ANP-2899NP
Revision 0

Fuel Design Evaluation for ATRIUM™ 10XM BWR Reload Fuel

April 2010

AREVA NP Inc.

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Nature of Changes

Item	Page	Description and Justification
1.	All	This is the initial release.

Contents

Tables

Figures

This document contains a total of 88 pages.

Nomenclature

AOO	anticipated operational occurrences
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BOL	beginning of life
Btu	British thermal units
BWR	boiling water reactor
cal/gm	calories per gram
CFR	Code of Federal Regulations
CHF	critical heat flux
CPR	critical power ratio
CRDA	control rod drop accident
CRWE	control rod withdrawal error
CUF	cumulative usage factor
ECCS	emergency core cooling system
EFPD	effective full power days
EOL	end of life
FA	fuel assembly
FC	fuel channel
FDL	fuel design limit
Gd	gadolinia
GWd/MTU	gigawatt days per metric ton of initial uranium
ID	inner diameter
klbm	pounds mass*1000
lbm	pounds mass
LHGR	linear heat generation rate
LOCA	loss- of- coolant accident
LTA (LUA)	lead test (use) assembly
LTP	lower tie plate
LTP	low temperature process
MAPLHGR	maximum average planar linear heat generation rate
MCPR	minimum critical power ratio
Mlbm/hr	million pounds mass per hour
MOX	mixed oxide
MTC	moderator temperature coefficient
MWd/kgU	Megawatt-days per kilogram of Uranium
MWR	metal-water reaction
MWt	Mega-Watt thermal

NRC	U.S. Nuclear Regulatory Commission
OD	outer diameter
OLMCPR	operating limit minimum critical power ratio
PCI	pellet-to-cladding interaction
PCT	peak cladding temperature
%F	percent flow
%P	percent power
PHTF	Portable Hydraulic Test Facility
PIE	post-irradiation examination
PLFR	part-length fuel rods
ppm	parts per million
psi	pounds per square inch
psia	pounds per square inch absolute
psid	pounds per square inch difference
QC	quality control
RPF	radial peaking factor
RXA	fully recrystallized annealed
SLMCPR	safety limit minimum critical power ratio
SRA	stress relieved annealed
SRP	Standard Review Plan
SST	stainless steel
UO_2	uranium dioxide
UTL	upper tolerance limit
UTP	upper tie plate

1.0 INTRODUCTION

This report provides the results of evaluations performed for the ATRIUM™* 10XM fuel design to demonstrate compliance with U. S. Nuclear Regulatory Commission (NRC) approved fuel licensing criteria defined in Reference 1. With Reference 1, the NRC approved a set of generic acceptance criteria to be satisfied by AREVA NP Inc. (AREVA) for new boiling water reactor (BWR) fuel designs. In accordance with the process described in Reference 1, new fuel designs or fuel design changes satisfying the generic acceptance criteria do not require explicit staff review. Satisfaction of the acceptance criteria is sufficient for approval by reference to the acceptance criteria.

Reference 1 (as clarified by References 2 and 3) requires that AREVA NP provide the NRC with an informational summary of the evaluation of the design against the NRC-approved acceptance criteria for a generic evaluation that is independent of plant and cycle. Documentation of analyses performed to demonstrate compliance with any criterion for a specific plant and/or cycle is provided to the Licensee as part of the normal reload licensing document package. AREVA's standard practice is to demonstrate compliance to the NRC-approved acceptance criteria on a plant- and cycle-specific basis.

The fuel design licensing process described in References 1 through 3 is the same process used to introduce and license AREVA's current ATRIUM-10 design and the previous ATRIUM-9 design. The evaluation results for the ATRIUM-10 and ATRIUM-9 designs were provided to the NRC in Reference 4 for information.

This report contains a detailed description of the ATRIUM 10XM design (Section 2.0); the ATRIUM 10XM evaluation results (Sections 3.0, 4.0, 5.0, and 7.0); a summary of the operating experience and the post-irradiation examination (PIE) results supporting the ATRIUM 10XM design features (Section 6.0); and a listing of the NRC-approved methods applicable to the ATRIUM 10XM design (Section 8.0).

* ATRIUM is a trademark of AREVA NP Inc.

2.0 DESIGN DESCRIPTION OF THE ATRIUM 10XM

2.1 Overview

A summary of the ATRIUM 10XM mechanical design that will be used in reload applications is given in this section. The ATRIUM 10XM design described herein shares many of the same proven design features of AREVA's ATRIUM-10 and ATRIUM-9 fuel designs that are in broad use in BWR plants. All materials used in the ATRIUM 10XM design have significant irradiation experience in the ATRIUM-10 and ATRIUM-9 designs.

In general, the design changes introduced with the ATRIUM 10XM design are evolutionary in nature and represent a [

]. The ATRIUM 10XM fuel bundle shares the same basic geometry as the current ATRIUM-10 fuel assembly design. This geometry consists of a 10x10 fuel lattice with a square internal water channel that displaces a 3x3 array of rods. Relative to the ATRIUM-10 fuel, the ATRIUM 10XM incorporates the following key design features:

[

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Table 2.1 lists the key design parameters of the ATRIUM 10XM fuel assembly and compares them to the current ATRIUM-10 design.

* ULTRAFLOW is a trademark of AREVA NP Inc.

2.2 Fuel Assembly

The ATRIUM 10XM fuel assembly consists of a lower tie plate (LTP) and upper tie plate (UTP), 91 fuel rods, 9 spacer grids, a central water channel with [ ], and miscellaneous assembly hardware. Of the 91 fuel rods, 12 are 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 central water channel. The lowest of the nine spacer grids 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 an illustration of the fuel bundle assembly is provided in Figure 2.1.

2.2.1 Spacer Grid

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Table 2.1 lists the main spacer grid attributes, and an illustration of the spacer grid is provided in Figure 2.2.

2.2.2 Water Channel

[

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Table 2.1 lists the main water channel attributes and are illustrated in Figure 2.1.

2.2.3 Lower Tie Plate (LTP)

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* FUELGUARD is a trademark of AREVA NP Inc.

Table 2.1 lists the main LTP attributes, and Figure 2.3 provides an illustration of the LTP.

2.2.4 Upper Tie Plate (UTP) and Connecting Hardware

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Table 2.1 lists the main UTP attributes, and Figure 2.4 provides an illustration of the UTP and locking components.

2.2.5 Fuel Rods

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Table 2.1 lists the main fuel rod attributes, and Figure 2.5 provides an illustration of the fuel rod components.

2.3 Fuel Channel and Components

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The fuel channel and fuel channel fastener are depicted in Figure 2.6 and Figure 2.7, respectively.

