ATTACHMENT 38

ANP-3409NP, Fuel-Related Emergent Regulatory Issues (Non-Proprietary)

Fuel-Related Emergent Regulatory Issues

ANP-3409NP
Revision 2

August 2015

AREVA Inc.

Nature of Changes

Item	Section(s) or Page(s)	Description and Justification
1	32	Updated reference list to latest revisions, References 5, 7, and 8.

Contents

Tables

Figures

NOMENCLATURE

Abbreviation	Description
ΔH	Deposited Enthalpy
ADS	Automatic Depressurization System
AST	Alternative Source Term
BOC	Beginning Of Cycle
BFN	Browns Ferry Nuclear Plant
BPWS	Banked Position Withdrawal Sequence
BWR	Boiling Water Reactor
cal/g	Calories per Gram
CFR	Code of Federal Regulations
CLTP	Current Licensed Thermal Power (3458 MWt)
CRDA	Control Rod Drop Accident
CWSR	Cold Work Stress Relieved
ECR	Equivalent Cladding Reacted
EOC	End-Of-Cycle
EPU	Extended Power Uprate
FGR	Fission Gas Release
GWd/MTU	Giga-Watt Days per Metric Ton of Uranium
HEX	Excess Clad Hydrogen
LAR	License Amendment Request
LOCA	Loss Of Coolant Accident
LPU	Licensed Power Uprate (3952 MWt)
MAPLHGR	Maximum Average Planar Linear Heat Generation Rate
MPa	Mega-Pascal
MWt	Mega-Watts thermal
NRC	Nuclear Regulatory Commission
OLTP	Original Licensed Thermal Power (3293 MWt)
PCMI	Pellet Clad Mechanical Interaction
PCT	Peak Clad Temperature
PHEX	Peak Hot Excess
ppm	Parts per million

Abbreviation	Description
RFI	Rod Failure Increase
RG	Regulatory Guide
RIA	Reactivity Initiated Accident
RPF	Radial Peaking Factor
RXA	Recrystallized Annealed
SRA	Stress Relieved and Annealed (aka CWSR)
SRP	Standard Review Plan
SSRF	Steady State Release Fraction
TFGR	Transient Fission Gas Release
TOTR	Total Release Fraction (SSRF + TFGR)
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
wppm	Parts per million weight percent

1.0 INTRODUCTION

This report addresses two emerging regulatory issues as part of the Browns Ferry Nuclear Plant (BFN) license amendment request (LAR) for extended power uprate (EPU). The BFN EPU LAR requests an increase from the current licensed thermal power (CLTP) level of 3458 MWt to a new licensed power uprate (LPU) of 3952 MWt. The LPU is approximately 120% of the original licensed thermal power (OLTP) for BFN.

The two issues being addressed are: (1) proposed new criteria for Loss Of Coolant Accident (LOCA) in the Code of Federal Regulations 10 CFR 50.46c; and (2) proposed changes to the acceptance criteria for Reactivity Initiated Accidents (RIA) in Section 4.2 of the Standard Review Plan (SRP) (Reference 1). These items are not part of the current Nuclear Regulatory Commission (NRC) regulatory requirements and therefore are not included in the current licensing basis for BFN. However, since both issues represent potential changes to regulatory requirements that may become effective during the NRC review of the BFN EPU LAR, the NRC staff has advised that their potential impact be addressed as part of the EPU LAR.

The approach taken in addressing these issues was to utilize a combination of approved methodologies as well as components of methods that are not part of the AREVA NRC approved methodologies.

2.0 PROPOSED NEW FUEL CRITERIA FOR LOCA (10 CFR 50.46c)

The NRC has been working to amend its regulations based on new research findings. The BWR Owner's Group developed the Reference 2 assessment to demonstrate that adequate margin exists within the BWR fleet relative to the proposed criteria. The BFN units were included in the Reference 2 margin assessment. The Reference 2 assessment is based on the FRAPCON hydrogen uptake model and the Baker-Just correlation for calculating metal-water reaction and clad oxidation. Use of the Baker-Just correlation is specified in 10 CFR 50 Appendix K.

The NRC could publish new fuel criteria while the BFN EPU LAR is under review by the NRC. In Reference 3 the NRC proposed criteria that the fuel cladding maintains a degree of post-quench ductility during a postulated LOCA event. The NRC has also provided draft guidance for the new rule, Reference 4. A re-assessment of the margins is being provided for the updated conditions of the BFN units to provide reasonable assurance that the BFN units will be able to meet the final 10 CFR 50.46c requirements at EPU conditions. The updated conditions are:

- EPU power
- Implementation of single-failures proof ADS logic
- Top-down cooling restriction
- Introduction of ATRIUM 10XM fuel

The LOCA results for these conditions are summarized in Reference 5.

The following four assessments have been performed:

1) Re-perform the Reference 2 assessment for the current BFN EPU LOCA analysis for ATRIUM 10XM fuel.
2) Perform the Reference 2 assessment using AREVA's hydrogen uptake model (Appendix A) based on the equilibrium cycle developed in support of the EPU LAR (Reference 6, Attachment 16 of the EPU LAR).
3) Perform the Reference 2 assessment using the Cathcart-Pawel correlation instead of the Baker-Just correlation.
4) Perform the Reference 2 assessment using the hydrogen uptake model (Appendix A) and the Cathcart-Pawel correlation.

