framatome

Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors
Supplement 2: Mechanical Methods

BAW-10247NP-A
Supplement 2NP-A
Revision 0

Topical Report

August 2018

Framatome Inc.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 15, 2018

Mr. Gary Peters, Director
Licensing and Regulatory Affairs
Framatome Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: FINAL SAFETY EVALUATION FOR TOPICAL REPORT BAW-10247P-A, SUPPLEMENT 2P, REVISION 0, "REALISTIC THERMAL-MECHANICAL FUEL ROD METHODOLOGY FOR BOILING WATER REACTORS, SUPPLEMENT 2: MECHANICAL METHODS" (CAC NO. MF7708; EPID L-2016-TOP-0005)

Dear Mr. Peters:

By letter dated April 29, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16124B057), Framatome Inc. (Framatome) (formerly AREVA Inc.) submitted Topical Report (TR) BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors, Supplement 2: Mechanical Methods," to the U.S. Nuclear Regulatory Commission (NRC) staff for review and approval. By letter dated October 26, 2017 (ADAMS Accession No. ML17244A077), an NRC draft safety evaluation (SE) regarding our approval of TR BAW-10247P-A, Supplement 2P, Revision 0, was provided for your review and comment. By letter dated January 19, 2018 (ADAMS Accession No. ML18024A459), Framatome provided comments on the draft SE. The NRC staff's disposition of the Framatome comments on the draft SE are discussed in the attachment (ADAMS Accession No. ML18040B298) to the final SE enclosed with this letter.

The NRC staff has found that TR BAW-10247P-A, Supplement 2P, Revision 0, is acceptable for referencing in licensing applications for nuclear power plants to the extent specified and under the limitations delineated in the TR and in the enclosed final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in licensing action requests, our review will ensure that the material presented applies to the specific plant involved. Requests for licensing actions that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that Framatome publish approved proprietary and non-proprietary versions of TR BAW-10247P-A, Supplement 2P, Revision 0, within 3 months of receipt of this letter. The approved versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The approved versions shall include an "-A" (designating approved) following the TR identification symbol.

As an alternative to including the RAIs and RAI responses behind the title page, if changes to the TR were provided to the NRC staff to support the resolution of RAI responses, and if the NRC staff reviewed and approved those changes as described in the RAI responses, there are two ways that the accepted version can capture the RAIs:

1. The RAIs and RAI responses can be included as an Appendix to the accepted version.
2. The RAIs and RAI responses can be captured in the form of a table (inserted after the final SE) which summarizes the changes as shown in the approved version of the TR. The table should reference the specific RAIs and RAI responses which resulted in any changes, as shown in the accepted version of the TR.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, Framatome will be expected to revise the TR appropriately or justify its continued applicability for subsequent referencing. Licensees referencing this TR would be expected to justify its continued applicability or evaluate their plant using the revised TR.

Sincerely,

Dennis C. Morey, Chief
Licensing Processes Branch
Division of Licensing Projects
Office of Nuclear Reactor Regulation

Project No. 728

Docket No. 99902041

Enclosure:
Final Safety Evaluation (Proprietary)

RESOLUTION OF COMMENTS BY THE OFFICE OF NUCLEAR REACTOR REGULATION ON DRAFT SAFETY EVALUATION FOR TOPICAL REPORT BAW-10247P-A, SUPPLEMENT 2P, REVISION 0, "REALISTIC THERMAL-MECHANICAL FUEL ROD METHODOLOGY FOR BOILING WATER REACTORS SUPPLEMENT 2: MECHANICAL METHODS"

FRAMATOME, INC.

PROJECT NO. 728/DOCKET NO. 99902041

This attachment provides the U.S. Nuclear Regulatory Commission (NRC) staff's review and disposition of the comments made by Framatome Inc. (formerly AREVA Inc.) on the draft safety evaluation for Topical Report BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods."

Page	Line	Proposed Change/Comment	NRC Response
		Please revise the draft safety evaluation to reflect approval of the use of Zry-2 recrystallized material for cladding and Z4B^{TM} material for water channels.	
		Please revise the burnup limitation for fuel assembly growth to be consistent with the additional information to be provided as discussed in the letter accompanying this summary table.	
1	24	Delete highlighted proprietary brackets, information is not proprietary	Brackets used to show inserted material, not proprietary information. Brackets not deleted as suggested in final SE.
2	22	Delete highlighted proprietary brackets, information is not proprietary	Brackets used to show inserted material, not proprietary information. Brackets not deleted as suggested in final SE.
3	5	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
3	26	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
3	29	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
3	34	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
3	35	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified

Page	Line	Proposed Change/Comment	NRC Response
			accordingly.
3	39	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
4	1	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
4	30	Delete highlighted proprietary brackets, information is not proprietary	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
4	38	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
4	41	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
5	31 to 34	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
5	36	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
5	39	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
6	15	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
6	39	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
6	42	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
6	43	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
7	1	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
7	8 to 11	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
7	37-38	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
8	17-18	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.

Page	Line	Proposed Change/Comment	NRC Response
8	20-21	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
8	29	Suggest replacing "non-ATRIUM" with "tie rod"	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
8	42	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
9	1	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
9	7	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
9	12	Suggest deleting "ATRIUM type"	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
9	16	Suggest deleting "ATRIUM type"	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
9	17-18	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
9	18-19	Suggest deleting "ATRIUM type"	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
9	20-21	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
10	1-2	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
10	5	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
10	7	Suggest deleting "ATRIUM type"	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
10	8-9	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
A1	5-6	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
A1	20	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
A3	14-	Highlighted information is	NRC staff agrees with suggested

Page	Line	Proposed Change/Comment	NRC Response
	15	proprietary, add proprietary brackets	editorial changes. Final SE modified accordingly.
A7	24	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
A9	18	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
A9	25	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
A10	1	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
A10	16	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
A10	24	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.
A10	26	Highlighted information is proprietary, add proprietary brackets	NRC staff agrees with suggested editorial changes. Final SE modified accordingly.

FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT BAW-10247P-A, SUPPLEMENT 2P, REVISION 0,

"REALISTIC THERMAL-MECHANICAL FUEL ROD METHODOLOGY FOR BOILING WATER REACTORS SUPPLEMENT 2: MECHANICAL METHODS"

PROJECT NO. 728/DOCKET NO. 99902014

1.0 INTRODUCTION

By letter dated April 29, 2016, Framatome Inc. (Framatome, formerly AREVA Inc.) submitted Topical Report (TR) BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," (Supplement 2P) for U.S. Nuclear Regulatory Commission (NRC) review and approval (Reference 6). The intent of this supplement is to replace the following legacy TRs:

- EMF-85-74(P), Revision 0, Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Model" (Reference 3) (clarification of exposure limit)
- XN-NF-32(P)(A), Supplements 1, 2, 3, & 4, "Computational Procedure for Evaluating Fuel Rod Bowing" (Reference 5)

Approval of this supplement allows Framatome to remove these legacy TRs when RODEX4 methods, described in BAW-10247P-A, Revision 0, are used for licensing in applications with Framatome boiling water reactor (BWR) fuel designs (Reference 1). Framatome further explains that "the [Supplement 2P] methodology includes peripheral mechanical methods which do not use any thermal-mechanical code such as RODEX4." That is, the mechanical methods described in Supplement 2P have no connection to the thermal-mechanical models previously approved in the base TR as implemented via the RODEX4 fuel performance code. As explained by Framatome, Supplement 2P is meant to provide "a well-defined licensing basis for BWR nuclear plants which have moved to AREVA's realistic fuel rod methodology."

Framatome also notes that "this supplement does not introduce any changes to AREVA's existing BWR methodology other than updates to correlations derived from operating experience data."

By letter dated February 23, 2018 (Reference 8), Framatome provided additional information, with appropriate TR updates (Reference 9), to provide the following:

- Clarification regarding the use of the mechanical models for re-crystallized annealed (RXA) Zircaloy-2 (Zry-2) cladding.
- A description of the material properties of Z4B™ and the mechanical analysis performed for the Z4B™ internal water channel assembly, and

Enclosure

- Additional Z4B™ water channel growth data to support extension of the fuel assembly growth correlation to a burnup of [ ].

Consequently, the NRC staff's review, discussed in detail in Section 3.0 below, is focused on ensuring the acceptability of:

- Updated correlations based on updated databases,
- The applicability of updated correlations to Framatome BWR fuel designs,
- The material properties of Z4B™ and mechanical analysis performed for the Z4B™ internal water channel assembly, and
- Updated mechanical methods compatibility with downstream safety analyses.

2.0 REGULATORY EVALUATION

The fuel system consists of arrays of fuel rods, including fuel pellets and tubular cladding, spacer grids, end plates, and reactivity control rods. The objectives of the fuel system safety review are to provide assurance that (1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs), (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. The NRC staff acceptance criteria are based on the NUREG-0800, Standard Review Plan (SRP), Section 4.2, "Fuel System Design." These criteria include three parts: (1) design bases that describe specified acceptable fuel design limits (SAFDLs) as depicted in General Design Criterion (GDC) 10 to Appendix A of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, (2) design evaluation that demonstrates that the design bases are met, and (3) testing, inspection, and surveillance plans that show that there are adequate monitoring and surveillance of irradiated fuel. The design bases include fuel system damage, fuel rod failure, and fuel coolability.

Framatome states that "the mechanical methods covered in this supplement will be limited to those that establish the design bases for the acceptance criteria as provided in SRP Section 4.2, II.1.A, "Fuel System Damage.""

Furthermore, Supplement 2P, Section 1.0, "Introduction," states that the "[Supplement 2P] methods are consistent with the underlying methods supporting the design criteria approved for generic application in Reference 3." Reference 3 of Supplement 2P is ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs" (Reference 2), which describes BWR thermal-mechanical, thermal-hydraulic, accident analysis, and nuclear licensing criteria. The purpose of the Framatome BWR licensing criteria or limits is to provide limiting values that prevent fuel damage or failure with respect to each damage mechanism. These previously approved licensing criteria or limits, along with certain definitions for fuel failure, constitute the SAFDLs required by GDC 10.

3.0 TECHNICAL EVALUATION

Since only correlations supporting the fuel rod bow, fuel rod axial growth, and fuel assembly axial growth methods have changed from the previously approved methodologies, these specific correlations were the focus of the NRC staff's review. Z4B™ material properties and the Z4B™ internal water channel mechanical analysis are also reviewed to confirm consistency with previously approved mechanical design criteria for Framatome BWR fuel designs as documented in Reference 2.

BWR Fuel Rod Bow Correlation

The applicable BWR thermal-hydraulic design criterion that is relevant to Supplement 2P is thermal margin performance (Reference 2), which is affected by fuel rod bowing.

The updated rod bow correlation provided in Supplement 2P, Appendix A, represents a deviation from the previously approved rod bow gap closure model documented in XN-75-32(P)(A), Supplement 1, Section 3.0, "Rod Bow Measurements." The updated correlation is now based on post-irradiation examination (PIE) measurements of rod-to-rod spacing from both legacy fuel designs (e.g., 7x7, 8x8, and 9x9) as well as the ATRIUM 10 fuel design, whereas the previous model [ ][1]. This is acceptable as there is [ ] based on the data presented in Figure A-1.

The statistical methodology for determining the [ ] is mostly consistent with what was previously approved in XN-75-32(P)(A), Supplement 1, Section 3.0. That is, gap closure measurements are first taken for each grid-to-grid span. Then, [ ]. As was previously approved, the correlation is based on [ ], which is conservative. Request for additional information (RAI) Question #2 (RAI 2, Reference 7) discussed concerns with rod growth data dependence based on the observation of distinct clusters in the data presented in Figure A-1 of Supplement 2P. However, Framatome clarified that the data shown in Figure A-1 is not representative of the data used to generate the correlation, which again, is based on [ ]. Therefore, data dependence is not an issue.

Differences in the new correlation compared to the old correlation include: (1) Fitting the data [ ] and (2) removal of the "1.5 multiplier to account for batch-batch variations." The NRC staff notes that the fuel rod data fit change reflects the preferable functional form at the time of XN-NF-32(P)(A), Supplements 1-4, approval. Removal of the 1.5 multiplier is also acceptable as batch-batch variability is already accounted for by basing the new methodology for correlation development on measurement data from several different BWR fuel designs with data from different reactors for a given fuel design.

The NRC staff documented the issues with the [ ] in RAI 2. However, based on the response to RAI 3 highlighting the conservatism of the fuel rod bow penalty, it is apparent that

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[1] The spacing measurements are translated into relative gap closure in percent.

substantial conservatism exists in downstream safety analyses to offset any potential issues identified in developing the fuel rod bow correlation [ ].

Framatome states that "when fuel rod-to-rod gap measurements are not taken, fuel rods are typically visually inspected for any signs of abnormal bow behavior." Framatome discusses how ATRIUM 11 lead test assemblies (LTAs) have shown to be "well-represented by the database as shown in recent examinations" for various assembly-average burnups ranging from [ ]. Visual inspections over a sufficient range of assembly-average burnups are acceptable to confirm that abnormal bow behavior is not occurring in LTAs; however, measurement data should be added to the database, and evaluated based on the specified surveillance frequency as discussed in the RAI 10c response to determine whether or not the [ ] needs to be updated.

Based on its review above, the NRC staff has reasonable assurance that the updated rod-to-rod gap closure correlation will provide an appropriate estimation of fuel rod-to-rod gap closure to be used as an input to minimum critical power ratio evaluations for all current and future Framatome BWR fuel designs up to an [ ] provided that the change process[2] described in Supplement 2P, Section 5.0, "Change Process," is followed.

Axial Irradiation Growth Correlations

The applicable BWR thermal-mechanical design criterion that is relevant to Supplement 2P relates to axial irradiation growth (Reference 2)[3]. The previously approved BWR fuel design licensing criteria requires that the fuel rods and other assembly components maintain clearances and engagements in the fuel assembly structure throughout the lifetime of the fuel up to currently approved burnup levels. As stated by Framatome, "the loss of clearance between the fuel rod and the upper tie plate has the potential to affect safety margins since interference may cause additional rod bow and lead to fuel failures."

During its review, the NRC staff noted that the analysis previously performed in EMF-85-74(P)(A), Supplement 2, Revision 0, evaluating the loss of lower tie plate (LTP) seal spring engagement with the channel was based on an upper limit of fuel assembly growth, which was not included as part of Supplement 2P. The NRC staff issued RAI 9 requesting an explanation for why the upper limit fuel assembly growth curve was not provided in Supplement 2P. The safety concern was with respect to the loss of engagement of the LTP seal spring, which limits the bypass coolant leakage rate between the LTP and fuel channel, potentially affecting thermal-hydraulic performance. The response to RAI 9 clarifies that loss of LTP seal spring engagement with the channel is still considered during design; however, it has been shown that additional leakage from losing seal spring engagement at end-of-life (EOL) is not enough to affect safety margins (Reference 4).

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[2] Additional clarification regarding the change process is provided in the responses to RAI 10.

[3] Supplement 2P describes both fuel rod and assembly internal water channel growth.

Framatome also clarifies in Reference 13 that:

> Per ANF-89-98(P)(A) Revision 1, external interfaces, including channel spacer/springs, are evaluated for all new fuel designs and for compatibility with co-resident fuel assemblies. When verifying channel spacer/spring engagement, the EOL upper tolerance value is derived from the nominal fuel assembly growth correlation presented in [Supplement 2P] Table C-1 by adding the tolerance value, T. This methodology confirms engagement between adjacent fuel assemblies considering [ ] and EOL axial growth conditions.

Consequently, the NRC staff finds it acceptable to use and maintain the upper limit fuel assembly growth curve to support external water channel spacer/spring engagement determination (i.e., by ensuring that the distance to loss of engagement is greater than or equal to T)[4], as part of the revised mechanical methods in Supplement 2P.

Regarding the remaining axial irradiation growth models, Framatome explains:

> To evaluate the minimum EOL clearance between the fuel rod and tie plates it is necessary to determine correlations for the fuel rod growth and the fuel assembly growth derived from post-irradiation length measurements. The initial nominal clearance between the fuel rod and upper tie plate can then be reduced by an accounting of fabrication tolerances and uncertainty in the growth correlations. This determines the design margin for growth.

Hence, the two correlations that have been developed to calculate the minimum EOL clearance between fuel rods and tie plates is a **[**] for fuel rod growth and a **[** **]** for fuel assembly growth. This is appropriate because the maximum fuel rod growth is subtracted from the minimum fuel assembly growth, thus ensuring that a clearance is maintained between the fuel rods and the assembly upper tie plate (UTP) at EOL.[5]

BWR Fuel Rod Growth Correlation

The original stress relief annealed (SRA) Zry-2 fuel rod growth correlation was based on data available up to 1998. The correlation has been updated in Supplement 2P and is now based on a more comprehensive database including PIE data since 1998. Comparing the pre-1998 data with the post-1998 data, the post-1998 data is within the scatter of the pre-1998 data.

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[4] This is equivalent to ensuring that the distance to loss of engagement is greater than or equal to the difference between the **[** **]** for the fuel assembly growth curve (or internal water channel growth curve) and the nominal fuel assembly growth curve.

[5] Note that the **[**

] However, the Supplement 2P update to the fuel assembly growth correlation does not apply to ATRIUM type fuel assemblies where assembly growth is controlled by tie-rods.

Data includes fuel rods from 7x7, 8x8, 9x9, and 10x10 arrays. Framatome notes that [ ]. Framatome explains that "these [ ]." Framatome further states that "this [ ]." Although the Supplement 2P data appears to be predictable based on burnup alone, growth of SRA Zry-2 depends on factors such as the amount of cold work (i.e., manufacturing process) and the presence of hydrogen or hydrides due to corrosion. Consequently, RAI 10a was issued to understand why the Supplement 2P correlation will be adequate to bound future fuel rod designs that may have different manufacturing processes, plant water chemistry, etc. The RAI 10a response clarifies that the fuel rod growth correlations are adequate to bound future fuel rod designs that are within the range of parameters defined by the existing database supporting the fuel rod growth correlations. Framatome states that any significant deviations in manufacturing processes, plant water chemistry, fuel designs, or the addition of new materials will require out of pile testing and lead test programs to acquire the necessary data to support evaluation methods. These types of changes would need to be submitted to the NRC for review and approval. Any performance drift caused by the accumulation of small changes in design or operation will be captured by Framatome's ongoing surveillance program by the continual collection of PIE data for assessment of growth correlation validity.

A [ ] model is used to estimate the best fit of the data; based on the observed fit in Figures B-1 and B-2, this functional form is appropriate and is consistent with the previously approved model in EMF-85-74(P)(A), Supplement 2, Revision 0 (Reference 3). The Siemens Power Corporation 7x7, 8x8, and 9x9 data is still used and is justified since [ ]

The new [ ] was initially found to be inappropriate based on the issues identified in RAI 2.[6] The main issue not clearly addressed in the RAI 2 response was why the current correlation remains valid in light of the strong data dependence. Framatome explains that there is conservatism in the correlation based on [ ]; however, the NRC staff does not agree that this treatment is completely conservative given that [ ], which would be non-conservative. Additionally, this does not address the data dependence concern, which if unaccounted for could manifest as an under prediction of the proposed [ ] due to underestimation of the uncertainty about the regression fit.

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[6] The data also appears to be [ ], which can also invalidate uncertainty estimates. The NRC staff thought this was possibly due to biased sampling. That is, undersampling lower burnup assemblies as evidenced by the number of points per assembly in Figure B-1 at higher burnup compared to lower burnup. This undersampling was also confirmed in the Appendix (see "Average samples per burnup bin"), but the NRC staff found that Framatome's proposed [ ] remained valid nonetheless.