Table 2.1 ATRIUM 10XM Key Design Parameters

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Table 2.1 ATRIUM 10XM Key Design Parameters ***(Continued)***

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		]

[

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Figure 2.1 ATRIUM 10XM Fuel Assembly (not to scale)

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Figure 2.2 ATRIUM 10XM ULTRAFLOW Spacer Grid

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Figure 2.3 ATRIUM 10XM FUELGUARD LTP

[

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Figure 2.4 ATRIUM 10XM UTP and Locking Hardware

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Figure 2.5 ATRIUM 10XM Fuel Rods

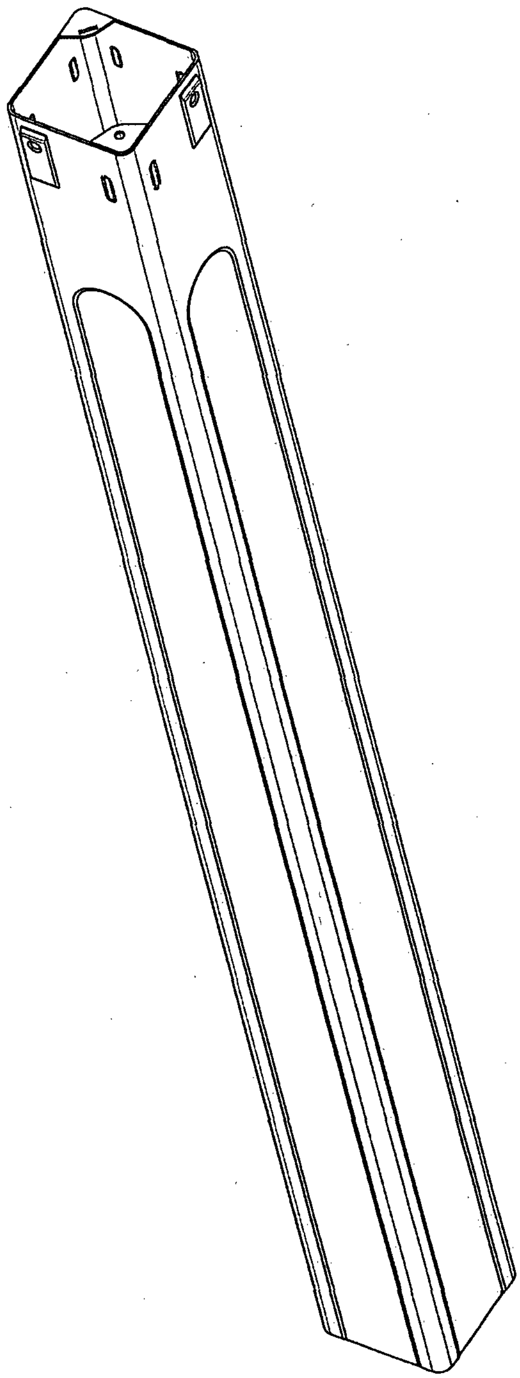

Figure 2.6 ATRIUM 10XM (or ATRIUM-10) Fuel Channel

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Figure 2.7 ATRIUM 10XM (or ATRIUM-10) Fuel Channel Fastener

3.0 FUEL SYSTEM DESIGN EVALUATION

3.1 Objectives

The objectives of building fuel assemblies (systems) to specific design criteria are to provide assurance that:

- The fuel assembly (system) shall not fail as a result of normal operation and anticipated operational occurrences (AOOs). The fuel assembly (system) dimensions shall be designed to remain within operational tolerances, and the functional capabilities of the fuels shall be established to either meet or exceed those assumed in the safety analysis.
- Fuel assembly (system) damage shall never prevent control rod insertion when it is required.
- The number of fuel rod failures shall be conservatively estimated for postulated accidents.
- Fuel coolability shall always be maintained.
- The mechanical design of fuel assemblies shall be compatible with co-resident fuel and the reactor core internals.
- Fuel assemblies shall be designed to withstand the loads from in-plant handling and shipping.

The first four objectives are those cited in the Standard Review Plan (SRP). The latter two objectives are to assure the structural integrity of the fuel and the compatibility with the existing reload fuel. To satisfy these objectives, the criteria are applied to the fuel rod and the fuel assembly (system) designs. Specific component criteria are also necessary to assure compliance. The criteria established to meet these objectives include those given in Chapter 4.2 of the SRP.

3.2 Fuel Rod Evaluation

The detailed fuel rod design evaluation entails such parameters as pellet diameter and density, cladding-pellet diametral gap, fission gas plenum size, and rod pre-pressurization level. The design evaluation also considers effects and physical properties of fuel rod components, which vary with burnup. The integrity of the fuel rods is ensured by designing to prevent excessive fuel temperatures, excessive rod internal pressures, and excessive cladding stresses and strains. This end is achieved by designing the fuel rods to satisfy the design criteria during normal operation and AOOs over the fuel lifetime. Summaries of the methods and codes used in the evaluation, along with the design criteria, are provided in the following sections. Details of the criteria and evaluation methodology can be found by consulting the referenced documents.

3.2.1 Internal Hydriding

The absorption of hydrogen by the cladding can result in cladding failure due to reduced ductility and formation of hydride platelets. Careful moisture control during fuel fabrication reduces the potential for hydrogen absorption on the inside of the cladding. The fabrication limit [ ] is verified by quality control (QC) inspection during fuel manufacturing.

3.2.2 Cladding Collapse

Creep collapse of the cladding and the subsequent potential for fuel failure is avoided in the AREVA fuel system design by limiting the gap formation due to fuel densification subsequent to pellet-clad contact. The size of the axial gaps that may form due to densification following the first pellet-clad contact shall be less than [ ].

The evaluation is performed using RODEX4. The design criterion and methodology are described in Reference 5. RODEX4 takes into account [ ]. A brief overview of RODEX4 and the statistical methodology is provided in Section 3.2.4.

3.2.3 Overheating of Cladding

The design basis to preclude fuel rod cladding overheating is that 99.9% of the fuel rods shall 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 AOOs. Compliance with this criterion is confirmed as part of the plant- and cycle-specific reload thermal-hydraulics analysis. An experimentally based, ATRIUM 10XM design-specific CHF correlation, which has been accepted by the NRC, is used in this evaluation (see Section 4.1.2).

3.2.4 Overheating of Fuel Pellets

Fuel failure from the overheating of the fuel pellets is not allowed. The centerline temperature of the fuel pellets must remain below melting during normal operation and AOOs. The melting point of the fuel includes adjustments for gadolinia (Gd) content. AREVA establishes the linear heat generation rate (LHGR) limit for each fuel system, which protects against fuel centerline melting during steady-state operation and during AOOs.

Fuel centerline temperature is evaluated using the RODEX4 (Reference 5) for both normal operating conditions and AOOs. A brief overview of the code and methodology follow.

RODEX4 evaluates the thermal-mechanical responses of the fuel rod surrounded by coolant. The fuel rod model considers the fuel column; gap region; cladding; gas plena and fill gas; and released fission gases. The fuel rod is divided into axial and radial regions, with conditions computed for each region. The operational conditions are controlled by [

].

The heat conduction in the fuel and clad is [

].

Mechanical processes include [

].

As part of the methodology, fuel rod power histories are generated [

].

Since RODEX4 is a best-estimate code, uncertainties [

]. Uncertainties taken into account in the analysis are summarized as:

- Power measurement and operational uncertainties: [ ].
- Manufacturing uncertainties: [ ].

- Model uncertainties: [

]．

3.2.5 <u>Stress and Strain Limits</u>

3.2.5.1 Pellet/Cladding Interaction

Cladding strain caused by transient-induced deformations of the cladding is calculated using the RODEX4 code and methodology, as described in Reference 5. See Section 3.2.4 for an overview of the code and method. [

].

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:

Category	Membrane	Bending
Primary	[	]
Secondary	[	]

Stresses are calculated at the cladding outer and inner diameter (ID) in the three principal directions for both beginning of life (BOL) and end of life (EOL) conditions. At EOL, the stresses due to mechanical bow and contact stress are decreased 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 weld stresses are evaluated for loadings from differential pressure, differential thermal expansion, rod weight, and plenum spring force.

The design limits are based on the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III and the minimum specified material properties.

