ATTACHMENT 27

ANP-3385NP, Mechanical Design Report for Browns Ferry Units 1, 2 and 3 Extended Power Uprate (EPU) ATRIUM-10 Fuel Assemblies (Non-Proprietary)

Mechanical Design Report for Browns Ferry Units 1, 2 and 3 Extended Power Uprate (EPU) ATRIUM-10 Fuel Assemblies

ANP-3385NP
Revision 1

Licensing Report

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AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description of Change
1.	All	Rev. 0, This is a new document.
2.	All	Rev. 1, Updated proprietary bracketing for consistency.
3.	Section 1	Rev. 1, Added trademark footnote for FUELGUARD.
4.	Section 1	Rev. 1, Added trademark symbol to ATRIUM and FUELGUARD.

Contents

1.0 Introduction and Summary . . . 1-1

2.0 Design Description . . . 2-1
- 2.1 Fuel Assembly . . . 2-1
  - 2.1.1 Spacer Grid . . . 2-1
  - 2.1.2 Water Channel . . . 2-1
  - 2.1.3 Lower Tie Plate . . . 2-2
  - 2.1.4 Upper Tie Plate and Connecting Hardware . . . 2-2
  - 2.1.5 Fuel Rods . . . 2-3
- 2.2 Fuel Channel and Components . . . 2-3

3.0 Fuel Design Evaluation . . . 3-1
- 3.1 Reactor Conditions . . . 3-1
- 3.2 Fuel Rod Evaluation Summary . . . 3-1
  - 3.2.1 Internal Hydriding . . . 3-2
  - 3.2.2 Cladding Collapse . . . 3-3
  - 3.2.3 Overheating of Cladding . . . 3-3
  - 3.2.4 Overheating of Fuel Pellets . . . 3-3
  - 3.2.5 Stress and Strain . . . 3-3
  - 3.2.6 Cladding Rupture . . . 3-5
  - 3.2.7 Fuel Rod Mechanical Fracturing . . . 3-5
  - 3.2.8 Fuel Densificatio and Swelling . . . 3-5
- 3.3 Fuel Assembly Evaluation Summary . . . 3-5
  - 3.3.1 Stress, Strain or Loads on Assembly Components . . . 3-6
  - 3.3.2 Fatigue . . . 3-7
  - 3.3.3 Fretting Wear . . . 3-8
  - 3.3.4 Oxidation, Hydriding and Crud Buildup . . . 3-8
  - 3.3.5 Rod Bow . . . 3-8
  - 3.3.6 Axial Irradiation Growth . . . 3-9
  - 3.3.7 Rod Internal Pressure . . . 3-10
  - 3.3.8 Assembly Liftoff . . . 3-10
  - 3.3.9 Fuel Assembly Handling . . . 3-10
  - 3.3.10 Miscellaneous Components . . . 3-11
- 3.4 Fuel Coolability . . . 3-11
  - 3.4.1 Cladding Embrittlement . . . 3-12
  - 3.4.2 Violent Expulsion of Fuel . . . 3-12
  - 3.4.3 Fuel Ballooning . . . 3-12
  - 3.4.4 Structural Deformations . . . 3-12
- 3.5 Fuel Channel and Fastener . . . 3-13
  - 3.5.1 Design Criteria for Normal Operation . . . 3-13

3.5.2 Design Criteria for Accident Conditions 3-14
4.0 Mechanical Tests 4-1
4.1 Fuel Assembly Axial Load Test 4-1
4.2 Spacer Grid Lateral Impact Strength Test 4-1
4.3 Tie Plate Strength Tests 4-2
4.4 Fuel Assembly Fretting Test 4-3
4.5 Fuel Assembly Static Lateral Deflectio Test 4-3
4.6 Fuel Assembly Lateral Vibration Tests 4-3
4.7 Fuel Assembly Impact Tests 4-4
5.0 References 5-1

Appendices

A Illustrations A-1

List of Tables

2.1 Fuel Assembly and Component Description 2-5
2.2 Fuel Channel and Fastener Description 2-7

3.1 Reactor Conditions Used in Analysis 3-16
3.2 Design Duty Cycles for Cyclic Fatigue Evaluation 3-16
3.3 Fuel Evaluation Results for ATRIUM-10 3-17
3.4 Evaluation Results for Fuel Channel 3-23
3.5 Evaluation Results for Channel Fasteners 3-25