To address the above concerns, the NRC staff performed a confirmatory analysis[7] using only the median value of the assembly-wise rod growth data to remove dependency amongst measurements from the same assembly at a given burnup, and to remove the non-conservative bias from assigning a higher burnup to lower burnup rods for a given fuel assembly.[8]

Based on the response to RAI 2, regarding development of a [ ] versus a [ ], and the NRC confirmatory analysis using the rod growth data supplied in the response to RAI 2f, the NRC staff has determined that the [ ] used, and subsequent correlation developed, is acceptable to be used for fuel rod to UTP clearance determination.

Regarding the possibility of a non-conservatively biased fuel rod growth database, Framatome explains in the RAI 4a response that:

> Generally, fuel assemblies are not necessarily chosen for the expressed purpose of taking fuel rod growth measurements. Fuel assemblies are generally chosen for a number of reasons. For example, assemblies could be chosen because they are limiting for corrosion after water chemistry changes, or because they are useful to explore the boundary of operating experience (e.g., burnup, time, fluence, etc.) in healthy fuel exams, or just because they are lead assemblies. Therefore, any bias would be conservative, i.e., toward the measurement of assemblies viewed as limiting in some aspect.

The RAI 4a response also states that all fuel rods in a fuel assembly are not always measured. Based on the fuel rod growth data provided in the RAI 2f response, it appears that [

].

To assess the impact of a potential bias in the proposed correlation, potentially caused by not sampling all fuel rods in a given assembly, the NRC staff performed a series of calculations whereby [ ] were generated and then compared for a series of cases where successively more assembly-wise data was removed starting with those data from fuel assemblies [ ]. By removing assembly-wise data with fewer measurements, this allows for a measure of [ ] sensitivity to potentially biased datasets. If there is a large [ ] sensitivity, then this may be indicative of a bias in Framatome's proposed [ ]; however, if little or no sensitivity is observed, this would provide reasonable assurance that there is no bias due to taking an insufficient number of samples from the population.

The cases analyzed were with [

].

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[7] See "Using the median value of the assembly-wise data" in the Appendix.

[8] Note that there are a few assemblies included in the fuel rod growth dataset that were measured at more than one fuel assembly average burnup – an attempt to treat this dependency was not considered.

There was minor variation in the [ ] when comparing these sets, and all were bounded by the [ ] proposed by Framatome. The minor variation between [ ] is a strong indicator that any bias that may exist is not likely to affect the proposed [ ] for rod growth, therefore the NRC staff finds the proposed [ ] for rod growth acceptable.

Framatome is also requesting approval of this updated correlation for [ ] fuel pellets with an [ ] applied to cover the [ ]. Framatome provides a mechanism for the [ ]." However, without data similar to that in Figure B-2 of Supplement 2P for [ ], this claim can't be quantitatively validated. In the response to RAI 6, Framatome explains that [ ].

Based on its review above, the NRC staff has reasonable assurance that the updated fuel rod growth correlation will provide an appropriate estimation of fuel rod growth to be used in determining a clearance between the fuel rod and UTP for all current and future Framatome BWR fuel designs that use Zry-2 cladding material up to an [ ] provided that the change process[9] described in Supplement 2P, Section 5.0, "Change Process," is followed.

BWR Fuel Assembly Growth Correlations

An updated assembly growth correlation is proposed:

> ...to be applied for the evaluation of AREVA BWR fuel assemblies where the axial growth is controlled by a central water channel made from a zirconium alloy. The new correlation is based on ATRIUM type fuel assembly growth data only, and excludes designs with load bearing tie rods as well as the European bundle in basket designs. The database includes water channels made from both Zircaloy-4 (Zry-4) and Z4B™ materials.

[9] Additional clarification regarding the change process is provided in the responses to RAI 10.

The measurement database only contains growth data from ATRIUM 10 and ATRIUM 11 internal water channels, therefore it is directly applicable to the intended application. Framatome states that they have not observed a [ ].

A [ ] model is used to estimate the best fit of the data; based on the observed fit in Figure C-1 of Supplement 2P, this functional form appears to be appropriate; however, it differs from the functional form of the previously approved model in Reference A.2 of EMF-85-74(P)(A), Supplement 2, Revision 0, which used a [ ] (Reference 3). The updated model only contains data from ATRIUM-10 and ATRIUM-11 fuel designs and no longer contains any data based on older SPC fuel channel designs, therefore the influence of tie rod fuel designs has been removed. The updated model is also more conservative compared to the previous model as seen by the [ ], which is [ ].

Additionally, the new [ ] was initially found to be inappropriate based on the issues identified in RAI 2. However, based on the response to RAI 2, regarding development of a [ ][10] versus a [ ], and a NRC staff confirmatory analysis, the NRC staff determined that the tolerance factor used and subsequent correlations developed are acceptable.

Based on its review above, the NRC staff has reasonable assurance that the updated fuel assembly growth correlations will provide for an appropriate estimation of fuel assembly growth to be used in determining: (1) external water channel spacer/spring engagement, and (2) clearances between fuel rods and UTPs for all ATRIUM type fuel designs up to ATRIUM 11 with Zry-4 or Z4B™ internal water channels up to an [ ] MWd/kgU provided that the change process[11] described in Supplement 2P, Section 5.0, "Change Process," is followed.

Z4B™ Material Properties

Framatome states that "a proprietary zirconium alloy has been developed, Z4B™, which optimizes the alloying element concentrations for improved corrosion and hydrogen pickup when used for a BWR fuel structural component." Framatome further states that: (1) Z4B™ is similar to Zircaloy-4 (Zry-4) except that Z4B™ has a slightly higher iron and chromium content, and (2) "the small differences in composition between Z4B™ and Zry-4 do not result in any significant differences in fabrication methods or processes."

---

[10] [ ]. The NRC staff confirmatory analysis in the Appendix, based on the assembly growth data supplied in RAI Response 2f, also provides additional confidence in the validity of the proposed correlation.

[11] Additional clarification regarding the change process is provided in the responses to RAI 10.

Operating experience in the open literature (Reference 10) supports Framatome's claim that increasing the iron and chromium content of niobium-free alloys such as Zry-4 by the amount specified in Supplement 2P, Appendix D, "Z4B™ Water Channel Assembly," has a corrosion rate reducing effect. Framatome also states that the improved corrosion and hydrogen uptake performance of Z4B™ relative to Zry-4 has been demonstrated through the Z4B™ spacer grid material test program and recent Z4B™ lead use fuel channel measurements (Reference 11). Based on past operating experience, as supported by the open literature, which shows improved corrosion and hydrogen pickup performance, the NRC staff finds the use of the Z4B™ acceptable for use as the internal water channel assembly material for Framatome BWR fuel assembly designs.

Z4B™ Internal Water Channel Assembly Mechanical Analysis

Supplement 2P, Appendix D, summarizes the relevant BWR design criteria applicable to the internal water channel assembly tie structure. The previously approved criteria in Reference 2 apply to:

- Stress, strain, or loading limits,
- Axial irradiation growth, and
- Fuel assembly handling.

The NRC staff confirmed that the criteria remain unchanged to those previously approved and therefore remain acceptable. Axial irradiation growth, or fuel assembly growth in this case, was discussed in detail in "Axial Irradiation Growth Correlations" above.

Framatome explains that due to the similarity of Zry-4 and Z4B™, differences in material properties such as elastic modulus, heat capacity, thermal expansion, thermal conductivity, density, etc., are negligible, therefore additional considerations with respect to these parameters are not taken into account. The NRC staff finds this explanation acceptable because the relatively small magnitude of the differences in material composition between Zry-4 and Z4B™ as provided in Supplement 2P, Appendix D, Table D-1, "Alloy Composition," are not expected to impact any of the relevant SRP acceptance criteria related to fuel system damage.

Framatome also explains that the minimum mechanical property requirements for Z4B™ as provided in Supplement 2P, Appendix D, Table D-4, "Unirradiated Strength Specifications for Zry-4 and Z4B™," are the same as or higher than those for Zry-4. Additionally, since it has been demonstrated that irradiated Z4B™ has higher corrosion resistance and lower hydrogen uptake relative to Zry-4, Framatome expects any strength reduction due to these factors to be bounded by what has been seen operationally with Zry-4 components. Consequently, the NRC staff confirms that Z4B™ unirradiated and irradiated strength specifications do not introduce any adverse change to the current analytical methods used to satisfy the mechanical design criteria approved in Reference 2.

Regarding material strain, Framatome compares unirradiated Zry-2, Zry-4, and Z4B™ strain test data and shows that [

].

Framatome implies that Zry-2 and Zry-4 strain test data would bound Z4B™ strain test data under irradiated conditions because the qualified fuel channel bulge analysis model described in

Reference 12 was found to be acceptable for irradiated external fuel assembly channels made of Zry-2 and Zry-4. Consequently, Framatome assumes the creep rate already defined for Zry-2 and Zry-4 channel material in Reference 12 will be representative of Z4B™. In consideration of the unirradiated Zry-2, Zry-4, and Z4B™ strain test data indicating comparable creep resistance, and the qualified fuel channel bulge analysis model described in Reference 12, the NRC staff finds that using the Zry-2 and Zry-4 channel material creep rate defined in Reference 12 for Z4B™ channel material is appropriate and therefore acceptable to demonstrate that the Z4B™ internal water channel assembly will meet the applicable design criteria in Reference 2.

In summary, Framatome evaluated the Z4B™ internal water channel mechanical design in Appendix D to Supplement 2P with respect to the generic BWR fuel design criteria applicable to the internal water channel tie structure during handling, normal operation, and anticipated operational occurrences (AOOs) as identified in Reference 2. All mechanical design analytical methods are described in Supplement 2P, Section 4, "Analytical Methodology." Supplement 2P, Section 4.1.1, "Stress, strain or loading limits," describes the general strength evaluations, including those performed for the internal water channel assembly to demonstrate that the stress, strain, and loading limit criteria (e.g., fuel handling) defined in Supplement 2P, Appendix D, Table D-3, "Generic BWR Design Criteria Applicable to the Tie Structure During Handling, Normal Operation, and AOO," are met. The results of the internal water channel mechanical design evaluations demonstrate that previously calculated design margins are maintained, therefore the internal water channel mechanical design evaluations are acceptable.

4.0 LIMITATIONS AND CONDITIONS

1. Approval of the updated correlations is limited to Framatome BWR fuel designs with Zry-2 cladding up to **[ ]**.

2. The updated fuel rod growth correlation only applies to: (1) standard (i.e, non-additive) fuel and (2) **[ ]** fuel when **[ ]**.

3. The updated fuel assembly growth correlation only applies to fuel designs with Zry-4 or Z4B™ internal water channels and does not apply to load bearing tie rod designs or the European bundle in basket design.

4. The RAI 10a response clarifies that the fuel rod growth correlations are adequate to bound future fuel rod designs that are within the range of parameters defined by the existing database supporting the fuel rod growth correlations. Any significant deviations in manufacturing processes, plant water chemistry, fuel designs, or the addition of new materials will require out of pile testing and lead test programs to acquire the necessary data to support evaluation methods, which will require NRC review and approval before use.

5. All post-irradiation data must be added to the measurement database supporting the respective correlation as it is acquired, and it must be evaluated as discussed in the RAI 10c response to determine whether or not limits need to be updated.

6. Using the same method as described in the response to RAI 2, new upper or lower limits supported by database updates must be calculated. If the new limits are outside the

envelope defined by the approved limits plus or minus one standard deviation, a new correlation must be submitted to the NRC for review and approval.

5.0 CONCLUSION

The NRC staff has reviewed the updated realistic thermal-mechanical fuel rod methodology for BWRs as described in Supplement 2 and concludes the following to the extent specified under the limitations and conditions delineated in Section 4.0 of this safety evaluation:

1. The fuel rod bow model is acceptable for Framatome BWR fuel designs with Zry-2 cladding up to [ ],

2. The fuel rod growth model is acceptable for Framatome BWR fuel rod designs with Zry-2 cladding up to [ ], and

3. The fuel assembly growth model is acceptable for fuel assembly designs with internal water channels made from Zry-4 and Z4B™ up to [ ].

6.0 REFERENCES

1. BAW-10247PA, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," AREVA Inc., April 2008 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML081340220).

2. ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels Corporation, May 1995 (ADAMS Accession Number ML081350281).

3. EMF-85-74(P) Revision 0, Supplement 1(P)(A) and Supplement 2(P)(A), "RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Models," Siemens Power Corporation, 1998. (ADAMS Accession Number ML15295A333).

4. Curet, H. D.,"NRC Request for Safety Assessment Related to Failed Seal Springs," Siemens Power Corporation, 1997.

5. XN-75-32(P)(A), Supplements 1, 2, 3, & 4, "Computational Procedure for Evaluating Fuel Rod Bowing," Exxon Nuclear Company Inc., 1983 (ADAMS Accession Number ML081710709).

6. BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," AREVA Inc., April 2016 (ADAMS Accession Number ML16125A033).

7. Peters, G., Response to Request for Additional Information Regarding AREVA Inc. Topical Report BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," April 2017 (ADAMS Accession No. ML17125A141).

8. Framatome, Inc., Additional Information Regarding BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors Supplement 2: Mechanical Methods," February 2018 (ADAMS Accession Number ML18058A781).

9. Framatome, Inc., BAW-10247NP-A, Suppl. 2Q2NP, Rev. 0, "Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors Supplement 2: Mechanical Methods Additional Information," February 2018 (ADAMS Accession Number ML18058A782).

10. Rudling, P., "Zr Alloy Corrosion and Hydrogen Pickup," A.N.T. International, December 2013 (ADAMS Accession Number ML15253A227).

11. ANP-10336P-A, Rev. 0, Z4B™ Fuel Channel Irradiation Program, AREVA Inc., June 2015 (ADAMS Accession Number ML17298A159).

12. EMF-93-177P-A, Rev. 1 Supplement 1P-A Rev. 0, "Mechanical Design for BWR Fuel Channels Supplement 1: Advanced Methods for New Channel Designs," AREVA Inc., September 2013 (ADAMS Accession Number ML14198A133).

13. Framatome, Inc., Additional Information Regarding BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors Supplement 2: Mechanical Methods," July 2018 (ADAMS Accession Number ML18193A453).

Attachment: Appendix

Principal Contributor: A. Patel, NRC/NRR/DSS/SNPB

Date: August 15, 2018

APPENDIX: BAW-10247P-A, SUPPLEMENT 2P CONFIRMATORY DATA ANALYSIS USING NON-PARAMETRIC REGRESSION TOLERANCE INTERVALS

Non-parametric tolerance intervals require less strict assumptions about the underlying data and are generally more conservative than [ ], therefore they have been used as an appropriate and independent means to confirm the validity of the 3 semi-empirical design limits proposed in BAW-10247P-A, Supplement 2P.

The tolerance package (Version 1.2.0) as part of the R statistical computing language (Reference Version 3.3.1) was used to support the confirmatory analysis.

Rod Bow Data

If the [ ] is used in the confirmatory UTL (designated as the upper bound curve of the darker grey area in the plot directly below)[1], as previously suggested to account for [ ], the confirmatory UTL remains in reasonable agreement with Framatome's proposed upper design limit (the red curve in the plot directly below)[2] despite it being somewhat above Framatome's limit for average assembly burnups in the range of approximately [ ] GWd/MTU.

Framatome's upper design limit is in reasonable agreement with the UTL determined in the NRC staff's confirmatory analysis, therefore it is acceptable.

---

[1] All lower/upper bound curves of darker grey areas in plots in this appendix correspond with the NRC staff's respective confirmatory limits.

[2] All red curves in plots in this appendix correspond with Framatome's respective design limits.

Rod Growth Data

Exploratory analysis

Data distributions and ranges

Comparison of assembly-wise burnup data can be made by creating a histogram of the rod growth measurements. The variation between assembly-wise data appears to be mostly normal; the quantiles of the data were also inspected to get a feel for the range of expected rod growth data for a specified burnup range. A summary table with the average measurement sample size is also provided at the end of this section to aid in understanding potential undersampling concerns as discussed in Footnote 5 of the SE.

There are some questionable groups of growth measurements, like those from Assembly 32B082 (see plot below with data from assembly burnups between [ ], which is clearly separated from the rest of the data in the respective group. Assembly 32B082 contains [ ] from what appears to be a 10x10 fuel assembly. [ ]. In this case, the data would not be expected to have a bias introduced from assembly undersampling, [ ] is questionable. However, because this is the only assembly-wise data set that appears to be questionable from a non-conservative standpoint, and since it is still within the range of the rest of data, there is no general concern.

[

]

Average sample size

[
]

Quantiles

[
]

[

]

Average sample size

[
]

Quantiles

[
]

[

]

Average sample size

[

]

Quantiles

[

]

[

]

Average sample size

[
]

Quantiles
[
]

[ ]

Average sample size

[ ]

Quantiles

[ ]

Average samples per burnup bin

[ ]

Assembly-wise data sample sizes

From the RAI 4a response, Framatome states that all fuel rods in a fuel assembly are not always measured. This could lead to a biased formulation of the [ ] if data sampling isn't close to random and/or if there are too few sampled data per assembly. The plot directly below shows the fuel rod growth data with the number of measured fuel rods per assembly indicated with red text directly above the corresponding assembly-wise data to provide additional information where data is densely packed.

[ ]

Based on the fuel rod growth data provided in the RAI 2f response, as plotted directly above, it appears that for approximately half of the data, [ ].[1]

[1] Since fuel assembly design types were not indicated in the dataset, the fraction of fuel rods sampled was estimated by assuming each assembly had the average number of fuel rods across the various assembly design types in the dataset.

Below a histogram of rod sampling fraction is shown along with sample quantiles and the data range in terms of number of rods sampled. This data supports the observations in the preceding paragraph.

[

]

Quantiles

[

]

Data range

[ ]

Using the median value of the assembly-wise data

Using the median value from the fuel-rod-wise data for a given fuel assembly will most likely produce the most reliable [ ][2]. This is because Framatome has [ ] as discussed in the RAI 2 response. This means that fuel rods [

]. Removing both the [ ] allows for an unbiased [ ] assessment. Additionally, using only a single

---

[2] This is not necessarily the most desirable [ ] because it removes the influence of the higher growth data. However, the influence of the lower growth data is also removed.

assembly-wise data point has the effect of reducing the effective sample size for the [ ] computation, increasing the tolerance factor, and adding conservatism.

Framatome's upper design limit was confirmed to be more conservative than the [ ] determined in the NRC staff's confirmatory analysis, therefore it is appropriate.

[ ]

Bias assessment

To assess the impact of a potential bias in the proposed correlation, potentially caused by not sampling all fuel rods in a given assembly, the NRC staff performed a series of calculations whereby [ ] were generated and then compared for a series of cases where successively more assembly-wise data was removed [ ]. By removing assembly-wise data with fewer measurements, this allows for a measure of [ ] sensitivity to potentially biased datasets. If there is a large [ ] sensitivity, then this may be indicative of a bias in Framatome's proposed [ ]. However, if little or no sensitivity is observed, this would provide reasonable assurance that there is no bias due to undersampling.

The cases analyzed were [ ].

There was minor variation in the [ ] when comparing these sets, and all were bounded by the [ ] proposed by Framatome. The minor variation between [ ] is a strong indicator that any bias that may exist is not likely to affect the proposed [ ] for rod growth, therefore the NRC staff finds the proposed [ ] for rod growth acceptable.

[ ]

Fuel Assembly Growth Data

The response to RAI 4b, regarding fuel assembly growth data collection, states the same basis for data sampling as that in RAI 4a. The NRC staff doesn't have a concern regarding a potential bias in the fuel assembly growth data [ ]. Furthermore, the fuel assembly growth correlation is supported by data from applicable fuel assemblies operating in the burnup range of interest.

Framatome's fuel assembly growth design limit is in close agreement with the [ ] determined by the NRC staff, therefore the fuel assembly growth design limit is acceptable.

AREVA

April 29, 2016
NRC:16:012

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Request for Review and Approval of BAW-10247P-A Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods"

AREVA Inc. (AREVA) requests the NRC's review and approval of the topical report BAW-10247P-A Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," dated April 2016, for referencing in licensing actions.

