Enclosure 9

AREVA Report ANP-3119(NP)

Mechanical Design Report for Monticello ATRIUM 10XM Fuel Assemblies

Revision 0

45 pages follow

ANP-3119NP
Revision 0

Mechanical Design Report for Monticello ATRIUM™ 10XM Fuel Assemblies

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AREVA NP Inc.

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Nature of Changes

Item	Revision Number	Section(s) or Page(s)	Description and Justification
1.	0	All	This is the initial release.

Contents

Tables

Figures

This document contains a total of 45 pages.

Nomenclature

Acronym	Definition
AFC	Advanced fuel channel
AOO	Anticipated operational occurrences
ASME	American Society of Mechanical Engineers
B&PV	Boiler and pressure vessel
BWR	Boiling water reactor
CRDA	Control rod drop accident
EOL	End of life
HDSFSR	High density spent fuel storage racks
LOCA	Loss-of-coolant accident
LTP	Lower tie plate
MWd/kgU	Megawatt-days per kilogram of Uranium
NRC	U. S. Nuclear Regulatory Commission
PLFR	Part-length fuel rods
psi	Pounds per square inch
S_m	Design stress intensity
SRA	Stress relief annealed
SRP	Standard review plan
S_u	Ultimate stress
S_y	Yield stress
UTP	Upper tie plate

1.0 Introduction

This report provides a design description, mechanical design criteria, fuel structural analysis results, and test results for the ATRIUM™* 10XM fuel assembly and 100/75 Advanced Fuel Channel (AFC) designs supplied by AREVA NP Inc. (AREVA) for use at the Monticello nuclear generating plant beginning with Cycle 28.

The scope of this report is limited to an evaluation of the structural design of the fuel assembly and fuel channel. The fuel assembly structural design evaluation is not cycle-specific so this report is intended to be referenced for each cycle where the fuel design is in use. Minor changes to the fuel design and cycle-specific input parameters will be dispositioned for future reloads. AREVA will confirm the continued applicability of this report prior to delivery of each subsequent reload of ATRIUM 10XM fuel at Monticello in a cycle specific compliance document.

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

Mechanical analyses have been performed using NRC-approved design analysis methodology (References 1, 2, 3 and 4). The methodology permits maximum licensed assembly and fuel channel exposures of 54 MWd/kgU (Reference 3).

* ATRIUM is a trademark of AREVA NP Inc.

2.0 Design Description

This report documents the structural evaluation of the ATRIUM 10XM fuel assembly and fuel channel described below. Reload-specific design information is available in the design package provided by AREVA for each reload delivery.

2.1 *Overview*

This ATRIUM 10XM fuel bundle geometry consists of a 10x10 fuel lattice with a square internal water channel that displaces a 3x3 array of rods.

Table 2-1 lists the key design parameters of the ATRIUM 10XM fuel assembly.

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 the appendix.

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 the appendix.

2.2.2 Water Channel

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Table 2-1 lists the main water channel attributes and the appendix provides an illustration of a section of the water channel.

2.2.3 Lower Tie Plate

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† FUELGUARD is a trademark of AREVA NP Inc.

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The appendix provides an illustration of the LTP.

2.2.4 Upper Tie Plate and Connecting Hardware

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The appendix 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 the appendix provides an illustration of the full length and part length fuel rods.

2.3 *Fuel Channel and Components*

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Table 2-2 lists the fuel channel component attributes. The fuel channel and fuel channel fastener are depicted in the appendix.

Table 2-1 Fuel Assembly and Component Description

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Table 2-2 Fuel Channel and Fastener Description

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3.0 Fuel Design Evaluation

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

3.1 *Objectives*

The objectives of designing fuel assemblies (systems) to specific 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 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 mechanical design report documents the fuel structural analyses only. The fuel rod evaluation will be documented in the Monticello fuel rod thermal-mechanical report.

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 or faulted 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.

All significant loads experienced during normal operation, AOOs, and under faulted conditions are evaluated to confirm the structural integrity of the fuel assembly components. Outside of faulted conditions, most structural components are under the most limiting loading conditions during fuel handling. See Section 3.3.9 for a discussion of fuel handling loads and Section 3.4.4 for the structural evaluation of faulted conditions. Although normal operation and AOO loads are often not limiting for structural components, a stress evaluation may be performed to confirm the design margin and to establish a baseline for adding accident loads. 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.

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See Table 3-1 for results from the component strength evaluations.

3.3.2 Fatigue

Fatigue of structural components is generally [

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3.3.3 Fretting Wear

Fuel rod failures due to grid-to-rod fretting shall not occur. [

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Fretting wear is evaluated by testing, as described in Section 4.5. 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 spacer cell locations, relaxed to end of life (EOL) conditions, provides further assurance that no significant fretting will occur at higher exposure levels.

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] 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

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

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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 criterion for fuel rod bowing is that [

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Rod bow is calculated using the approved model described in Reference 4. 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 predicted rod-to-rod gap closure due to bow is assessed for impact on thermal margins.

