Browns Ferry Unit 2 ATRIUM 11 Lead Test Assemblies Design & Licensing Summary Report

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AREVA Inc.

Nature of Changes

Item	Revision Number	Section(s) or Page(s)	Description and Justification
1.	0	All	Initial Issue

Contents

Tables

This document contains a total of 15 pages (10 page body text and 5 pages front matter).

Nomenclature

3GFG	3rd Generation FUELGUARD
AOO	Anticipated Operational Occurrences
BWR	Boiling Water Reactor
KATHY	Karlstein Thermal Hydraulic Test Loop
LHGR	Linear heat generation rate
LOCA	Loss of Coolant Accident
LTA	Lead Test Assembly
LTP	Lower tie plate
NRC	Nuclear Regulatory Commission
PHTF	Portable Hydraulic Test Facility
PLFR	Part Length Fuel Rods
SPCB	Siemens Power Critical Heat Flux Correlation for BWRs
SRA	Stress relieved annealed cladding
WC	Water channel
UTP	Upper tie plate

1.0 INTRODUCTION

Tennessee Valley Authority (TVA) is conducting an irradiation demonstration program consisting of eight (8) ATRIUM™* 11 Lead Test Assemblies (LTA) supplied by AREVA Inc. (AREVA), in Browns Ferry Unit 2 Cycle 19 (BFE2-19). Each ATRIUM 11 LTA is supplied with a Z4B™*-BQ (Zircaloy BWR beta-quenched) lead fuel channel. The BFE2-19 LTA program is a component of AREVA's comprehensive testing and irradiation program designed to qualify the ATRIUM 11 fuel assembly design for reload supply within the U.S. AREVA's roadmap for new fuel design introduction is described in ANF-89-98, Nuclear Regulatory Commission (NRC) approved "Generic Mechanical Design Criteria for BWR Fuel Designs" topical report (Reference 1). Per Reference 1, the AREVA process for introducing new fuel design features involves prototype testing and/or lead test assemblies prior to full reload implementation and continuing irradiation surveillance programs including post irradiation examinations to confirm fuel assembly performance. In accordance with the NRC guidance in Supplement 1 to Generic Letter 90-02 (Reference 2), the provision has been made in the Browns Ferry Unit 2 Technical Specifications to accept a limited number of lead test assemblies that have not completed representative testing provided they are placed in non-limiting core regions. For the materials or features outside the current NRC approved codes and methods, the licensing analyses demonstrate that modeling of this small number of test fuel assemblies with NRC approved codes and methods produces either a conservative result or has a negligible impact with respect to cycle specific licensing analyses.

The ATRIUM 11 LTAs are introduced according to the provisions of 10 CFR 50.59. The cycle specific analyses provided by AREVA in the reload licensing documentation support the 10 CFR 50.59 evaluation for the BFE2-19 reload and the corresponding core operating limits report. Based on the licensing analyses performed by AREVA, the Browns Ferry ATRIUM 11 LTAs meet the relevant design criteria of Reference 1 and are suitable for irradiation in Cycle 19 and beyond for Browns Ferry Unit 2.

This report is provided to the NRC for information purposes to provide: a description of the Browns Ferry ATRIUM 11 LTAs; a summary of the licensing analyses performed for the LTAs using NRC-approved methods which demonstrate compliance to the approved design criteria; and a description of the recommended post irradiation examinations to be considered with respect to the LTAs. The scope of this report is limited to the BFE2-19 ATRIUM 11 LTA program. Per Reference 1, prior to full reload implementation of the ATRIUM 11, AREVA will provide the NRC with a comprehensive report demonstrating compliance with the Reference 1 design criteria.

2.0 ATRIUM 11 MECHANICAL DESIGN

Table 2-1 lists the key design parameters of the ATRIUM 11 fuel assembly and compares them to the current ATRIUM 10XM design.

* ATRIUM and Z4B are trademarks of AREVA Inc.

The ATRIUM 11 fuel bundle geometry consists of an 11x11 fuel lattice with a square internal water channel that displaces a 3x3 array of rods which provides desired moderation characteristics. 92 full length fuel rods, 8 long part length rods and 12 short part length fuel rods.

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The ATRIUM 11 utilizes a variant of the ULTRAFLOW™* spacer grid used on the ATRIUM 10XM. The ATRIUM 11 ULTRAFLOW spacer grid is constructed entirely from Alloy 718 sheet strip. The new ATRIUM 11 spacers provide the rod support configuration with integral springs and dimples rather than separate spring and dimple elements.