The NRC draft guidance establishes an oxidation limit that is a function of the pre-transient hydrogen content of the cladding which is presented in Figure 2-1.

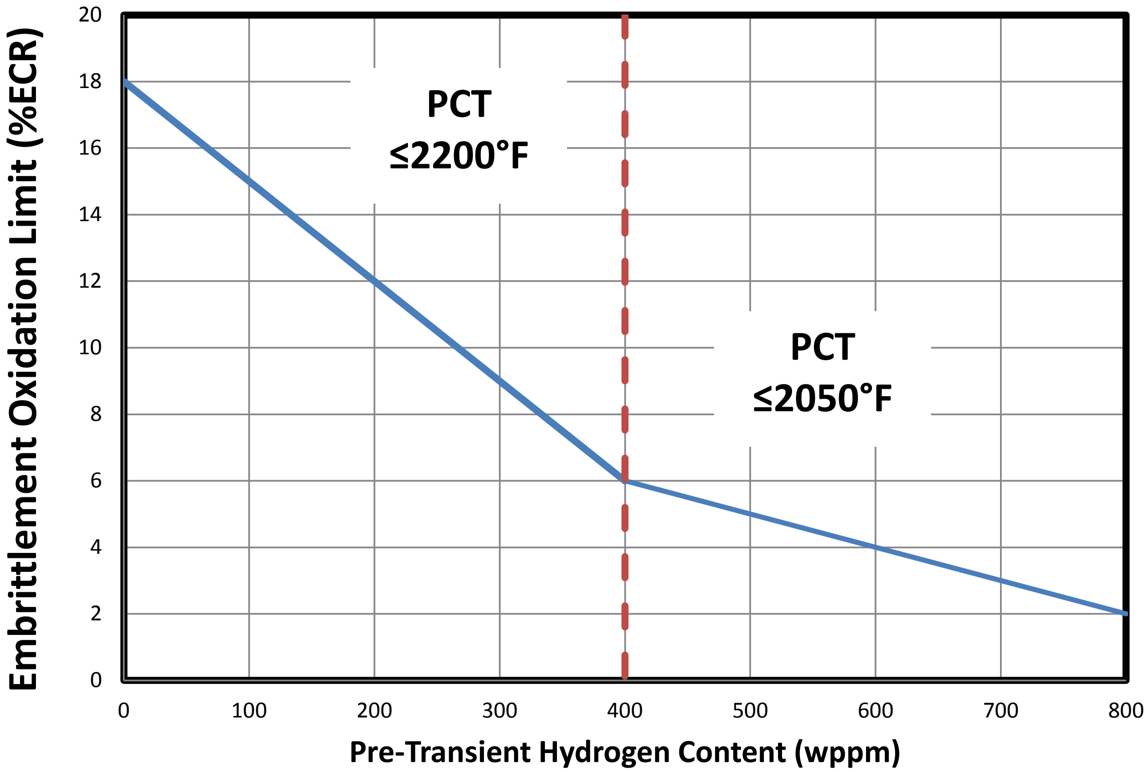

Figure 2-1 NRC Draft Guidance (Reference 4)

The above oxidation limit is applicable when the Cathcart-Pawel correlation is used to calculate the oxidation.

2.1 *Re-Perform the Reference 2 Assessment for the EPU LOCA (Reference 5)*

The MAPLHGR limits established to protect the LOCA acceptance criteria are established as a function of planar exposure, Figure 2.1 of Reference 5**.** The maximum PCT is 2008 ^{o}F and occurs at 0.0 GWd/MTU. The maximum local cladding oxidation at this exposure is 1.90% (as calculated by the Baker-Just correlation). In this case the cladding did not rupture, so the oxidation was only calculated on the outside of the cladding. In the Reference 2 assessment the hydrogen accumulation was based on an assumed exposure of 62 GWd/MTU where

FRAPCON results indicated 160 ppm hydrogen and the resulting oxidation limit was 12.6%. Also, in the Reference 2 assessment the oxidation calculated on the outside was doubled before comparing to the oxidation criterion. Therefore, when reperforming the Reference 2 assessment for the EPU LOCA (Reference 5) the margin to the oxidation limit would be 8.8% (12.6% - (2 X 1.9%)).

2.2 *Perform the Reference 2 Assessment Using AREVA's Hydrogen Uptake Model (Appendix A) Based on the Equilibrium Cycle Developed in Support of the EPU LAR*

The maximum local cladding oxidation at 0.0 GWd/MTU, as calculated by the Baker-Just correlation, is 1.90% (Table 5.2 in Reference 5). At this exposure there would be no hydrogen uptake and the oxidation limit from Figure 2-1 would be 18%. Therefore, the margin to the oxidation limit is 16.1%.

The maximum local cladding oxidation at any exposure, as calculated by the Baker-Just correlation, is 2.16% at 25.0 GWd/MTU (Table 5.2 in Reference 5). Rod failure was calculated at this exposure, so 2.16% includes the oxidation on the outside and the inside of the cladding. The AREVA hydrogen uptake model was applied for the BFN EPU equilibrium cycle. At an assembly planar exposure of 25 GWd/MTU, the hydrogen uptake was predicted to be less than **[ ]** . The corresponding oxidation limit from Figure 2-1 is **[ ]**. Therefore, the margin to the oxidation limit would be **[ ]**.