List of Figures

1.1 LHGR Limits for Normal Operation and AOO 1-2

3.1 RODEX2A Fuel Rod Power History Input 3-26
3.2 Calculated Fuel Centerline Temperatures, Normal Operation and AOO for UO2 Fuel Rod 3-27
3.3 Calculated Fuel Centerline Temperatures, Normal Operation and AOO for Gadolinia Fuel Rod 3-28
3.4 Calculated Fuel Centerline Temperatures, Normal Operation and AOO for PLFR 3-29
3.5 Calculated Fuel Rod Cladding Steady-State Strain 3-30
3.6 Calculated Fuel Rod Internal Pressure 3-31

Total number of pages: 61

Nomenclature

Acronym	Definitio
AFC	Advanced Fuel Channel
ALC	Advanced Load Chain
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	Definitio
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 Confidenc Level
UTL	Upper Tolerance Limit
UTP	Upper Tie Plate

1.0 Introduction and Summary

The purpose of this report is to show that the AREVA Inc. (AREVA) fuel mechanical design criteria are satifie at Browns Ferry EPU conditions (120 percent of the original licensed thermal power (OLTP)). This report provides a design description, mechanical design criteria, fuel structural analysis results, and test results for the ATRIUM™*-10 fuel assembly and 100/75 Advanced Fuel Channel (AFC) designs supplied by AREVA for use at Browns Ferry Units 1, 2 and 3.

ATRIUM-10 design features include the [ ] fuel pellet design, [ ] fuel rod cladding, [ ] advanced fuel channels, the [ ], upper tie plate, winged channel fastener, the improved FUELGUARD™† (IFG) lower tie plate and [ ] lower tie plate seal spring.

Many of the structural analyses of the fuel assembly are performed on a generic basis. However, the increase in core power for EPU is also associated with an increase in core pressure drop which does have an effect on some mechanical analyses. This increase in pressure specificall affects the fuel assembly liftoff analyses and the calculated stress and deformation of the fuel channel and water channel. These analyses were revisited and shown to maintain design margin.

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:

- [ ] assembly average exposure
- [ ] rod average exposure (full-length fuel rod)

The analyses demonstrate that the mechanical criteria applicable to the design are satisfie 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 Units 1, 2 and 3. This includes compatibility with both co-resident fuel and the reactor core internals.

*ATRIUM is a trademark of AREVA Inc.
†FUELGUARD is a trademark of AREVA Inc.

[

]

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, [ ] spacer grids, a central water channel, and miscellaneous assembly hardware. Of the 91 fuel rods, [ ] 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 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. [

].

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

2.1.2 Water Channel

[

]

[

]

*ULTRAFLOW is a trademark of AREVA Inc.

[

]

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 LTP FUELGUARD† [

]

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

[

]

[

]

† FUELGUARD is a trademark of AREVA Inc.

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

[

]

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 assembly also includes channel spacers and channel fasteners. [
]

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

[	
	]

(continued next page)

Table 2.1 Fuel Assembly and Component Description (continued)

[	
	]

Table 2.2 Fuel Channel and Fastener Description

[	
	]

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 satisfie 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.

[

]

[

]

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 [ ] 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 [ ]. 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.** [ ]
- **Stress and Strain Limits.** The uniform cladding strain during a transient must be less than [ ]. 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 [ ]. 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 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 Densificatio and Swelling.** There are no specifi limits on combined fuel densificatio and swelling. Instead, the densificatio 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 [ ] hydrogen is specifie 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 influenc of external pressure, fast neutron flux and temperature. The gap conditions are evaluated after the firs [
]. 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). [
].