3.2.6 Cladding Rupture

According to Code of Federal Regulations 10 CFR 50 Appendix K, the cladding rupture must not be underestimated when analyzing a loss-of-coolant accident (LOCA). NRC-approved cladding ballooning and rupture models are used by AREVA in the evaluation of cladding rupture. The specific models are those presented in NUREG-0630. There is no explicit limit on the deformation. However, the calculations with the deformation models must satisfy the event criteria given in 10 CFR 50.46. This analysis is performed as part of the reload licensing and is evaluated for each plant reload on a cycle-specific basis. (See Section 4.2.)

3.2.7 Fuel Rod Mechanical Fracturing

A mechanical fracture refers to a defect in a fuel rod caused by an externally applied force, such as loads due to earthquakes and postulated pipe breaks. These externally applied forces therefore include hydraulic loads and loads derived from core-plate motion. See Section 3.4.4 for a discussion of this accident evaluation.

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 RODEX4 fuel rod performance code.

3.3 Fuel System Evaluation

The detailed fuel system design evaluation is performed to ensure the structural integrity of the design under normal operation, AOO, faulted conditions, handling operations, and shipping. The analysis methods are based on fundamental mechanical engineering techniques—often employing finite element analysis, prototype testing, and correlations based on in-reactor performance data. Summaries of the major assessment topics are described in the sections that follow.

3.3.1 Stress, Strain, or Loading Limits on Assembly Components

The structural integrity of the fuel assemblies is assured by setting design limits on stresses and deformations due to various handling, operational and accident loads. AREVA uses Section III of the ASME B&PV Code as a guide to establish acceptable stress, deformation, and load limits for standard assembly components. These limits are applied to the design and evaluation of the UTP, LTP, spacer grids, springs, and load chain components, as applicable. The fuel assembly structural component criteria under faulted conditions are based on Appendix F of the ASME

B&PV Code Section III with some criteria derived from component tests. Table 7.1 in Section 7.0 lists the specific criteria.

The stress calculations use conventional, open-literature equations. A general-purpose, finite element stress analysis code, such as ANSYS, may be used to calculate component stresses. Section 3.2.5.2 discusses the cladding stress and strain evaluations.

3.3.2 Fatigue

Fatigue of structural components is generally [

]. Section 3.2.4 provides an overview of the code and method. Each fuel rod history is evaluated for power changes. The allowable number of cycles for every power change is determined from the cyclic stress calculated by RODEX4, along with a design fatigue S-N curve for zircaloy. The CUF (cumulative usage factor) is summed for all of the axial regions of the fuel rod using Miner's rule. The axial region with the highest CUF is used in the subsequent [

] is determined. The maximum CUF for the cladding must remain below [ ] to satisfy the design criterion.

3.3.3 Fretting Wear

Fretting wear is evaluated by testing, as described in Section 6.1.1.5. The testing is conducted by [

]. The inspection measurements for wear are documented. The lack of significant wear demonstrates adequate rod restraint geometry at the contact locations. Also, the lack of significant wear at the relaxed spacer cell locations provides further assurance that no significant fretting will occur at higher exposure levels.

[

] and has operated successfully without incidence of grid-to-rod fretting in more than 20,000 fuel assemblies.

3.3.4 Oxidation, Hydriding, and Crud Buildup

Corrosion of structural components must be conservatively bound in strength calculations to account for the material loss that occurs due to oxidation. This is most significant for [

].

Cladding external oxidation is calculated using RODEX4. Section 3.2.4 includes an overview of the code and method. The corrosion model includes an enhancement factor that is derived from poolside measurement data to obtain a fit of the expected oxide thickness. An uncertainty on the model enhancement factor also is determined from the data. The model uncertainty is included as part of the [

].

In the event abnormal crud is expected for a plant, a specific analysis is required to address the higher crud level. An abnormal level of crud is defined by a formation that increases the calculated fuel average temperature by 25°C above the design basis calculation. The formation of crud is not calculated within RODEX4; instead, an upper bound of expected crud is input by the use of the crud heat transfer coefficient. The corrosion model also takes into consideration the effect of the higher thermal resistance from the crud on the corrosion rate. A higher corrosion rate is, therefore, included as part of the abnormal crud evaluation. A similar specific analysis is required if higher corrosion instead of crud is expected for a plant.

The maximum oxide on the fuel rod cladding shall not exceed [ ]. The limit is evaluated such that greater than [ ].
Currently, there is no hydrogen limit, and no hydrogen uptake is reported.

3.3.5 Rod Bow

Differential expansion between the fuel rods and cage structure, and lateral thermal and flux gradients can lead to lateral creep bow of the rods in the spans between spacer grids. This lateral creep bow alters the pitch between the rods and may affect the peaking and local heat transfer. The AREVA design basis for fuel rod bowing is that [

].

Rod bow is calculated using the approved model described in Reference 6. The model has been shown to be conservative for application to the ATRIUM-10 fuel design. Less rod bow is predicted for the ATRIUM 10XM compared to the ATRIUM-10 due to a larger diameter fuel rod and a reduced distance between most spacer grids. [

]. The rod-to-rod gap closure due to bow is assessed for impact on thermal margins, as described in Section 4.1.4.

3.3.6 Axial Irradiation Growth

Three growth calculations are considered for the ATRIUM 10XM design:

- minimum fuel rod clearance between the LTP and UTP
- minimum engagement of the fuel channel with the LTP seal spring
- external interfaces (e.g., channel fastener springs)

Rod growth, assembly growth, and fuel channel growth are calculated using correlations derived from post-irradiation data. The evaluation of initial engagements and clearances accounts for the combination of fabrication tolerances on individual component dimensions.

The SRA fuel rod growth correlation was established from [

]. 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 resulting growth correlations are described in Reference 7. The minimum and maximum [ ], as appropriate, are used to obtain EOL growth values.

3.3.7 Rod Internal Pressure

Fuel rod internal pressure is calculated using the RODEX4 code and methodology, as described in Reference 5. Section 3.2.4 provides an overview of the code and method. The maximum rod pressure is calculated under steady-state conditions and taking into account slow transients.

Rod internal pressure is limited to [ ]. The expected upper bound of rod pressure [ ] is calculated for comparison to the limit.

3.3.8 Assembly Lift-off

Fuel assembly lift-off is evaluated 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 adding 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 assembly lift-off. 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 assembly lift-off are considered on a cycle-specific basis, as determined by the plant and other fuel types. Analyses to date indicate a large margin to assembly lift-off under normal operating conditions. Therefore, fuel lift-off in BWRs under normal operating conditions is considered to be a small concern.

For faulted conditions, [

]. The up-lift 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.

3.3.9 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 to an axial tensile force greater than [ ]. An acceptable test shows no yielding after loading. The testing is described further in Section 6.1.1.1.

Also, the plenum spring [ ]. This spring force requirement is demonstrated through a combination of design calculations and testing.

3.3.10 Miscellaneous Component Criteria

3.3.10.1 Compression Spring Forces

The ATRIUM 10XM has a single large compression spring mounted on the central water channel. The compression spring serves the same function as 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 consideration is required for fuel rod buckling loads.

3.3.10.2 LTP Seal Spring

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. [ ] 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. Chapter 4.2 of the SRP provides several specific areas important to fuel coolability, as discussed below.

3.4.1 Cladding Embrittlement

The requirements on cladding embrittlement relate to the LOCA requirements of 10 CFR 50.46. AREVA demonstrates compliance with the Part 50.46 limits (2200°F peak cladding temperature, local and core-wide oxidation, and long-term coolability). The models to compute the temperatures and oxidation are those prescribed by Appendix K of 10 CFR 50. These models are in the approved AREVA emergency core cooling system (ECCS) evaluation model. The LOCA analysis is performed on a plant-specific basis.