The largest hydrogen uptake calculated at the end of cycle (EOC) in the BFN ATRIUM 10XM equilibrium cycle is **[ ]** . With this hydrogen uptake, the oxidation limit from Figure 2-1 would be **[ ]**. The assembly with this maximum hydrogen uptake has a maximum planar exposure of 52.6 GWd/MTU. Exposure dependent heatup calculations were performed at 50.0 and 55.0 GWd/MTU.

The margin to the oxidation limit is lower when evaluating the heatup results at 50.0 GWd/MTU. At this exposure rod failure was calculated, so the 0.76% oxidation calculated with the Baker-Just correlation includes the oxidation on the outside and the inside of the cladding. **[**

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2.3 *Perform the Reference 2 Assessment Using the Cathcart-Pawel Correlation Instead of the Baker-Just Correlation*

Reference 5 reports the PCT and maximum local cladding oxidation as a function of planar exposure in Table 5.2. The results are calculated with the Baker-Just correlation. The cladding temperature as a function of time for each exposure dependent heatup calculation was used to calculate the maximum oxidation with the Cathcart-Pawel correlation. In this approach the difference in heat generated that would result from the different in oxidation between Baker-Just and Cathcart-Pawel was not accounted for during the calculation of cladding temperature as a function of time.

The maximum PCT is 2008 ^{o}F and occurs at 0.0 GWd/MTU. The maximum local cladding oxidation at this exposure is 2.1% when calculated with the Cathcart-Pawel correlation. Since the cladding did not rupture, the oxidation was only calculated on the outside of the cladding. In the Reference 2 assessment the hydrogen accumulation was based on an assumed exposure of 62 GWd/MTU where FRAPCON results indicated 160 ppm hydrogen and the resulting oxidation limit was 12.6%. Also, in the Reference 2 assessment the oxidation calculated on the outside was doubled before comparing to the oxidation criterion. Therefore, when reperforming the Reference 2 assessment for the EPU LOCA (Reference 5) the margin to the oxidation limit would be 8.4% (12.6% - (2 X 2.1%)).

2.4 *Perform the Reference 2 Assessment Using the Reference Appendix A Hydrogen Uptake Model and the Cathcart-Pawel Correlation*

The largest hydrogen uptake calculated at the end of cycle (EOC) in the BFN equilibrium cycle for ATRIUM 10XM fuel is [ ] . With this hydrogen uptake, the oxidation limit from Figure 2-1 would be [ ]. The assembly with this maximum hydrogen uptake has a maximum planar exposure of 52.6 GWd/MTU. Exposure dependent heatup calculations where performed at 50.0 and 55.0 GWd/MTU. The margin to the oxidation limit is lower when evaluating heatup results at 55.0 GWd/MTU and the oxidation is calculated to be < 1% with the Cathcart-Pawel correlation. [

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2.5 *Conclusion*

The evaluations above utilized the updated hydrogen uptake model presented in Appendix A, which is the correlation that AREVA intends to use for 50.46c compliance evaluations once the rule is effective. The effect of changing from the Baker-Just correlation to the Cathcart-Pawel correlation, which will be required by the revised rulemaking, was evaluated. The evaluations demonstrate that sufficient margin exists to the oxidation limits in the proposed rulemaking. Therefore, there is high confidence BFN units can meet proposed rulemaking at EPU conditions.

In Reference 2 the BFN units had been the limiting plant in Group 4a (BWR/4 with a single failure that could result in a loss of all ADS). Now that this potential single failure has been eliminated, the BFN units are in Group 4. Based on the evaluations documented previously, the BFN units may become the limiting plant in Group 4.

3.0 PROPOSED FUEL CRITERIA FOR REACTIVITY INITIATED ACCIDENT

RIAs are postulated accidents involving the sudden and rapid insertion of positive reactivity. For Boiling Water Reactors (BWR), the limiting RIA scenario is the Control Rod Drop Accident (CRDA). In support of the BFN EPU LAR, evaluations of the BWR CRDA have been performed to address the current BFN licensing basis using AREVA NRC approved methodology, as summarized in Section 3.1.1.

In a January 2015 pre-application meeting between TVA and the NRC with respect to the BFN EPU LAR, TVA provided information to demonstrate compliance with the current plant licensing basis for the CRDA. The NRC advised TVA to consider the impact of the portions of the proposed regulatory acceptance criteria for the BFN EPU LAR. Subsequent to this January meeting, the Reference 9 NRC Memorandum was issued documenting proposed criteria and guidance for the RIA. Additional evaluations have been performed as documented in this report to address this NRC advisement. The proposed criteria from Reference 9 which are addressed in this document are described in Section 3.1.2 and the evaluations are subdivided into two primary areas: 1) Fuel Melt, and 2) Fuel Failures.

The potential for cladding failure from fuel melt in the startup range is precluded in this evaluation by demonstrating that the incipient fuel melt temperature is not reached. The evaluation for fuel melt is addressed in Section 3.2 of this report, including impact on core coolability criterion 1 and 2 of Table 3-2.

