BSEP 11-0031
Enclosure 4

AREVA Report ANP-2992NP, Revision 0,
AREVA Response to Additional RAI on the Brunswick RODEX4 LAR

ANP-2992NP
Revision 0

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March 2011

AREVA NP Inc.

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Nature of Changes

Item	Page	Description and Justification
1.		This is a new document.

Contents

Tables

There are no tables in this document.

Figures

This document contains a total of 25 pages.

Nomenclature

AOO	anticipated operational occurrence
AREVA NP	AREVA NP Inc.
BSEP	Brunswick Steam Electric Plant
BWR	boiling water reactor
GWd/tU	Gigawatt days per tonne
LOCA	loss-of-coolant accident
LTP	low temperature process
MPa	Mega Pascal
NFIR	Nuclear Fuel Industry Research
NRC	Nuclear Regulatory Commission
PCMI	pellet/cladding mechanical interaction
PNNL	Pacific Northwest National Laboratory
ppm	parts per million
PWR	pressurized water reactor
RAI	request for additional information
RIA	reactivity initiated accident
RXA	recrystallization annealed
SAFDL	specified acceptable fuel design limits
SRA	stress relief annealed
SRP	Standard Review Plan
Zry	zircaloy

1.0 Introduction

The Nuclear Regulatory Commission (NRC) provided an additional Request for Additional Information (RAI), which was received from Progress Energy for the Brunswick Steam Electric Plant (BSEP) on March 9, 2011 (Reference 1). The questions in this additional RAI relate to the corrosion impact on meeting specified acceptable fuel design limits (SAFDL) with respect to cladding strain. In addition, the oxide limit approved for RODEX4 was questioned from the point of view of [ ]

Responses to these questions, comprising proposed corrosion-related limits and evidence that, for AREVA NP Inc.'s (AREVA NP) ATRIUM™* 10XM fuel design (as well as other AREVA NP fuel designs, as the cladding material is the same), the cladding complies with these limits, as presented below in Section 2.0.

2.0 NRC additional RAI Question and AREVA Response

2.1 *NRC RAI Question*

The following additional question was received from the NRC on March 9, 2011:

REQUEST FOR ADDITIONAL INFORMATION
BRUNSWICK STEAM ELECTRIC PLANT UNITS 1 AND 2
REQUEST FOR LICENSE AMENDMENTS
ADDITION OF ANALYTICAL METHODOLOGY TOPICAL REPORTS
TO TS 5.6.5, CORE OPERATING LIMITS REPORT
(ME3858 and ME3859)

"Localized cladding defects (e.g., spallation, hydride blisters) could significantly impact fuel rod stress and strain calculations and ultimately the ability to accurately predict cladding failure. In responding to the Nuclear Regulatory Commission (NRC) staff's RAI, the licensee indicated that [ ] were observed at [ ] microns oxidation level. Additionally, the NRC staff determined that there is not adequate justification within the licensee's application for demonstrating that cladding strain would meet the specified acceptable fuel design limits (SAFDL) in Appendix A to 10 CFR 50 with the amount of hydrogen at the oxidation limit of [ ] microns for ATRIUM 10XM. The NRC staff has determined that the licensee needs to establish (1) an upper bound on cladding

* ATRIUM is a trademark of AREVA NP Inc.

oxide thickness corresponding to [

] and (2) an upper bound on cladding hydrogen content corresponding to the fuel rod cladding strain SAFDL assumed in the AOO overpower analyses.

Please provide to the NRC staff how the licensee will define and document these limits, the basis for the limits, and what processes the licensee will use to comply with these limits."

2.2 *AREVA Response*

The response is structured in two sub-sections, as defined in the NRC question in Section 2.1.

The experimental data and theoretical considerations and analysis presented herein comes from AREVA NP Inc. (AREVA NP), as well as from studies performed in the frame of industry-wide research programs and reported by different organizations in the open literature.

The conclusions of the review of this body of experimental data are that the following upper bounds can be justified:

- [ ] peak oxide, which conservatively bounds the onset of spallation with the potential consequence of hydride localization beneath spalled off areas. The calculation of span peak oxide by the NRC-approved BAW-10247PA (Reference 2) RODEX4 corrosion model for comparison to this analytical limit is biased to include nodular corrosion peaks such that this normal corrosion feature is accounted for by the model.
- [ ] ppm hydrogen pick-up, which conservatively bounds the ductility required to support the anticipated operational occurrences (AOOs) strain criterion.