This topical report contains updated rod bow and growth correlations supporting the fuel mechanical analyses. These correlations include data necessary to implement the use of chromia-doped fuel into AREVA's suite of BWR licensing methods. Many of the topical reports that make up AREVA's suite of BWR methods are currently being reviewed for approval by the NRC. Topical report BAW-10247P-A Supplement 2P, Revision 0 is an integral part of the BWR suite of methods. The use of chromia-doped fuel has the goal of enhancing safe operation at BWR plants by reducing PCI failures.

In support of the Office of Nuclear Reactor Regulation's prioritization efforts, the Topical Report Prioritization Scheme is included as an enclosure with this letter.

AREVA would appreciate the NRC approval of this topical report by the end of the fourth quarter of calendar year 2017.

AREVA considers some of the material contained in the enclosed document to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support the withholding of the information from public disclosure. A proprietary version and a non-proprietary version of the report are enclosed.

There are no commitments within this letter or its enclosures.

AREVA INC.

3315 Old Forest Road, Lynchburg, VA 24501
Tel.: 434 832 3000 - www.areva.com

If you have any questions related to this information, please contact Mr. Alan B. Meginnis by telephone at (509) 375-8266, or by e-mail at Alan.Meginnis@areva.com.

Sincerely,

Gary A. Peters

Gary A. Peters, Director
Licensing & Regulatory Affairs
AREVA Inc.

cc: J. G. Rowley
Project 728

Enclosures:

1. Proprietary copy of topical report BAW-10247P-A Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors Supplement 2: Mechanical Methods"
2. Non-Proprietary copy of topical report BAW-10247NP-A Supplement 2NP, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors Supplement 2: Mechanical Methods"
3. BAW-10247P-A Supplement 2P, Revision 0 Priority Form
4. Notarized Affidavit

TR Prioritization Scheme			
Title: BAW-10247P-A Supplement 2P Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors Supplement 2: Mechanical Methods"			
Expect submitting FY	**TAC**	**PM**	**Today's Date:4/29/2016**
Technical Review Division(s)		**Technical Review Branch(s)**	
Factors	**Select the Criteria That the TR satisfies**	**Points can be Assigned for Each Criteria**	**Assigned Points**
TR Classification (Select one only)	Resolve Generic Safety Issue (GSI)	6	2
	Emergent NRC Technical Issue	3	
	New technology improves safety	2	
	TR Revision reflecting current requirements or analytical methods.	2	
	Standard TR	1	
TR Applicability (Select one only)	Potential industry-wide applications	3	2
	Potentially applicable to entire groups of licensees.	2	
	Intended for only partial groups of licensees.	1	
TR Implementation Certainty (Select one only)	Industry-wide Implementation expected	3	1
	Expected implementation by an entire group of licensees (BWROG, PWROG, BWRVIP, etc.) who sponsored the TR.	2	
	Docketed intent by U.S. plant(s) but no formal LAR schedule yet	1	
	No US plants have indicated strong intent on docket to implement yet.	0	
Tie to a LAR (Select if applicable)	A SE is requested by a certain date (less than two years) to support a licensing activity or renewal date (note it in Comments)	3	3
Review Progress (Points are cumulative as applicable)	Accepted for review	0.3	
	RAI issued	0.5	
	RAI responded	1.2	
	SE Drafted	2.0	
Management (LT/ET) discretion adjustment		-3 to +3	
Total Points (Add the total points from each factor and total here):			8
Comments: Request Approval by December 2017 for LAR Reference.			

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for AREVA Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in the report BAW-10247P-A Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," dated April 2016 and referred to herein as "Document." Information contained in this Document has been classified by AREVA as proprietary in accordance with the policies established by AREVA for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is

requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(a), 6(b), 6(c), 6(d) and 6(e) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 28th day of April, 2016.

SUSAN K MCCOY
NOTARY PUBLIC - WASHINGTON
MY COMMISSION EXPIRES 01-14-2020

Susan K. McCoy
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 1/14/2020

Received 9-7-16
T4.12.1 – NRC-IC-16-020
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 2, 2016

Mr. Gary Peters, Director
Licensing and Regulatory Affairs
AREVA Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: ACCEPTANCE FOR REVIEW OF AREVA INC. TOPICAL REPORT BAW-10247P-A, SUPPLEMENT 2P, REVISION 0, "REALISTIC THERMAL-MECHANICAL FUEL ROD METHODOLOGY FOR BOILING WATER REACTORS SUPPLEMENT 2: MECHANICAL METHODS" (TAC NO. MF7708)

Dear Mr. Peters:

By letter dated April 29, 2016 (Agencywide Documents Access and Management System Accession Number ML16124B057), AREVA Inc. (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review Topical Report (TR) BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods." The NRC staff has performed an acceptance review of TR BAW-10247P-A, Supplement 2P, Revision 0. We have found that the material presented is sufficient to begin our comprehensive review. NRC staff expects to issue its request for additional information by March 10, 2017, and issue its draft safety evaluation (SE) by September 29, 2017. This schedule information takes into consideration NRC's current review priorities and available technical resources, and may be subject to change. If modifications to these dates are deemed necessary, we will provide appropriate updates to this information. NRC staff estimates that the review will require approximately 200 staff hours, including project management and contractor time. The review schedule milestones and estimated review costs were discussed and agreed upon in a telephone conference between AREVA Product Licensing Manager, Alan Meginnis, and NRC staff on August 16, 2016.

Section 170.21 of Title 10 of the *Code of Federal Regulations* requires that TRs are subject to fees based on the full cost of the review. You did not request a fee waiver; therefore, NRC staff hours will be billed accordingly.

As with all topical reports, the SE will be reviewed by NRC's Office of the General Counsel (OGC) to determine whether it falls within the scope of the Congressional Review Act (CRA). During the course of this review, OGC considers whether any endorsement or acceptance of a TR by the NRC amounts to a rule as defined in the CRA. If this initial review concludes that the SE, with its accompanying TR, may be a rule, NRC will forward the package to the Office of Management and Budget (OMB) for further review and consideration. Any review by OMB would impact the schedule for the issuance of the final SE.

If you have questions regarding this matter, please contact Jonathan G. Rowley at (301) 415-4053.

Sincerely,

Kevin Hsueh

Kevin Hsueh, Chief
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 728

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 23, 2017

Mr. Gary Peters, Director
Licensing and Regulatory Affairs
AREVA Inc.
3315 Old Forest Road
Lynchburg, VA 24501

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION RE: AREVA INC. TOPICAL REPORT BAW-10247P-A, SUPPLEMENT 2P, REVISION 0, "REALISTIC THERMAL-MECHANICAL FUEL ROD METHODOLOGY FOR BOILING WATER REACTORS SUPPLEMENT 2: MECHANICAL METHODS" (CAC NO. MF7708)

Dear Mr. Peters:

By letter dated April 29, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16124B057), AREVA Inc. (AREVA) submitted for U.S. Nuclear Regulatory Commission (NRC) staff review and approval Topical Report BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods." Upon review of the information provided, the NRC staff has determined that additional information is needed to complete the review. On January 11, 2017, Alan Meginnis, AREVA Product Licensing Manager, and I agreed that the NRC staff will receive the response to the enclosed request for additional information (RAI) questions within 60 days from the date of this letter.

If you have any questions regarding the enclosed RAI questions, please contact me at 301-415-4053.

Sincerely,

Jonathan Rowley

Jonathan G. Rowley, Project Manager
Licensing Processes Branch
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 728

Enclosure:
RAI Questions (Proprietary)

NOTICE: Enclosure transmitted herewith contains proprietary information. When separated from Enclosure, this transmittal document is decontrolled.

REQUEST FOR ADDITIONAL INFORMATION

RELATED TO TOPICAL REPORT BAW-10247P-A, SUPPLEMENT 2P, REVISION 0

"REALISTIC THERMAL-MECHANICAL FUEL ROD METHODOLOGY FOR BOILING WATER

REACTORS SUPPLEMENT 2: MECHANICAL METHODS"

AREVA INC.

(CAC NO. MF7708)

RAI 1

Regarding measurements supporting the rod-to-rod gap closure correlation, the topical report (TR) states that the fuel rod-to-rod gap measurements are generally performed at each span between spacer grids and for each gap. The text should be changed to indicate that each span and gap must be measured so that the correlation remains unbiased (i.e., not measuring each span and every gap could invalidate the derived 95/95 upper tolerance limit (UTL)). Revise the text describing the methodology or justify not doing so. Also, describe any changes to the measurement tool and techniques to acquire the rod bow data relative to XN-75-32(P)(A), Supplement 1.

RAI 2

The various models defined in Appendices A, B, and C of the TR are based on [ ]. However, Equation A-3 includes a [ ] of Reference A-1 in the TR. This table was not meant to be used for [ ]. Reference A-1, includes information for [ ]

Furthermore, use of statistical tolerance factors for a [ ], as discussed in Reference 1, assumes: (1) the random error follows a normal distribution with mean 0 and some standard deviation, and (2) the observations to be statistically independent of each other, i.e., the correlation of y_i and y_j for i not equal to j is zero. Looking at the data in Figure A-1, "[Boiling Water Reactor] BWR Fuel Rod Bow Correlation," it appears that the assumption of statistical independence is not valid as there are distinct clusters in the data marked by vertical lines. Furthermore, it is not clear that the random error follows a normal distribution. Consequently, use of the one-sided statistical tolerance factor as described in the TR does not appear to be justified.

a. Update Equation A-3 based on the discussion above or explain why the model remains appropriate.

b. Regarding Equation A-2, what are [ ] and [ ] defined as? An [ ] is specified but not used; update the equation accordingly.

Enclosure

c. Similar to Part a., update Equation B-1 or explain why the model specified by Equation B-1 remains appropriate. Also note that Figure B-1, "BWR Fuel Rod Growth Correlation for [stress relief annealed] SRA Cladding," appears to show a [ ], and the tolerance factor used does not account for this. The U.S. Nuclear Regulatory Commission (NRC) staff notes that this may be caused by oversimplification of the model due to the lumping of subgroups into a single group.

d. Similar to Part a., update Equation C-1 or explain why the model specified by Equation C-1 remains appropriate.

e. The "s" term (i.e., the standard deviation) in Equations B-2 and C-2 are inconsistent with the analogous term used in the fuel rod bow model given in Equation A-2. Revise the "s" term in Equations B-2 and C-2 to be consistent with the [ ] in Equation A-2 or explain why using the standard deviation is appropriate.

f. Provide the data supporting the fuel rod bow, fuel rod growth, and fuel assembly growth correlations in tabular format so that the NRC staff can perform confirmatory calculations to either verify the validity of: (1) the corresponding [ ] presented in the TR, or (2) of any model updates in response to Parts a. through e. of this request for additional information. [

]

RAI 3

The TR does not provide a discussion of how the updated rod-to-rod gap closure correlation is applied in downstream safety analysis methods -- it is only mentioned that "the rod-to-rod gap closure predicted as a function of fuel assembly exposure is used as an input to thermal limit evaluations (i.e., MCPR) for AREVA BWR fuel designs." Describe how the rod bow empirical model is used in downstream safety analyses. Consider the following NRC staff observations for additional context:

The discussion at the end of "Accepted Version of Exxon Nuclear Licensing Topical Report, XN-NF-85-67(P)(A), 'Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081760201), Section 3.4.9, "Fuel Rod Spacing and Rod Bow" states:

[

]

Does this mean that spacings have never been reduced enough to warrant a minimum critical power ratio (MCPR) penalty? Is this still true? If so, what is the latest licensing basis that states this?

Further, TR ANP-2637, Revision 6, "BWR Licensing Methodology Compendium" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15282A224), Section 2.2.6, 'Rod Bowing,' states:

> Rather than placing design limits on the amount of bowing that is permitted, the effects of bowing are included in the cladding overheating analysis by limiting fuel rod powers when bowing exceeds a predetermined amount. AREVA uses an approved methodology (Reference 2-9) to determine a rod-to-rod clearance closure limit below which a penalty is addressed on the MCPR and above which no reduction in MCPR is necessary. The methodology is based on empirical data (Reference 2-2) to calculate minimum end of life rod-to-rod spacing. The potential effect of this rod bow on thermal margin is negligible. Rod bow at extended burnup does not affect thermal margins due to the lower powers achieved at high exposure.

What approved TR describes how "the effects of bowing are included in the cladding overheating analysis by limiting fuel rod powers when bowing exceeds a predetermined amount"?

Reference 2-9 mentioned in the quoted passage above is XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Supplement 1, "Extended Burnup Qualification of ENC 9x9 BWR Fuel," Advanced Nuclear Fuels Corporation, May 1988. This reference does not give a formulation for the MCPR penalty that would be applied if rod-to-rod closure is greater than the 95 percent UTL given by the corresponding correlation. What approved topical report describes the MCPR penalty formulation?

RAI 4

a. Describe the process for fuel assembly selection when fuel rod growth measurement data is generated for the measurement database supporting the corresponding correlation to ensure that the correlation remains unbiased (i.e., inconsistent data generation could invalidate the derived [ ]). Is growth measurement data entered into the database for all fuel rods in a given fuel assembly selected for fuel rod growth measurement?

b. Similarly, describe the process for fuel assembly selection when internal water channel growth measurement data is generated for the measurement database supporting the corresponding correlation.

RAI 5

Is there any [ ] clad fuel rod growth data included in Figure B-1 of the TR? Including [ ] clad fuel rod growth data would be inappropriate since EMF-85-74(P)(A), Supplement 2, Revision 0, noted that fuels with [ ] and inclusion of this data could bias the data non-conservatively.

RAI 6

Regarding the fuel rod growth enhancement factor that accounts for the presence of chromia-doped fuel, the following statement is made: "[

]." It is understood that the mechanism for increased axial growth is the same; however, it is not clear that the *magnitude* of the effect will be the same. Provide data similar to that in Figure B-2 for SRA cladding to support the claim that the enhancement factor will be the same for fuel with either RXA or SRA cladding.

RAI 7

The summary regarding the BWR fuel rod growth correlation in Appendix B of the TR, states: "Based on the data and similarity in manufacturing processes, the BWR rod growth correlation is fully applicable to AREVA BWR fuel rod designs with SRA [Zircaloy-2] Zry-2 cladding."

a. Has a similar correlation been developed and implemented for RXA cladding? If so, where is this discussed?

b. Confirm that the RODEX4 rod growth model is unaffected by the updated fuel rod growth database in the TR. For example, determination of the rod free volume depends on the rod growth model. This rod growth model, described in Section 4.2.6, "Rod Axial Elongation" of TR BAW-10247PA, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," doesn't currently include the effects of chromia-doped fuel which exhibits more rod growth compared to non-doped fuel. Also, describe any other equation constants and tuning parameters derived in the base topical report that are potentially affected by the new data provided in the TR. If AREVA believes that impacts to RODEX4 are beyond the scope of the TR review, explain where these issues have been addressed or will be addressed (e.g., in other supplements that have been previously approved or are currently under review).

RAI 8

It appears that some ATRIUM-10 data from the previously approved fuel assembly growth model in EMF-85-74(P)(A), Supplement 2, Revision 0, has been removed when comparing Figure C-1 of the TR and the figure in Reference A.2 of EMF-85-74(P)(A), Supplement 2, Revision 0. In particular, the 2 points around [ ] with values of approximately [ ] are no longer present in Figure C-1. Provide justification for why data points were removed from either the fuel assembly or fuel rod growth model development process if this is the case.

RAI 9

Why isn't an upper bound maximum fuel channel growth curve included in the TR as was done for the previously approved evaluation of fuel channel overlay with the lower tie plate seal spring in EMF-85-74(P)(A), Supplement 2, Revision 0? A value of [ ] was determined at a burnup of [ ] previously and it appears that the new data presented in Figure C-1 would cause a significant increase in the upper bound curve.

RAI 10

The update process for the models described in the TR is described in Section 5.0, "UPDATE PROCESS."

a. Although the TR data appears to be predictable based on burnup alone, growth of SRA Zry-2 depends on factors such as the amount of cold work (i.e., manufacturing process) and the presence of hydrogen or hydrides due to corrosion. Explain why the TR correlation will be adequate to bound future fuel rod designs that may have different manufacturing processes, plant water chemistry, etc.

b. During the acceptance review, AREVA stated that fuel rod growth is independent of fuel design and that cladding material drives the need for different growth correlations. However, the need for an [ ], demonstrate otherwise. Given the provided data, explain why fuel rod growth will remain independent of future fuel designs (e.g., ATRIUM 11 and other evolutions of this design that may or may not contain fuel additives). This may be covered under Section 5.0, "Update Process," of the TR.

c. Although the section states that models will be reviewed against a growing post irradiation examination database, it does not specify with what frequency. If the frequency is too low, data may be added that could non-conservatively invalidate current models without having to submit updated models for NRC review and approval. Specify an appropriate minimum frequency.

d. The following statement does not contain a sufficient level of specificity: "The threshold for submittal of the growth and bow correlations is an increase of the correlation tolerance limits by one standard deviation." To fully understand the criterion, provide additional specificity. For example: (1) Provide the mathematical definition of the standard deviation being referred to and why it is appropriate (e.g., Why not use standard error?), and (2) Does the increase have to be observed over the entire burnup range, some subset of the burnup range, or something else?

April 27, 2017
NRC:17:022

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Response to Request for Comment on AREVA Inc. Topical Report BAW-10247PA, Revision 0, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods"

Ref. 1: Letter, Gary Peters (AREVA Inc.) to Document Control Desk (NRC), "Request for Review and Approval for BAW-10247P-A, Revision 0, Supplement 2P, Revision 0, 'Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods'," NRC:16:012, April 29, 2016.

Ref. 2: Letter, Jonathan G. Rowley (NRC) to Gary Peters (AREVA Inc.), "Request for Additional Information Re: AREVA Inc. Topical Report BAW-10247P-A, Supplement 2, Revision 0, 'Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods'," (CAC NO. MF7708), February 23, 2017.

In Reference 1, AREVA Inc. (AREVA) requested the NRC's review and approval of the topical report BAW-10247P-A, Revision 0, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods." The NRC provided a request for additional information (RAI) in Reference 2. AREVA's response to the RAI is enclosed to this letter. Also enclosed to this letter is a DVD including data referenced in the enclosed response to the NRC's RAI 2-f.

AREVA considers some of the information contained in the enclosures to be proprietary. As required by 10 CFR 2.390(b), an affidavit is enclosed to support withholding the information from public disclosure. Proprietary and non-proprietary versions of the RAI responses are provided.

There are no new commitments within this letter or its enclosures.

AREVA INC.

3315 Old Forest Road, Lynchburg, VA 24501
Tel.: 434 832 3000 - www.areva.com

If you have any questions related to this letter please contact Mr. Morris E. Byram, Product Licensing Manager, by telephone at 434-832-4665 or by e-mail at Morris.Byram@areva.com.

Sincerely,

Gary Peters, Director
Licensing & Regulatory Affairs
AREVA Inc.

cc: J. G. Rowley
Project 728

Enclosures:

1. A Proprietary Copy of the Report, BAW-10247PA, Revision 0, Supplement 2Q1P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Mechanical Methods – Responses to NRC Request for Additional Information"
2. A Non-Proprietary Copy of the Report, BAW-10247PA, Revision 0, Supplement 2Q1P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Mechanical Methods – Responses to NRC Request for Additional Information"
3. DVD: "Rod Bow, Fuel Assembly Growth, and Fuel Rod Growth Data Referenced in RAI 2f Response of BAW-10247PA, Rev. 0, Supp. 2 Q1P Rev. 0"
4. Notarized Affidavit

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Morris Byram. I am Manager, Product Licensing, for AREVA Inc. (AREVA) and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by AREVA to determine whether certain AREVA information is proprietary. I am familiar with the policies established by AREVA to ensure the proper application of these criteria.

