the fuel rod cladding, assembly structural integrity, and fuel assembly liftoff. Restricting fuel uplift and limiting fuel channel deformation under accident conditions permit insertion of the control blades.

The ATRIUM-10 fuel assembly has been evaluated for integrity during external loading by testing and analysis. Testing is done to obtain the dynamic characteristics of the fuel assembly and spacer grids. The stiffnesses, natural frequencies and damping values derived from the tests are used as inputs for dynamic mechanical models of the fuel assembly and fuel channel. Tests are done with and without a fuel channel. In addition, the dynamic models are compared to the test results to ensure an accurate characterization of the fuel. See Section 4.0 for descriptions of testing.

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In general, the testing and analyses have shown the dynamic response of the ATRIUM-10 design to be very similar to BWR fuel designs that have the same basic channel configuration and weight. This includes the previously analyzed GNF fuel at Browns Ferry. In addition, the original or revised seismic/LOCA reactor pressure vessel analyses performed to determine maximum core accelerations, deflections, and loads will apply to the ATRIUM-10 because of the dynamic similarity with past designs. The dynamic response of the channeled ATRIUM-10 fuel assembly is primarily dependent on the fuel channel stiffness and the fuel assembly mass. Because the fuel assembly weight and channel stiffness do not vary significantly from prior AREVA fuel designs (or other co-resident fuel types), the maximum loads and deflections for the ATRIUM-10 fuel assembly will be essentially unchanged from before.

For fuel lift-off, a comparative evaluation is performed with the previously analyzed fuel design by taking into consideration differences in mass and pressure drop between the fuel types. The net force on fuel assemblies, considering contributions from gravity, buoyancy, hydraulic forces and momentum, are calculated and compared. The relative amount of uplift is assessed for the ATRIUM-10 design [

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3.5 Fuel Channel and Fastener

The fuel channel and fastener design criteria are summarized below, and evaluation results are summarized in Table 3.4 and Table 3.5. The analysis methods are described in detail in Reference 2.

3.5.1 Design Criteria for Normal Operation

Steady-State Stress Limits. The stress limits during normal operation are obtained from the ASME B&PV Code, Section III, Division 1, Subsection NG for Level A Service. The calculated stress intensities are due to the differential pressure across the channel wall. The pressure loading includes the normal operating pressure plus the increase during AOO. The unirradiated properties of the fuel channel material are used since the yield and ultimate tensile strength increase during irradiation.

As an alternative to the elastic analysis stress intensity limits, a plastic analysis may be performed as permitted by paragraph NB-3228.3 of the ASME B&PV Code.

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Fuel Channel Fatigue. Cyclic changes in power and flow during operation impose a duty loading on the fuel channel. [

]. The fatigue life is based on the O'Donnel and Langer curve (see Reference 10), which includes a factor of 2 on stress amplitude or a factor of 20 on the number of cycles, whichever is more conservative.

Corrosion and Hydrogen Concentration. Corrosion reduces the material thickness and results in less load-carrying capacity. The fuel channels have thicker walls than other components (e.g., fuel rods), and the normal amounts of oxidation and hydrogen pickup are not limiting provided: the alloy composition and impurity limits are carefully selected; the heat treatments are also carefully chosen; and the water chemistry is controlled. [

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Long-Term Creep Deformation. Changes to the geometry of the fuel channel occur due to creep deformation during the long term exposure in the reactor core environment. Overall deformation of the fuel channel occurs from a combination of bulging and bowing. Bulging of the side walls occurs because of the differential pressure across the wall. Lateral bowing of the channel is caused primarily from the neutron flux and thermal gradients. Too much deflection may prevent normal control blade maneuvers and it may increase control blade insertion time above the technical specification limits. [
].

3.5.2 Design Criteria for Accident Conditions

Fuel Channel Stresses and Limit Load. The criteria are based on the ASME B&PV Code, Section III, Appendix F, for faulted conditions (Level D Service). Component support criteria for elastic system analysis are used as defined in paragraphs F-1332.1 and F-1332.2. The unirradiated properties of the fuel channel material are used since the yield and ultimate tensile strength increase during irradiation.

Stresses are alternatively addressed by the plastic analysis collapse load criteria given in paragraph F-1332.2(b). For the plastic analysis collapse load, the permanent deformation is limited to twice the deformation the structure would undergo had the behavior been entirely elastic.

The amount of bulging remains limited to [ ].

Fuel Channel Gusset Load Rating. The strength of the fuel channel gussets is established by the load-rating method. Testing is done using a fixture that simulates an UTP. This fixture is inserted into a short section of a fuel channel. The bottom end of the channel section is restrained while the simulated tie plate fixture is pulled upwards against the gussets.