The ATRIUM 11 UTP is a modular construction that utilizes a spacer-like Alloy 718 grid to provide radial constraint to the tops of the full length fuel rods. The construction of the grid provides a uniform matrix of strips across the entire lower surface of the UTP.

The 3rd Generation FUELGUARD™* (3GFG) inlet debris filter was developed for use on the ATRIUM 11 fuel design to protect against the entry of wire debris. The ATRIUM 11 lower tie plate continues to use AREVA's snap-in seal spring design for the lower tie plate to fuel channel interface.

ATRIUM 11 employs a similar version of the harmonized advanced load chain (HALC) used by the ATRIUM 10XM design. The upper and lower tie plates retain all features of the ATRIUM 10XM necessary for compatibility with reactor internal structures, fuel storage racks, external channels, and all fuel-handling equipment. The water channel is constructed of annealed Z4B material.

The ATRIUM 11 Advanced Fuel Channel (AFC) is fabricated from Z4B-BQ sheet and [

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* ULTRAFLOW and FUELGUARD are trademarks of AREVA Inc.

Table 2-1 ATRIUM 10XM and ATRIUM 11 Key Design Parameters

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Table 2-1 ATRIUM 10XM and ATRIUM 11 Key Design Parameters (cont.)

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3.0 LEAD TEST ASSEMBLY TECHNICAL EVALUATION

The reload analysis to support operation of the fresh reload fuel including the ATRIUM 11 LTAs is documented in Reference 4. The nuclear, thermal hydraulic, transient and accident analyses have explicitly modeled the ATRIUM 11 fuel with NRC-approved analytical methods.

3.1 Fuel Mechanical Design Analysis

The ATRIUM 11 LTAs rely on the use of structural and fuel rod components similar to those in current operation with the ATRIUM 10XM fuel design. The mechanical design of the LTAs was evaluated according to the AREVA BWR generic mechanical design criteria (Reference 1). The generic design criteria have been approved by the U.S. NRC and the criteria are applicable to the subject design.

Since the basic fuel rod and mechanical design of the LTAs are similar to the ATRIUM 10XM design currently in operation, the mechanical analyses for the LTAs have been performed using NRC-approved design analysis methodology (Reference 3). The methodology permits [

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A new, proprietary Zirconium alloy is being implemented on the ATRIUM 11 water channels. Z4B is similar to Zry-4 except for elevated amounts of iron and chromium. Previous operating experience of this material on fuel channels and BWR spacer grids has shown improved corrosion performance compared to Zry-4. Since the recrystallized heat treatment is the same as used for Zry-4, the growth is expected to be in the bounds of the approved assembly growth models. Therefore approved analysis methods may be applied to LTAs with Z4B water channels.

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

3.2 Thermal Hydraulic Analysis

3.2.1 Thermal Hydraulic Compatibility

Consistent with the AREVA approved methodology, the ATRIUM 11 fuel assembly has undergone a pressure drop test in AREVA's Portable Hydraulic Test Facility (PHTF). The component loss coefficients from this pressure drop test have been used to explicitly model the ATRIUM 11 design in both the neutronic and safety analysis for the current and future cycles. Thus, consistent with the AREVA NRC-approved methodology, the thermal hydraulic characteristics of the bundle have been explicitly modeled in all analyses.

The ATRIUM 11 LTAs have been determined to be hydraulically compatible with the co-resident fuel designs for the entire range of the licensed power-to-flow operating map. Core bypass flow (defined as leakage flow through the LTP flow holes, channel seal, core support plate, and LTP-fuel support interface) is not adversely affected by the ATRIUM 11 LTAs.

3.2.2 Critical Power Performance

The critical power performance of the ATRIUM 11 LTAs is calculated based on a conservative application of the NRC-approved SPCB [ ] critical power correlation (Reference 6). The SPCB correlation was used in design and licensing calculations and is used for monitoring the LTAs during reactor operation.

The critical power performance of ATRIUM 11 has been measured in AREVA's Karlstein Thermal Hydraulic Test Facility (KATHY). [ ] critical power tests were performed with a full scale electrically heated ATRIUM 11 assembly. The test data demonstrates that the critical power performance of ATRIUM 11 [ ]. Evaluation of the test data also demonstrated that the SPCB correlation can be used [ ] the critical power performance of ATRIUM 11 LTAs.