Fuel failure assessment against the proposed acceptance criteria is the subject of Section 3.3. The fuel failure subsections address various evaluations needed to support the overall fuel failure assessment as summarized below:

- Section 3.3.1: Radial (Assembly) Enthalpy - This section addresses extending the determination of deposited enthalpy from the bundles immediately in the error cell to bundles outside of this cell. This is required since the proposed criteria lowers failure thresholds increasing the potential for failures to occur outside of the error cell.
- Section 3.3.2: PCMI Cladding Failure – Addresses failures due to pellet clad mechanical interaction.
- Section 3.3.3: High Temperature Failure Threshold – Addresses potential for high temperature failures, including impact of internal rod pressure.
- Section 3.3.4: Dose Consequence – Addresses the impact of including the transient fission gas release into the isotopic release fractions. This can impact the existing dose

assessment. A method of adjusting the number of calculated rod failures is provided to compensate and maintain the current dose assessment basis.

- Section 3.3.5: Rod Failure Assessment – Combines the results of the previous subsections into a total number of fuel failures using the proposed criteria.

This evaluation utilizes a combination of methodologies and analyses to specifically address the NRC advisement with respect to the proposed regulatory acceptance criteria for CRDA. In addition, significant conservatisms have been included as summarized in Section 3.4 of this report.

3.1 *Regulatory Acceptance Criteria*

3.1.1 Current Licensing Basis Criteria

The current licensing basis for CRDA is discussed in Section 2.8.5.4.4, "Spectrum of Rod Drop Accidents" of the FUSAR (Reference 7) included as Attachment 8 of the EPU LAR.

The current Regulatory Guide (RG) 1.77 RIA criterion is that the peak enthalpy must remain below a value of 280 cal/g. As described in Section 14.6.2 of the Updated Final Safety Analysis Report (UFSAR), analyses are performed for each reload cycle to ensure that a peak enthalpy of 280 cal/g is not exceeded. Furthermore, any rods that exceed a value of 170 cal/g are assumed to fail. The total number of failed rods is required to remain below 850 rods assumed by the Alternative Source Term (AST) CRDA dose assessment.

Analyses have been performed using AREVA's NRC approved methodology to support the EPU LAR. One of the cores analyzed is an ATRIUM 10XM equilibrium cycle with the results provided in the FUSAR (Reference 7, Attachment 8 of the EPU LAR). A separate transition cycle analysis was also provided as part of the reload analysis report (Reference 8, Attachment 20 of the EPU LAR). These results are summarized in Table 3-1.

Table 3-1 Summary of CRDA Results with Current Methodology

	ATRIUM 10XM Equilibrium Cycle	Representative Transition Cycle	Criteria
Peak Fuel Enthalpy (cal/g)	142	156	< 280 cal/g*
Number of Failed Rods	0	0	< 850 (Number of rods exceeding 170 cal/g)

* The current regulatory requirement is to limit the peak fuel enthalpy to < 280 cal/g. For the licensing analyses submitted as part of the EPU LAR, a 230 cal/g limit has been utilized due to the emerging changes to RIA acceptance criteria. The use of the lower enthalpy limit is conservative with respect to the current licensing requirement.

3.1.2 **Proposed New Criteria**

Reference 9 is an NRC technical basis document intended to support proposed revisions to SRP Section 4.2 (Reference 1). The focus of this technical basis document is revisions to RIA acceptance criteria. The proposed criteria addressed in this report are summarized below:

Table 3-2 Summary of Proposed RIA Acceptance Criteria

<table>
<tr><td rowspan="3">Fuel Cladding Failure</td><td>Pellet Clad Mechanical Interaction (PCMI)</td><td>Figure 3-1<table><tr><th>Excess Clad Hydrogen, HEX (wppm)</th><th>Peak Radial Ave Enthalpy (cal/g)</th></tr><tr><td>≤ 117</td><td>150</td></tr><tr><td>> 117</td><td>406 – 53.8 ln(HEX)</td></tr></table></td></tr>
<tr><td>High Cladding Temperature Failure</td><td>Figure 3-2<table><tr><th>Clad Differential Pressure (Mpa)</th><th>Peak Radial Ave Enthalpy (cal/g)</th></tr><tr><td>ΔP ≤ 1.0</td><td>170</td></tr><tr><td>1.0 > ΔP < 4.5</td><td>170–(ΔP–1.0)*20</td></tr><tr><td>ΔP ≥ 4.5</td><td>100</td></tr></table></td></tr>
<tr><td>Fuel Melt</td><td>Rod Failure Assumed if fuel temperature anywhere in the pellet exceeds incipient fuel melting conditions.</td></tr>
<tr><td>Core Coolability</td><td></td><td>1. Peak radial average fuel enthalpy < 230 cal/g.
2. A limited amount of fuel melt is acceptable provided it is restricted to: (1) fuel centerline region, and (2) less than 10% of pellet volume. For the outer 90 % of the pellet volume, peak fuel temperature must remain below incipient fuel melting conditions.</td></tr>
</table>