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. [
]

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 satisfie the strain limit of [ ]. 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 finit 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	[ ]	[ ]
Secondary	[ ]	[ ]

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 specifie 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 Densificatio and Swelling

Fuel densificatio 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 densificatio 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 significan cyclic loads to be less than [ ]
- **Fretting Wear.** The design basis for fretting wear is that fuel rod failures due to fretting shall not occur. There is no specifi wear limit. The acceptability of fretting resistance is verifie by a [ ].
- **Oxidation, Hydriding, and Crud Buildup.** [ ]. 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 significan loads experienced by the components during normal operation, AOOs, and under faulted conditions. Those components that are subjected to significan 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 finit 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 [ ] to the end fitting . Stress limits are derived from the ASME B&PV Code (Reference 9) using the minimum-specifie 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, [

] 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. The testing was conducted by [

].

The inspection measurements for wear were documented. [

]

3.3.4 Oxidation, Hydriding and Crud Buildup

RODEX2A is used to calculate fuel rod cladding external oxidation. [

]

[

]

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

Because of the low amount of corrosion on fuel assembly components, [

].

3.3.5 Rod Bow

Rod bow is calculated using the approved model described in Reference 11. [

]

[

] 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 [

]. The maximum rod growth is predicted using the [ ] of the data at the EOL exposure and fluenc level.

Assembly growth is dictated by the water channel growth. The growth of the water channel and the fuel channel is based on [ ]. These data and the correlation of growth are described in Reference 7. The minimum and maximum [
], 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, [ ].

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 fl w, difference in flui fl w entrance and exit momentum, and buoyancy. The calculated net force is confi med to be in the downward direction, indicating no liftoff. Maximum hot channel conditions are used in the calculation because the greater two-phase fl w losses produce a higher uplift force.

Mixed core conditions for liftoff are considered on a specifi 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 [ ]. An acceptable test shows no yielding after loading. 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 deflectio and specifie 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 similar to previous designs used on AREVA fuel. Flow testing is used to confi m acceptable bypass fl w characteristics. The seal spring is designed with adequate deflectio to accommodate the maximum expected channel bulge while maintaining acceptable bypass fl w. [ ] 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 finit 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 specifi 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 fl w 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 influenc 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.

[

]

[

]. Table 3.3 lists the margins for the fuel assembly components at the maximum acceleration allowed for the channel design. Component load and stress limits are derived using the ASME B&PV Code, Section III, Division 1, Appendix F, and SRP Section 4.2, Appendix A. Specifie tensile properties or testing are used to establish the limits.

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 configu ation 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, deflection , 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 significantl from prior AREVA fuel designs (or other co-resident fuel types), the maximum loads and deflection for the ATRIUM-10 fuel assembly will be essentially unchanged from before.

For fuel lift-off, [

]. The uplift is limited to be less than the axial engagement such that the fuel assembly neither becomes laterally displaced nor blocks insertion of the control blade.

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.

In the case of AOOs, the amount of bulging due to yielding is limited to that value which will permit control blade movement. During normal operation, any significan permanent deformation due to yielding is precluded by restricting the maximum stresses at the inner and outer faces of the channel to be less than the yield strength.

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

]

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. [

]

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 flu and thermal gradients. Too much deflectio may prevent normal control blade maneuvers and it may increase control blade insertion time above the technical specificatio limits. The total channel deformation must not stop free movement of the control blade.

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 define 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 that value which will permit control blade insertion.

Fuel Channel Gusset Load Rating. [

]

Table 3.1 Reactor Conditions Used in Analysis

Parameter	Value
Core thermal power, MWt	3952
System pressure, psia	1050
Total number of assemblies in core	764
Nominal total core fl w 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

[	
	]

Table 3.3 Fuel Evaluation Results for ATRIUM-10

Criteria Section	Description	Criteria	Result
3.2	Fuel Rod Criteria		
3.2.1	Internal hydriding	[ ]	Verifie by QC inspection.
3.2.2	Cladding collapse	[ ]	[ ]
3.2.3	Overheating of cladding	[ ]	Covered separate from this report.
3.2.4	Overheating of fuel pellets	[ ]	[ ] See Figure 3.2, Figure 3.3 and Figure 3.4.
3.2.5	Stress and strain limits		
3.2.5.1	Pellet/cladding interaction	Cladding steady-state strain [ ]	Transient and steady-state strains meet the criteria. (See Figure 3.5 for steady-state strain results and Section 3.2.4 of this table for fuel temperature.)

NOTE: Results are presented for a bounding analysis.