3. I am familiar with the AREVA information contained in the report BAW-10247PA, Revision 0, Supplement 2Q1P, Revision 0, entitled "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 1: Mechanical Methods Responses to NRC Request for Additional Information," and the DVD files named "Fuel Assembly Growth.pdf', "Fuel Rod Growth.pdf', and "Rod Bow.pdf" referred to herein as "Documents." Information contained in these documents has been classified by AREVA as proprietary in accordance with the policies established by AREVA Inc. for the control and protection of proprietary and confidential information.

4. These documents contain information of a proprietary and confidential nature and is of the type customarily held in confidence by AREVA and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in these documents as proprietary and confidential.

5. These documents have been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in these documents

be withheld from public disclosure. The request for withholding of proprietary information is made in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by AREVA to determine whether information should be classified as proprietary:

(a) The information reveals details of AREVA's research and development plans and programs or their results.

(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for AREVA.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for AREVA in product optimization or marketability.

(e) The information is vital to a competitive advantage held by AREVA, would be helpful to competitors to AREVA, and would likely cause substantial harm to the competitive position of AREVA.

The information in these documents is considered proprietary for the reasons set forth in paragraphs 6(c), 6(d), and 6(e) above.

7. In accordance with AREVA's policies governing the protection and control of information, proprietary information contained in these documents has been made available, on a limited basis, to others outside AREVA only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 26th

day of April, 2017.

HAILEY M SIEKAWITCH
Commission Expires
NOTARY
PUBLIC
Sept. 28, 2020
STATE OF WASHINGTON

AREVA

Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors
Supplement 2: Mechanical Methods Responses to NRC Request for Additional Information

BAW-10247NP-A
Supplement
2Q1NP
Revision 0

April 2017

AREVA Inc.

Nature of Changes

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

Contents

Figures

INTRODUCTION

AREVA is consolidating the miscellaneous mechanical methods in BAW-10247P-A, Supplement 2 (Reference I-1) which were not updated in the base topical report. As part of this consolidation effort, the most recent operating experience data is provided in order to update the correlations for fuel rod bow, fuel rod growth, and fuel assembly growth. Supplement 2 does not introduce any changes to AREVA's existing BWR methodology with the exception of the correlation updates.

BAW-10247P-A, Supplement 2 was submitted for approval to the NRC in Reference I-2. The NRC subsequently requested additional information in Reference I-3 in order to complete the review. This document supplies AREVA's responses to the NRC's request for additional information.

References:

I-1. BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods." April 2016.

I-2. Letter, Gary Peters (AREVA, Inc.) to NRC Document Control Desk, "Request for Review and Approval of BAW-10247P-A Supplement 2P, Revision 0, Realistic Thermal Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods," April 29, 2016.

I-3. Letter, Kevin Hsueh (NRR) to Gary Peters (AREVA, Inc.), "Request for Additional Information Re: AREVA Inc. Topical Report BAW-10247P-A, Supplement 2P, Revision 0, 'Realistic Thermal Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods'," February 23, 2017.

RAI-1:

Regarding measurements supporting the rod-to-rod gap closure correlation, the topical report (TR) states that the fuel rod-to-rod gap measurements are generally performed at each span between spacer grids and for each gap. The text should be changed to indicate that each span and gap must be measured so that the correlation remains unbiased (i.e., not measuring each span and every gap could invalidate the derived 95/95 upper tolerance limit (UTL)). Revise the text describing the methodology or justify not doing so. Also, describe any changes to the measurement tool and techniques to acquire the rod bow data relative to XN-75-32(P)(A), Supplement 1.

AREVA Response RAI-1:

The text in question from Attachment A reads as follows:

Fuel rod-to-rod gap measurements are typically taken at each span between spacer grids (usually 8 spans) at mid-span for each fuel rod-to-rod gap.

This text is replaced by:

Fuel rod-to-rod gap measurements are taken at each span between spacer grids. All accessible rod-to-rod gaps are measured. Some internal locations, such as those behind the water channel, are not always accessible by the tool. It is expected that the internal rods are lower power and less likely to be limiting with respect to rod bow.

The corresponding text changes to Appendix A, Pages A-1 and A-2, of Supplement 2 are shown in Attachment 1.

If measurements are missing within a fuel assembly AREVA risks introducing bias into the calculated tolerance limit. A review of the database determined that the following fuel assemblies are missing data from at least one span:

Fuel Assembly	Exposure (MWD/kgU)	Measured spans
KG-30993L	0	2 to 8
KG-30995G	39.588	3,6,7
C776	28.225	2,3
BZ705	37.672	2 to 7
BC205	24.52	1 to 6
BC191	23.15	1,2,3,6,7

If we remove these fuel assemblies from the database, this leads to the following fitted parameters:

[ ]

The corresponding 95/95 UTL curves (see Equation A-4) are the following:

[ ]

Figure 1-1: Worst span, 95/95 UTL curve

The curve obtained by removing the Fuel Assemblies bounds the original curve. The maximum difference is [ ]. This difference, which comes from the lower amount of data creating a new curve with an increased k factor (T= k•s), is negligible compared to the 20% extra margin included in the correlation presented in the topical report (Reference 1-1). Therefore, it is not necessary to change the correlation initially submitted.

The measurement tool and techniques have not changed over time.

RAI-1 Response References:

1-1 BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods." April 2016.

RAI-2:

The various models defined in Appendices A, B, and C of the TR are based on [ ]. However, Equation A-3 includes a [ ] of Reference A-1 in the TR. This table was not meant to be used for [ ]. Reference A-1, includes information for [ ]

Furthermore, use of statistical tolerance factors for a [ ], as discussed in Reference 1, assumes: (1) the random error follows a normal distribution with mean 0 and some standard deviation, and (2) the observations to be statistically independent of each other, i.e., the correlation of y_i and y_j for i not equal to j is zero. Looking at the data in Figure A-1, "[ ] BWR Fuel Rod Bow Correlation," it appears that the assumption of statistical independence is not valid as there are distinct clusters in the data marked by vertical lines. Furthermore, it is not clear that the random error follows a normal distribution. Consequently, use of the one-sided statistical tolerance factor as described in the TR does not appear to be justified.

a. *Update Equation A-3 based on the discussion above or explain why the model remains appropriate.*

b. *Regarding Equation A-2, what are [ ] and [ ] defined as? An [ ] is specified but not used; update the equation accordingly.*

c. *Similar to Part a., update Equation B-1 or explain why the model specified by Equation B-1 remains appropriate. Also note that Figure B-1, "BWR Fuel Rod Growth Correlation for [ ] SRA Cladding," appears to show a [ ], and the tolerance factor used does not account for this. The U.S. Nuclear Regulatory Commission (NRC) staff notes that this may be caused by oversimplification of the model due to the lumping of subgroups into a single group.*

d. *Similar to Part a., update Equation C-1 or explain why the model specified by Equation C-1 remains appropriate.*

e. The "s" term (i.e., the standard deviation) in Equations B-2 and C-2 are inconsistent with the analogous term used in the fuel rod bow model given in Equation A-2. Revise the "s" term in Equations B-2 and C-2 to be consistent with the [ ] in Equation A-2 or explain why using the standard deviation is appropriate.

f. Provide the data supporting the fuel rod bow, fuel rod growth, and fuel assembly growth correlations in tabular format so that the NRC staff can perform confirmatory calculations to either verify the validity of: (1) the corresponding [ ] presented in the TR, or (2) of any model updates in response to Parts a. through e. of this request for additional information. [ ]

<u>AREVA Response RAI-2</u>:

Contrary to what is implied in the report text, the bow and growth models are not the result of a [ ]. Instead, [ ] are presented in Appendices A, B and C which are based on [ ]. The following general approach, which was also used for model parameter uncertainties in RODEX4 (Ref. BAW-10247, Section 5.4), was employed in [ ] of the three correlations in Appendices A, B and C.

First, [ ]

[

].

Specific responses to sub-questions are as follows:

AREVA Response RAI-2a:

[

].

AREVA Response RAI-2b:

Indeed, Equation A-2 is equivalent to the general equation (4) and the [

].

Page A-4 in Attachment 1, shows the revised equation A-2.

AREVA Response RAI-2c:

The same arguments as for item ‘a’ apply here. [

]:

[ ]

Figure 2-1: Frequency Distribution of Rod Growth Residuals

AREVA Response RAI-2d:

The response to sub-part 'a' applies here as well.

AREVA Response RAI-2e:

The justification for items 'a' and 'd' applies here, too. As explained above, [

].

Page A-4 in Attachment 1 shows the revision of "The standard error of the square root function with curve fitted parameters with respect to the data..." to "The standard deviation of the difference between the measured and predicted values for the entire data set...".

AREVA Response RAI-2f:

Due to the amount of supporting data, the requested data has been provided in digital format on a DVD. The DVD contains three Excel files, one file for each correlation which are labeled accordingly.

The rod bow file provides the fuel assembly ID (FA_ID) in the first column, the fuel assembly average burnup (Burnup) in MWd/kgU in the second column, and the gap closure data in the third column.

The fuel rod growth file provides the fuel assembly ID (Assy ID) in the first column, the rod number (Rod) in the second column, the fuel assembly average burnup (FA Average BU) in MWd/kgU in the third column, and the rod growth in percent in the last column.

The fuel assembly growth file provides the fuel assembly ID (Assembly ID) in the first column, the fuel assembly burnup (Assy Exposure) in MWd/kgU in the second column, and the fuel assembly growth in percent in the third column.

RAI-2 Response References:

2-1 BAW-10247P-A, Supplement 2P, Revision 0, “Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods.” April 2016.

RAI-3:

The TR does not provide a discussion of how the updated rod-to-rod gap closure correlation is applied in downstream safety analysis methods -- it is only mentioned that "the rod-to-rod gap closure predicted as a function of fuel assembly exposure is used as an input to thermal limit evaluations (i.e., MCPR) for AREVA BWR fuel designs." Describe how the rod bow empirical model is used in downstream safety analyses. Consider the following NRC staff observations for additional context:

The discussion at the end of "Accepted Version of Exxon Nuclear Licensing Topical Report, XN-NF-85-67(P)(A), 'Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081760201), Section 3.4.9, "Fuel Rod Spacing and Rod Bow" states:

> *[*
>
> *]*

Does this mean that spacings have never been reduced enough to warrant a minimum critical power ratio (MCPR) penalty? Is this still true? If so, what is the latest licensing basis that states this?

Further, TR ANP-2637, Revision 6, "BWR Licensing Methodology Compendium" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15282A224), Section 2.2.6, 'Rod Bowing,' states:

> *Rather than placing design limits on the amount of bowing that is permitted, the effects of bowing are included in the cladding overheating analysis by limiting fuel rod powers when bowing exceeds a predetermined amount. AREVA uses an approved methodology (Reference 2-9) to determine a rod-to-rod clearance closure limit below which a penalty is addressed on the MCPR and above which no reduction in MCPR is necessary. The methodology is based on empirical data (Reference 2-2) to calculate minimum end of life rod-to-rod spacing. The potential effect of this rod bow on thermal margin is negligible.*

Rod bow at extended burnup does not affect thermal margins due to the lower powers achieved at high exposure.

What approved TR describes how "the effects of bowing are included in the cladding overheating analysis by limiting fuel rod powers when bowing exceeds a predetermined amount"?

Reference 2-9 mentioned in the quoted passage above is XN-NF-82-06(P)(A), Supplement 1, Revision 2, "Qualification of Exxon Nuclear Fuel for Extended Burnup," Supplement 1, "Extended Burnup Qualification of ENC 9x9 BWR Fuel," Advanced Nuclear Fuels Corporation, May 1988. This reference does not give a formulation for the MCPR penalty that would be applied if rod-to-rod closure is greater than the 95 percent UTL given by the corresponding correlation. What approved topical report describes the MCPR penalty formulation?

AREVA Response RAI-3:

The rod bow MCPR penalty is determined [

].

AREVA's BWR rod bow CPR penalty was derived using open literature data (Reference 3-4, Attachment 1). Based on this data, it was concluded that thermal margins were not substantially reduced for closures as low as 0.06 inch (59% closure). Based on the available data, a conservative model of CPR as a function of rod spacing was described (Reference 3-4, Attachment 1).

AREVA's rod bow model application to ATRIUM 10 type fuel was presented in an informational submittal to the NRC (Reference 3-1) and the application for ATRIUM 10XM fuel was presented in Reference 3-2. The CPR penalty (decrease in CPR) versus rod bow (% closure) is presented in Figure 3-1, below. To confirm the conservatism of this model, AREVA ran a critical power test on an ATRIUM-10 design test assembly

where two rods were bowed to touch (i.e. 100% gap closure). The maximum measured CPR penalty in the test was [ ] (as shown in Figure 3-1). The conservatism of the model was confirmed.

[ ]

Figure 3-1: CPR Penalty vs Test Data

From Figure 3-1, the CPR penalty begins at [ ]. It is evident when examining Figure A-1 of Reference 3-5 that the rod bow impact on CPR does not begin until exposures of approximately [ ] are reached. [ ].

AREVA has implemented guidelines (Reference 3-3, Section 6.4.3) to ensure the application of MCPR penalties for % gap closures above [ ]. The appropriate rod bow penalty has been included in the MCPR operating limits in MICROBURN-B2 for core monitoring.

RAI-3 Response References

3-1. EMF-95-52(P) Revision 1, Fuel Design Evaluation for Siemens Power Corporation ATRIUM™-10 BWR Reload Fuel, April 1998, transmitted to the NRC by Siemens Power Corporation Letter, "Design Evaluations for SPC ATRIUM™-9B and ATRIUM™-10 Fuel", April 8, 1998, (NRC:98:021).

3-2. ANP-3289P Revision 0, Responses to RAI from SNPB on MNGP Transition to AREVA Fuel, February 2014.

3-3. EMF-2001(P), Guidelines for BWR Safety Analysis, P110,3201, "MCPRp Limits and LHGRFACp Multipliers (Heat Flux Ratio)," AREVA NP, December 2015.

3-4. XN-NF-82-06(P)(A) Supplement 1 Revision 2, Qualification of Exxon Nuclear Fuel for Extended Burnup, Supplement 1, Extended Burnup Qualification of ENC 9x9 BWR Fuel, May 1980.

3-5. BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods." April 2016.

RAI-4:

a. *Describe the process for fuel assembly selection when fuel rod growth measurement data is generated for the measurement database supporting the corresponding correlation to ensure that the correlation remains unbiased (i.e., inconsistent data generation could invalidate the derived [ ]). Is growth measurement data entered into the database for all fuel rods in a given fuel assembly selected for fuel rod growth measurement?*

b. *Similarly, describe the process for fuel assembly selection when internal water channel growth measurement data is generated for the measurement database supporting the corresponding correlation.*

AREVA Response RAI-4a:

Generally, fuel assemblies are not necessarily chosen for the expressed purpose of taking fuel rod growth measurements. Fuel assemblies are generally chosen for a number of reasons. For example, assemblies could be chosen because they are limiting for corrosion after water chemistry changes, or because they are useful to explore the boundary of operating experience (e.g., burnup, time, fluence, etc.) in healthy fuel exams, or just because they are lead assemblies. Therefore, any bias would be conservative, i.e., toward the measurement of assemblies viewed as limiting in some aspect. When an assembly has been chosen for fuel rod growth measurements, not all fuel rods in the assembly are always measured. However, all rods that are measured during the inspection are included in the database. The change process as described in RAI 10 will be followed when updating the growth correlations after incorporation of new measurements into the database.

AREVA Response RAI-4b:

The process for fuel assembly selection for water channel growth measurements are the same as the selection for rod growth.

RAI-5:

Is there any [ ] clad fuel rod growth data included in Figure B-1 of the TR? Including [ ] clad fuel rod growth data would be inappropriate since EMF-85-74(P)(A), Supplement 2, Revision 0, noted that fuels with [ ] and inclusion of this data could bias the data non-conservatively.

<u>AREVA Response RAI-5</u>:

[ ] cladding is included in the growth database, or described in Reference 5-1.

RAI 5 Response References

5-1. BAW-10247PA, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," April 2008.

RAI-6:

Regarding the fuel rod growth enhancement factor that accounts for the presence of chromia-doped fuel, the following statement is made: "[

]." It is understood that the mechanism for increased axial growth is the same; however, it is not clear that the magnitude of the effect will be the same. Provide data similar to that in Figure B-2 for SRA cladding to support the claim that the enhancement factor will be the same for fuel with either RXA or SRA cladding.

<u>AREVA Response RAI-6</u>:

The net rod axial growth is the combined result of irradiation growth and axial creep due to axial PCMI. As mentioned in response to RAI-7a, [

].

The extension to [

].

RAI-7:

The summary regarding the BWR fuel rod growth correlation in Appendix B of the TR, states: "Based on the data and similarity in manufacturing processes, the BWR rod growth correlation is fully applicable to AREVA BWR fuel rod designs with SRA [ ] Zry-2 cladding."

a. *Has a similar correlation been developed and implemented for RXA cladding? If so, where is this discussed?*

b. *Confirm that the RODEX4 rod growth model is unaffected by the updated fuel rod growth database in the TR. For example, determination of the rod free volume depends on the rod growth model. This rod growth model, described in Section 4.2.6, "Rod Axial Elongation" of TR BAW-10247PA, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," doesn't currently include the effects of chromia-doped fuel which exhibits more rod growth compared to non-doped fuel. Also, describe any other equation constants and tuning parameters derived in the base topical report that are potentially affected by the new data provided in the TR. If AREVA believes that impacts to RODEX4 are beyond the scope of the TR review, explain where these issues have been addressed or will be addressed (e.g., in other supplements that have been previously approved or are currently under review).*

<u>AREVA Response RAI-7a</u>:

A similar fuel rod growth correlation for BWR RXA cladding has not been developed or implemented for application in the U.S. at this time. [

].

AREVA Response RAI-7b:

The rod growth model, mentioned in Section 4.2.6, "Rod Axial Elongation" of Reference I-1, is fully described in the RODEX4 Theory Manual, Reference 7-2. As stated in both documents, [

].

The SRA Zry-2 data that have been used for RODEX4 verification and validation are 2008 vintage and include a large fraction of the "new data" of Appendix B of Reference 7-3. This has been documented in response to RAI-20b in Reference I-1.

Moreover, as can be seen from Figure B-1 of Reference 7-3, [

].

[

].

RAI 7 Response References

7-1 BAW-10247PA, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors," April 2008.

7-2 EMF-2994(P), Revision 6, "RODEX4: Thermal-Mechanical Fuel Rod Performance Code Theory Manual," February 2012.

7-3 BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods." April 2016.

7-4 ANP-10340P, Revision 0, "Incorporation of Chromia-Doped Fuel Properties in AREVA Approved Methods," April 2016.

7-5 Griffiths, M., Gilbert, R. W., and Fidleris, V., "Accelerated Irradiation Growth of Zirconium Alloys," Zirconium in the Nuclear Industry: Eighth International Symposium, ASTM STP 1023, 1989, pp. 658-677.

RAI-8:

It appears that some ATRIUM-10 data from the previously approved fuel assembly growth model in EMF-85-74(P)(A), Supplement 2, Revision 0, has been removed when comparing Figure C-1 of the TR and the figure in Reference A.2 of EMF-85-74(P)(A), Supplement 2, Revision 0. In particular, the 2 points around [ ] with values of approximately [ ] are no longer present in Figure C-1. Provide justification for why data points were removed from either the fuel assembly or fuel rod growth model development process if this is the case.

AREVA Response RAI-8:

We have performed a thorough analysis of the fuel assembly growth database and the two data points referenced above were excluded. Measurement of the same assemblies after subsequent cycles showed lower growth, and therefore the two earlier data points were deemed inaccurate. These two data points were not included in the database, or correlation. In addition, fuel rod data from these assemblies are not applicable to the U.S. BWR fuel rod growth database as they are made of RXA material.