3.3 Neutronic Analysis

The current approved CASMO-4/MICROBURN-B2 neutronic methodology (Reference 7) remains applicable to the ATRIUM 11 LTAs. All Reference 7 SER restrictions continue to be met when the methodology is applied to the ATRIUM 11 design. AREVA has completed comparisons of ATRIUM 11 results from higher order methods to those from CASMO-4/MICROBURN-B2 as required by AREVA procedures and verified that the results are acceptable.

The core has been designed such that the ATRIUM 11 LTAs are in non-limiting core bundle power locations. The core loading plan is designed to ensure that the LTAs have more margin to the core MCPR, LHGR, and APLHGR limits than the core limiting fuel assemblies.

Calculations of core shutdown margin explicitly model the LTAs for all cycles and show adequate shutdown margin. The LTA locations are not shutdown margin limiting locations.

3.4 Safety Analysis

Current NRC-approved methods have been applied in the safety analyses performed for the LTAs. The geometric, nuclear, and hydraulic characteristics of the LTAs have been explicitly modeled and are supported by the current NRC-approved safety analysis methods.

3.4.1 Anticipated Operational Occurrences

Based on the modeling of the ATRIUM 11 LTAs, their impact has been explicitly incorporated into the cycle specific analyses. The results of these analyses have been used to establish power and flow dependent MCPR limits and LHGRFAC multipliers to assure the applicable acceptance criteria are met. The analysis results for ATRIUM 11 and the co-resident fuel types as well as the resulting MCPR limits and LHGRFAC multipliers are reported in Reference 4.

3.4.2 Accident Analyses

3.4.2.1 LOCA Analysis

LOCA analyses have been performed for the LTAs. Results of these analyses are reported in Reference 4 and show that the PCT and peak local metal-water reaction results for the ATRIUM 11 fuel are calculated to be lower than those calculated for ATRIUM 10XM fuel.

3.4.2.2 Control Rod Drop Analysis

The cycle specific analysis for control rod drop has explicitly included the modeling of the LTAs and has demonstrated that the maximum deposited enthalpy is less than the NRC limit of 280 cal/gm and that the radiological release from the number of fuel rods which exceed the damage threshold is within that used in performing the radiological assessment for this event. The maximum deposited enthalpy is also less than 230 cal/gm. Due to the placement of the LTAs in low power locations, none of the rods exceeding the failure threshold of 170 cal/gm in the CRDA analysis are in the LTAs.

3.4.2.3 Fuel Handling / Cask Drop Accident

An explicit analysis has been performed for the LTAs to demonstrate that the radiological doses are well within the guidelines specified in 10 CFR 100 as well as the radiological safety limits defined in NUREG-0800 Section 15.7.4 for the fuel handling accident and well within the acceptance criteria for the cask drop accident.

3.4.3 Stability Analyses

Browns Ferry has implemented BWROG Long Term Stability Solution Option III (Oscillation Power Range Monitor-OPRM). Reload validation and stability based operating limit determination have been performed in accordance with the NRC-approved Reference 13 methodology, based on relative change in CPR as a function of hot channel oscillation magnitude (HCOM) calculations performed with the RAMONA5-FA code in accordance with Reference 14. The NRC-approved STAIF computer code (Reference 5) has been used in the core hydrodynamic stability analyses to define the BSP region boundaries. These cycle specific analyses have included explicit modeling of the LTAs and have shown acceptable results.

In addition to inclusion of the LTAs in the cycle specific analysis, a comparative stability analysis has been performed to assess the relative stability performance of the ATRIUM 11 fuel to the current ATRIUM 10XM fuel. The result of the comparative analysis is that the ATRIUM 11 fuel design is at least as stable as the ATRIUM 10XM fuel design.

4.0 Z4B-BQ FUEL CHANNELS

Excessive control blade friction due to fuel channel bow remains a significant technical challenge to the boiling water reactor (BWR) industry. As part of its efforts to resolve this issue, AREVA has developed lead fuel channel programs (References 11 and 12) to implement a new material that has demonstrated improved performance relative to Zry-4 fuel channels. The channels used on the ATRIUM 11 LTAs are made from Z4B material that has received a beta-quench (BQ) heat treatment. Eight Z4B-BQ fuel channels have been placed on ATRIUM 11 fuel assemblies for irradiation in Browns Ferry Unit 2 Cycle 19 (BFE2-19).

4.1 Technical Background

Z4B represents the culmination of AREVA's extensive research and experience with Zry-2 and Zry-4 alloys. While Z4B is closely related to Zry-4, some of the alloying elements (iron and chromium) are purposely set outside the ASTM-specified range for Zry-4 to achieve the desired performance. A comparison of alloying elements in Zry-2, Zry-4, and Z4B as specified by AREVA is shown in Table 4-1.