(continued next page)

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

Criteria Section	Description	Criteria	Result
3.2	Fuel Rod Criteria (Continued)		
3.2.5.2	Cladding stress		BOL Cold / BOL Hot / EOL Hot
	Primary membrane stress	[ ]	[ ]
	Primary membrane + bending	[ ]	[ ]
	Primary + secondary	[ ]	[ ]
	End cap stress		
	Primary membrane + bending	[ ]	[ ]
	Primary + secondary	[ ]	[ ]
3.2.6	Cladding rupture	[ ]	Covered separate from this report.
3.2.7	Mechanical fracturing	ASME Section III, App. F	See 3.4.4 of this table.
3.2.8	Densificatio and swelling	Sections 3.2.4, 3.2.5.1 and 3.3.7 of this table.	Models included in accepted fuel performance codes.

NOTE: Results are presented for a bounding analysis.

(continued next page)

*BOL hot conditions are satisfie by a limit analysis.

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

Criteria Section	Description	Criteria	Result
3.3	Fuel System Criteria		
3.3.1	Stress, strain and loading limits on assembly components (normal operation)		(See table sections 3.3.9 for handling and 3.4.4 for accident conditions.)
	Spacer grid	Lateral load < load limit	See 3.4.4 of this table.
	Water channel		
	Channel strength	The pressure including AOO is limited to [ ] according to ASME B&PV Code, Section III. The pressure is also limited such that [ ]	[ ]
	UTPs and LTPs	[ ]	[ ]
3.3.2	Fatigue	[ ]	CUF: [ ] [ ] [ ]

NOTE: Results are presented for a bounding analysis.

(continued next page)

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

Criteria Section	Description	Criteria	Result
3.3	Fuel System Criteria (Continued)		
3.3.3	Fretting wear	[ ]	Fretting was evaluated by testing. Testing indicates [ ]
3.3.4	Oxidation, hydriding, and crud buildup	[ ]	Approved fuel rod performance code accounts for [ ].
3.3.5	Rod bow	Protect thermal limits	NRC accepted model for rod closure due to rod bow assessed for impact on thermal margins in a separate report.
3.3.6	Axial irradiation growth		
	Upper end cap clearance	Clearance always exists	[ ]
	Seal spring engagement	Remains engaged	[ ]
3.3.7	Rod internal pressure	[ ]	UO_2 Rod = [ ]
3.3.8	Assembly liftoff		
	Normal operation (including AOOs)	No liftoff from fuel support	Net force on assembly is downward.
	Postulated accident	No disengagement from fuel support	Fuel assembly LTP nozzle remains engaged with fuel support.

NOTE: Results are presented for a bounding analysis.

(continued next page)

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

Criteria Section	Description	Criteria	Result
3.3	Fuel System Criteria (Continued)		
3.3.9	Fuel assembly handling	Assembly withstands [ ]	Verifie by test to be [ ]
3.3.10	Miscellaneous components		
3.3.10.1	Compression spring forces	Support weight of UTP and fuel channel at EOL [ ]	Compression spring force of [ ]
3.3.10.2	LTP seal spring	Accommodate fuel channel deformation, adequate corrosion, and withstand operating stresses	The design criteria are met.
3.4	Fuel Coolability		
3.4.1	Cladding embrittlement	Include in LOCA analysis	Covered separate from this report.
3.4.2	Violent expulsion of fuel	[ ]	Covered separate from this report.
3.4.3	Fuel ballooning	Consider impact on fl w blockage in LOCA analysis	Covered separate from this report.

NOTE: Results are presented for a bounding analysis.

(continued next page)

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

Criteria Section	Description	Criteria	Result
3.4	Fuel Coolability (Continued)		
3.4.4	Structural deformations	Maintain coolable geometry and ability to insert control blades. SRP 4.2, App. A, and ASME Section III, App. F.	See results below for individual components. [ ]
	Fuel rod stresses	[ ]	[ ]
	Spacer grid lateral load	[ ]	[ ]
	Water channel load limit	[ ] according to ASME B&PV Code Section III, App. F. [ ]	[ ]
	UTP lateral load	[ ]	[ ]
	LTP lateral load	[ ]	[ ]

NOTE: Results are presented for a bounding analysis.