RAI-9:

Why isn't an upper bound maximum fuel channel growth curve included in the TR as was done for the previously approved evaluation of fuel channel overlay with the lower tie plate seal spring in EMF-85-74(P)(A), Supplement 2, Revision 0? A value of [ ] was determined at a burnup of [ ] previously and it appears that the new data presented in Figure C-1 would cause a significant increase in the upper bound curve.

<u>AREVA Response RAI-9</u>:

In the cited reference, the [ ]. As described on Page 4-7 of the supplement 2 topical report, Reference 9-1 "...only the loss of clearance between the fuel rod and the upper tie plate has the potential to affect safety margins since interference may cause additional rod bow and lead to fuel rod failures." This means that methods for evaluating other dimensional changes in the fuel assembly are not presented in this current report for NRC approval since the evaluations do not impact safety margins. For the specific evaluation of the fuel channel overlay of the lower tie plate seal spring, the safety margin calculation is reliant on having an accurate or conservative estimation of the leak rate to the bypass. The accuracy required for leak rate at end of life is so low that seal spring coverage becomes irrelevant. Analyses have shown, such as the one sent to the NRC in Reference 9-2, that the additional leakage from losing seal spring contact at end of life is not enough to affect the safety margins.

RAI-9 Response References

9-1. BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods." April 2016.

9-2. Letter, H. D. Curet (SPC) to NRC Document Control Desk, ATTN: Mr. T. E. Collins, Subject: “NRC request for Safety Assessment Related to Failed Seal Springs,” ML# 9710140031, HDC:97:108, dated October 3, 1997.

RAI-10:

The update process for the models described in the TR is described in Section 5.0, "UPDATE PROCESS."

a. *Although the TR data appears to be predictable based on burnup alone, growth of SRA Zry-2 depends on factors such as the amount of cold work (i.e., manufacturing process) and the presence of hydrogen or hydrides due to corrosion. Explain why the TR correlation will be adequate to bound future fuel rod designs that may have different manufacturing processes, plant water chemistry, etc.*

b. *During the acceptance review, AREVA stated that fuel rod growth is independent of fuel design and that cladding material drives the need for different growth correlations. However, the need for an [*

],

demonstrate otherwise. Given the provided data, explain why fuel rod growth will remain independent of future fuel designs (e.g., ATRIUM 11 and other evolutions of this design that may or may not contain fuel additives). This may be covered under Section 5.0, "Update Process," of the TR.

c. *Although the section states that models will be reviewed against a growing post irradiation examination database, it does not specify with what frequency. If the frequency is too low, data may be added that could non-conservatively invalidate current models without having to submit updated models for NRC review and approval. Specify an appropriate minimum frequency.*

d. *The following statement does not contain a sufficient level of specificity: "The threshold for submittal of the growth and bow correlations is an increase of the correlation tolerance limits by one standard deviation." To fully understand the criterion, provide additional specificity. For example: (1) Provide the mathematical definition of the standard deviation being referred to and why it is appropriate (e.g., Why not use standard error?), and (2) Does the increase have to be observed over the entire burnup range, some subset of the burnup range, or something else?*

AREVA Response RAI-10a and RAI-10b:

The fuel rod growth correlations are not adequate to bound future fuel rod designs with significantly different manufacturing processes, plant water chemistry, fuel designs, or new materials. In such cases, out of pile testing and leads programs are undertaken to acquire the necessary data to support evaluation methods which are presented to the NRC for approval. The new chromia-doped fuel product is an example of such a program. However, in the absence of significant changes the current fuel rod growth correlations are adequate to bound future performance since they are based on well-populated databases with data from multiple fuel assembly designs and varying plant chemistries. The factors which affect rod growth are well known and AREVA will not introduce changes to manufacturing or operation that may result in performance that deviates from the established database without a test or demonstration program. It is possible that performance may drift over time with the accumulation of small changes in design or operation. This is monitored through the acquisition and evaluation of new post-irradiation data with modifications of the growth correlation subject to the Update Process described in Section 5.0 of the topical report.

AREVA Response RAI-10c

All post-irradiation data is reviewed as it is acquired with thresholds available to site personnel to indicate data which requires urgent review by engineering. The exam reports sent to the customer summarize the data and evaluate whether it was within expectations based on current approved models and analyses of record supporting fuel licensing. If the data is outside the predictions, an internal condition report will be created and evaluated for potential Part 21 reporting. Assuming the acquired data is within the predicted range, the database would only be reviewed annually subject to the Update Process described in Section 5.0 of the topical report.

AREVA Response RAI-10d

New measurement data, as described in response to 10c, will be added to the applicable databases. Using the same method as described in RAI-2, the new upper or lower limits supported by the updated database will be calculated. If the new limits are outside the envelope defined by the approved limits plus or minus one standard deviation, a new correlation will be submitted to the NRC for approval. This is the same threshold for resubmittal as previously approved for similar correlations used in Reference 10-1 and 10-2. If the new limits are within the envelope, then AREVA can use the updated correlation without prior approval by the NRC.

RAI-10 Response References

10-1. Letter from H. N. Berkow (NRC) to R. L. Gardner (AREVA), "Final Safety Evaluation for Framatome ANP (FANP), Topical Report (TR) EMF-93-177(P) Revision 1, Mechanical Design for BWR [Boiling Water Reactor] Fuel Channels, (TAC NO. MC5665)," August 23, 2005, Section 4.0 of Enclosure.

10-2. EMF-85-74(P)(A) Revision 0 with Supplement 1 (P)(A) and Supplement 2 (P)(A), RODEX2A (BWR) Fuel Rod Thermal-Mechanical Evaluation Models, Siemens Power Corporation Nuclear Division, Richland, WA (98Feb.), Supplement 2 page A-2.

ATTACHMENT 1:

Planned Marked Up Pages For
BAW-10247PA Supplement 2P Revision 0

[ ]

framatome

February 23, 2018
NRC:18:007

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Additional Information Regarding BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors Supplement 2: Mechanical Methods"

Ref. 1: Letter, Gary A. Peters (AREVA Inc.) to Document Control Desk (NRC), "Request for Review and Approval of BAW-10247P-A, Supplement 2P, Revision 0, 'Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods'," NRC:16:012, April 29, 2016.

Ref. 2: Letter, Dennis C. Morey (NRC) to Gary Peters (AREVA Inc.), "Draft Safety Evaluation for AREVA Inc. Topical Report BAW-10247P-A, Supplement 2P, Revision 0, 'Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors, Supplement 2: Mechanical Methods' (CAC No. MF7708)," October 26, 2017.

Framatome Inc. (Framatome, formerly AREVA Inc.) requested the NRC's review and approval of the topical report BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods" in Reference 1.

Framatome is requesting approval to use Zry-2 recrystallized (RXA) material for cladding and Z4B™ material for water channels in addition to approval for the use of various mechanical models with this topical report. A telephone call was held with the NRC staff on December 15, 2017 and Framatome agreed to provide the following information:

1) Additional text in the topical report to clarify the use of the mechanical models for Zry-2 RXA cladding.
2) A description of the material properties of Z4B™ and the mechanical analysis that would be performed for Z4B™ water channels in addition to fuel assembly growth.
3) A summary of the additional data that Framatome has gathered for Z4B™ water channel fuel assembly growth and of the recalculation of the growth model.

Framatome understands that this information will be used to support a modification of the draft safety evaluation (Reference 2) to reflect approval 1) to use Zry-2 recrystallized material for cladding, 2) to use Z4B™ material for water channels, and 3) a higher burnup limit for the fuel assembly growth model for Z4B™ water channels. The additional information is provided in the enclosures to this letter. Framatome would appreciate the issuance of the final SE by no later than June 2018.

Framatome Inc.
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Framatome considers some of the material contained in the enclosed documents to be proprietary. As required by 10 CFR 2.380(b), an affidavit is enclosed to support the withholding of information form public disclosure.

There are no commitments contained within this letter or its enclosures.

If you have any questions related to this information, please contact Ms. Gayle Elliott, Product Licensing Manager, by telephone at (434) 832-3347, or by e-mail at Gayle.Elliott@framatome.com.

Sincerely,

Gary Peters, Director
Licensing & Regulatory Affairs
Framatome Inc.

cc: J. G. Rowley
Project 728

Enclosures

1. BAW-10247P-A, Supplement 2Q2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors Supplement 2: Mechanical Methods"
2. BAW-10247NP-A, Supplement 2Q2NP, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors Supplement 2: Mechanical Methods"
3. Notarized Affidavit

AFFIDAVIT

STATE OF WASHINGTON)
) ss.
COUNTY OF BENTON)

1. My name is Alan B. Meginnis. I am Manager, Product Licensing, for Framatome Inc. and as such I am authorized to execute this Affidavit.

2. I am familiar with the criteria applied by Framatome to determine whether certain Framatome information is proprietary. I am familiar with the policies established by Framatome to ensure the proper application of these criteria.

3. I am familiar with the Framatome information contained in the report BAW-10247P-A Supplement 2Q2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors Supplement 2: Mechanical Methods Additional Information," dated February 2018 and referred to herein as "Document." Information contained in this Document has been classified by Framatome as proprietary in accordance with the policies established by Framatome for the control and protection of proprietary and confidential information.

4. This Document contains information of a proprietary and confidential nature and is of the type customarily held in confidence by Framatome and not made available to the public. Based on my experience, I am aware that other companies regard information of the kind contained in this Document as proprietary and confidential.

5. This Document has been made available to the U.S. Nuclear Regulatory Commission in confidence with the request that the information contained in this Document be withheld from public disclosure. The request for withholding of proprietary information is made

in accordance with 10 CFR 2.390. The information for which withholding from disclosure is requested qualifies under 10 CFR 2.390(a)(4) "Trade secrets and commercial or financial information."

6. The following criteria are customarily applied by Framatome to determine whether information should be classified as proprietary:

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(b) Use of the information by a competitor would permit the competitor to significantly reduce its expenditures, in time or resources, to design, produce, or market a similar product or service.

(c) The information includes test data or analytical techniques concerning a process, methodology, or component, the application of which results in a competitive advantage for Framatome.

(d) The information reveals certain distinguishing aspects of a process, methodology, or component, the exclusive use of which provides a competitive advantage for Framatome in product optimization or marketability.

(e) The information is vital to a competitive advantage held by Framatome, would be helpful to competitors to Framatome, and would likely cause substantial harm to the competitive position of Framatome.

The information in the Document is considered proprietary for the reasons set forth in paragraphs 6(b), 6(d) and 6(e) above.

7. In accordance with Framatome's policies governing the protection and control of information, proprietary information contained in this Document have been made available, on a limited basis, to others outside Framatome only as required and under suitable agreement providing for nondisclosure and limited use of the information.

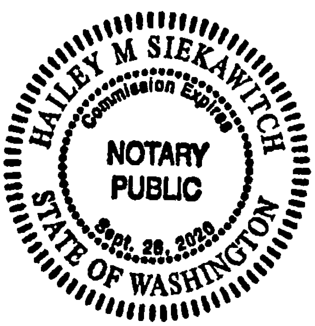
8. Framatome policy requires that proprietary information be kept in a secured file or area and distributed on a need-to-know basis.

9. The foregoing statements are true and correct to the best of my knowledge, information, and belief.

SUBSCRIBED before me this 21st

day of February, 2018.

Hailey M Siekawitch
NOTARY PUBLIC, STATE OF WASHINGTON
MY COMMISSION EXPIRES: 9/28/2020



framatome

Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors
Supplement 2: Mechanical Methods Additional Information

BAW-10247NP-A
Supplement 2Q2NP
Revision 0

Topical Report

February 2018

Framatome Inc.

Summary

Framatome Inc. (f/k/a AREVA Inc.) is requesting approval to use Zry-2 recrystallized (RXA) material for cladding and Z4B™ material for water channels in addition to approval for the use of various mechanical models with this topical report. A telephone call was held with the NRC staff on December 15, 2017 and Framatome agreed to provide the following additional information:

1) Additional text in the topical report to clarify the use of the mechanical models for Zry-2 RXA cladding.

2) A description of the material properties of Z4B™ and the mechanical analysis that would be performed for Z4B™ water channels in addition to fuel assembly growth.

3) A summary of the additional data that Framatome has gathered for Z4B™ water channel fuel assembly growth and of the recalculation of the growth model.

The additional information is provided in the form of markup pages for the topical report. To address item 1 above, the application of the rod bow model and rod growth model to RXA cladding has been clarified with markups to Appendices A and B, respectively. A new Appendix D is provided to address item 2. Markup pages for Appendix C are provided to address item 3. Markups of various pages in the topical report are provided to clarify that we will use both Zry-2 CWSR and RXA for cladding and Zry-4 and Z4B™ will be used for water channels.

MARKUP PAGES

- Page ii
- Page iii
- Page 1-2
- Page 2-2
- Page 4-1
- Page 4-2
- Page 4-4
- Page A-1
- Page B-1
- Page B-4
- Page C-2
- Page C-4
- Added Appendix D (Pages D-1 through D-9)

Contents

List of Tables

List of Figures

fuel rod exposure limit for the realistic fuel rod thermal-mechanical methodology has been established in Reference 1.

The mechanical methods described in this report will be applied to all BWR fuel designs. The fuel rod cladding will be either Zry-2 SRA or RXA cladding. The water channel will be made from either Zry-4 or Z4B™[1] material.

[1] Z4B is a trademark of AREVA Inc.

correlations for fuel rod bow, fuel rod growth, and fuel assembly growth due to the incorporation of recent operating experience data. Note that the fuel rod growth correlation presented in this supplement is based on the total axial rod growth measured in post-irradiation exams, and therefore does not affect the RODEX4 stress-free irradiation growth model in Reference 1.

The rod bow correlation is constructed from fuel rod-to-rod gap closure data measured on a broad selection of AREVA BWR fuel designs in varied operating environments. The rod-to-rod gap closure predicted as a function of fuel assembly exposure is used as an input to thermal limit evaluations (i.e. MCPR) for AREVA BWR fuel designs.

The BWR rod growth correlation is updated with the most recent data from AREVA's Zircaloy-2 stress-relief annealed (SRA) cladding, [

] . The BWR fuel assembly growth correlation is built from post-irradiation length measurement data taken from ATRIUM™[1] fuel assemblies. This model is applicable to all ATRIUM™ fuel assembly designs for which assembly growth is controlled by the water channel growth, including the ATRIUM™ 11 with Z4B™~~[2]~~ water channels. The combination of the fuel rod and assembly growth correlations is used to define the maximum fuel rod length which will not interfere with the upper tie plate at end of life. This is the only mechanical method defined in this report which is limiting at end of life, and the growth databases support the maximum requested fuel assembly exposure limit.

---

1 ATRIUM is a trademark of AREVA Inc.
~~2 Z4B is a trademark of AREVA Inc.~~

4.0 ANALYTICAL METHODOLOGY

4.1 *Mechanical Methods*

The methods described in this section cover the topics included in SRP Section 4.2 II.1.A for fuel system damage except those already addressed in the base topical report (Reference 1). These methods support AREVA's generic BWR fuel design criteria approved in Reference 3 throughout the design lifetime of the fuel and cover handling, normal operation and AOO conditions. The only changes to the previously approved BWR fuel mechanical methods are updates to fuel rod bow, fuel rod growth, and fuel assembly growth correlations and extension to Zry-2 RXA cladding and Z4B™ water channels.

4.1.1 Stress, strain or loading limits

The strength of the fuel assemblies and fuel rods is assured by evaluating the margin to conservative stress and deformation design limits under various shipping, handling and operational loads. The loads are applied to the fuel rod cladding, upper and lower tie plates, grid spacers, water channel (or tie rods) and connecting hardware, fuel assembly cage and springs where applicable. AREVA defines a maximum axial handling design load equivalent to [ ].

As described in Reference 3, AREVA uses Section III of the ASME Boiler and Pressure Vessel Code as guidance for establishing the acceptable stress, strain, or load criteria for assembly components and the corresponding analysis methods which may be used to evaluate those criteria. These methods include elastic and plastic analysis techniques as well as load rating from prototype testing. Analysis methods include use of conventional, open-literature equations, elasticity formulations, general purpose finite element stress analysis codes such as ANSYS, or testing.

The minimum specified yield and ultimate strength for unirradiated material are used in the analyses. This is a conservative assumption since strength will increase under

irradiation. Since loads often stay the same or decrease over time, the beginning of life (BOL) strength evaluations tend to be the most limiting. This is true even when the material loss due to oxidation that would be expected at end of life (EOL) is factored in. The oxide is either insignificant, as observed on stainless steel and nickel alloy components; or the oxide is on relatively thick components such as the ~~Zircaloy~~ zirconium alloy water channel and fuel channel. Zircaloy fuel rod cladding (SRA or RXA) is the only component with a specific EOL analysis requiring the assumed loss of material due to corrosion. However, even in this case the EOL analysis is not limiting due to the reduced loads at EOL.

While all load bearing fuel assembly components have some analysis or test to validate that criteria are met for the given design loads, a few of the components have standard evaluations as described below.

4.1.1.1 Fuel rod cladding

Various normal operation and AOO loads create stresses on the fuel rod cladding. Each individual stress is calculated at the inner and outer surfaces of the cladding at both the mid-span between spacer grids and at the spacer grid. The stresses at each location are then combined to determine the maximum stress intensities. The analysis is performed at BOL and EOL and at cold and hot conditions with unirradiated material strength. The stress analysis assumes maximum fuel rod power, minimum fill gas pressure, and the most conservative fuel rod geometry including a reduced wall thickness at EOL due to oxidation. The methods are applicable to both Zry-2 SRA and RXA cladding.

[

] The cladding stress analysis method has not changed from what was documented in Section 3.4.3 of Reference 4. The stress calculations use conventional,

Flow testing is used to confirm acceptable bypass flow characteristics. The seal spring is designed with adequate deflection range for accommodating the maximum expected channel bulge while maintaining an acceptable leakage rate. Seal spring stresses are analyzed using a finite element method or handbook equations.

4.1.2 Strain fatigue

Fatigue of structural components is generally low because the cyclic loadings on the structural components typically have either a small number of cycles (i.e. reactor startup) or small amplitude (i.e. flow-induced vibration). Cyclic loading associated with relatively large changes in power can cause cumulative damage which may eventually lead to fatigue failure. The O'Donnell and Langer fatigue curves are used in the analysis of ~~Zircaloy~~ zirconium alloy components (Reference 7). These fatigue curves incorporate the NRC recommended "2 or 20" safety factor. This safety factor reduces the stress amplitude by a factor of two or reduces the number of cycles by a factor of twenty, whichever is more conservative. The fatigue curves provide the maximum allowed number of cyclic loadings for each stress amplitude. The fatigue usage factor is the number of expected cycles divided by the number of allowed cycles. The total cumulative usage factor is the sum of the individual usage factors for each duty cycle.

4.1.3 Fretting wear

Fretting wear is a concern for the fuel rod cladding. Fretting wear may occur on the fuel cladding surfaces in contact with the spacer grids if there is a reduction in grid spacer spring loads in combination with small amplitude, flow induced, vibratory forces.

[

]

APPENDIX A BWR FUEL ROD BOW CORRELATION

Introduction

AREVA has gathered post-irradiation rod-to-rod gap closure measurements from a variety of BWR fuel designs as shown in Figure A-1. Both the absolute and the percent gap closure data for all measured AREVA BWR designs (7x7, 8x8, 9x9, and 10x10) reveal [

] This new correlation will provide a bounding estimation of fuel rod-to-rod gap closure that will be used for all current and future AREVA BWR fuel designs in the United States for both SRA or RXA cladding.

Measurement Description

AREVA has conducted PIE campaigns both in the U.S. and in Europe to collect fuel rod-to-rod gap measurements. This global database contains fuel rods with both SRA and RXA cladding. The fuel rod-to-rod gap database includes AREVA's legacy designs with 7x7, 8x8, and 9x9 rod arrays which are no longer in operation, and also the ATRIUM™-10 design still in use today.

