ATTACHMENT 29

ANP-3388NP, Fuel Rod Thermal-Mechanical Evaluation for Browns Ferry Extended Power Uprate (Non-Proprietary)

AREVA

Fuel Rod Thermal-Mechanical Evaluation for Browns Ferry Extended Power Uprate

ANP-3388NP
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Licensing Report

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Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	All	Initial Issue

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Nomenclature

Acronym	Definition
AOO	anticipated operational occurrences
ASME	American Society of Mechanical Engineers
B&PV	Boiler and Pressure Vessel
BOL	beginning of life
BWR	boiling water reactor
CGU	commercial grade uranium
CRWE	control rod withdrawal error
CUF	cumulative usage factor
EOL	end of life
EPU	extended power uprate
FDL	fuel design limit
ID	inside diameter
MWd/kgU	megawatt days per kilogram of initial uranium
LFA	lead fuel assembly
LHGR	linear heat generation rate
NRC	Nuclear Regulatory Commission, U. S.
OD	outside diameter
OLTP	original licensed thermal power
PCI	pellet-to-cladding-interaction
PLFR	part length fuel rod
ppm	parts per million
SRA	stress relieved annealed
S-N	stress amplitude versus number of cycles
UTL	upper tolerance limit

1.0 INTRODUCTION

Results of fuel rod thermal-mechanical analyses are presented to demonstrate that the applicable design criteria are satisfied. The analyses are for the AREVA Inc. (AREVA) ATRIUM™ 10XM* fuel. These evaluations assess fuel rod performance at Extended Power Uprate (EPU) conditions that are assumed to first occur in Cycle 19 of Browns Ferry Unit 3. This unit is to be considered as a proxy for all three units. A first batch of ATRIUM 10XM fuel is inserted in Cycle 18, which is assumed to operate at the currently licensed thermal power (105% of the Original Licensed Thermal Power—OLTP), and a second batch of ATRIUM 10XM fuel is inserted in Cycle 19. For Cycle 19, as well as subsequent cycles, the thermal power is assumed to 120% of OLTP. The evaluations are based on methodologies and design criteria approved by the U. S. Nuclear Regulatory Commission (NRC). Performance for cycles beyond Cycle 19 is assessed using an equilibrium cycle comprised exclusively of ATRIUM 10XM fuel assemblies and operating at EPU conditions.

The analysis results are evaluated according to the generic fuel rod thermal and mechanical design criteria contained in ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 1) along with design criteria provided in the RODEX4 fuel rod thermal-mechanical topical report (Reference 2). The cladding external oxidation limit was reduced according to a regulatory commitment made to the NRC when RODEX4 was first implemented (Reference 3).

The RODEX4 fuel rod thermal-mechanical analysis code is used to analyze the fuel rod for fuel centerline temperature, cladding strain, rod internal pressure, cladding collapse, cladding fatigue and external oxidation. The code and application methodology are described in the RODEX4 topical report (Reference 2). The cladding steady-state stress and plenum spring design methodology are summarized in Reference 1.

The fuel rod design is very similar to that used for the current ATRIUM-10 design in the Browns Ferry units. The fuel rod outside diameter is approximately **[ ]** than the ATRIUM-10 fuel rod and the cladding diameter and pellet diameter were scaled in a way that preserves the extensive operating experience and performance history of the ATRIUM-10 rod design. Also, the rod design is nearly identical to the design used for the first U.S. ATRIUM

* ATRIUM is a trademark of AREVA Inc.

10XM Lead Fuel Assemblies, the reload fuel currently supplied to two other U.S. BWR/4 reactors, and a first reload of ATRIUM 10XM fuel planned for insertion into Cycle 19 of Browns Ferry Unit 2. These ATRIUM 10XM fuel assemblies are loaded with pellets composed of either [ ] powder for the manufacture of the fuel pellets or Commercial Grade Uranium (CGU) powder. [

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The following sections describe the fuel rod design, design criteria and methodology with reference to the source topical reports. Results from the analyses are summarized for comparison to the design criteria.

2.0 SUMMARY AND CONCLUSIONS

Table 2-1 compares the results of the licensing basis analysis for ATRIUM 10XM fuel for each of the design criteria at full-power operation (120% OLTP). For each criterion, the more limiting of the cycle-specific or equilibrium cycle results are shown. Results for each of the contributory scenarios, whether from steady state operation or steady-state with AOO transients, are shown in Section 3.0.

The analyses support a maximum fuel rod discharge exposure of 62 MWd/kgU.

Fuel rod criteria applicable to the design are summarized in Section 3.0. Analyses show the criteria are satisfied when the fuel is operated at or below the LHGR (linear heat generation limit) presented in Figure 2-1.