The AREVA NP oxide database of current Zircaloy-2 (Zry-2) low temperature process (LTP) cladding that did not exhibit an onset of oxide spallation, either stress-relief annealed (SRA) or recrystallization annealed (RXA), is bounded by a peak oxide of [ ] µm. This peak oxide upper bound is consistent with the calculation of span maximum values described by the Reference 2 RODEX4 methodology.

With respect to the hydrogen pick-up limit, the AREVA NP database can be used to express compliance with the [ ] ppm limit in terms of an exposure limit. The statistical analysis of the high burnup sub-set of the hydrogen pick-up versus exposure data shows that the 95/95 upper bound of the hydrogen pick-up is approximately [ ] ppm at [ ] GWd/tU rod average exposure. Therefore, the [ ] ppm limit can be supported up to a rod average exposure of [ ] GWd/tU.

The fuel exposure range of applicability for the NRC-approved RODEX4 methodology is up to [ ] rod average exposure. As noted, the 95/95 upper bound H pick-up below this existing exposure limit is [ ] ppm. Therefore, no additional limit beyond the existing [ ] RODEX4 range of applicability for rod average exposure is required to monitor a H pick-up limit of [ ] ppm for AOO analyses.

2.2.1 Oxide Limit

2.2.1.1 General Considerations

The NRC requests the definition of an oxide upper bound value that would prevent the formation of [ ], which are identified as:

[

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Items a and b are indeed localized oxide surface non-uniformities or defects, while item c refers to a hydride lens formation beneath the surface over a confined small domain.

In fact, item c is dependent on item b, as hydride lens formation could be a consequence of oxide spallation. Oxide spallation is known as a process whereby the oxide scale formed on the surface of the cladding is removed. Oxide spallation can occur, but it is generally confined to a small sector of the cladding circumference and limited in axial extent. The result of oxide spalling is the removal of the thermal barrier of the oxide layer and, therefore, the creation of a cold spot on the clad outer surface at the location of oxide spallation due to the direct contact with the primary coolant. The effects of this cold spot due to oxide spallation can be summarized as follows:

- Reduction in oxidation rate
- Migration of H to the cold spot and potential formation of hydride lenses just below the clad OD surface

While the first effect is beneficial for fuel performance, the second can have detrimental consequences because the hydride lens is more brittle and can promote cracks from the outside of the cladding. This consequence has been invoked, especially in relation to fuel behavior during accidental regimes, such as loss-of-coolant accident (LOCA) or reactivity initiated accident (RIA). In fact, the propensity to crack initiation and lowering of the enthalpy failure threshold during RIA has been proposed to explain failure of the rod REP-Na1 in the CABRI test reactor RIA test program (Reference 3).

However, during normal operation AOO events, the pellet/cladding mechanical interaction (PCMI) process is deformation driven by the larger expansion of the pellet. As documented in Reference 4, under these conditions, the hydride lens is less deleterious than under pressure (i.e., load)-driven conditions, such as those encountered during LOCA, under gas internal pressurization.

Moreover, oxide spallation occurs later in the irradiation life when the oxide reaches the critical thickness for spallation to occur. Therefore, there is limited time for hydrogen migration to the cold spot beneath oxide spallation in order to form the potentially deleterious hydride lens.

The hydride lens-driven cracking mechanism, as a result of oxide spallation, has never been observed to have caused failure during normal operation of fuel rods in a boiling water reactor (BWR). Past failures in early fuel designs caused by massive hydride lenses (i.e., hydride blisters) have been shown to be caused by internal H source in the pellet. Preventive action has since been taken to limit the H content of the fuel rod, as prescribed by the Standard Review Plan (SRP) 4.2 for internal hydriding.

Past failures that have been attributed to run-away oxidation mechanism have been shown to occur at oxide thicknesses larger than about [ ] µm. This would constitute a maximum upper bound for the oxide layer thickness.

In the Reference 2 methodology (i.e., RODEX4), the metal consumed to form the oxide layer is subtracted from the cladding thickness; therefore, spalled or not, the mechanical impact of the brittle oxide layer is accounted for in the calculation of margin to mechanical fuel design limits.

Nevertheless, it is recognized that spallation should be prevented as a precautionary measure against crack initiation by hydride lenses that could be formed at oxide spallation spots. However, it should be noted that this is mainly related to accident conditions.