Fuel rod-to-rod gap measurements are typically taken at each span between spacer grids (usually 8 spans) and at each fuel rod-to-rod gap. In addition, measurements can

APPENDIX B BWR FUEL ROD GROWTH CORRELATION

Introduction

The fuel rod growth correlation was most recently approved in 1998 using Zircaloy-2 (Zry-2) Stress Relief Annealed (SRA) growth data obtained from post irradiation examination (PIE) campaigns. This correlation has been updated to include the Zry-2 SRA rod growth data obtained from PIE campaigns since 1998. It is also conservative to use this rod growth correlation for Zry-2 RXA cladding since it has been observed to have less axial growth than SRA cladding.

Measurement Description

Figure B-1 shows the fuel rod growth correlation containing the data presented in the rod growth correlation from 1998, the data in open blue markers, as well as the new data collected since 1998, the closed red markers. The data include fuel rods from 7x7, 8x8, 9x9 and 10x10 arrays (ATRIUM™-10). [

]

Correlation Development

The fuel rod growth data (expressed as percent of active fuel length) versus assembly average burnup is shown in Figure B-1. [

]

Summary

The results of the fuel rod growth linear correlation are summarized in Table B-1. The maximum fuel assembly exposure level represented by the data is [

] Based on the data and similarity in manufacturing processes, the BWR rod growth correlation is fully applicable to AREVA BWR fuel rod designs with SRA and RXA Zry-2 cladding.

[

]

Table C-1 BWR Fuel Assembly Growth Correlation

[

]

[

]

Figure C-1 BWR Fuel Assembly Growth Correlation

APPENDIX C REFERENCES

C-1 Factors for One-Sided Tolerance Limits and For Variables Sampling Plans, D. B. Owen. Sandia Corporation Monograph (SRC-607), March 1963.

APPENDIX D Z4B™ WATER CHANNEL ASSEMBLY

Introduction

BWR fuel assemblies have used a zirconium alloy tie structure for several decades. The tie structure can consist of fueled tie rods, water rods, a water channel, or a fuel channel. Most BWR fuel designs have used either Ziraloy-2 or Zircaloy-4 alloys as defined in ASTM B352/B352M (Reference D-1) for fuel channels with acceptable performance. However, the behavior of these alloys can be improved for use in the tie structure. More corrosion occurs on Zircaloy-4 than Zircaloy-2 in a BWR coolant environment, and there is a higher hydrogen pickup fraction in Zircaloy-2 than Zircaloy-4. The ideal alloy would have both low corrosion and low hydrogen pickup. A proprietary zirconium alloy has been developed, Z4B™, which optimizes the alloying element concentrations for improved corrosion and hydrogen pickup when used for a BWR fuel structural component. This appendix provides additional information on Z4B™ and its application to a water channel assembly used as part of a BWR fuel assembly tie structure.

Alloy Composition

The composition of Z4B™ is shown in Table D-1 and is similar to that of Zircaloy-4 (Zry-4) as defined in ASTM B352/B352M (Reference D-1), though Z4B™ has slightly higher iron (Fe) and chromium (Cr) contents. [

] Both alloys are composed of about 98 wt% zirconium and have a hexagonal crystal structure at room and service temperatures. The small differences in composition between Z4B™ and Zry-4 do not result in any significant differences in fabrication methods or processes.

Table D-1 Alloy Composition

Element	Composition (weight percent)	
	Zry-4 (UNS R60804)	Z4B™
Zirconium (Zr)	~ 98	~ 98
Tin (Sn)	1.20 – 1.70	[ ]
Iron (Fe)	0.18 – 0.24	[ ]
Chromium (Cr)	0.07 – 0.13	[ ]
Nickel (Ni)	-	[ ]

The motivation for increasing Fe and Cr in Z4B™ is to improve the corrosion resistance and hydrogen uptake relative to Zry-4. Industry experience indicates that increases in Fe and Cr act to reduce the corrosion rate of Nb-free alloys such as Zry-4 and Z4B™ (Reference D-2). Table D-2 indicates that the "best alloy content" for Fe is greater than 0.3% and for Cr is above 0.15 % with respect to corrosion resistance and hydrogen pickup fraction. These values compare well with the ranges listed above for Z4B™ (Table D-1). As indicated in Table D-2, the solubility of Fe and Cr in the zirconium matrix is very low, which means that these elements exist primarily in second phase particles (SPPs). Given the similarity in composition and crystal structure of Z4B™ and Zry-4, that the solubility of Fe and Cr in the alloy matrix is similar for these alloys, the additional concentrations of these elements in Z4B™ could result in a larger number of SPPs, a larger average SPP size, or both depending on the details of material processing. The processing of Z4B™ targets [

]. The superior corrosion and hydrogen uptake performance of Z4B™ relative to Zry-4 has been demonstrated through a material test program for Z4B™ spacer grids and recent measurements collected on Z4B™ Lead Use Fuel Channels (Reference D-3).

Table D-2 Effect of Alloying Elements on Corrosion of Zirconium Alloys (Reference D-2)

Element	Solubil. (%)	Best alloy content (%)	Out-pile corrosion	In-PWR corr.	LiOH corr.	In-BWR corr.	HPUF
Sn	2	0/>1	—	=	+ +	+	0
Nb	0.5	0.5/>2	+ +	+ +	0	—/+	+/0
Fe	<0.01	≥0.3	+ +	+ +	+ +	+	0/+
Cr	<0.01	≥0.15	+/—	+	+ +	+	+ (>0.15)
Ni	<0.01	0.05	+ +	+		+	=/0
V	<0.01	≥0.15	+/—	+	+ +	+	+
Cu	<0.1	≥0. 5	+				0
0: no effect, — increase, = strong increase, + reduction, ++ strong reduction, 0/+ effect differs in different environments.							

Water Channel Assembly

The water channel is made from sheet material and formed into the shape of a square duct with rounded corners. End fittings are welded to the ends of the water channel which allow the upper and lower tie plates to be secured to the water channel assembly. Inlet and outlet holes in the end fittings permit the flow of single-phase water through the water channel. Spacer stops are welded to the sides of the water channel to control the axial position of the spacer grids. The water channel plus the spacer stops and end fittings constitute the water channel assembly. See Figure D-1 for an illustration. Although not shown in the figure, some water channel assemblies also have 'crowns' which are thin metallic strips welded to the water channel with the purpose of diverting single-phase water toward the fuel rods. All water channel assembly components are made from a zirconium alloy, i.e. Z4B™.

The structural tie between the lower tie plate (LTP) and the upper tie plate (UTP) is provided by the water channel assembly. Within the ATRIUM™ family of BWR fuel designs, there have been a few variations of the water channel assembly design. Currently the upper end fitting contains an integrated connecting rod which extends

from the water channel up to the UTP locking hardware. The LTP is secured to the water channel lower end fitting by a threaded connection. Large cross-sectional threaded fasteners and connecting hardware ensure a strong connection between the two tie plates.

[

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Figure D-1 Water Channel Assembly

Design Loads and Requirements

The generic design criteria for BWR fuel designs have been defined in Reference D-4. Requirements for the fuel design have been developed based on the guidance in the Standard Review Plan, including some specific requirements related to the tie structure. The specific requirements applicable to the tie structure have been defined in Section 3.3.1 of Reference D-4 for stress, strain or loading limits, Section 3.3.6 for axial growth, and Section 3.3.9 for fuel assembly handling (see Table D-3 which quotes the criteria from Reference D-4).

Table D-3 Generic BWR Design Criteria Applicable to the Tie Structure during Handling, Normal Operation, and AOO (Reference D-4)

Section in Ref. D-4	Topic	Criteria
3.3.1	Stress, Strain, or Loading Limits	[ ]
3.3.6	Axial Irradiation Growth	[ ]
3.3.9	Fuel Assembly Handling	[ ]

The primary mechanical function of the BWR fuel assembly tie structure is to allow the fuel assembly to be lifted by a grapple attached to the upper tie plate. However, the Fuel Handling Accident evaluation assumes the fuel assembly is dropped over the core which could occur either from a crane failure or a break of the fuel assembly tie structure itself. In order to provide substantial design margin against such an accident occurring, Section 3.3.9 of Reference D-4 provides a fuel assembly handling requirement that a test or analysis of the assembly must not [

].

Of more significance to safety performance is that the tie structure maintains an acceptable dimensional configuration in the core during normal operation and anticipated operational occurrences (AOO). The design loads are small and consist mostly of component weight, friction forces transmitted through spacer grids, and hydraulic differential pressure. These loads are analyzed against the strength requirements listed in Table D-3. In addition, a dimensional analysis must be performed to ensure the deformation caused by these loads (including the effects of thermal expansion, growth, and creep) do not significantly affect design engagements and clearances. There are no significant cyclic loads so a break due to fatigue is not a concern.

Analytical Methods

The mechanical analytical methods have been described in Section 4 of this topical report. Section 4.1.1 describes the general strength evaluations which include those performed for the water channel assembly to demonstrate that the stress, strain, and loading limit criteria (including fuel handling) defined above in Table D-3 are met. The use of Z4B™ material has no significant impact on either the method or calculated margins for these evaluations.

As discussed in Section 2 of this topical report, the composition of Z4B™ differs from ASTM Zry-4 only by slight increases in Fe and Cr. Both alloys are composed of about 98 wt% zirconium and have a hexagonal crystal structure at room and service temperatures. Based on these considerations, there are negligible differences in basic material properties (e.g., elastic moduli, heat capacity, thermal expansion, thermal conductivity, density, etc.) between Z4B™ and Zry-4. The minimum specified strengths are used in the analyses. As shown in Table D-4, the minimum mechanical property requirements for Z4B™ water channel strip are the same as or higher than those for Zry-4 water channel strip. There is no difference in the specification limits for end plug barstock. Therefore, the use of Z4B™ in place of Zry-4 for these components will not result in a reduction in strength margins. The strength analyses are also dependent on corrosion and related wall thinning. As discussed in Section 2, the compositional increases in Fe and Cr result in superior corrosion resistance and lower hydrogen uptake of Z4B™ relative to Zry-4. Therefore, the corrosion and hydrogen embrittlement behavior of Z4B™ components will be bounded by that of Zry-4 components.

Table D-4 Unirradiated Strength Specifications for Zry-4 and Z4B™

[

]

General fit-up and deformation analyses are performed at BOL and EOL conditions to demonstrate that adequate engagements and clearances are maintained throughout the fuel assembly lifetime as described in Section 4.1.5. The methodology for axial growth is described in Section 4.1.5.2 which demonstrates that the criteria for axial irradiation

growth defined in Table D-3 are met. The assembly growth correlation with Z4B™ water channels is documented in Appendix C.

Testing of unirradiated, recrystallized Zry-2, Zry-4, and Z4B™ has shown that differences in alloy composition between these materials have no significant effect on creep rate. [

]. Overall, the difference is not judged to be significant, and it remains conservative to use the creep rate already defined for Zry-2 and Zry-4 fuel channel material in Reference D-5.

APPENDIX D REFERENCES

D-1. ASTM B352/B352M, *Standard Specification for Zirconium and Zirconium Alloy Sheet, Strip, and Plate for Nuclear Application.*

D-2. Rudling, P., Zr Alloy Corrosion and Hydrogen Pickup. A.N.T. International. NRC Accession Number: ML15253A227. 2013.

D-3. ANP-10336P-A, Revision 0, "Z4B™ Fuel Channel Irradiation Program," AREVA NP, July 2017.

D-4. ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels, April 1995.

D-5. EMF-93-177 Revision 1 Supplement 1(P)(A), "Mechanical Design for BWR Fuel Channels, Supplement 1: Advanced Methods for New Channel Designs," AREVA NP, September 2013.

framatome

July 10, 2018
NRC:18:027

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Additional Information Regarding BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods"

Ref. 1: Letter, Gary A. Peters (AREVA Inc.) to Document Control Desk (NRC), "Request for Review and Approval of BAW-10247P-A Supplement 2P, Revision 0, 'Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods'," NRC:16:012, April 29, 2016.

Framatome Inc. (Framatome, formerly AREVA Inc.) requested the NRC's review and approval of the topical report BAW-10247P-A, Supplement 2P, Revision 0, "Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors Supplement 2: Mechanical Methods" in Reference 1.

The NRC provided an additional question on BAW-10247P-A Supplement 2P in a telephone call on June 19, 2018. A response to the additional question is provided in the enclosure to this letter. This letter will be included in the approved version of the topical report when it is issued.

There are no commitments within this letter or its enclosures.

If you have any questions related to this submittal, please contact Ms. Gayle F. Elliott, Deputy Director, Licensing and Regulatory Affairs, by telephone at 434-832-3347, or by e-mail at Gayle.Elliott@framatome.com.

Sincerely,

Gary Peters, Director
Licensing & Regulatory Affairs
Framatome Inc.

cc: J. G. Rowley
Project 728

Enclosures:

1. Response to NRC Question Regarding BAW-10247P-A Supplement 2P

Enclosure 1:
Response to NRC Question Regarding BAW-10247P-A, Supplement 2P

NRC Reviewer Question:

If Framatome is not including the upper tolerance limit for fuel assembly maximum axial growth in BAW-10247 Sup 2, how will co-resident compatibility be analyzed with fuel assemblies at the channel spacer/spring interface, considering BOL and EOL axial growth conditions?

Response:

Per ANF-89-98(P)(A) Revision 1, external interfaces, including channel spacer/springs, are evaluated for all new fuel designs and for compatibility with co-resident fuel assemblies. When verifying channel spacer/spring engagement, the EOL upper tolerance value is derived from the nominal fuel assembly growth correlation presented in the BAW-10247 Supplement 2 Table C-1 by adding the tolerance value, T. This methodology confirms engagement between adjacent fuel assemblies considering BOL and EOL axial growth conditions.

AREVA

Realistic Thermal-Mechanical Fuel Rod Methodology For Boiling Water Reactors
Supplement 2: Mechanical Methods

BAW-10247NP-A
Supplement 2NP
Revision 0

Topical Report

April 2016

AREVA Inc.

Nature of Changes

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

Contents

List of Tables

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Nomenclature

Acronym	Definition
AOO	Anticipated Operational Occurrence
BOL	Beginning of Life
CFR	Code of Federal Regulations
DNB	Departure from Nucleate Boiling
EOL	End of Life
FMM	Fuel Management Manual
GDC	General Design Criteria
LTL	Lower Tolerance Limit
LTP	Lower Tie Plate
PIE	Post Irradiation Examination
QAP	Quality Assurance Program
RXA	Recrystallized Annealed
SRA	Stress-Relief Annealed
SRP	Standard Review Plan
UTL	Upper Tolerance Limit
UTP	Upper Tie Plate
Z4B™	AREVA Proprietary Zirconium Alloy
Zry-2	Zircaloy-2 Alloy
Zry-4	Zircaloy-4 Alloy

ABSTRACT

This supplement covers the balance of the BWR fuel mechanical methods which were not updated in the base topical report. The purpose of consolidating these miscellaneous mechanical methods into this supplement is to remove the existing ties to legacy methodology reports. This provides a well-defined licensing basis for BWR nuclear plants which have moved to AREVA's realistic fuel rod methodology.

As part of this consolidation effort, the most recent operating experience data is provided in order to update the correlations for fuel rod bow, fuel rod growth, and fuel assembly growth. This data supports raising the fuel assembly and fuel channel exposure limit to a value that will not restrict the fuel design from reaching the fuel rod exposure limit established in the base topical report.

1.0 INTRODUCTION

AREVA has been upgrading the entire suite of Boiling Water Reactor (BWR) fuel rod analysis methods to the RODEX4 realistic fuel rod thermal-mechanical methodology. This methodology includes peripheral mechanical methods which do not use any thermal-mechanical code such as RODEX4. This supplement is intended to clearly define the entire set of mechanical methods which are applicable to AREVA BWR fuel designs licensed with the RODEX4 methodology.

The thermal-mechanical fuel rod methods which were approved by the NRC in Reference 1 covered the bulk of the fuel rod evaluation requirements in Section 4.2 of the Standard Review Plan (SRP) (Reference 2). However, as noted in the last paragraph of Section 3.1.1 of Reference 1, a small selection of supplemental analysis methods are still based on approval of a topical report tied to legacy methods. AREVA intends to eliminate the need for this legacy reference with the approval of this supplement report. Not only does this provide more cohesive licensing documentation, it allows AREVA the opportunity to update the operating experience database based on our current knowledge of fuel rod performance.

This supplement does not introduce any changes to AREVA's existing BWR methodology other than updates to correlations derived from operating experience data. The described methods are consistent with the underlying methods supporting the design criteria approved for generic application in Reference 3. Recent operating experience data is provided for fuel rod bow, fuel rod growth, and fuel assembly growth; and the correlations are adjusted accordingly. This data is used to establish a fuel assembly exposure limit of [ ]. Since fuel channels are licensed for one fuel assembly lifetime, this represents an incremental increase in exposure for channels as well. This fuel assembly exposure limit is high enough to allow fuel rods to achieve the currently approved fuel rod exposure limit. The

fuel rod exposure limit for the realistic fuel rod thermal-mechanical methodology has been established in Reference 1.

The mechanical methods described in this report will be applied to all BWR fuel designs. The fuel rod cladding will be either Zry-2 SRA or RXA cladding. The water channel will be made from either Zry-4 or Z4B™[1] material.

[1] Z4B is a trademark of AREVA Inc.

2.0 SUMMARY

This report describes the peripheral BWR mechanical methods within AREVA's realistic thermal-mechanical fuel rod methodology which had been included by reference in the base topical report (Section 3.1.1 of Reference 1). The legacy references are therefore superseded by this supplement; specifically

- 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.
- 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.)

Approval of this supplement allows AREVA to remove these legacy topical reports when RODEX4 methods are used for licensing. New analyses based on RODEX4 methods will implement this supplement and the described methods. However, existing analyses based on RODEX4 methods and analyses based on older methodologies, such as RODEX2A, will not be updated after approval of this supplement.

The information provided in this supplement justifies a maximum fuel assembly exposure limit of [ ] . This is a small increase over the previously approved fuel assembly burnup limit in order to allow fuel rods to achieve the currently approved fuel rod exposure limit. Since fuel channels are licensed for one fuel assembly lifetime, this represents an incremental increase in exposure for channels as well. The fuel rod exposure limit for the realistic fuel rod thermal-mechanical methodology has been established in Reference 1.

A review of the design methodology indicates no changes are required in the existing approved methodology or design criteria. However, there are adjustments to the

correlations for fuel rod bow, fuel rod growth, and fuel assembly growth due to the incorporation of recent operating experience data. Note that the fuel rod growth correlation presented in this supplement is based on the total axial rod growth measured in post-irradiation exams, and therefore does not affect the RODEX4 stress-free irradiation growth model in Reference 1.

The rod bow correlation is constructed from fuel rod-to-rod gap closure data measured on a broad selection of AREVA BWR fuel designs in varied operating environments. The rod-to-rod gap closure predicted as a function of fuel assembly exposure is used as an input to thermal limit evaluations (i.e. MCPR) for AREVA BWR fuel designs.

The BWR rod growth correlation is updated with the most recent data from AREVA's Zircaloy-2 stress-relief annealed (SRA) cladding, [

]. The BWR fuel assembly growth correlation is built from post-irradiation length measurement data taken from ATRIUM™[1] fuel assemblies. This model is applicable to all ATRIUM™ fuel assembly designs for which assembly growth is controlled by the water channel growth, including the ATRIUM™ 11 with Z4B™ water channels. The combination of the fuel rod and assembly growth correlations is used to define the maximum fuel rod length which will not interfere with the upper tie plate at end of life. This is the only mechanical method defined in this report which is limiting at end of life, and the growth databases support the maximum requested fuel assembly exposure limit.