Table 4-1 Alloying elements of Zirconium alloys

Element		Composition range, wt%		
		Zry-2	Zry-4	Z4B
Tin	(Sn)	1.20 – 1.50*	1.20 – 1.50*	[ ]
Iron	(Fe)	0.14 – 0.20	0.18 – 0.24	[ ]
Chromium	(Cr)	0.05 – 0.15	0.07 – 0.13	[ ]
Nickel	(Ni)	0.03 – 0.08	-	-
Iron + Cr	(Fe+Cr)	-	0.28 – 0.37	[ ]
Oxygen	(O)	0.09 – 0.13	0.09 – 0.13	[ ]
Silicon	(Si)	0.008 – 0.012	0.008 – 0.012	[ ]
Carbon	(C)	0.012 – 0.020	0.012 – 0.020	[ ]

** The allowable range of Tin specified in ASTM B352 is 1.20-1.70 wt%.*

4.2 Licensing Assessment

The AREVA fuel channel topical report (References 9 and 10) has been approved by the NRC with the restriction of using either Zry-2 or Zry-4. Therefore, the Z4B fuel channels are treated as Lead Use Channels (LUC) in accordance with AREVA's approved methodology (Reference 1) for introducing new products. Under this methodology, AREVA has demonstrated that all performance criteria are met by the LUCs and examination programs have confirmed the satisfactory performance of the LUC design under irradiation. TVA has inserted the ATRIUM 11 LTAs with Z4B channels according to the provisions of 10 CFR 50.59 on the basis of reload licensing documentation performed by AREVA.

The Z4B channels do not affect any neutronic, thermal-hydraulic, or safety analyses. Strength, corrosion and distortion are bounded by existing methods. Fuel channel bow can indirectly affect Minimum Critical Power Ratio (MCPR) safety limits if the channel bow is outside predictions. However, channel bow for the Z4B channels has been predicted based on Zry-4 fuel channel performance, which post irradiation data have shown bounds Z4B-BQ fuel channel performance. The channels' performance will also be monitored via in-service testing as a precautionary measure to detect bow beyond the amount assumed in the safety analyses. As an additional conservatism to ensure dimensional compatibility, the minimum fuel channel growth has been assumed to be zero.

5.0 POST-IRRADIATION EXAMINATIONS

A key objective of ATRIUM 11 LTA programs at host reactors is to obtain performance data for these fuel assemblies, Reference 8. The proposed scope of the ATRIUM 11 post-irradiation examination is provided in the table below. The final program will be developed and coordinated with the host utilities.

Table 5-1 Post-Irradiation Examination of ATRIUM 11 LTAs

Inspection Description	Scope	Comments
1. Visual Inspection – Fuel channel removed – Brush off loose crud	Up to two bundles after each cycle	High resolution video and photography
2. Extended Visual – Lift select fuel rods	Several peripheral rods in one bundle after first and last cycle	Remove upper tie plate and lift select rods to allow view of elevations within spacer grids
3. Fuel Rod Oxide and Profilometry – Extract select rods	Several rods in one or two bundles after first and last cycle	Set aside one or more bundles after first cycle if inspections cannot be conducted during refueling outage
4. Fuel Rod Growth	Within one bundle after first and last cycle (using same bundle)	Remove select fuel rods and install reference for rod measurements
5. Fuel Bundle Growth	Up to two bundles after first and last cycle (using same bundles)	Track Water Channel data
6. Channel oxide	Measure up to two channels after 2nd and 3rd cycles	Track Z4B-BQ Fuel Channel data
7. Channel bow and bulge	Measure channels after 2nd and 3rd cycles	Track Z4B-BQ Fuel Channel data
8. Rod-to-rod spacing	At least one bundle per cycle.	Examination of Rod-to-Rod spacing. Qualitative first cycle (measurement of obvious gap closures) and quantitative for remaining cycles.
9. Poolside Gamma Scan	Several rods after first and last cycles	Can be conducted in sequence with Profilometry

6.0 REFERENCES

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4. ANP-3354 Revision 2, *Browns Ferry Unit 2 Cycle 19 Reload Analysis*, AREVA Inc., July 2015.
5. EMF-CC-074(P)(A) Volume 4 Revision 0, *BWR Stability Analysis - Assessment of STAIF with Input from MICROBURN-B2*, Siemens Power Corporation, August 2000.
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