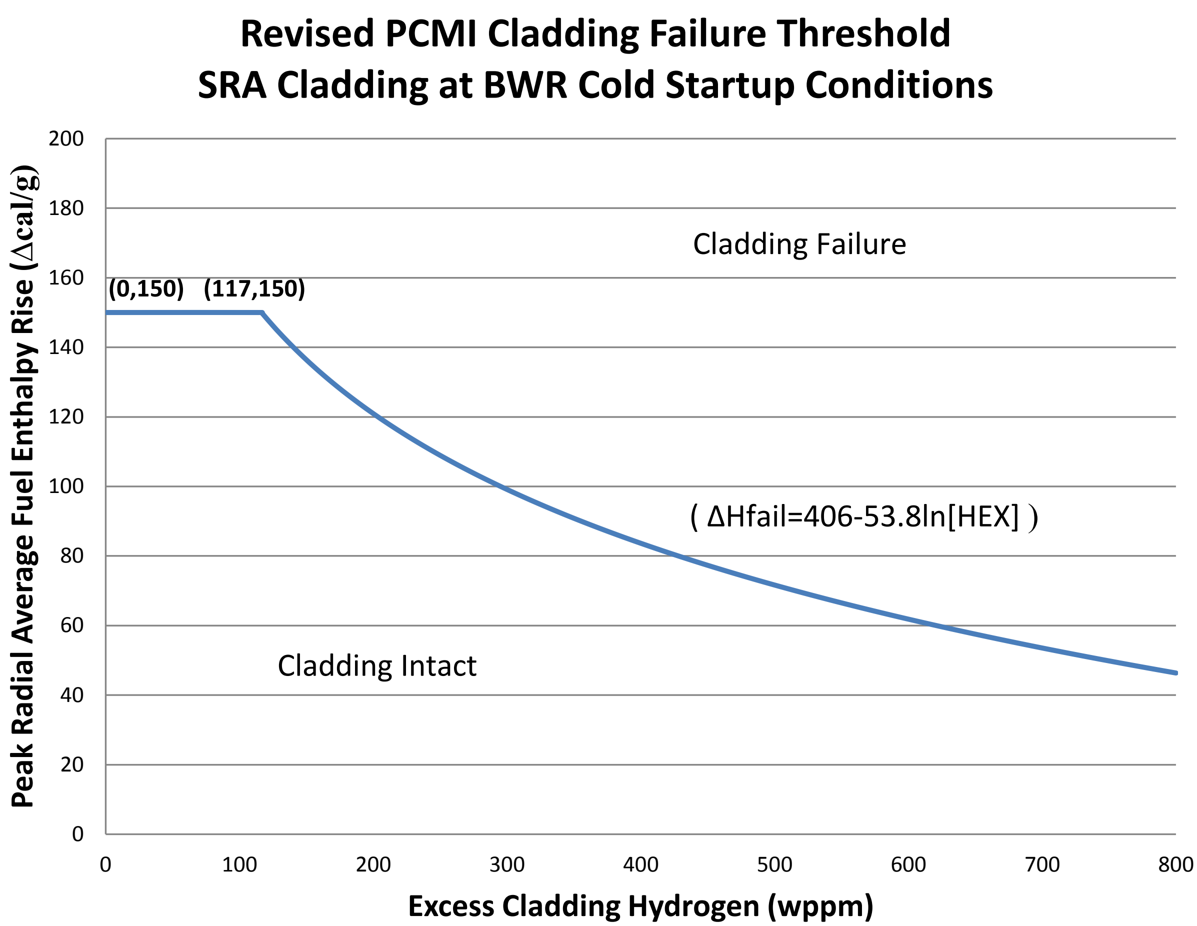

Figure 3-1 BWR PCMI Failure Threshold for SRA Cladding

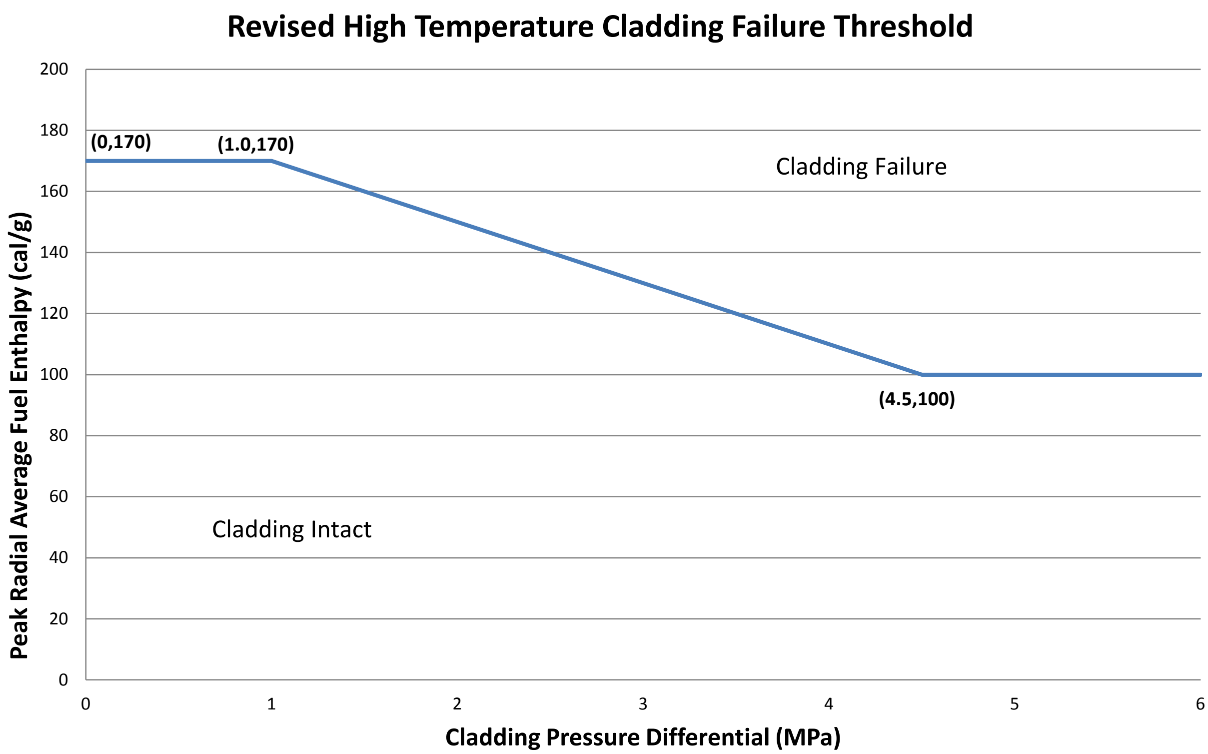

Figure 3-2 High Temperature Cladding Failure Threshold

3.2 *Core Coolability*

This section addresses the core coolability criteria 1 and 2 of Table 3-2.

Core coolability criterion 1 is addressed by limiting the peak radial enthalpy to 230 cal/g. The results provided in Table 3-1 demonstrate that this criterion is met for the BFN ATRIUM 10XM equilibrium and transition cycles.

To address core coolablity criterion 2, a fuel melt curve was established utilizing the RODEX 4 methodology (Reference 12). The RODEX 4 code was used to establish the steady state pellet radial power profile as a function of exposure. **[**

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(Definition of prompt pulse is consistent with SRP Section 4.2 (Reference 1) Appendix B, and is the radial average fuel enthalpy rise at the time corresponding to one pulse width after the peak of the prompt pulse).

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As a consequence of the buildup of fissile plutonium isotopes due to the resonance capture of epithermal neutrons by U238 at the pellet surface, the pellet radial power profile becomes increasingly peaked at relatively high exposures. The temperature versus enthalpy relationship for UO2 is defined in the RODEX 4 methodology. Given this relationship, a fuel melt enthalpy is determined. **[**

] The resulting fuel melt threshold is provided in Figure 3-3.

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Figure 3-3 Maximum Radial Prompt Average Enthalpy versus Rod Exposure

The current AREVA CRDA methodology (Reference 10) is based on adiabatic conditions. It is well understood that the pellet heats up in a nearly adiabatic condition during the power deposition; therefore variations in the gap thermal conductivity do not have a significant impact on the peak pellet temperature. [ ] The pellet radial temperature profile peaked at the RODEX4 melting temperature is shown in Figure 3-4.

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Figure 3-4 Radial and Exposure Dependence of the Pellet Temperature Distribution

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The AREVA COTRANSA2 (Reference 13) simulated startup range rod drops with ATRIUM 10XM fuel were evaluated. The enthalpy at the time corresponding to one pulse width after the peak of the prompt pulse was compared to the COTRANSA2 final enthalpy [

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3.3 *CRDA Fuel Rod Failure Evaluation*

As discussed in the previous section, maintaining a radial average enthalpy less than **[**

] The proposed RIA criteria evaluated in this report includes rod failure thresholds for pellet clad mechanical interaction and high temperature. The following subsections address these criteria as well as adjustments to the number of calculated rod failures to address the impact of including transient fission gas release within the BFN licensing basis.