Table 2-1 Summary of ATRIUM 10XM Fuel Rod Design Evaluation Results

Criteria Section*	Description	Criteria	Result, Margin† or Comment
3.2	**Fuel Rod Criteria**		
3.2.1	Internal hydriding	**[**	**]**
(3.1.1)	Cladding collapse	**[**	**]**
(3.1.2)	Overheating of fuel pellets	No fuel melting margin to fuel melt > 0. °C	**[ ]**
3.2.5	Stress and strain limits		
(3.1.1) (3.1.2)	Pellet-cladding interaction	**[**	**]**
3.2.5.2	Cladding stress	**[**	**]**
3.3	**Fuel System Criteria**		
(3.1.1)	Fatigue	**[**	**]**
(3.1.1)‡	Oxidation, hydriding, and crud buildup	**[**	**]**
(3.1.1) (3.1.2)	Rod internal pressure	**[**	**]**
3.3.9	Fuel rod plenum spring (fuel handling)	Plenum spring to **[**	**]**

* Numbers in the column refer to paragraph sections in the generic design criteria document, ANF-89-98(P)(A) Revision 1 and Supplement 1 (Reference 1). A number in parentheses is the paragraph section in the RODEX4 fuel rod topical report (Reference 2).

† Margin is expressed as (limit – result)

‡ The cladding external oxidation limit is restricted to **[ ]** µm by Reference 3.

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Figure 2-1 ATRIUM 10XM LHGR Limit (Normal Operation)

3.0 FUEL ROD DESIGN EVALUATION

Summaries of the design criteria and methodology are provided in this section along with analysis results in comparison to criteria. Both the fuel rod criteria and fuel system criteria as directly related to the fuel rod analyses are covered.

The fuel rod analyses cover normal operating conditions and AOOs (anticipated operational occurrences). The fuel centerline temperature analysis (overheating of fuel) and cladding strain analysis take into account slow transients at rated operating conditions.

Other fuel rod-related topics of overheating of cladding, cladding rupture, fuel rod mechanical fracturing, rod bow, axial irradiation growth, cladding embrittlement, violent expulsion of fuel and fuel ballooning are evaluated as part of the respective fuel assembly structural analysis, thermal hydraulic analyses, or LOCA analyses and are reported elsewhere. The evaluation of fast transients and transients at off-rated conditions also are reported separately.

3.1 Fuel Rod Design

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] plenum spring on the upper end of

the fuel column **[**

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Table 3-1 lists the main parameters for the fuel rod and components.

3.2 Summary of Fuel Rod Design Evaluation

Results from the analyses are listed in Table 3-2 through Table 3-4. Summaries of the methods and codes used in the evaluation are provided in the following paragraphs. The design criteria also are listed along with references to the sections of the design criteria topical reports (References 1 and 2).

The fuel rod thermal and mechanical design criteria are summarized as follows.

- **Internal Hydriding**. The fabrication limit **[** **]** to preclude cladding failure caused by internal sources of hydrogen (Section 3.2.1 of Reference 1).
- **Cladding Collapse**. Clad creep collapse shall be prevented. **[** **]** (Section 3.1.1 of Reference 2).
- **Overheating of Fuel Pellets**. The fuel pellet centerline temperature during anticipated transients shall remain below the melting temperature (Section 3.1.2 of Reference 2).

- **Stress and Strain Limits**. **[ ]** during normal operation and during anticipated transients (Sections 3.1.1 and 3.1.2 of Reference 2).

  Fuel rod cladding steady-state stresses are restricted to satisfy limits derived from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code (Section 3.2.5.2 of Reference 1).

- **Cladding Fatigue**. The fatigue cumulative usage factor for clad stresses during normal operation and design cyclic maneuvers shall be below **[ ]** (Section 3.1.1 of Reference 2).

- **Cladding Oxidation, Hydriding and Crud Buildup**. Section 3.1.1 of Reference 2 limits the maximum cladding oxidation to less than **[ ]** μm to prevent clad corrosion failure. The oxidation limit is further reduced to **[ ]** μm consistent with a regulatory commitment made to the NRC during the first application of the RODEX4 methodology (Reference 3).

- **Rod Internal Pressure**. The rod internal pressure is limited **[ ]** to assure that significant outward clad creep does not occur and unfavorable hydride reorientation on cooldown does not occur (Section 3.1.1 of Reference 2).

- **Plenum Spring Design (Fuel Handling)**. The rod plenum spring must maintain a force against the fuel column stack **[ ]** (Section 3.3.9 of Reference 1).

The cladding collapse, overheating of fuel, cladding transient strain, cladding cyclic fatigue, cladding oxidation, and rod pressure are evaluated **[ ]**. Cladding stress and the plenum spring are evaluated on a design basis.

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 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 design by limiting the gap formation due to fuel densification subsequent to pellet-clad contact. The size of the axial gaps which may form due to densification following first pellet-clad contact shall be less than **[ ]**.