2.2.1.2 Spallation: Onset and Prior Experience on Partial Spallation

In the prior response to an RAI (Reference 5), AREVA indicated the onset of spallation to occur at [ ] µm. This value was provided in the context of general industry observations and drawn primarily from pressurized water reactor (PWR) experience, including poor past corrosion performance that was isolated to a tubing vendor and manufacturing process with an inadequate annealing parameter.

Moreover, in that context, the [ ] µm oxide thickness was associated with incipient spallation, i.e., a part of the oxide layer (at the outside) was delaminated and, in some cases, completely spalled off. This is explained and illustrated in Reference 6, which states that "at oxide thicknesses greater than approximately 60-70 µm, the start of oxide delamination can occasionally be observed." This is illustrated in Figure 6 of Reference 6, which is reproduced below as Figure 1.

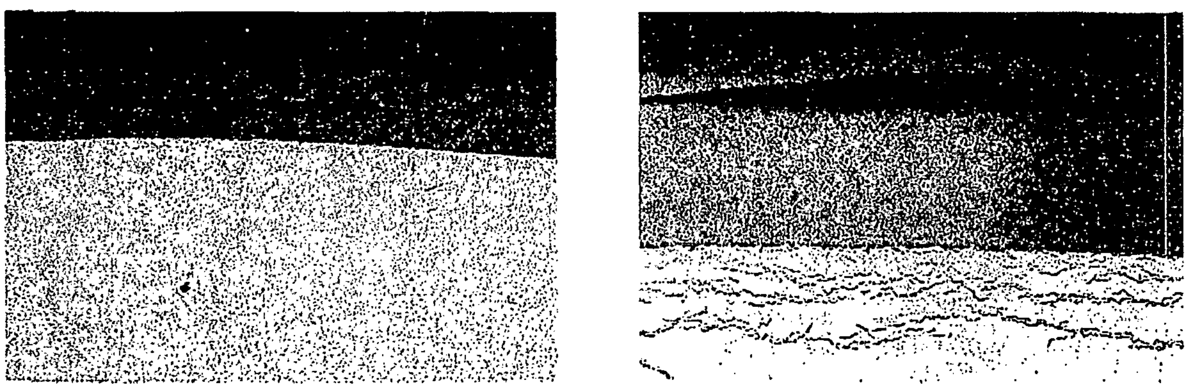

Figure 1 Example of partial delamination of oxide outer sub-layers

It is the experience of AREVA NP that, at higher oxide thicknesses, only the outer sub-layers of the oxide scale spall-off and a sub-layer of [ ] µm remains on the metal. This holds for both PWR and BWR conditions and materials.

A similar remark was made in the discussion of a paper presented by A. M. Garde from ABB CE (currently Westinghouse) in Reference 7. In a clarification response on page 429 of Reference 7, it is stated that spallation of the oxide layer "was observed for oxide thickness generally greater than 100 µm."

An illustration of AREVA NP metallography is presented below in Figure 2, for a local oxide of up to almost [ ] µm on a BWR fuel rod. This is the result of shadow corrosion, but it shows that no spallation has occurred even at this oxide layer thickness.

[

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Figure 2 Thick oxide layer with no spallation

Similar examples cited from prior PWR experience are presented in Figure 3 through Figure 5, below. These figures show the oxide condition in the case of partial and full delamination.

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Figure 3 Example of partial spallation on a PWR rod

[

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Figure 4 Example of almost full spallation with incipient hydride lens on a PWR rod

[

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Figure 5 Example of partial spallation and no hydride lens on a PWR rod

2.2.1.3 Review of AREVA NP Historical Cladding Oxide Database for Zircaloy-2 Cladding Type

A review of the historical oxide database of AREVA NP (Richland) SRA Zircaloy-2 (Zry-2) cladding is illustrated in Figure 6. The visual examination reports on record do not indicate any oxide spallation for any of the data points. The maximum measured oxide thickness was [ ] µm, followed by the second largest oxide thickness of [ ] µm and the bulk of the data below [ ] µm.

Figure 7 illustrates the lift-off oxide database of AREVA NP GmbH (Germany) RXA Zry-2 cladding, presented as Figure 6.11 in Reference 7. No spallation was observed for this database either.

Based on theoretical considerations, and as confirmed by measurements, AREVA NP has established in Reference 2 that the corrosion performance of SRA and RXA Zry-2 material types is the same.