3.3.6 Axial Irradiation Growth

Fuel assembly components, including the fuel channel, shall maintain clearances and engagements, as appropriate, throughout the design life. Three specific 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 3. The upper and lower [ ], as appropriate, are used to obtain EOL growth values.

The minimum EOL rod growth clearance and EOL fuel channel engagement with the seal spring are listed in Table 3-1. The channel fastener spring axial compatibility is reported in Table 3-3.

3.3.7 Rod Internal Pressure

This will be addressed in the Monticello fuel rod thermal-mechanical report.

3.3.8 Assembly Lift-off

Fuel assembly lift-off is evaluated under both normal operating conditions (including AOOs) and under faulted conditions. 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.

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, [

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[ ]. 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.

3.3.9 Fuel Assembly Handling

The fuel assembly shall withstand, without permanent deformation, all normal axial loads from shipping and fuel handling operations. Analysis or testing shall show that the fuel is capable of [ ].

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 4.1.

There are also handling requirements for the fuel rod plenum spring which are addressed in the Monticello fuel rod thermal-mechanical report.

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

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.

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 LOCA evaluation is addressed in the Monticello LOCA MAPLHGR analysis for ATRIUM 10XM fuel report.

3.4.2 Violent Expulsion of Fuel

Results for the CRDA analysis are presented in the Monticello ATRIUM 10XM fuel transition report and the subsequent cycle-specific Monticello reload licensing report.

3.4.3 Fuel Ballooning

The LOCA evaluation is addressed in the Monticello LOCA MAPLHGR analysis for ATRIUM 10XM fuel report.

3.4.4 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 Section 4.2, Appendix A. The limits for each ATRIUM 10XM structural component are derived from analyses and/or component load tests.

Testing is performed 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 analytical models of the fuel assembly and fuel channel. [

]. See Section 4.0 for descriptions of the testing.

The methodology for analyzing the channeled fuel assembly under the influence of accident loads is described in Reference 2. Evaluations performed for the fuel under accident loadings include mechanical fracturing of the fuel rod cladding, assembly structural integrity, and fuel assembly liftoff.

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Assembly liftoff under accident conditions is described in Section 3.3.8.

3.4.4.1 Fuel Storage Seismic Qualification

The High Density Spent Fuel Storage Racks (HDSFSR) analysis accounts for the fuel as added mass in calculating the structural integrity under postulated seismic loads. The weights of legacy fuel assembly designs at Monticello encompass the weight of the ATRIUM 10XM fuel design. Therefore, the HDSFSR remains qualified with the introduction of the ATRIUM 10XM fuel design.

3.5 *Fuel Channel and Fastener*

The fuel channel and fastener design criteria are summarized below, and evaluation results are summarized in Table 3-2 and Table 3-3. 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 Service Level A. 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 (Reference 8).

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 is limited to that value which will permit control blade movement. During normal operation, any significant 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 flow during operation impose a duty loading on the fuel channel. The cyclic duty from pressure fluctuations is limited to less than the fatigue lifetime of the fuel channel. The fatigue life is based on the O'Donnel and Langer curve (see Reference 5), which includes a factor of 2 on stress amplitude or a factor of 20 on the number of cycles, whichever is more conservative.

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

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Long-Term Creep Deformation. Changes to the geometry of the fuel channel occur due to creep deformation during the long term exposure in the reactor core environment. Overall deformation of the fuel channel occurs from a combination of bulging and bowing. Bulging of the side walls occurs because of the differential pressure across the wall. Lateral bowing of the channel is caused primarily from the neutron flux and thermal gradients. Too much deflection may prevent normal control blade maneuvers and it may increase control blade insertion time above the Technical Specification limits. 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 (Service Level D). Component support criteria for elastic system analysis are used as defined in paragraphs F-1332.1 and F-1332.2. The unirradiated properties of the fuel channel material are used since the yield and ultimate tensile strength increase during irradiation.

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

The amount of bulging remains limited to that value which will permit control blade insertion.

Fuel Channel Gusset Load Rating. [

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Table 3-1 Results for ATRIUM 10XM Fuel Assembly

Criteria Section	Description	Criteria	Results
3.3	**Fuel System Criteria**		
3.3.1	Stress, strain and loading limits on assembly components	The ASME B&PV Code Section III 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.	[ ].
	• Water channel	[ ].	[ ].
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 continued on next page)

Table 3-1 Results for ATRIUM 10XM Fuel Assembly *(Continued)*

Criteria Section	Description	Criteria	Results
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	N/A	Not covered in structural report
3.3.8	Assembly liftoff		
	• Normal operation (including AOOs)	No liftoff from fuel support	[ ].
	• Postulated accident	No disengagement from fuel support. No liftoff from fuel support.	[ ].
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.