3.4.2 Violent Expulsion of Fuel

In a reactivity initiated severe accident, the deposition of energy in the fuel is the critical item. A large deposition could result in melting, fragmentation, and dispersal of fuel. The NRC has established a guideline in Regulatory Guide 1.77 and the SRP that restricts the radially averaged energy deposition. The guideline requires the hottest axial deposition to be less than 280 calories/gram (cal/gm). AREVA uses the 280 cal/gm as a design criteria.

3.4.3 Fuel Ballooning

During a LOCA, the cladding swelling and burst strain can result in flow blockage. Therefore, the LOCA analysis must consider the cladding swelling and burst strain impacts on the flow. As discussed in Section 3.2.6, AREVA uses the models in NUREG-0630. This swelling and rupture model is an integral part of the LOCA evaluation.

3.4.4 Structural Deformations

The AREVA seismic analysis methodology provides a way to evaluate the consequences of core support plate motion. AREVA limits the combined stresses from postulated accidents to the stress limits given in ASME B&PV Code, Section III, Appendix F for faulted conditions. The stress limits are derived from analyses and/or component load tests. For plants with existing seismic/LOCA analyses, [

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[

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Figure 3.1 Example LHGR Limit for the ATRIUM 10XM Design

4.0 THERMAL AND HYDRAULIC DESIGN EVALUATION

Thermal-hydraulic analyses are performed to verify that design criteria are satisfied and to help establish thermal operating limits with acceptable margins of safety during normal reactor operation and AOOs. The design criteria that are applicable to the ATRIUM 10XM fuel design are described in Reference 1. To the extent possible, these analyses are performed on a generic fuel design basis. However, due to reactor and cycle operating differences, many of the analyses supporting these thermal-hydraulic operating limits are performed on a plant- and cycle-specific basis and are documented in plant- and cycle-specific reports.

4.1 Thermal-Hydraulic Design Criteria

The thermal-hydraulic design criteria are summarized below:

- **Hydraulic compatibility.** The hydraulic flow resistance of the reload fuel assemblies shall be sufficiently similar to the existing fuel in the reactor such that there is no significant impact on total core flow or the flow distribution among assemblies in the core.
- **Thermal margin performance.** Fuel assembly geometry, including spacer design and rod-to-rod local power peaking, should minimize the likelihood of boiling transition during normal reactor operation, as well as during AOOs. The fuel design should fall within the bounds of the applicable empirically based boiling transition correlation approved for AREVA reload fuel. Within other applicable mechanical, nuclear, and fuel performance constraints, the fuel design should achieve good thermal margin performance.
- **Fuel centerline temperature.** Fuel design and operation shall be such that fuel centerline melting is not projected for normal operation and AOOs.
- **Rod bow.** The anticipated magnitude of fuel rod bowing under irradiation shall be accounted for in establishing thermal margin requirements.
- **Bypass flow.** The bypass flow characteristics of the reload fuel assemblies shall not differ significantly from the existing fuel in order to provide adequate flow in the bypass region.
- **Stability.** Reactors fueled with new fuel designs must be stable in the approved power and flow operating region. The stability performance of new fuel designs will be equivalent to or better than existing (approved) AREVA fuel designs.
- **LOCA analysis.** LOCAs are analyzed in accordance with Appendix K modeling requirements using NRC-approved models. The criteria are defined in 10 CFR 50.46.
- **Control rod drop accident (CRDA) analysis.** The deposited enthalpy must be less than 280 cal/gm for fuel coolability based on the NRC limits defined in NUREG-0800, Section 15.4.9.

- **ASME overpressurization analysis.** ASME pressure vessel code requirements must be satisfied.
- **Seismic/LOCA lift-off.** Under accident conditions, the assembly must remain engaged in the fuel support.

4.1.1 Hydraulic Compatibility

The methodology and constitutive relationships used by AREVA for the calculation of pressure drop in BWR fuel assemblies are presented in Reference 9 and are implemented in the XCOBRA code. The XCOBRA code predicts the steady-state thermal-hydraulic performance of BWR cores at various operation conditions and power distributions. XCOBRA received NRC approval in Reference 10. The NRC reviewed the inclusion of the water rod models presented in Reference 11 and provided acceptance in Reference 12. As described in Section 6.1.2, a series of pressure drop tests with an ATRIUM 10XM fuel assembly was performed in AREVA's Portable Hydraulic Test Facility (PHTF). The component loss coefficients derived from these pressure drop tests are used to explicitly model the ATRIUM 10XM hydraulic performance in both the neutronic and safety evaluations. The application of the two-phase multipliers was confirmed with full-scale prototypic two-phase test results. Thus, consistent with the AREVA NRC-approved methodology (Reference 9), the thermal-hydraulic characteristics of the bundle are explicitly modeled in all analyses.

While thermal-hydraulic compatibility is demonstrated on a plant-specific basis, results of an example compatibility analysis are presented to demonstrate that the ATRIUM 10XM fuel design is compatible with the previously approved ATRIUM-10 fuel design for an example BWR/4 core. The analysis includes calculations for a [

]. Analyses were performed at rated and off-rated operating conditions. The transition core loadings and the operating conditions used in the compatibility analyses are shown in Table 4.1. Summary results showing how the pressure drop, bypass flow and hot assembly flow distributions are impacted by the transition from a full core of ATRIUM-10 fuel to a full core of ATRIUM 10XM fuel are presented in Table 4.2 and Table 4.3. Hot assembly results for the transition core analyses with 1/3 ATRIUM 10XM fuel are presented in Table 4.4 and Table 4.5. The results demonstrate that, for the conditions analyzed, the changes in core pressure drop, bypass flow, core flow distribution to the average powered and hot assemblies, and critical power ratio (CPR) performance are not significant as the core transitions to a full core of ATRIUM 10XM fuel.

Based on the results shown, the ATRIUM 10XM design is considered hydraulically compatible with the ATRIUM-10 fuel design. While the results presented are for a mid-peaked axial power shape, analyses with bottom- and top-peaked axial power shapes support the conclusion that the fuel designs are compatible. As noted earlier, thermal-hydraulic compatibility analyses are performed and reported on a plant-specific basis.

4.1.2 Thermal Margin Performance

Operation of a BWR requires protection against fuel damage during normal reactor operation and AOOs. A rapid decrease in heat removal capacity associated with boiling transition can potentially result in high transient temperatures in the cladding. Deterioration of mechanical properties associated with the elevated temperature may result in a loss of the fuel rod integrity. Protection of the fuel against boiling transition assures that such degradation is avoided. This protection is accomplished by determining the operating limit minimum CPR (OLMCPR) for each fuel assembly in the reactor core for each cycle. The THERMEX thermal limits methodology, described in Reference 10, consists of a series of analyses—the core limiting safety limit MCPR (SLMCPR) and the limiting transient ΔCPR—that establish the OLMCPR.

The calculation of the fuel assembly critical power performance is established by means of an empirical correlation based upon results of boiling transition test programs. The applicable critical power correlation is used to determine the operating limits and, for consistency, is also used to monitor the core. The approved AREVA CPR correlation for the ATRIUM 10XM is the ACE Critical Power Correlation. The basic form of the ACE Critical Power Correlation was approved by the NRC in Reference 13. The ATRIUM 10XM design specific ACE correlation was approved by the NRC in Reference 14.