Based on these two databases, it is concluded that below [ ] µm, no spallation occurs and between [ ] µm and [ ] µm, limited flaking occurs in some cases. A visual examination was performed on four lead assemblies that had a maximum local peak oxide of [ ] µm. The four lead assemblies were located in symmetric locations in the core and are at approximately the same exposure. Figure 8, on the right-hand side, shows the visual appearance of representative fuel rods from one of the four lead assemblies. The left-hand side of Figure 8 shows the visual appearance of a fuel rod with a high peak oxide, which was irradiated in a different BWR plant. Some nodular corrosion was noted during the visual examination of these fuel rods, but no evidence of spallation was found. It should be noted that this [ ] μm value is larger than a node average oxide layer thickness, both because it is a span maximum versus a node average value, and because its basis is consistent with a lift-off measurement that includes the combined thickness of the oxide layer and crud.

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Figure 6 Zircaloy-2 SRA oxide historical lift-off database

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Figure 7 Zircaloy-2 RXA lift-off oxide database

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Figure 8 Typical visual aspect of BWR rods with higher peak oxide, showing no spallation

2.2.1.4 Nodular Corrosion and Oxide Data Reduction

Nodular corrosion is a normal form of oxidation and it is not a cladding surface defect. Nodules are localized, thicker oxide patches, but they are dense oxide layers with no spallation and, moreover, [ ] (e.g., see Figure 6.8 on p. 6-16 of Reference 8).

Another dense nodule example is illustrated in Figure 9. As illustrated in Figure 9, the nodules are typically of an [ ]

[ ]

Figure 9 Example of nodules as very localized higher oxides spots superimposed on the general lower uniform corrosion layer

The database presented in Section 2.2.1.3 is based on reducing the measured data as peak oxide value over a span. In this manner, the higher oxide thickness of the nodules is fully captured, as illustrated in Figure 10.

[

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Figure 10 Measured liftoff oxide thickness as biased by the nodular corrosion peaks

As noted above, the metal consumed to form the oxide layer is subtracted from the cladding thickness by the fuel rod licensing methodology; therefore, spalled or not, the mechanical impact of the brittle oxide layer is accounted for.

With respect to H pick-up, it is known that [ ] Therefore, nodular corrosion is of no concern with respect to spallation and of even less concern with respect to hydride localization.

2.2.2 Hydrogen Pick-up Limit

The RAI requests to establish an acceptance criterion for H pick-up in relation to the fuel rod cladding strain SAFDL assumed in the AOO overpower analyses.

No criterion for H pick-up is currently required by SRP 4.2 guidance. Consistent with this guidance, the NRC-approved RODEX4 methodology and predecessor fuel codes and methodologies assume that the dominant phenomenon impacting the cladding ductility during in-reactor operation is irradiation hardening, which overshadows the effect of H when it is precipitated as hydrides.

2.2.2.1 Experimental Data on Ductility of Irradiated and Hydrided Zircaloy

The statement in Section 2.2.2 is valid and applicable to normal operating and AOO conditions for which the cladding temperature is at the operating level of ~ 300 C for BWR conditions. This is the domain required for licensing design calculations of AOO events, as described by SRP 4.2.

The basis for the Section 2.2.2 statement that irradiation hardening overshadows the effect of H when it is precipitated as hydrides is the large body of experimental data acquired through international programs (such as NFIR), or individual testing programs. Burst or ring tests (most relevant for the hoop direction, but also axial tensile tests) have been performed on irradiated and hydrided (i.e., as–irradiated or charged with extra H) cladding, mostly Zry-4 in the SRA or RXA metallurgical condition. This data is fully relevant to Zry-2, because the small difference in the alloying composition has no impact on the ductility characteristic.

The most relevant and recent papers presenting ductility of irradiated and hydrided cladding are References 9 and 10.

Both papers conclude that H content of the cladding at operating temperatures has negligible impact on cladding ductility and strength (being overshadowed by irradiation hardening) up to very high H concentration levels. This conclusion is summarized in Reference 10, as described below.