1 ATRIUM is a trademark of AREVA Inc.

3.0 APPLICABLE REGULATORY GUIDANCE

Regulatory guidance for the review of fuel system designs and adherence to applicable General Design Criteria is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants", Section 4.2, "Fuel System Design" (Reference 2). In accordance with the Standard Review Plan (SRP) Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

- The fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),
- Fuel system damage is never so severe as to prevent control rod insertion when it is required,
- The number of fuel rod failures is not underestimated for postulated accidents, and
- Fuel coolability is always maintained.

The mechanical methods covered in this supplement will be limited to those that establish the design bases for the acceptance criteria as provided in SRP Section 4.2 II.1.A, "Fuel System Damage". These SRP acceptance criteria have been translated into specific requirements defined for AREVA BWR fuel designs in Reference 3. As shown in Table 3-1, only correlations supporting the fuel rod bow and axial growth methods have changed from previously approved methodology.

Table 3-1 Methods Supporting the Standard Review Plan Criteria

1. Design Bases	Item	Topic	Assessment	Location in this Report
A. Fuel System Damage	i	Stress, strain, or loading limits	There is no change from the previously approved fuel rod cladding stress methodology as described in Section 3.4.3 of Reference 4. For structural components (not cladding), the methods are included in this supplement but there is also no change from previously approved methods.	Section 4.1.1
	ii	Fatigue	There is no change from the previously approved fuel rod cladding fatigue methodology as described in Reference 1. For structural components (not cladding), the methods are included in this supplement but there is also no change from previously approved methods.	Section 4.1.2
	iii	Fretting wear	The methodology for evaluating fretting is included in the supplement, but there is no change from previously approved methods.	Section 4.1.3
	iv	Oxidation, hydriding, crud	There is no change from the previously approved fuel rod cladding corrosion methodology as described in Reference 1. For structural components (not cladding), the methods are included in this supplement but there is also no change from previously approved methods.	Section 4.1.4
	v	Dimensional changes	Fuel rod bow and axial growth methods are included in the supplement with updated correlations based on recent data. Fuel channel bow and bulge methods are covered in References 5 and 6.	Section 4.1.5
	vi	Rod internal gas pressure	There is no change from the previously approved fuel rod internal pressure methodology as described in Reference 1.	This topic is not addressed in this supplement.
	vii	Assembly liftoff	The methodology for evaluating assembly liftoff is included in the supplement, but there is no change from previously approved methods.	Section 4.1.6
	viii	Control rod reactivity and insertability	Fuel channel bow and bulge methods are covered in References 5 and 6. The fuel channel methodology ensures control rod insertability.	This topic is not addressed in this supplement.

4.0 ANALYTICAL METHODOLOGY

4.1 *Mechanical Methods*

The methods described in this section cover the topics included in SRP Section 4.2 II.1.A for fuel system damage except those already addressed in the base topical report (Reference 1). These methods support AREVA's generic BWR fuel design criteria approved in Reference 3 throughout the design lifetime of the fuel and cover handling, normal operation and AOO conditions. The only changes to the previously approved BWR fuel mechanical methods are updates to fuel rod bow, fuel rod growth, and fuel assembly growth correlations and extension to Zry-2 RXA cladding and Z4B™ water channels.

4.1.1 Stress, strain or loading limits

The strength of the fuel assemblies and fuel rods is assured by evaluating the margin to conservative stress and deformation design limits under various shipping, handling and operational loads. The loads are applied to the fuel rod cladding, upper and lower tie plates, grid spacers, water channel (or tie rods) and connecting hardware, fuel assembly cage and springs where applicable. AREVA defines a maximum axial handling design load equivalent to [ ].

As described in Reference 3, AREVA uses Section III of the ASME Boiler and Pressure Vessel Code as guidance for establishing the acceptable stress, strain, or load criteria for assembly components and the corresponding analysis methods which may be used to evaluate those criteria. These methods include elastic and plastic analysis techniques as well as load rating from prototype testing. Analysis methods include use of conventional, open-literature equations, elasticity formulations, general purpose finite element stress analysis codes such as ANSYS, or testing.

The minimum specified yield and ultimate strength for unirradiated material are used in the analyses. This is a conservative assumption since strength will increase under

irradiation. Since loads often stay the same or decrease over time, the beginning of life (BOL) strength evaluations tend to be the most limiting. This is true even when the material loss due to oxidation that would be expected at end of life (EOL) is factored in. The oxide is either insignificant, as observed on stainless steel and nickel alloy components; or the oxide is on relatively thick components such as the zirconium alloy water channel and fuel channel. Zircaloy fuel rod cladding (SRA or RXA) is the only component with a specific EOL analysis requiring the assumed loss of material due to corrosion. However, even in this case the EOL analysis is not limiting due to the reduced loads at EOL.

While all load bearing fuel assembly components have some analysis or test to validate that criteria are met for the given design loads, a few of the components have standard evaluations as described below.

4.1.1.1 Fuel rod cladding

Various normal operation and AOO loads create stresses on the fuel rod cladding. Each individual stress is calculated at the inner and outer surfaces of the cladding at both the mid-span between spacer grids and at the spacer grid. The stresses at each location are then combined to determine the maximum stress intensities. The analysis is performed at BOL and EOL and at cold and hot conditions with unirradiated material strength. The stress analysis assumes maximum fuel rod power, minimum fill gas pressure, and the most conservative fuel rod geometry including a reduced wall thickness at EOL due to oxidation. The methods are applicable to both Zry-2 SRA and RXA cladding.

[

] The cladding stress analysis method has not changed from what was documented in Section 3.4.3 of Reference 4. The stress calculations use conventional,

open-literature equations. A general purpose, finite element stress analysis code such as ANSYS may be used to calculate the stress due to spacer spring contact forces and the stress at the fuel rod end cap weld location.

4.1.1.2 Plenum spring

The internal fuel rod plenum spring provides an axial load on the fuel stack that is sufficient to assist in the closure of any gaps caused by handling, shipping, and densification. After fuel densification, the plenum spring has no functional requirements. Therefore, an EOL analysis is not required. The spring criteria are evaluated with coil spring handbook equations and validation testing.

4.1.1.3 Compression spring

The compression spring supports the upper tie plate (UTP) and fuel channel. The spring is evaluated with coil spring handbook equations and validation testing based on the deflection and specified spring force requirements. Irradiation-induced relaxation is taken into account to ensure the minimum compression spring force is greater than the combined weight for the UTP and fuel channel, including channel fastener hardware. Since the compression spring does not interact with the fuel rods, no evaluation is required for fuel rod buckling loads.

4.1.1.4 Seal spring

The lower tie plate (LTP) seal spring limits the bypass coolant leakage rate between the LTP and fuel channel. The seal spring accommodates the expected channel deformation while remaining in contact with the fuel channel. In addition, the seal spring must have adequate corrosion resistance and be able to withstand the operation stresses without yielding.

Flow testing is used to confirm acceptable bypass flow characteristics. The seal spring is designed with adequate deflection range for accommodating the maximum expected channel bulge while maintaining an acceptable leakage rate. Seal spring stresses are analyzed using a finite element method or handbook equations.

4.1.2 Strain fatigue

Fatigue of structural components is generally low because the cyclic loadings on the structural components typically have either a small number of cycles (i.e. reactor startup) or small amplitude (i.e. flow-induced vibration). Cyclic loading associated with relatively large changes in power can cause cumulative damage which may eventually lead to fatigue failure. The O'Donnell and Langer fatigue curves are used in the analysis of zirconium alloy components (Reference 7). These fatigue curves incorporate the NRC recommended "2 or 20" safety factor. This safety factor reduces the stress amplitude by a factor of two or reduces the number of cycles by a factor of twenty, whichever is more conservative. The fatigue curves provide the maximum allowed number of cyclic loadings for each stress amplitude. The fatigue usage factor is the number of expected cycles divided by the number of allowed cycles. The total cumulative usage factor is the sum of the individual usage factors for each duty cycle.

4.1.3 Fretting wear

Fretting wear is a concern for the fuel rod cladding. Fretting wear may occur on the fuel cladding surfaces in contact with the spacer grids if there is a reduction in grid spacer spring loads in combination with small amplitude, flow induced, vibratory forces.

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4.1.4 Oxidation, hydriding, and crud

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

[

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Oxidation, hydriding, and crud are the greatest at EOL. However, the effects of corrosion are not limiting at EOL. Design analyses have shown that irradiation increases material strength more than wall thinning due to oxidation reduces strength on fuel assembly structural components; [

].

4.1.5 Dimensional changes

The dimensions of fuel assembly components will change during irradiation which has the potential to affect safety margins. As detailed in the SRP, the thermal-hydraulic safety limits could be affected by rod bow. This includes the rod bow which would occur if the differential irradiation growth between the fuel rod and fuel assembly was enough to cause interference between the rod and upper tie plate. Fuel channel deformation due to bow and bulge is also a concern as it can affect control blade insertability if there is too much interference between the channel and blade. The topics of rod bow and axial growth are covered below. Fuel channel deformation is covered in References 5 and 6. AREVA uses empirical models to determine the expected bow and growth.

4.1.5.1 Rod bow

Differential expansion between the fuel rods, as well as 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. Therefore, AREVA has developed a correlation for predicting BWR

fuel rod-to-rod gap closure as a function of assembly burnup for use in the thermal-hydraulic safety analyses. Data shows that [ ] and is provided in Appendix A. This observation, combined with the fact that [ ]. The rod bow correlation has been updated and is provided in Appendix A.

4.1.5.2 Axial irradiation growth

AREVA sizes the fuel assembly components to have clearances or engagements which are sufficient to accommodate differential growth through EOL. There are a handful of interfaces which change due to axial irradiation growth such as the engagement between channel springs on adjacent assemblies, engagement of the fuel channel with the seal spring, engagement of the fuel rods in the spacer grids, and clearance between the fuel rod and the upper tie plate. ([ ]). While all of these evaluations are considered during design, only the loss of clearance between the fuel rod and the upper tie plate has the potential to affect safety margins since interference may cause additional rod bow and lead to fuel rod failures.

To evaluate the minimum EOL clearance between the fuel rod and tie plates it is necessary to determine correlations for the fuel rod growth and the fuel assembly growth derived from post-irradiation length measurements. The initial nominal clearance between the fuel rod and upper tie plate can then be reduced by an accounting of fabrication tolerances and uncertainty in the growth correlations. This determines the design margin for growth.

EOL calculations are limiting, and data has been provided which covers the maximum fuel assembly exposure limit. The fuel rod growth and fuel assembly growth

correlations have been updated in Appendices B and C, respectively, including an upper bound for the fuel rod growth and a lower bound for the fuel assembly growth. Note that the fuel rod growth correlation presented in this supplement is based on the total axial growth measured in post-irradiation exams which includes both stress-free growth and stress-induced growth. Only stress-free irradiation growth data is used to support the growth model in RODEX4 (Reference 1). Therefore, this new growth correlation has no impact on the RODEX4 stress-free irradiation growth model.

4.1.6 Assembly liftoff

AREVA requires that the fuel assembly not levitate due to hydraulic loads during normal operation and AOO conditions. The criterion covers both cold and hot conditions and uses the Technical Specification limits on flow.

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 component pressure drop coefficients are determined from flow testing. The calculated net force is confirmed to be in the downward direction, indicating no assembly liftoff. Maximum hot channel conditions are used in the calculation because the greater two-phase flow losses produce higher uplift force. At higher exposures, the lower reactivity results in lower two-phase pressure drop. This will result in a smaller overall lift force such that high exposure fuel assemblies are non-limiting.

Analyses to date indicate a large margin to assembly liftoff under normal operating conditions and AOO. Therefore, fuel liftoff in BWRs under normal operating conditions and AOO is considered to be of no concern.

5.0 UPDATE PROCESS

AREVA plans to continue to acquire rod bow and growth data on irradiated BWR fuel designs. As post-irradiation examination data are obtained, the AREVA PIE database will be expanded. Periodically, the models shown in the appendices will be reviewed against the growing database. If the data support a modification to these models, the internal AREVA design change process will be followed. This change process includes documentation and justification of the change and evaluation of the impact on future design analyses. Any changes to the models will be maintained in an internal AREVA document. A summary of any updates made to the models will be provided to the NRC in a letter for information only, unless the change exceeds the threshold required for submittal. The threshold for submittal of the growth and bow correlations is an increase of the correlation tolerance limits by one standard deviation.

The update process ensures that design margins are maintained, and it ensures compliance with any limitations specified in the NRC's Safety Evaluation Report. If the updates are outside of the NRC's Safety Evaluation Report limitations, then one of the following actions will be taken:

- No credit taken for the update, or
- Update documented for NRC review and approval.

The bow and growth correlations provided in the appendices are based on an extensive database covering several fuel designs operating in many different reactors. There is not expected to be a significant change in any of these correlations unless a significantly different material or fuel design is introduced. In such as case, the lead assembly process would be followed prior to reload supply to justify continued use of these correlations.

6.0 QUALITY ASSURANCE PROGRAM

Licensees and vendors use a variety of methods to evaluate the thermal, mechanical, and materials design of the fuel system. The NRC staff reviews these methods to ensure that they provide a realistic or conservative result and that they adhere to the requirements of the Code of Federal Regulations (CFR) and General Design Criteria (GDC). Regulations, which are applicable to thermal, mechanical and material design of the fuel system, are found in 10 CFR 50.46; GDC 10, 27 and 35; 10 CFR 50 Appendix K; and 10 CFR 100. Additionally, because the result of the transient and accident analysis methods are important to the safety of the nuclear power plants, these methods must be maintained under a quality assurance program (QAP) which meets the criteria set for in 10 CFR 50 Appendix B. The AREVA QAP is documented in Reference 8.

The AREVA QAP covers the procedures for design control, document control, software configuration control and testing, and error identification and corrective actions used in the development and maintenance of the evaluation model. The program also ensures adequate training of personnel involved with code development and maintenance, as well as those who perform the analyses.

7.0 REFERENCES

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

2. USNRC Standard Review Plan, Section 4.2 "Fuel System Design", NUREG-0800, Revision 3, March 2007.

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

4. ANF-88-152(P)(A) with Amendment 1 and Supplement 1, "Generic Mechanical Design for Advanced Nuclear Fuels 9x9-5 BWR Reload Fuel," Advanced Nuclear Fuels Corporation, August 1990.

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

6. EMF-93-177P-A Revision 1 Supplement 1P-A Revision 0, "Mechanical Design for BWR Fuel Channels, Supplement 1: Advanced Methods for New Channel Designs," AREVA NP Inc., September 2013.

7. W. J. O'Donnell and B. F. Langer, "Fatigue Design Basis for Zircaloy Components," Nuclear Science and Engineering, Volume 20, Number 1, September 1964.

8. FMM Revision 6, AREVA Mining Front –End Business Group Fuel Management Manual, effective December 2015.

APPENDIX A BWR FUEL ROD BOW CORRELATION

Introduction

AREVA has gathered post-irradiation rod-to-rod gap closure measurements from a variety of BWR fuel designs as shown in Figure A-1. Both the absolute and the percent gap closure data for all measured AREVA BWR designs (7x7, 8x8, 9x9, and 10x10) reveal [

] This new correlation will provide a bounding estimation of fuel rod-to-rod gap closure that will be used for all current and future AREVA BWR fuel designs in the United States for both SRA or RXA cladding.

Measurement Description

AREVA has conducted PIE campaigns both in the U.S. and in Europe to collect fuel rod-to-rod gap measurements. This global database contains fuel rods with both SRA and RXA cladding. The fuel rod-to-rod gap database includes AREVA's legacy designs with 7x7, 8x8, and 9x9 rod arrays which are no longer in operation, and also the ATRIUM™-10 design still in use today.

Fuel rod-to-rod gap measurements are taken at each span between spacer grids. All accessible rod-to-rod gaps are measured. Some locations, such as those behind the

water channel, are not always accessible by the tool. It is expected that the higher power rods tend to be on the outside of the assemblies, and these locations are generally accessible for measurement. In addition, measurements can be taken when the tool is inserted and withdrawn, for two measurements of every gap. For example, there are 120 fuel rod-to-rod gaps in a 9x9 assembly with a central water channel. Therefore, in this particular assembly, there could be up to 1920 measurements if two measurements per gap are recorded. For the ATRIUM™-10 family of fuel designs, there are 156 fuel rod-to-rod gaps which could lead to a total of up to 2496 measurements for the entire bundle if two measurements per gap are recorded.

[

]

Visual Inspections

When fuel rod-to-rod gap measurements are not taken, fuel rods are typically visually inspected for any signs of abnormal bow behavior. The ATRIUM™ 11, currently supplied as lead test assemblies, is also well-represented by the database as shown in recent visual examinations. [

] These inspections confirm that ATRIUM™ 11 is performing similar to past operating experience in 7x7, 8x8, 9x9, and 10x10 designs.

Table A-1 ATRIUM™ 11 Leads Visual Inspections

[

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Correlation Development

[

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[

] To include additional conservatism, the obtained 95/95 UTL function is multiplied by a 1.2 factor. This 1.2 factor was previously suggested in Section 2.5 of Reference A-2 to account for changes in rod bow at hot operating conditions with respect to the cold conditions where the measurements are taken.

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Table A-2 BWR Fuel Rod Bow Equation

[

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Summary

The comparison between the fuel rod bow correlation and the AREVA BWR rod-to-rod gap closure database is presented in Figure A-1. The chart contains the 95/95 UTL percent gap closure data per span for AREVA BWR fuel assembly designs, including 7x7, 8x8, 9x9, and ATRIUM™-10. The double yellow line represents the correlation as described by Equation A-4.

Visual exams on ATRIUM™ 11 have not revealed any unusual fuel rod bow behavior [

] Therefore, ATRIUM™ 11 has been shown to have minimal rod bow which can be conservatively bound with the new rod-to-rod gap closure correlation.

AREVA will continue to monitor the mechanical performance of its fuel designs, including ATRIUM 11 leads in the U.S. and in Europe. In addition to visual examinations where rod bow behavior is specifically addressed, fuel rods are routinely visually inspected when possible during PIE campaigns to identify unusual trends in rod bow behavior that may require additional measurements to characterize the performance.

[

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Figure A-1 BWR Fuel Rod Bow Correlation

APPENDIX A REFERENCES

A-1 Factors for One-Sided Tolerance Limits and For Variables Sampling Plans, D. B. Owen. Sandia Corporation Monograph (SRC-607), March 1963.

A-2 Memorandum from D. F. Ross and D. G. Eisenhut, NRC, to D. B. Vassallo and K. R. Goller, Subject: Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors, dated February 16, 1977.

APPENDIX B BWR FUEL ROD GROWTH CORRELATION

Introduction

The fuel rod growth correlation was most recently approved in 1998 using Zircaloy-2 (Zry-2) Stress Relief Annealed (SRA) growth data obtained from post irradiation examination (PIE) campaigns. This correlation has been updated to include the Zry-2 SRA rod growth data obtained from PIE campaigns since 1998. It is also conservative to use this rod growth correlation for Zry-2 RXA cladding since it has been observed to have less axial growth than SRA cladding.

Measurement Description

Figure B-1 shows the fuel rod growth correlation containing the data presented in the rod growth correlation from 1998, the data in open blue markers, as well as the new data collected since 1998, the closed red markers. The data include fuel rods from 7x7, 8x8, 9x9 and 10x10 arrays (ATRIUM™-10). [

]

Correlation Development

The fuel rod growth data (expressed as percent of active fuel length) versus assembly average burnup is shown in Figure B-1. [

]

[

]

Table B-1 BWR Fuel Rod Growth Equation

[

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[

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Summary

The results of the fuel rod growth linear correlation are summarized in Table B-1. The maximum fuel assembly exposure level represented by the data is [

] Based on the data and similarity in manufacturing processes, the BWR rod growth correlation is fully applicable to AREVA BWR fuel rod designs with SRA and RXA Zry-2 cladding.