The determination of the SLMCPR, which ensures that 99.9% of the fuel rods do not experience boiling transition during AOOs and steady-state operation, is obtained from a series of Monte Carlo calculations in which the variables used to determine the onset of boiling transition are randomly varied. For a given SLMCPR, the number of rods predicted to experience boiling transition is determined for each Monte Carlo trial. The SLMCPR methodology (Reference 15) explicitly accounts for the effects of channel bow. The SLMCPR is evaluated on a cycle-specific basis and is reported in the reload licensing report. The ACE/ATRIUM 10XM critical power correlation is applied in the SLMCPR methodology the same way as the ACE/ATRIUM-10 correlation.

The limiting NUREG-0800 Chapter 15 events that result in the limiting ΔCPRs are analyzed on a cycle-specific basis, and the results are included in the reload licensing report. COTRANSA2 (Reference 16), XCOBRA-T (Reference 17), XCOBRA (Reference 10) and CASMO-4/MICROBURN-B2 (Reference 18) are the major codes used in the AOO analyses. COTRANSA2 is a system transient simulation code that includes an axial one-dimensional neutronics model that captures the effects of axial power shifts associated with the system transients. XCOBRA-T is a transient thermal-hydraulics code used in the analysis of thermal margin for the limiting fuel assembly. XCOBRA is used in steady-state analyses. As noted earlier, the ACE critical power correlation is used to evaluate thermal margin for the ATRIUM 10XM fuel.

4.1.3 Fuel Centerline Temperature

Fuel design and operation shall be such that fuel centerline melting is not projected for normal operation and AOOs. This analysis is performed as part of the plant- and cycle-specific fuel mechanical design analysis (Section 3.2.4).

4.1.4 Rod Bow

The bases for rod bow are discussed in Section 3.3.5. [

].

4.1.5 Bypass Flow

Total core bypass flow is defined as leakage flow through the LTP flow holes, channel seal, core support plate, and LTP-fuel support interface. Plant-specific analyses are performed to demonstrate that core bypass flow is not adversely affected by the introduction of new assembly designs. These analyses use the explicit thermal-hydraulic characteristics of the ATRIUM 10XM fuel design. The results from the Section 4.1.1 compatibility example problem show that the bypass flow at rated conditions [

].

4.1.6 Stability

Each new fuel design is analyzed to demonstrate that the stability performance is equivalent to or better than an existing (NRC-approved) AREVA fuel design. The stability performance is a function of the core power, core flow, core power distribution and, to a lesser extent, the fuel design. [

]. A comparative stability analysis was performed with the NRC-approved STAIF code (Reference 19). The study shows that the ATRIUM 10XM fuel design has decay ratios equivalent to or better than other approved AREVA fuel designs.

As stated above, the stability performance of a core is strongly dependent on the core power, core flow, and power distribution in the core. Therefore, core stability is evaluated on a cycle-specific basis and addressed in the reload licensing report.

4.2 Loss-of-Coolant Accident (LOCA) Analysis

Hypothetical LOCAs are analyzed in accordance with Appendix K modeling requirements using the ECCS models described in References 20, 21, 22 and 23. The ECCS analyses provide peak cladding temperature (PCT) and peak local metal-water reaction (MWR) values and are used to define maximum average planar LHGR (MAPLHGR) limits that ensure that the 10 CFR 50.46 criteria are met. [

].

For each plant, the limiting break and ECCS single failure are determined by evaluating a spectrum of potential break locations, sizes, and ECCS single failures. Break spectrum analyses are documented in a plant-specific break spectrum report. For each AREVA fuel type, LOCA calculations are performed with the limiting break and limiting ECCS single failure to establish the maximum planar power at which the fuel may be operated over its exposure life without violating the criteria specified in 10 CFR 50.46. The results of these analyses are documented in a plant- and fuel-specific MAPLHGR report. LOCA MAPLHGR calculations are repeated when the nuclear design of the reload fuel changes. ATRIUM 10XM LOCA MAPLHGR calculations will be performed for the limiting break with the NRC-approved methodology, and the impact of changes in nuclear design will be reported in the reload licensing report.

4.3 Control Rod Drop Accident (CRDA) Analysis

The CRDA has been analyzed and parameterized on a generic basis, as documented in Reference 25. CRDA analyses have been performed for 9x9 and 10x10 fuel designs to show that the generic CRDA methodology remains applicable to all AREVA fuel designs. Calculations have also been performed that demonstrate this methodology is applicable to a wide variety of other vendor product lines and is essentially independent of fuel design. The key parameters in the generic CRDA methodology are the dropped control rod worth, local four-bundle peaking, Doppler coefficient, and delayed neutron fraction. The dropped control rod worth and the local four-bundle peaking are the parameters that are calculated for each cycle-specific analysis using the AREVA core simulator methodology. The Doppler coefficients have been calculated for representative lattice enrichment and gadolinia designs for each fuel type, including AREVA 8x8, 9x9-2, 9x9-5, 10x10-8B, ATRIUM-9, ATRIUM-10, and ATRIUM 10XM assemblies. Doppler coefficients for other vendor product lines—including GE-8x8, GE9, GE10, GE11, GE13, GE14, and SVEA-96 fuel—have also been determined. For all designs, the variation in Doppler coefficient from design to design [ ]. From these results, conservative Doppler coefficients are chosen for use in the cycle-specific CRDA analysis. The delayed neutron fraction is calculated for each reactor core on a cycle-specific basis. The delayed neutron fraction is primarily a function of the core average exposure. The CRDA analysis will demonstrate that the maximum deposited enthalpy is less than 280 cal/gm, and the calculated number of fuel rod failures does not exceed the allowable value for the reactor. The CRDA results will be reported in the cycle-specific reload licensing report.

4.4 ASME Overpressurization Analysis

An overpressurization analysis is performed on a cycle-specific basis to assure the vessel pressure requirements of the ASME B&PV Code are satisfied. The purpose of the analysis is to determine if safety/relief valves in the steam lines have sufficient capacity and performance to prevent the vessel pressure from reaching the established transient pressure limit, which is 110% of the vessel design pressure. The analysis, which presumes failure of all non-safety grade components, does not contribute to the determination of thermal margin requirements. The ASME overpressurization event is analyzed with NRC-approved methods (Reference 16). Per the Reference 16 methodology, fuel design-specific hydraulic and neutronic information are accounted for in the ASME overpressurization analysis. The most severe overpressurization event is generally analyzed each cycle, and results are reported in the cycle-specific reload licensing report.

4.5 Seismic/LOCA Lift-off

Levitation of a fuel assembly could result in the assembly becoming disengaged from the fuel support and interfering with control rod movements. For accident conditions, the fuel design will be such that the normal hydraulic loads plus additional accident loads shall not cause the assembly to become disengaged from the fuel support (see Section 3.3.8).