Table 3.1 Reactor Conditions Used in Analyses

Parameter	Value
Core thermal power, MWt	3952
System pressure, psia	1050
Total number of assemblies in core	764
Nominal total core flow rate, Mlbm/hr	102.5
Core inlet enthalpy, Btu/lbm	523.2
Fraction of heat from fuel rods	[ ]
Peak assembly burnup, MWd/kgU	[ ]
Peak rod burnup, MWd/kgU	[ ]

Table 3.2 Design Duty Cycles for Cyclic Fatigue Evaluation

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Table 3.3 Fuel Evaluation Results for ATRIUM-10

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Table 3.3 Fuel Evaluation Results for ATRIUM-10 (continued)

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Table 3.3 Fuel Evaluation Results for ATRIUM-10 (continued)

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Table 3.3 Fuel Evaluation Results for ATRIUM-10 (continued)

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Table 3.3 Fuel Evaluation Results for ATRIUM-10 (continued)

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Table 3.3 Fuel Evaluation Results for ATRIUM-10 (continued)

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Table 3.4 Evaluation Results for Fuel Channel

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Table 3.4 Evaluation Results for Fuel Channel (continued)

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Table 3.5 Evaluation Results for Channel Fasteners

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Figure 3.1 RODEX2A Fuel Rod Power History Input

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Figure 3.2 Calculated Fuel Centerline Temperatures, Normal Operation and AOO for UO2 Fuel Rod

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Figure 3.3 Calculated Fuel Centerline Temperatures, Normal Operation and AOO for Gadolinia Fuel Rod

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Figure 3.4 Calculated Fuel Centerline Temperatures, Normal Operation and AOO for PLFR

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Figure 3.5 Calculated Fuel Rod Cladding Steady-State Strain

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Figure 3.6 Calculated Fuel Rod Internal Pressure

4.0 Mechanical Tests

The AREVA testing and inspection requirements are essential elements in assuring conformance to the design criteria. The component parameters either directly demonstrate compliance with the criteria or are input for the design calculations.

Testing performed to qualify the mechanical design or evaluate assembly characteristics includes:

1. Fuel assembly axial load structural strength test
2. Spacer grid lateral impact strength test
3. Tie plate lateral load strength tests and LTP axial compression test
4. Fuel assembly fretting test
5. Fuel assembly static lateral deflection test
6. Fuel assembly lateral vibration tests
7. Fuel assembly impact tests

The torsional stiffness of the fuel assembly is not measured since it is not a significant factor in either the dynamic testing or the analytical model. Summary descriptions of the tests are provided below.

4.1 Fuel Assembly Axial Load Test

An axial load test was conducted by applying an axial tensile load between the LTP grid and UTP handle of a fuel assembly cage specimen. The load was slowly applied while monitoring the load and deflection. [

]. See Reference 14 for the ALC and Reference 15 for the HALC for more information.

4.2 Spacer Grid Lateral Impact Strength Test

Spacer grid impact strength was determined by a transverse dynamic load test. A vibration machine is used for the test. The machine consists of a large horizontal table guided by linear bearings on parallel shafts. A large hydraulic cylinder, which is attached to the table, provides the input excitation. A short section of a fuel assembly is mounted on the table of the vibration machine.

[ ]. The ends of the assembly are attached to the table with brackets. Impact rails are situated on both sides of the spacer grid in the center of the assembly. The equipment and test assembly are instrumented such that forces and displacements can be recorded.

[ ].

4.3 Tie Plate Strength Tests

Three separate tests were conducted on the tie plates: (1) The UTP was loaded laterally to obtain a limit for accident conditions, (2) the LTP was subjected to an axial compressive load to simulate handling loads, and (3) the LTP was loaded laterally at the nozzle to determine a load limit for accident conditions.

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Results from the testing were adjusted, accounting for reactor operating conditions, to determine the load limits reported in Table 3.3.

4.4 Fuel Assembly Fretting Test

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4.5 Fuel Assembly Static Lateral Deflection Test

A lateral deflection test was performed to determine the fuel assembly stiffness. The stiffness is obtained by supporting the fuel assembly at the two ends in a vertical position, applying a side displacement at the central spacer location, and measuring the corresponding force. Results from this test are input to the fuel assembly structural model.

4.6 Fuel Assembly Lateral Vibration Tests

A vibration machine was used in determining the natural frequencies, damping, and spacer grid impact stiffness for a fuel assembly. Testing was performed on a full-scale fuel assembly in both air and water, and with and without a fuel channel. Two principal tests were done in the vibration machine: lateral vibration tests and a fuel assembly impact test.

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Results from the test were used as a basis for selecting fuel assembly stiffness values and damping for the structural model.

4.7 Fuel Assembly Impact Tests

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Measured impact loads were used in establishing the spacer in-grid stiffness.

5.0 References

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Appendix A Illustrations

The following table lists the fuel assembly and fuel channel component illustrations found in this section.

These illustrations are for descriptive purposes only. Please refer to the current Parts List for production dimensions.

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