[

]

Figure B-1 BWR Fuel Rod Growth Correlation for SRA Cladding

[

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Figure B-2 UO2 and Chromia-doped Fuel Rod Growth in BWR Reactor C22 with RXA Cladding

APPENDIX B REFERENCES

B-1 Factors for One-Sided Tolerance Limits and For Variables Sampling Plans, D. B. Owen. Sandia Corporation Monograph (SRC-607), March 1963.

APPENDIX C BWR FUEL ASSEMBLY GROWTH CORRELATION

Introduction

This appendix provides an assembly growth correlation to be applied for the evaluation of AREVA BWR fuel assemblies where the axial growth is controlled by a central water channel made from a zirconium alloy. The new correlation is based on ATRIUM™ type fuel assembly growth data only, and excludes designs with load bearing tie rods as well as the European bundle in basket designs. The database includes water channels made from both Zircaloy-4 (Zry-4) and Z4B™ materials.

Measurement Description

Figure C-1 only includes growth data from ATRIUM™-10 assemblies, solid green data points, and ATRIUM™ 11 assemblies, open orange data points, made from Zry-4 and Z4B™ water channels, excluding designs with load bearing tie rods and the European bundle in basket design. [

]

Correlation Development

The fuel assembly growth data (expressed as percent of active fuel length) versus assembly average burnup is shown in Figure C-1. [

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[

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Table C-1 BWR Fuel Assembly Growth Correlation

[

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[

]

The results of the fuel assembly growth correlation are summarized in Table C-1. The maximum assembly exposure level represented by the data is [ ] Based on the data and similarity in manufacturing processes, the BWR fuel assembly growth correlation is fully applicable to ATRIUM™ type fuel assembly designs, including ATRIUM™ 11, with water channels made from Zry-4 and Z4B™. This correlation does not apply to load bearing tie rod designs or the European bundle in basket design.

[

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Figure C-1 BWR Fuel Assembly Growth Correlation

APPENDIX C REFERENCES

C-1 Factors for One-Sided Tolerance Limits and For Variables Sampling Plans, D. B. Owen. Sandia Corporation Monograph (SRC-607), March 1963.

APPENDIX D Z4B™ WATER CHANNEL ASSEMBLY

Introduction

BWR fuel assemblies have used a zirconium alloy tie structure for several decades. The tie structure can consist of fueled tie rods, water rods, a water channel, or a fuel channel. Most BWR fuel designs have used either Ziraloy-2 or Zircaloy-4 alloys as defined in ASTM B352/B352M (Reference D-1) for fuel channels with acceptable performance. However, the behavior of these alloys can be improved for use in the tie structure. More corrosion occurs on Zircaloy-4 than Zircaloy-2 in a BWR coolant environment, and there is a higher hydrogen pickup fraction in Zircaloy-2 than Zircaloy-4. The ideal alloy would have both low corrosion and low hydrogen pickup. A proprietary zirconium alloy has been developed, Z4B™, which optimizes the alloying element concentrations for improved corrosion and hydrogen pickup when used for a BWR fuel structural component. This appendix provides additional information on Z4B™ and its application to a water channel assembly used as part of a BWR fuel assembly tie structure.

Alloy Composition

The composition of Z4B™ is shown in Table D-1 and is similar to that of Zircaloy-4 (Zry-4) as defined in ASTM B352/B352M (Reference D-1), though Z4B™ has slightly higher iron (Fe) and chromium (Cr) contents. [

] Both alloys are composed of about 98 wt% zirconium and have a hexagonal crystal structure at room and service temperatures. The small differences in composition between Z4B™ and Zry-4 do not result in any significant differences in fabrication methods or processes.

Table D-1 Alloy Composition

Element	Composition (weight percent)	
	Zry-4 (UNS R60804)	Z4B™
Zirconium (Zr)	~ 98	~ 98
Tin (Sn)	1.20 – 1.70	[ ]
Iron (Fe)	0.18 – 0.24	[ ]
Chromium (Cr)	0.07 – 0.13	[ ]
Nickel (Ni)	-	[ ]

The motivation for increasing Fe and Cr in Z4B™ is to improve the corrosion resistance and hydrogen uptake relative to Zry-4. Industry experience indicates that increases in Fe and Cr act to reduce the corrosion rate of Nb-free alloys such as Zry-4 and Z4B™ (Reference D-2). Table D-2 indicates that the "best alloy content" for Fe is greater than 0.3% and for Cr is above 0.15 % with respect to corrosion resistance and hydrogen pickup fraction. These values compare well with the ranges listed above for Z4B™ (Table D-1). As indicated in Table D-2, the solubility of Fe and Cr in the zirconium matrix is very low, which means that these elements exist primarily in second phase particles (SPPs). Given the similarity in composition and crystal structure of Z4B™ and Zry-4, that the solubility of Fe and Cr in the alloy matrix is similar for these alloys, the additional concentrations of these elements in Z4B™ could result in a larger number of SPPs, a larger average SPP size, or both depending on the details of material processing. The processing of Z4B™ targets [ ]. The superior corrosion and hydrogen uptake performance of Z4B™ relative to Zry-4 has been demonstrated through a material test program for Z4B™ spacer grids and recent measurements collected on Z4B™ Lead Use Fuel Channels (Reference D-3).

Table D-2 Effect of Alloying Elements on Corrosion of Zirconium Alloys (Reference D-2)

Element	Solubil. (%)	Best alloy content (%)	Out-pile corrosion	In-PWR corr.	LiOH corr.	In-BWR corr.	HPUF
Sn	2	0/>1	—	=	+ +	+	0
Nb	0.5	0.5/>2	+ +	+ +	0	—/+	+/0
Fe	<0.01	≥0.3	+ +	+ +	+ +	+	0/+
Cr	<0.01	≥0.15	+/—	+	+ +	+	+ (>0.15)
Ni	<0.01	0.05	+ +	+		+	=/0
V	<0.01	≥0.15	+/—	+	+ +	+	+
Cu	<0.1	≥0. 5	+				0
0: no effect, — increase, = strong increase, + reduction, ++ strong reduction, 0/+ effect differs in different environments.							

Water Channel Assembly

The water channel is made from sheet material and formed into the shape of a square duct with rounded corners. End fittings are welded to the ends of the water channel which allow the upper and lower tie plates to be secured to the water channel assembly. Inlet and outlet holes in the end fittings permit the flow of single-phase water through the water channel. Spacer stops are welded to the sides of the water channel to control the axial position of the spacer grids. The water channel plus the spacer stops and end fittings constitute the water channel assembly. See Figure D-1 for an illustration. Although not shown in the figure, some water channel assemblies also have 'crowns' which are thin metallic strips welded to the water channel with the purpose of diverting single-phase water toward the fuel rods. All water channel assembly components are made from a zirconium alloy, i.e. Z4B™.

The structural tie between the lower tie plate (LTP) and the upper tie plate (UTP) is provided by the water channel assembly. Within the ATRIUM™ family of BWR fuel designs, there have been a few variations of the water channel assembly design. Currently the upper end fitting contains an integrated connecting rod which extends

from the water channel up to the UTP locking hardware. The LTP is secured to the water channel lower end fitting by a threaded connection. Large cross-sectional threaded fasteners and connecting hardware ensure a strong connection between the two tie plates.

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Figure D-1 Water Channel Assembly

Design Loads and Requirements

The generic design criteria for BWR fuel designs have been defined in Reference D-4. Requirements for the fuel design have been developed based on the guidance in the Standard Review Plan, including some specific requirements related to the tie structure. The specific requirements applicable to the tie structure have been defined in Section 3.3.1 of Reference D-4 for stress, strain or loading limits, Section 3.3.6 for axial growth, and Section 3.3.9 for fuel assembly handling (see Table D-3 which quotes the criteria from Reference D-4).

Table D-3 Generic BWR Design Criteria Applicable to the Tie Structure during Handling, Normal Operation, and AOO (Reference D-4)

Section in Ref. D-4	Topic	Criteria
3.3.1	Stress, Strain, or Loading Limits	[ ]
3.3.6	Axial Irradiation Growth	[ ]
3.3.9	Fuel Assembly Handling	[ ]

The primary mechanical function of the BWR fuel assembly tie structure is to allow the fuel assembly to be lifted by a grapple attached to the upper tie plate. However, the Fuel Handling Accident evaluation assumes the fuel assembly is dropped over the core which could occur either from a crane failure or a break of the fuel assembly tie structure itself. In order to provide substantial design margin against such an accident occurring, Section 3.3.9 of Reference D-4 provides a fuel assembly handling requirement that a test or analysis of the assembly must not [

].

Of more significance to safety performance is that the tie structure maintains an acceptable dimensional configuration in the core during normal operation and anticipated operational occurrences (AOO). The design loads are small and consist mostly of component weight, friction forces transmitted through spacer grids, and hydraulic differential pressure. These loads are analyzed against the strength requirements listed in Table D-3. In addition, a dimensional analysis must be performed to ensure the deformation caused by these loads (including the effects of thermal expansion, growth, and creep) do not significantly affect design engagements and clearances. There are no significant cyclic loads so a break due to fatigue is not a concern.

Analytical Methods

The mechanical analytical methods have been described in Section 4 of this topical report. Section 4.1.1 describes the general strength evaluations which include those performed for the water channel assembly to demonstrate that the stress, strain, and loading limit criteria (including fuel handling) defined above in Table D-3 are met. The use of Z4B™ material has no significant impact on either the method or calculated margins for these evaluations.

As discussed in Section 2 of this topical report, the composition of Z4B™ differs from ASTM Zry-4 only by slight increases in Fe and Cr. Both alloys are composed of about 98 wt% zirconium and have a hexagonal crystal structure at room and service temperatures. Based on these considerations, there are negligible differences in basic material properties (e.g., elastic moduli, heat capacity, thermal expansion, thermal conductivity, density, etc.) between Z4B™ and Zry-4. The minimum specified strengths are used in the analyses. As shown in Table D-4, the minimum mechanical property requirements for Z4B™ water channel strip are the same as or higher than those for Zry-4 water channel strip. There is no difference in the specification limits for end plug barstock. Therefore, the use of Z4B™ in place of Zry-4 for these components will not result in a reduction in strength margins. The strength analyses are also dependent on corrosion and related wall thinning. As discussed in Section 2, the compositional increases in Fe and Cr result in superior corrosion resistance and lower hydrogen uptake of Z4B™ relative to Zry-4. Therefore, the corrosion and hydrogen embrittlement behavior of Z4B™ components will be bounded by that of Zry-4 components.

Table D-4 Unirradiated Strength Specifications for Zry-4 and Z4B™

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General fit-up and deformation analyses are performed at BOL and EOL conditions to demonstrate that adequate engagements and clearances are maintained throughout the fuel assembly lifetime as described in Section 4.1.5. The methodology for axial growth is described in Section 4.1.5.2 which demonstrates that the criteria for axial irradiation

growth defined in Table D-3 are met. The assembly growth correlation with Z4B™ water channels is documented in Appendix C.

Testing of unirradiated, recrystallized Zry-2, Zry-4, and Z4B™ has shown that differences in alloy composition between these materials have no significant effect on creep rate. [

]. Overall, the difference is not judged to be significant, and it remains conservative to use the creep rate already defined for Zry-2 and Zry-4 fuel channel material in Reference D-5.

APPENDIX D REFERENCES

D-1. ASTM B352/B352M, *Standard Specification for Zirconium and Zirconium Alloy Sheet, Strip, and Plate for Nuclear Application.*

D-2. Rudling, P., Zr Alloy Corrosion and Hydrogen Pickup. A.N.T. International. NRC Accession Number: ML15253A227. 2013.

D-3. ANP-10336P-A, Revision 0, "Z4B™ Fuel Channel Irradiation Program," AREVA NP, July 2017.

D-4. ANF-89-98(P)(A) Revision 1 and Supplement 1, "Generic Mechanical Design Criteria for BWR Fuel Designs," Advanced Nuclear Fuels, April 1995.

D-5. EMF-93-177 Revision 1 Supplement 1(P)(A), "Mechanical Design for BWR Fuel Channels, Supplement 1: Advanced Methods for New Channel Designs," AREVA NP, September 2013.

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fuel rod exposure limit for the realistic fuel rod thermal-mechanical methodology has been established in Reference 1.

correlations for fuel rod bow, fuel rod growth, and fuel assembly growth due to the incorporation of recent operating experience data. Note that the fuel rod growth correlation presented in this supplement is based on the total axial rod growth measured in post-irradiation exams, and therefore does not affect the RODEX4 stress-free irradiation growth model in Reference 1.

The rod bow correlation is constructed from fuel rod-to-rod gap closure data measured on a broad selection of AREVA BWR fuel designs in varied operating environments. The rod-to-rod gap closure predicted as a function of fuel assembly exposure is used as an input to thermal limit evaluations (i.e. MCPR) for AREVA BWR fuel designs.

The BWR rod growth correlation is updated with the most recent data from AREVA's Zircaloy-2 stress-relief annealed (SRA) cladding, [

] . The BWR fuel assembly growth correlation is built from post-irradiation length measurement data taken from ATRIUM™[1] fuel assemblies. This model is applicable to all ATRIUM™ fuel assembly designs for which assembly growth is controlled by the water channel growth, including the ATRIUM™ 11 with Z4B™[2] water channels. The combination of the fuel rod and assembly growth correlations is used to define the maximum fuel rod length which will not interfere with the upper tie plate at end of life. This is the only mechanical method defined in this report which is limiting at end of life, and the growth databases support the maximum requested fuel assembly exposure limit.

1 ATRIUM is a trademark of AREVA Inc.
2 Z4B is a trademark of AREVA Inc.

4.0 ANALYTICAL METHODOLOGY

4.1 *Mechanical Methods*

The methods described in this section cover the topics included in SRP Section 4.2 II.1.A for fuel system damage except those already addressed in the base topical report (Reference 1). These methods support AREVA's generic BWR fuel design criteria approved in Reference 3 throughout the design lifetime of the fuel and cover handling, normal operation and AOO conditions. The only changes to the previously approved BWR fuel mechanical methods are updates to fuel rod bow, fuel rod growth, and fuel assembly growth correlations.

4.1.1 Stress, strain or loading limits

The strength of the fuel assemblies and fuel rods is assured by evaluating the margin to conservative stress and deformation design limits under various shipping, handling and operational loads. The loads are applied to the fuel rod cladding, upper and lower tie plates, grid spacers, water channel (or tie rods) and connecting hardware, fuel assembly cage and springs where applicable. AREVA defines a maximum axial handling design load equivalent to [ ].

As described in Reference 3, AREVA uses Section III of the ASME Boiler and Pressure Vessel Code as guidance for establishing the acceptable stress, strain, or load criteria for assembly components and the corresponding analysis methods which may be used to evaluate those criteria. These methods include elastic and plastic analysis techniques as well as load rating from prototype testing. Analysis methods include use of conventional, open-literature equations, elasticity formulations, general purpose finite element stress analysis codes such as ANSYS, or testing.

The minimum specified yield and ultimate strength for unirradiated material are used in the analyses. This is a conservative assumption since strength will increase under irradiation. Since loads often stay the same or decrease over time, the beginning of life

(BOL) strength evaluations tend to be the most limiting. This is true even when the material loss due to oxidation that would be expected at end of life (EOL) is factored in. The oxide is either insignificant, as observed on stainless steel and nickel alloy components, or the oxide is on relatively thick components such as the Zircaloy water channel and fuel channel. Zircaloy fuel rod cladding is the only component with a specific EOL analysis requiring the assumed loss of material due to corrosion. However, even in this case the EOL analysis is not limiting due to the reduced loads at EOL.

While all load bearing fuel assembly components have some analysis or test to validate that criteria are met for the given design loads, a few of the components have standard evaluations as described below.

4.1.1.1 Fuel rod cladding

Various normal operation and AOO loads create stresses on the fuel rod cladding. Each individual stress is calculated at the inner and outer surfaces of the cladding at both the mid-span between spacer grids and at the spacer grid. The stresses at each location are then combined to determine the maximum stress intensities. The analysis is performed at BOL and EOL and at cold and hot conditions with unirradiated material strength. The stress analysis assumes maximum fuel rod power, minimum fill gas pressure, and the most conservative fuel rod geometry including a reduced wall thickness at EOL due to oxidation.

[

] The cladding stress analysis method has not changed from what was documented in Section 3.4.3 of Reference 4. The stress calculations use conventional, open-literature equations. A general purpose, finite element stress analysis code such

4.1.2 Strain fatigue

Fatigue of structural components is generally low because the cyclic loadings on the structural components typically have either a small number of cycles (i.e. reactor startup) or small amplitude (i.e. flow-induced vibration). Cyclic loading associated with relatively large changes in power can cause cumulative damage which may eventually lead to fatigue failure. The O'Donnell and Langer fatigue curves are used in the analysis of Zircaloy components (Reference 7). These fatigue curves incorporate the NRC recommended "2 or 20" safety factor. This safety factor reduces the stress amplitude by a factor of two or reduces the number of cycles by a factor of twenty, whichever is more conservative. The fatigue curves provide the maximum allowed number of cyclic loadings for each stress amplitude. The fatigue usage factor is the number of expected cycles divided by the number of allowed cycles. The total cumulative usage factor is the sum of the individual usage factors for each duty cycle.

4.1.3 Fretting wear

Fretting wear is a concern for the fuel rod cladding. Fretting wear may occur on the fuel cladding surfaces in contact with the spacer grids if there is a reduction in grid spacer spring loads in combination with small amplitude, flow-induced, vibratory forces.

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AREVA has gathered post-irradiation rod-to-rod gap closure measurements from a variety of BWR fuel designs as shown in Figure A-1. Both the absolute and the percent gap closure data for all measured AREVA BWR designs (7x7, 8x8, 9x9, and 10x10) reveal [

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AREVA has conducted PIE campaigns both in the U.S. and in Europe to collect fuel rod-to-rod gap measurements. The fuel rod-to-rod gap database includes AREVA's legacy designs with 7x7, 8x8, and 9x9 rod arrays which are no longer in operation, and also the ATRIUM™-10 design still in use today.

Fuel rod-to-rod gap measurements are typically taken at each span between spacer grids (usually 8 spans) and at each fuel rod-to-rod gap. In addition, measurements can be taken when the tool is inserted and withdrawn, for two measurements of every gap.

For example, there are 120 fuel rod-to-rod gaps in a 9x9 assembly with a central water channel. Therefore, in this particular assembly, there could be up to 1920 measurements if two measurements per gap are recorded. For the ATRIUM™-10 family of fuel designs, there are 156 fuel rod-to-rod gaps which could lead to a total of up to 2496 measurements for the entire bundle if two measurements per gap are recorded.

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]

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[

]

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The fuel rod growth correlation was most recently approved in 1998 using Zircaloy-2 (Zry-2) Stress Relief Annealed (SRA) growth data obtained from post irradiation examination (PIE) campaigns. This correlation has been updated to include the Zry-2 SRA rod growth data obtained from PIE campaigns since 1998.

Measurement Description

Figure B-1 shows the fuel rod growth correlation containing the data presented in the rod growth correlation from 1998, the data in open blue markers, as well as the new data collected since 1998, the closed red markers. The data include fuel rods from 7x7, 8x8, 9x9 and 10x10 arrays (ATRIUM™-10). [

]

Correlation Development

The fuel rod growth data (expressed as percent of active fuel length) versus assembly average burnup is shown in Figure B-1. [

]

Summary

The results of the fuel rod growth linear correlation are summarized in Table B-1. The maximum fuel assembly exposure level represented by the data is [

] Based on the data and similarity in manufacturing processes, the BWR rod growth correlation is fully applicable to AREVA BWR fuel rod designs with SRA Zry-2 cladding.

[

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Table C-1 BWR Fuel Assembly Growth Correlation

[

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[

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Figure C-1 BWR Fuel Assembly Growth Correlation

APPENDIX C REFERENCES

C-1 Factors for One-Sided Tolerance Limits and For Variables Sampling Plans, D. B. Owen. Sandia Corporation Monograph (SRC-607), March 1963.