3.3.1 Radial (Assembly) Enthalpy

The current approved AREVA CRDA methodology is formulated based on the energy deposition within the four fuel bundles of a core control cell containing the assumed dropped rod. **[**

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The resulting correlation:

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Where,

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The data correlation representation is provided in Figure 3-5.

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Figure 3-5 [ ]

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3.3.2 PCMI Cladding Failure

The BWR SRA cladding failure threshold provided in Figure 3-1 (from Figure 3.2.2-24 of Reference 9) is used for this evaluation. In order to establish the failure threshold the cladding hydrogen content is required. The cladding hydrogen content is calculated using an AREVA hydrogen uptake model discussed in Appendix A. The cladding content is calculated for each pin and each nodal segment for all assemblies in the core. The rod failure criteria selected for a given assembly is based on the maximum pin nodal segment value at the time in cycle for the given assembly. This is a conservative application of the hydrogen pickup in that the actual pin cladding hydrogen content varies axially as well as radially from one pin to another. In this evaluation, PCMI failure occurred at BOC in three assemblies in two different rod drops.

3.3.3 High Temperature Failure Threshold

The current approved methodology does not provide a direct method for addressing this failure threshold. To address this, a series of RODEX 4 (Reference 12) calculations were run to generate the internal rod pressure for various power histories, power levels, and irradiation periods. The example results for internal rod pressure at end of cycle conditions are provided in Figure 3-6. [

] Table 3-3 also indicates the maximum rod exposure peaking factor that is supported with this formulation.

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Figure 3-6 Rod Internal Pressure without TFGR for EOC Rod Drop

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Figure 3-7 Rod Differential Pressure with TFGR versus Burnup

Table 3-3 High Temperature Fuel Failure Threshold with Assembly Burnup

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3.3.4 **Dose Consequence**

The dose consequence for the CRDA was determined for EPU conditions at BFN and is documented in the BFN UFSAR. The current licensing basis dose evaluation assumed that 850 rods fail, this is equivalent to 1.79% of the core since it assumed 62 rods per assembly. Therefore, failing less than 1.79% of the core meets the dose consequences for BFN*.