**Table 4.1 BWR/4 Sample Problem
Thermal-Hydraulic Design Conditions**

Reactor Conditions	100%P / 100%F*	60%P / 45%F
Core power level, MWt	2923.0	1753.8
Core exit pressure, psia	1058.3	993.1
Core inlet enthalpy, Btu/lbm	528.3	505.2
Total core coolant flow, Mlbm/hr	77	34.65
Axial power shape	Middle-peaked	Middle-peaked

* Note: %P = % power, %F = % flow

	Number of Assemblies	
	Central Region	Peripheral Region
First Transition Core Loading		
[		]
[		]
Second Transition Core Loading		
[		]
[		]

Table 4.2 BWR/4 Sample Problem Thermal-Hydraulic Results at Rated Conditions (100%P / 100%F) for Transition to ATRIUM 10XM Fuel

[

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Table 4.3 BWR/4 Sample Problem Thermal-Hydraulic Results at Off-Rated Conditions (60%P / 45%F) for Transition to ATRIUM 10XM Fuel

[

]

**Table 4.4 BWR/4 Sample Problem
First Transition Core Thermal-Hydraulic Results at
Rated Conditions (100%P / 100%F)**

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* [

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Table 4.5 BWR/4 Sample Problem
First Transition Core Thermal-Hydraulic Results at
Off-Rated Conditions (60%P / 45%F)

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5.0 NUCLEAR DESIGN EVALUATION

The nuclear design analyses are subdivided into two parts: a nuclear fuel assembly design analysis and a core design analysis. The fuel bundle nuclear design analysis is assembly-specific and changes only as features affecting the nuclear characteristics of the fuel change, i.e., rod enrichments, burnable absorber content, etc. The core nuclear design analysis is specific to the core configuration and changes on a cycle-specific basis. Nuclear fuel and core analyses are performed using NRC-approved methodology (References 26 and 18) to assure that the new fuel assembly and/or design features meet the nuclear design criteria established for the fuel and core.

The fuel bundle nuclear design characteristics are considered for each AREVA fuel bundle design added to the core. The key characteristics affecting the nuclear design analysis include the following items:

- Assembly average enrichment
- Radial and axial enrichment distribution
- Burnable poison content and distribution
- Nature and location of non-fueled rods

These key characteristics establish the fuel (local) and core power distributions and the kinetic parameters that are used in the thermal-hydraulic, mechanical, and nuclear safety evaluations. These fuel assembly characteristics are evaluated on a reload cycle-specific basis during the neutronic and thermal-hydraulic safety analyses to assure that FDLs are not exceeded during either normal operation or AOOs, and that the effects of postulated reactivity accidents will neither cause significant damage to the reactor coolant pressure boundary nor impair the capability to cool the core.

The accuracy of the approved methodology (References 26 and 18) has been demonstrated for the ATRIUM-10 fuel assembly design by comparison to measurements taken with operating reactors for many years at many reactors. Relative to the ATRIUM 10XM fuel design, the minor changes in fuel density and cladding diameter are specifically accounted for in the methodology and have no impact on the applicability of the methodology.

The ATRIUM 10XM does not depart from the current orthogonal lattice designs and does not exceed the gadolinia or U^{235} enrichment limits, as required in the SER of Reference 18.

5.1 Power Distribution

Comparison of the calculated LHGR and MCPR to the corresponding LHGR and MCPR limits is performed during the cycle-specific design and during core monitoring. Power history generation for evaluation of the LHGR Limit (FDL) is discussed in Section 3.2.4. This evaluation is performed explicitly for each cycle of operation.

The new lattice design has been evaluated with CASMO-4 and MCNP to demonstrate that the local power distribution uncertainties are consistent with the uncertainties described in the methodology report (Reference 18). The standard deviation of the local power distribution for various void fractions was determined to be [ ], which is consistent with the value [ ] in Table 2.1 of Reference 18. These results also demonstrate equivalent uncertainties in reactivity so the assembly radial power uncertainties are also consistent with the approved methodology.

5.2 Kinetic Parameters

Design criteria for the reactivity coefficients are as follows:

- Void Reactivity Coefficient due to boiling in the active channels shall be negative
- Doppler Coefficient shall be negative at all operating conditions
- Power Coefficient shall be negative at all operating power levels
- Moderator Temperature Coefficient shall be negative at all operating levels

The void reactivity coefficient for ATRIUM 10XM is always negative. The Doppler coefficient for ATRIUM 10XM fuel is always negative. The power coefficient for ATRIUM 10XM fuel is always negative due to the Doppler and void coefficients. Clarification of some of the Reference 1 design criteria was provided to the NRC in November 1997 (Reference 27). The Moderator Temperature Coefficient (MTC), in particular, was stated as being a component of the power coefficient, which is dominated by the Doppler and void reactivity components. The MTC as a separate component, particularly in the power operating range (run mode), is not meaningful. Therefore, AREVA concludes that the negative MTC criterion is satisfied if the power coefficient is shown to be negative. This conclusion is consistent with Chapter 4.3 of the SRP and satisfies GDC 11 which requires that, in the power operating range, the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

5.3 Stability

The design of the ATRIUM 10XM assembly includes sufficient design features to assure overall core stability. [

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Monitoring of local power range monitors will provide the capability to detect oscillatory behavior. Suppression of oscillations is provided by a reduction in core power, increase in core flow, and/or the insertion of negative reactivity. See Section 4.1.6 for additional discussion.

5.4 Control Rod Reactivity

The design of the assembly is such that the technical specification shutdown margin will be maintained. Specifically, the assemblies and the core are designed to remain subcritical with the highest reactivity-worth control rod fully withdrawn and the remaining control rods fully inserted. Plant- and cycle-specific design calculations are performed to demonstrate that adequate hot excess reactivity and cold-shutdown margin exist throughout the cycle for each reactor.

6.0 TESTING, INSPECTION AND SURVEILLANCE

The AREVA testing and inspection requirements are essential elements in assuring conformance to the design criteria. The AREVA QC program provides assurance that the components satisfy the product specifications. The AREVA Quality Assurance manual (Reference 29) controls and maintains this program. The NRC has reviewed and accepted this manual as being in compliance with Appendix B of 10 CFR 50.

The specific QC inspections performed by AREVA include component parts, pellets, rods and assemblies, as well as process control inspections. In addition, AREVA reviews and over-checks inspections performed by vendors. These AREVA and vendor inspections provide verification that the manufactured fuel is consistent with the fuel design.

Surveillance programs of the irradiated fuel provide confirmation of the design adequacy. AREVA has performed extensive poolside examinations of irradiated fuel designs. These surveillance programs have confirmed the good performance of the AREVA fuel.

6.1 Design Verification Testing

6.1.1 Mechanical Testing

AREVA performs testing to demonstrate compliance with the design criteria or to generate input parameters for the design calculations. Testing performed to qualify the mechanical design or evaluate assembly characteristics includes:

- Fuel assembly axial load structural strength test
- Spacer grid lateral impact strength test
- Tie plate lateral load strength tests and LTP axial compression test
- Debris filter efficiency test
- Fuel assembly fretting test
- Fuel assembly static lateral deflection test
- Fuel assembly lateral vibration tests
- Fuel assembly impact tests

Mechanical testing of the Improved FUELGUARD demonstrates compliance with design criteria for both the ATRIUM-10 and ATRIUM 10XM applications. Summary descriptions of the tests are provided below.

6.1.1.1 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 significant permanent deformation was detected for loads in excess
[

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6.1.1.2 Spacer Grid Lateral Impact Strength Test

Spacer grid impact strength was determined by a [

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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.

6.1.1.3 Tie Plate Strength Tests

In addition to the axial tensile tests, a lateral load test is performed on the UTP and LTP, and a compressive load test is performed on the LTP.

The UTP lateral load test was conducted on a test machine, which applied an increasing load until a measurable amount of plastic deformation was detected. [

].

For the Improved FUELGUARD LTP compressive load test, the tie plate was supported by the nozzle to simulate the fuel support conditions. A uniform, compressive axial load was applied to the grid. [

].

The LTP lateral load test was conducted by attaching the grid of the tie plate to a rigid vertical plate and applying a side load to the cylindrical part of the nozzle. [

].