(Table continued on next page)

Table 3-1 Results for ATRIUM 10XM Fuel Assembly *(Continued)*

Criteria Section	Description	Criteria	Results
3.4	**Fuel Coolability**		
3.4.1	Cladding embrittlement	N/A	Not covered in structural report
3.4.2	Violent expulsion of fuel	N/A	Not covered in structural report
3.4.3	Fuel ballooning	N/A	Not covered in structural report
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	[ ].	[ ].
	UTP lateral load	[ ]	[ ]
	LTP lateral load	[ ]	[ ]

Table 3-2 Results for Advanced Fuel Channel

Criteria Section	Description	Criteria	Results
3.5.1	**Advanced Fuel Channel – Normal Operation**		
	Stress due to pressure differential	The pressure load including AOO is limited to [ ] according to ASME B&PV Code, Section III. The pressure load is also limited such that [ ].	The deformation during AOO remains within functional limits for normal control blade operation and the collapse load requirement is met with [ ]. There is no significant plastic deformation during normal operation [ ].
	Fatigue	Cumulative cyclic loading to be less than the design cyclic fatigue life for Zircaloy. [ ].	Expected number of cycles [ ] is less than allowable.
	Oxidation and hydriding	Oxidation shall be accounted for in the stress and fatigue analyses	The maximum expected oxidation is low in relation to the wall thickness. Oxidation was accounted in the stress and fatigue analyses.
	Long-term deformation (bulge creep and bow)	Bulge and bow shall not interfere with free movement of the control blade	Margin to a stuck control blade remains positive.

(Table continued on next page)

Table 3-2 Results for Advanced Fuel Channel *(Continued)*

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

Table 3-3 Results for Channel Fastener

Criteria Section	Description	Criteria	Results
3.5	**Channel Fastener**		
	Compatibility	Spring height must extend to the middle of the control cell to ensure contact with adjacent spring.	All compatibility requirements are met. The spring will extend beyond the cell mid-line.
		Spring axial location must be sufficient to ensure alignment with adjacent spring at all exposures.	The axial location of the spring flat will always be in contact with an adjacent spring; even if a fresh ATRIUM 10XM 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 fastener.	All ASME stress criteria are met for the spring and cap screw. In addition, the spring will not yield under the maximum deflection.

4.0 Mechanical Testing

Prototype testing is an essential element of AREVA's methodology for demonstrating compliance with structural design requirements. Results from design verification testing may directly demonstrate compliance with criteria or may be used as input to design analyses.

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

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 significant permanent deformation was detected for loads in excess [

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4.2 *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.

4.3 *Tie Plate Strength Tests*

In addition to the axial tensile tests described above, 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. This provides a limiting lateral load for accident conditions. [

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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. [

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To determine a limiting lateral load for accident conditions, 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. [

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4.4 *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.

4.5 *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 on fuel rods in contact with spacer springs relaxed to EOL conditions. 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.

4.6 *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.

4.7 *Fuel Assembly Lateral Vibration Tests*

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

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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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4.8 *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.

5.0 Conclusion

The fuel assembly and channel meet all the mechanical design requirements identified in References 1 and 2. Additionally, the fuel assembly and channel meet the mechanical compatibility requirements for use in Monticello. This includes compatibility with both co-resident fuel and the reactor core internals.

6.0 References

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

2. EMF-93-177(P)(A) Revision 1, *Mechanical Design for BWR Fuel Channels,* Framatome ANP Inc., August 2005.

3. EMF-85-74(P) Revision 0 Supplement 1(P)(A) and Supplement 2(P)(A), *RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model,* Siemens Power Corporation, February 1998.

4. XN-NF-75-32(P)(A) Supplements 1 through 4, *Computational Procedure for Evaluating Fuel Rod Bowing*, Exxon Nuclear Company, October 1983. (Base document not approved.)

5. W. J. O'Donnell and B. F. Langer, *Fatigue Design Basis for Zircaloy Components*, Nuclear Science and Engineering, Volume 20, January 1964.

6. XN-NF-81-51(P)(A), *LOCA - Seismic Structural Response of an Exxon Nuclear Company BWR Jet Pump Fuel Assembly*, Exxon Nuclear Company, May 1986.

7. XN-NF-84-97(P)(A), LOCA - Seismic Structural Response of an ENC 9x9 BWR Jet Pump Fuel Assembly, Exxon Nuclear Company, August 1986.

8. Huan, P. Y., Mahmood, S. T., and R. Adamson, R. B. "Effects of Thermomechanical Processing on In-Reactor Corrosion and Post-Irradiation Properties of Zircaloy-2", *Zirconium in the Nuclear Industry: Eleventh International Symposium*, ASTM STP 1295, E. R. Bradley and G. P. Sabol, Eds., American Society for Testing and Materials, 1996, pp. 726-757.

Appendix A Illustrations

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

These illustrations are for descriptive purpose only. Please refer to the current reload design package for product dimensions and specifications.

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Figure A-1 ATRIUM 10XM Fuel Assembly

(not to scale)

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Figure A-2 UTP with Locking Hardware

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Figure A-3 Improved FUELGUARD LTP

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Figure A-4 ATRIUM 10XM ULTRAFLOW Spacer Grid

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fsjm00410

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Figure A-5 Full and Part-Length Fuel Rods

(not to scale)

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Figure A-6 Advanced Fuel Channel

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Figure A-7 Fuel Channel Fastener Assembly