As part of the proposed acceptance criteria, revised release fractions are proposed to account for transient fission gas release (TFGR). The transient release terms (from Reference 9) expressed as a fraction are:

Peak Pellet BU < 50 GWd/MTU: $$\text{TFGR} = \frac{\left[\left(0.26\ *\ \Delta H\right)\ -\ 13\right]}{100} \geq 0$$

Peak Pellet BU ≥ 50 GWd/MTU: $$\text{TFGR} = \frac{\left[\left(0.26\ *\ \Delta H\right)\ -\ 5\right]}{100} \geq 0$$

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* As noted, the CRDA dose evaluation assumed 62 rods per assembly. TVA elected to maintain the number of rod failures at 850 which corresponds to **[ ]**of a full core of ATRIUM 10XM with **[ ]** equivalent full length rods per assembly **[ ]**.

Where,

TFGR is transient fission gas release (must be ≥ 0)

ΔH fuel enthalpy increase (cal/g)

The transient release term is combined with the existing steady state release fraction to produce revised release fractions for the CRDA.

The actual release fractions utilized for the BFN CRDA licensing basis dose evaluations are:

I-131	I-132	Kr85	Other Nobles	Other Halogens	Alkali Metals
0.10	0.10	0.10	0.10	0.05	0.12

It is recognized that there is consideration of potential changes in the steady state source term release factors. However, this evaluation addresses the increase in the source term with the inclusion of the TFGR relative to the actual release fractions in the BFN licensing basis.

To account for the inclusion of the TFGR within the release fractions, a conservative approximation of the axial enthalpy shape was utilized to determine the increase in the release for a rod based on the maximum deposited enthalpy and establish modified rod release fractions. The axial enthalpy shape is assumed to be **[** **]** remainder of the nodes. Based on a review of rod drop evaluations this assumption provides a conservative representation of the total enthalpy deposited in the fuel rod for evaluating the transient fission gas release.

An example of the axial enthalpy distribution for eight rod drops is shown in Figure 3-8. The corresponding TFGR fractions for the eight rod drops based on the actual deposited enthalpy are given in Figure 3-9. A comparison of the total gas release based on the actual enthalpy to that **[** **]** is provided in Table 3-5.

The actual total rod release fraction for a given nuclide group can then be reformulated as:

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Where,

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The steady state release fraction for other Halogens is the smallest steady state release fraction used in the BFN licensing basis, therefore the TFGR term has the largest impact on this nuclide group. The source term increase factor, the ratio of the source term with transient fission gas release to that used in the BFN licensing basis for the CRDA can be used to represent the Rod Failure Increase (RFI) factor. Using Other Halogens group, the RFI factor can be determined as:

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The number of actual rod failures can be increased by this factor for evaluation with respect to the BFN current licensing basis.

The application of the RFI is bundle dependent based on the maximum deposited enthalpy.

The actual application of the RFI factors is provided in Section 3.3.5.

The current radiological consequences determined for BFN are provided in Table 3-4 along with the fraction of the regulatory limit.

Table 3-4 BFN CRDA Radiological Consequences

	TEDE Dose (REM) Receptor Location		
	CR	EAB	LPZ
BFN AST Dose EPU	0.26	1.17	0.7
Allowable TEDE Limit	5	6.3	6.3
Fraction of Limit	0.05	0.19	0.11
Fraction to 75% of limit	0.07	0.25	0.15

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Figure 3-8 Axial Enthalpy Profile for Eight Representative Rod Drops

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Figure 3-9 Nodal Transient Fission Gas Release Fractions

Table 3-5 Actual TFGR Fraction versus [ ] Basis

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3.3.5 **Rod Failure Assessment**

The rod drop results were evaluated against the revised failure thresholds established in Sections 3.3.2 and 3.3.3.

There were only three assemblies at beginning of cycle (BOC) which had rod failures identified due to PCMI failures. For this core the hydrogen up-take is relatively low with the exception of the peripheral assemblies. This is due to the large batch size and high power density during EPU operation.

Several fuel failures were identified based on the high temperature differential pressure failure threshold. This is primarily attributed to the conservative formulation of the failure threshold. Detailed modeling of the actual pin pressure to the time of the event would have resulted in fewer failures than utilizing the failure threshold based on assembly burnup. The combined failures from both PCMI and High Temperature failures were tabulated for BOC and peak hot excess reactivity (PHEX) conditions. No failures were identified at EOC. Twelve rod drops resulted in fuel failures at BOC (Table 3-6) and thirteen resulted in fuel failures at PHEX (Table 3-7). In addition to identification of the failure mode, the equivalent number of rod failures in terms of the current licensing basis source term for BFN is provided utilizing the correlations from Section 3.3.4.

The rod failures provided in Table 3-6 and Table 3-7 are based upon a 10x10 fuel design. From these tables, one of the BOC rod drops results in the maximum number of failures with a combined value of 816 rods when adjusted to the revised release fraction basis. The current BFN licensing basis allows 850 rod failures. While the analysis shows that for this evaluation the current licensing basis remains supported, the potential remains that 850 rods could be exceeded for future cores if the same approach were utilized. The current licensing basis of 850 rods is based upon distributing the source term using a 62 rod/assembly basis, the equivalent of an older 8x8 fuel design. When adjusted to an ATRIUM 10XM 10x10 assembly design, the number of equivalent full length rods to maintain the same source term would be **[ ]** *.