6.1.1.4 Debris Filter Efficiency Test

Debris filtering tests were performed for the Improved FUELGUARD lower tie plate to evaluate its debris filtering efficiency. These tests evaluated the ability of the Improved FUELGUARD to protect the fuel rod array from a wide set of debris forms. In particular, testing was performed using small filamentary debris (e.g., wire brush debris) as this form is known to be a cause of debris fretting fuel failures. When testing the small filamentary debris forms, the debris filter is placed in a hydraulic test loop above a debris chamber. After insertion of a debris set in the debris chamber, the loop pump is started to circulate water in the loop for a given amount of time. Multiple pump shutdowns and restarts are then simulated. At the end of the test, the location of all debris is recorded and the filtering efficiency is determined. These debris filtering tests demonstrate that the Improved FUELGUARD is effective at protecting the fuel rod array from all high-risk debris forms.

6.1.1.5 Fuel Assembly Fretting Test

A fretting test was conducted on a full-size test assembly to evaluate the ATRIUM 10XM fuel rod support design. Spacer springs were relaxed in selected locations to simulate irradiation relaxation. [

]. After the test, the assembly was inspected for signs of fretting wear. No significant wear was found. The results agree with past test results on BWR designs where no noticeable wear was found on the fuel rods or other interfacing components following exposure to coolant flow conditions.

6.1.1.6 Fuel Assembly Static Lateral Deflection Test

A lateral deflection test was performed to determine the fuel assembly stiffness, both with and without the fuel channel. 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 for seismic analysis.

6.1.1.7 Fuel Assembly Lateral Vibration Tests

The lateral vibration testing consists of both a free vibration test and a forced vibration test [

]. Results from these tests are used as input to the fuel assembly structural model for seismic analysis.

The test setup for the free vibration test is similar to the lateral deflection test described above. The fuel assembly is deflected to a specific displacement and then released. Displacement data are recorded at several spacer locations. The assembly natural frequencies and damping ratios are derived from the recorded motion. The test is repeated for several initial displacements.

The forced vibration testing [

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6.1.1.8 Fuel Assembly Impact Tests

Impact testing was performed in a similar manner to the lateral deflection test. The unchanneled assembly is supported in a vertical position with both ends fixed. The assembly is displaced a specified amount and then released. A load cell is fixed to a rigid structure and located adjacent to a mid-assembly spacer. The fuel assembly impacts the load cell and the resulting impact force is recorded as a function of the initial displacement. The measured impact loads are used in establishing the spacer grid stiffness.

6.1.2 Thermal-Hydraulic Testing

A single-phase pressure drop test was performed on a prototypical ATRIUM 10XM fuel assembly to evaluate its hydraulic characteristics. The test was conducted in the PHTF, a single-phase, closed re-circulating pressurized water loop. The PHTF is designed for 300°F and 200 psig operating conditions at volumetric flows up to 3000 gpm.

Differential pressure measurements were taken for the test assembly setup [

].

Results from two-phase flow tests with a full-scale prototypic ATRIUM 10XM assembly were performed and used to demonstrate the applicability of the two-phase multipliers.

6.2 Operating Experience

The ATRIUM 10XM design is currently being operated in reload quantities in Europe and as lead test assemblies (LTAs) in the U.S. [

].

Relative to previous ATRIUM fuel designs, no new materials are introduced with the ATRIUM 10XM (see Table 6.1). From a materials perspective, the most important change compared to the current ATRIUM-10 design is the switch from zircaloy to alloy 718 for the spacer grid. Multiple LTA programs, and extensive post-irradiation measurement campaigns, including poolside and hot cell examinations, have been performed over the last decade with a key objective to characterize and validate the performance of alloy 718 spacer grids. [

].

As of August 2009, [

]. The experience is gained under conditions representing the whole BWR fleet: different reactor types, different loading patterns (control cell core and scatter), and various operating modes and cycle lengths (with and without control rod sequence in 12 months, 18 months or 24 months).

Table 6.1 Irradiation Experience for Materials Used in U.S. ATRIUM Designs

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Table 6.2 AREVA Experience with Advanced ATRIUM Designs

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6.3 Summary of Lead Test Assembly (LTA) Programs

6.3.1 Initial Alloy 718 Spacer Grid Test Program

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6.3.2 ATRIUM 10XP and ATRIUM 10XM

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Figure 6.1 ATRIUM 10XP/XM Surveillance Program

]. A discussion of the obtained PIE results is provided in Section 6.4.

6.4 Poolside Examination Results

6.4.1 Performance of Alloy 718 Spacer Grid Material

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6.4.1.1 General Performance of Alloy 718 as Spacer Structural Material

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Figure 6.2 Visual Aspect of Alloy 718 Spacer Grids

6.4.1.2 Shadow Corrosion of Fuel Rods Induced by Alloy 718 Spacer Grids

6.4.1.2.1 Visual Inspection

The shadow corrosion phenomenon induced on the Zircaloy-2 cladding tubes by alloy 718 is typically observed during fuel assembly inspection, especially at low burnup, when the general oxide layer is still thin. Figure 6.3 shows a typical example of spacer grid induced shadow corrosion on fuel rods.

[

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Figure 6.3 Shadow Corrosion in Vicinity of Alloy 718 Spacer Grid

A better evaluation is possible when rods are withdrawn and visually inspected. In Figure 6.4, different degrees of shadow corrosion are presented. Note that the [

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[

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Figure 6.4 Typical Appearance of Spacer Spring Contact Points (4 cycles)

6.4.1.2.2 Shadow Corrosion Measurement Database

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[

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Figure 6.5 Measurement Traces for Shadow Corrosion Evaluation

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]. The complete database is given in Figure 6.6.

[

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Figure 6.6 Shadow Corrosion Database of ATRIUM Fuel Rods (lift-off measurement including the enhancement by shadow corrosion)

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6.4.1.2.3 Metallographic Examination

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Figure 6.7 Circumferential Shadow Corrosion (at spacer elevation)

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Figure 6.8 Comparison of Hydride Distribution between Mid-Span and at Spacer Position

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[

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Figure 6.9 Evaluation of Shadow Corrosion on Two-Cycle ATRIUM 10XP Fuel Rod

Table 6.3 Comparison of Shadow Corrosion Obtained by Eddy Current Measurements and Metallography

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				]

6.4.1.3 Fuel Channel Shadow Corrosion Induced by Alloy 718 Spacer Grids

[

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[

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Figure 6.10 Spacer-Induced Shadow Corrosion Inside Fuel Channels

6.4.1.4 Conclusions

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6.4.2 Performance of 0.4047-Inch Fuel Rod

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6.4.2.1 Corrosion (lift-off measurements)

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[

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Figure 6.11 Corrosion Lift-Off Database

6.4.2.2 Uniform Corrosion

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[

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Figure 6.12 Uniform Corrosion (obtained by metallography)

6.4.2.3 Hydrogen Uptake

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Figure 6.13 Hydrogen Concentration Database

6.4.2.4 Fuel Rod Growth

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[

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Figure 6.14 Impact of Fuel Rod Diameter on Fuel Rod Growth

6.4.2.5 Diameter Change

Diameter change under irradiation is a consequence of cladding creep and fuel swelling. Before gap closure, the diameter change is governed by the cladding creep properties. Afterwards the swelling fuel leads to a re-straining of the cladding. [

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[

]

Figure 6.15 Fuel Rod Diameter Change

6.4.2.6 Fission Gas Release

Fission gas release is an important value to characterize the behavior of a fuel rod, because the fission gas accumulated in the plenum region of the rod will increase the internal pressure and influence the heat transfer. [

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[

]

Figure 6.16 Fission Gas Release Database

6.4.2.7 Conclusions

Pool and hot cell examinations are performed in order to establish a database on the irradiation properties of fuel rods with 0.4047-inch OD. Based on these examinations, there is no difference in behavior compared to the former design with 0.3957-inch-diameter fuel rods.