The evaluation is performed using RODEX4. The design criterion and methodology are described in Reference 2. RODEX4 takes into account the **[**

]. A brief overview of RODEX4 and the statistical methodology is provided in the next section.

Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.3 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 content. AREVA establishes an LHGR limit to protect against fuel centerline melting during steady-state operation and during AOOs.

Fuel centerline temperature is evaluated using the RODEX4 code (Reference 2) 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 the 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 **[**

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The heat conduction in the fuel and clad is **[**

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Mechanical processes include **[**

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

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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 – **[** **]**.

Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.4 Stress and Strain Limits

3.2.4.1 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 2. See Section 3.2.3 for an overview of the code and method. **[**

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Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.4.2 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 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 cladding-to-end cap weld stresses are evaluated for loadings from differential pressure, differential thermal expansion, rod weight, and plenum spring force.

The design limits are derived from the ASME (American Society of Mechanical Engineers) Boiler and Pressure Vessel (B&PV) Code Section III (Reference 4) and the minimum specified material properties.

Table 3-4 lists the results in comparison to the limits for hot, cold, BOL and EOL conditions.

3.2.5 Fuel Densification and Swelling

Fuel densification and swelling are limited by the design criteria for fuel temperature, cladding strain, cladding collapse, and rod internal 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.

3.2.6 Fatigue

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]. 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.

Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.7 Oxidation, Hydriding, and Crud Buildup

Cladding external oxidation is calculated using RODEX4. Section 3.2.3 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 **[** **]**.

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In the event abnormal crud is discovered or 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 a plant experiences higher corrosion instead of crud.

Eddy current liftoff measurements at the Browns Ferry units **[**

] for all of the Browns Ferry units.

The maximum oxide on the fuel rod cladding shall not exceed **[ ]** µm. Previously, a **[ ]** µm limit was approved as part of the RODEX4 methodology (Reference 2). Concerns were raised on the effect of non-uniform corrosion, such as spallation, and localized hydride formations on the ductility limit of the cladding. As a result, a regulatory commitment was made to reduce the limit to **[ ]** µm (Reference 3).

Currently, there is **[ ]**. However, as noted above, the **[ ]** µm was established, in part, as a means of **[**
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The oxide limit is evaluated such that greater than **[**
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Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.8 Rod Internal Pressure

Fuel rod internal pressure is calculated using the RODEX4 code and methodology as described in Reference 2. Section 3.2.3 provides an overview of the code and method. The maximum rod pressure is calculated under steady-state conditions and also takes into account slow transients. Rod internal pressure is limited to **[** **]**. The expected upper bound of rod pressure **[** **]** is calculated for comparison to the limit.

Table 3-2 and Table 3-3 list the results for equilibrium and cycle-specific conditions, respectively.

3.2.9 Plenum Spring Design (Fuel Assembly Handling)

The plenum spring must maintain a force against the fuel column to **[** **]**. This is accomplished by designing and verifying the spring force in relation to the fuel column weight. The plenum spring is designed such that the **[** **]**.

Table 3-1 Key ATRIUM 10XM Fuel Rod Design Parameters

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Table 3-2 RODEX4 ATRIUM 10XM Fuel Rod Results for Equilibrium Cycle Conditions

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* Margin is defined as (limit – result).

Table 3-3 RODEX4 ATRIUM 10XM Fuel Rod Results for Browns Ferry 3 Cycle 19 Operation*

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Table 3-4 ATRIUM 10XM Cladding and Cladding-End Cap Steady-State Stresses

Description, Stress Category	Criteria	Result		
Cladding stress		[		]
P_m (primary membrane stress)	[			]
$P_m + P_b$ (primary membrane + bending)	[			]
P + Q (primary + secondary)	[			]
Cladding-End Cap stress				
$P_m + P_b$	[			]

* Note that Cycle 19 results are provided up to the end of Cycle 19. The minimum margin for the Cycle 18 and Cycle 19 batches are shown.

† Fatigue result is extrapolated to three cycles of operation based on the Cycle 19 result.

4.0 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. BAW-10247PA Revision 0, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, AREVA NP Inc., February 2008.

3. Letter from Farideh E. Saba (NRC) to Joseph W. Shea (TVA), "BROWNS FERRY NUCLEAR PLANTS, UNITS 1, 2 AND 3 – ISSUANCE OF AMENDMENTS REGARDING TECHNICAL SPECIFICATION (TS) CHANGE TS-478 ADDITION OF ANALYTICAL METHODOLOGIES TO TS 5.6.5 AND REVISION OF TS 2.1.1.2 FOR UNIT 2 (TAC NOS. MF0878 AND MF0879), ML14108A334," dated July 31, 2014.

4. *ASME Boiler and Pressure Vessel Code*, Section III, "Rules for Construction of Nuclear Power Plant Components," 1977.

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