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Table 3-6 BOC Fuel Rod Failures with Assumed CRDA Criteria

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* The rod failures are for a 10x10 ATRIUM 10XM fuel design. The corresponding BFN AST licensing basis allowing 850 rod failures is based upon an 8x8 assembly. The equivalent source release would occur for **[ ]** ATRIUM 10XM rods.

† The 'Adjusted' column refers to the equivalent number of rods required to bound the source released by the number of rods in the 'Total' column if TFGR impact on current licensing basis release fractions were considered.

Table 3-7 PHEX Fuel Rod Failures with Assumed CRDA Criteria

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* The rod failures are for a 10x10 ATRIUM 10XM fuel design. The corresponding BFN AST licensing basis allowing 850 rod failures is based upon an 8x8 assembly. The equivalent source release would occur for **[ ]** ATRIUM 10XM rods.

† The 'Adjusted' column refers to the equivalent number of rods required to bound the source released by the number of rods in the 'Total' column if TFGR impact on current licensing basis release fractions were considered.

3.4 *Summary of Inherent Conservatisms*

The CRDA evaluation with AREVA's current approved methodology and BFN current licensing basis resulted in no fuel rod failures, Table 3-1. With the evaluation of the CRDA in consideration of the proposed criteria, a significant number of failures are indicated as a result of the lower failure thresholds. Although this evaluation illustrates a significant increase in fuel rod failures, there are several conservatisms which could be addressed to demonstrate compliance with the BFN CRDA current licensing basis or the actual RG 1.183 (Reference 15) dose criteria.

BFN has adopted the Banked Position Withdrawal Sequence (BPWS) rules (Reference 16) to mitigate the consequences of the CRDA by minimizing the worth of individual control rods. Rod worth minimization is achieved through a combination of defining distributed rod groups and the enforcement of banking of these pre-defined groups. The net effect is to distribute changes in control rod density and the corresponding reactivity changes which results in lower effective localized rod worths. The reduced rod worths resulting from the application of BPWS rules helps mitigate the magnitude of the core response to a CRDA. A detailed discussion of the BPWS is contained in Reference 16.

The evaluation of the rod drop involves numerous conservatisms:

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The actual dose consequence evaluation contains conservatism in the calculation and in the margin to the regulatory requirement.

- The number of rod failures (i.e. 850) is based on an 8X8 assembly with 62 fuel rods per assembly. This is equivalent to 1.79% of the core. [

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3.5 *Conclusion*

Given the localized nature of the CRDA and the utilization of BPWS rules (Reference 16) at BFN, it is not anticipated that the dose consequences established for BFN will be exceeded even with the use of the conservative currently approved rod drop methodology.

It has been shown, as summarized in Table 3-1, that the use of the currently approved methodology will continue to protect the current licensing basis when applied to BFN cores at the proposed EPU operating condition.

Furthermore, this document has addressed the impact of the proposed acceptance criteria of Reference 9 in the startup range if it were to be implemented during BFN EPU operation. In summary, it was found that:

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It is therefore concluded that continued evaluation of BFN reload cycles at the proposed EPU conditions with the currently approved methodology will continue to ensure that the current licensing basis and dose limits are met. Furthermore, based on the results of the evaluations above, it is reasonable to conclude that the BFN will comply with the proposed new acceptance criteria for the CRDA at EPU conditions.

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4.0 REFERENCES

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APPENDIX A: HYDROGEN UPTAKE MODEL

Hydrogen Model

AREVA has submitted a hydrogen pick-up (HPU) model as part of the RODEX4 Thermal Mechanical Code Licensing Technical Report (LTR) (Reference A). The U. S. Nuclear Regulatory Commission (NRC) judged that the database was insufficient to allow approval of this model, and it was therefore excluded from the Safety Evaluation Report (SER). Subsequently, AREVA has established an enlarged hydrogen pick-up database from new hot cell examinations to determine hydrogen pick-up levels of liner cladding consisting of recrystallized annealed (RXA) Zircaloy-2 material, from fuel rods irradiated in European power reactors. In the US, AREVA uses Zircaloy-2 in the cold worked stress relieved (CWSR) condition. The equivalence of the two separate lift-off based oxidation databases for the RXA and CWSR metallurgical conditions support the assertion that corrosion of Zircaloy-2 is the same regardless of the metallurgical state. Because hydrogen uptake is intimately linked to oxidation, hydrogen uptake is also considered equivalent for the two metallurgical states of Zircaloy-2.

The currently available industry data are typically presented as a function of burnup (BU), which is roughly proportional to the correct independent variable, namely irradiation time. The hydrogen is created from the oxidation reaction between the zircaloy cladding and the coolant, with a certain percentage, the hydrogen pick-up factor (HPUF), being absorbed by the clad and leading to the final hydrogen wppm (parts per million by weight) content in the clad. The oxidation rate is also believed to be enhanced at high burnup. This HPU evolution is typically illustrated as a plot of hydrogen content versus BU, which exhibits an accelerating level of hydrogen content with burnup. Typical hydrogen values at end-of-life BU (at ~62 GWd/MTU rod average BU) that were reported in the past ranged up to **[ ]** and exhibited a great deal of scatter **[ ]**

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The result of this assessment was the development of a new hydrogen uptake model which [ ] The new model, Reference C, was developed to specifically account for:

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The detailed discussion of the model theory and calibration is provided in Reference C. The comparison of the model to results is provided in Figure A-1 for information.

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Figure A-1 Calculated and Measured HPU

References

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