6.4.3 Fuel Assembly Growth

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6.4.4 Visual Appearance of Water Channel Crowns and Other Components

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Figure 6.17 UTP Inspection

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Figure 6.18 Bottom Nozzle and Seal Spring Inspection

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Figure 6.19 Water Channel Crown Inspection

6.5 Conclusions

Operational experience and PIE of ATRIUM 10XM fuel assemblies and components is broad and spans the range of operating environments and reactor types. Data have been collected at burnup levels exceeding current U.S licensed limits. A key focus of the exams is the performance of the alloy 718 spacer grid strip material, which has been demonstrated to perform in a manner consistent with expectations. The fuel rod behavior and fuel assembly growth are well represented by the existing measurement database. Based on a significant number of visual, poolside and metallographic inspections, the expected performance of the ATRIUM 10XM fuel design is confirmed.

7.0 GENERIC DESIGN CRITERIA EVALUATION FOR ATRIUM 10XM

Table 7.1 Generic Design Criteria

Criteria Section	Description	Criteria	Disposition
ANF-89-98(P)(A) Revision 1 and Supplement 1			
3.2	**Fuel Rod Criteria**		
3.2.1	Internal hydriding	[	]
3.2.2	Cladding collapse	[	]
3.2.3	Overheating of cladding	99.9% of rods not to exceed CHF	Evaluated on a cycle-specific basis
3.2.4	Overheating of fuel pellets	No fuel melting	Evaluated on a cycle-specific basis
3.2.5	Stress and strain limits		
3.2.5.1	PCI	[	]
3.2.5.2	Cladding stress		[ ]
	P_m (Primary membrane stress)	[	]
	$P_m + P_b$ (Primary membrane + bending)	[	]
	P + Q (Primary + secondary)	[	]
3.2.6	Cladding rupture	Not underestimated during LOCA and used in determination of 10 CFR 50.46 criteria	Evaluated on a plant-specific basis
3.2.7	Mechanical fracturing	ASME Section III, App. F	See 3.4.4 below
3.2.8	Densification and swelling	[	]

Table 7.1 Generic Design Criteria *(Continued)*

Criteria Section	Description	Criteria	Disposition
ANF-89-98(P)(A) Revision 1 and Supplement 1 *(Continued)*			
3.3	**Fuel System Criteria**		
3.3.1	Stress, strain, and loading limits on assembly components.	The ASME B&PV Code is used to establish acceptable stress levels or load limits for assembly structural components. The design limits for accident conditions are derived from Appendix F of Section III.	[ ].
3.3.2	Fatigue	[	].
3.3.3	Fretting wear	[	].
3.3.4	Oxidation, hydriding, and crud buildup	[	].
3.3.5	Rod bow	Protect thermal limits	[ ]

Table 7.1 Generic Design Criteria *(Continued)*

Criteria Section	Description	Criteria	Disposition
ANF-89-98(P)(A) Revision 1 and Supplement 1 *(Continued)*			
3.3	**Fuel System Criteria** *(Continued)*		
3.3.6	Axial irradiation growth		
	Upper end cap clearance	Clearance always exists	[ ]
	Seal spring engagement	Remains engaged with channel	[ ]
3.3.7	Rod internal pressure	[	]
3.3.8	Assembly lift-off		
	Normal operation (including AOOs)	No lift-off from fuel support	Evaluated on a cycle-specific basis
	Postulated accident	No disengagement from fuel support	Evaluated on a cycle-specific basis
3.3.9	Fuel assembly handling	[	]
3.3.10	Miscellaneous components		
3.3.10.1	Compression spring forces	Support weight of UTP and fuel channel throughout design life	The design criteria are met
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	Evaluated on a plant-specific basis
3.4.2	Violent expulsion of fuel	< 280 cal/g coolability	Evaluated on a cycle-specific basis
3.4.3	Fuel ballooning	Consider impact on flow blockage in LOCA analysis	Evaluated on a plant-specific basis

Table 7.1 Generic Design Criteria *(Continued)*

Criteria Section	Description	Criteria	Disposition
ANF-89-98(P)(A) Revision 1 and Supplement 1 *(Continued)*			
3.4	**Fuel Coolability** *(Continued)*		
3.4.4	Structural deformations	Maintain coolable geometry and ability to insert control blades	This is a plant-specific analysis
4.1	**Thermal and Hydraulic Criteria**		
4.1.1	Hydraulic compatibility	Hydraulic flow resistance shall be sufficiently similar to existing fuel such that there is no significant impact on total core flow or flow distribution among assemblies	Evaluated on a plant-specific basis
4.1.2	Thermal margin performance	Fuel design shall be within the limits of applicability of an approved CHF correlation. < 0.1% of the rods in boiling transition.	Evaluated on a cycle-specific basis
4.1.3	Fuel centerline temperature	No centerline melting	Evaluated on a cycle-specific basis (see 3.2.4)
4.1.4	Rod bow	Rod bow must be accounted for in establishing thermal margins	[ ]
4.1.5	Bypass flow	Bypass flow characteristics shall be similar among assemblies to provide adequate bypass flow	Evaluated on a plant-specific basis
4.1.6	Stability	See Section 5.3	Evaluated on a cycle-specific basis
4.2	**LOCA**	LOCAs are analyzed in accordance with Appendix K modeling requirements. Criteria defined in 10 CFR 50.46	Evaluated on a plant-specific basis
4.3	**CRDA**	< 280 cal/gm for coolability	Evaluated on a cycle-specific basis

Table 7.1 Generic Design Criteria *(Continued)*

Criteria Section	Description	Criteria	Disposition
ANF-89-98(P)(A) Revision 1 and Supplement 1 *(Continued)*			
4.4	**ASME Overpressurization Analysis**	ASME pressure vessel code requirements shall be satisfied	Evaluated on a cycle-specific basis
4.5	**Seismic/LOCA Lift-off**	Assembly remains engaged in fuel support	Evaluated on a plant-specific basis
5.0	**Neutronics Criteria**		
5.1	**Power Distribution**	LHGR and MCPR limits	Evaluated on a cycle-specific basis
5.2	**Kinetics Parameters**		
5.2.1	Void Reactivity Coefficient	Negative coefficient due to boiling in the active channel	The design criterion is met
5.2.2	Doppler Reactivity Coefficient	Negative at all operating conditions	The design criterion is met
5.2.3	Power Coefficient	Negative at all operating power levels	The design criterion is met
5.2.4	Moderator Temperature Coefficient (MTC)	Component of negative power coefficient in Run mode	The design criterion is met
5.3	**Stability**		
5.3.1	Relative Fuel Design Stability	Stability performance of new fuel designs, or of modifications to existing designs, will be equivalent to (or better than) existing (approved) AREVA fuel designs	The ATRIUM 10XM stability performance was demonstrated to be better than existing (approved) AREVA fuel designs
5.3.2	Fuel Cycle Stability	New fuel designs are stable in the approved power and flow operating region	Evaluated on a cycle-specific basis
5.4	**Control Rod Reactivity**	Technical Specification shutdown margin maintained	Evaluated on a cycle-specific basis

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