Table 3.4 Evaluation Results for Fuel Channel

Criteria Section	Description	Criteria	Result
3.5	Fuel Channel Normal Operation		
	Steady-state stress limits	[ ] according to ASME B&PV Code, Section III. [ ]	[ ] There is no significan plastic deformation during normal operation [ ]
	Cumulative cyclic loading to be less than the design cyclic fatigue life for Zircaloy	[ ]	Expected number of cycles [ ] is less than allowable.
	[ ]	–	The maximum expected oxidation is low in relation to the wall thickness. [ ]
	Long-term creep deformation (bulge and bow)	Bulge and bow shall not interfere with free movement of the control blade	Margin to a stuck control blade remains positive.

(continued next page)

Table 3.4 Evaluation Results for Fuel Channel (continued)

Criteria Section	Description	Criteria	Result
3.5	Fuel Channel Accident Conditions		
	Fuel channel stresses and load limit	The pressure load is limited to the [ ] according to ASME B&PV Code, Section III, Appendix F. The pressure load is also limited such that deformation remains within functional requirements.	The deformation during blowdown does not interfere with control blade insertion [ ]
	Channel bending from combined horizontal excitations	Allowable bending moment based on ASME B&PV Code, Section III, Appendix F plastic analysis collapse load.	[ ]
	Fuel channel gusset load rating	ASME allowable [ ] of one gusset is [ ]	[ ]

Table 3.5 Evaluation Results for Channel Fasteners

Criteria Section	Description	Criteria	Result
3.5	Channel Fastener		
–	Compatibility	Spring height must extend to the middle of the control cell to ensure contact with adjacent spring. Spring axial location must be sufficien to ensure alignment with adjacent spring at all exposures.	All compatibility requirements are met. The spring will extend beyond the cell mid-line at hot conditions. The axial location of the spring fla will always be in contact with an adjacent spring, even if a fresh ATRIUM-10 is placed adjacent to an EOL co-resident assembly.
–	Strength	Spring must meet ASME stress criteria and not yield beyond functional limit. Cap screw must meet ASME criteria for threaded fasteners.	All ASME stress criteria are met for the spring and cap screw. In addition, the spring will not yield under the maximum deflection

[

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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 deflectio 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 significan 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 No significan permanent deformation was detected for loads applied up to [

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4.2 Spacer Grid Lateral Impact Strength Test

Spacer grid impact strength was determined by a [

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[

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The maximum force prior to the onset of buckling was determined from the testing. The results were adjusted to reactor operating temperature conditions to establish an allowable lateral load.

4.3 Tie Plate Strength Tests

Three separate tests were conducted on the tie plates: (1) [
] (2) [
] and (3) [
].

The UTP [

].

For the FUELGUARD LTP [

].

A FUELGUARD LTP [

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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

A fretting test was conducted on a full-size test assembly to evaluate the ATRIUM-10 fuel rod support design. [

] After the test, the assembly was inspected for signs of fretting wear. [

]

4.5 Fuel Assembly Static Lateral Deflectio Test

A lateral deflectio 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

[

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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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3. XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4, and 5, *Qualificatio of Exxon Nuclear Fuel for Extended Burnup*, Exxon Nuclear Company, October 1986.

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8. XN-NF-81-58(P)(A) Revision 2, Supplements 1 and 2, *RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model*, Exxon Nuclear Company, March 1984.

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14. EMF-CQR-310619 Revision 1, *Component Qualificatio Report, ATRIUM-10XP+ Upper and Lower Load Chain Modification* , Framatome ANP, April 2004.

15. A1C-1333587-0, *Tensile Loading Test at Room Temperature on the Harmonized Advanced Load Chain (Upper Part Including the FA Design Upper Tie Plate)*, AREVA NP, May 2006

Appendix A Illustrations

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

Description	Page
Fuel Bundle Assembly	A-2
Upper Cage Assembly	A-3
Lower Cage Assembly	A-4
ULTRAFLOW Spacer Grid	A-5
Fuel Rods (Full- and Part-Length)	A-6
Fuel Channel	A-7
Fuel Channel Winged Fastener Assembly	A-8

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

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