The studies presented in References 9 and 10 are illustrated in Figure 6 of Reference 9 and Figure 7 of Reference 10, which are reproduced below as Figure 11 and Figure 12. These studies show that sufficient [

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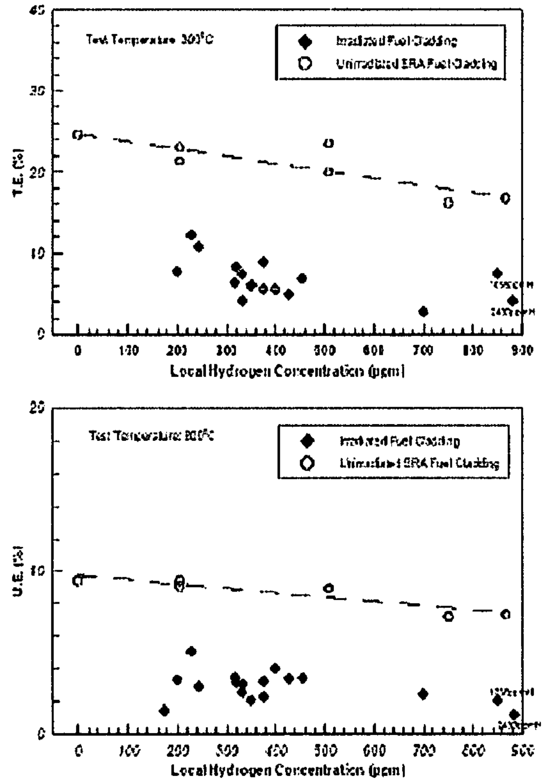

Figure 6 Tensile strains for unirradiated and irradiated fuel cladding specimens as a function of local hydrogen concentrations, UTT at 300°C

Figure 11 Ductility of irradiated cladding vs. H concentration (Reference 9)

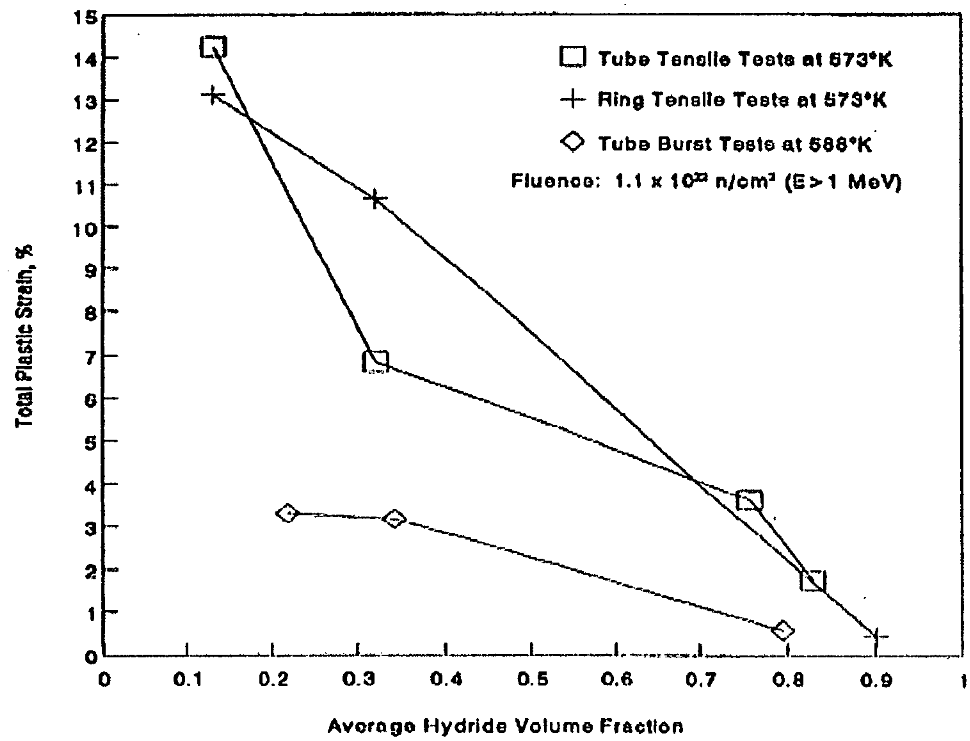

Figure 12 Ductility of irradiated cladding vs. H concentration (Reference 10)

2.2.2.2 Ductility Model from PNNL-17700 Report

The above data is confirmed by the review study performed by Pacific Northwest National Laboratory (PNNL) for the NRC and reported in PNNL-17700, July 2008 (Reference 11). The PNNL report includes the description of a model to estimate remaining cladding ductility for a given H concentration.

Estimations of the clad ductility based on that PNNL model developed for the NRC in Reference 11 were made and presented below. The model estimates the uniform hoop strain as dependent on H concentration and temperature. Assuming a conservative lower bound temperature of the clad of 554 K, Figure 13 below shows the results.

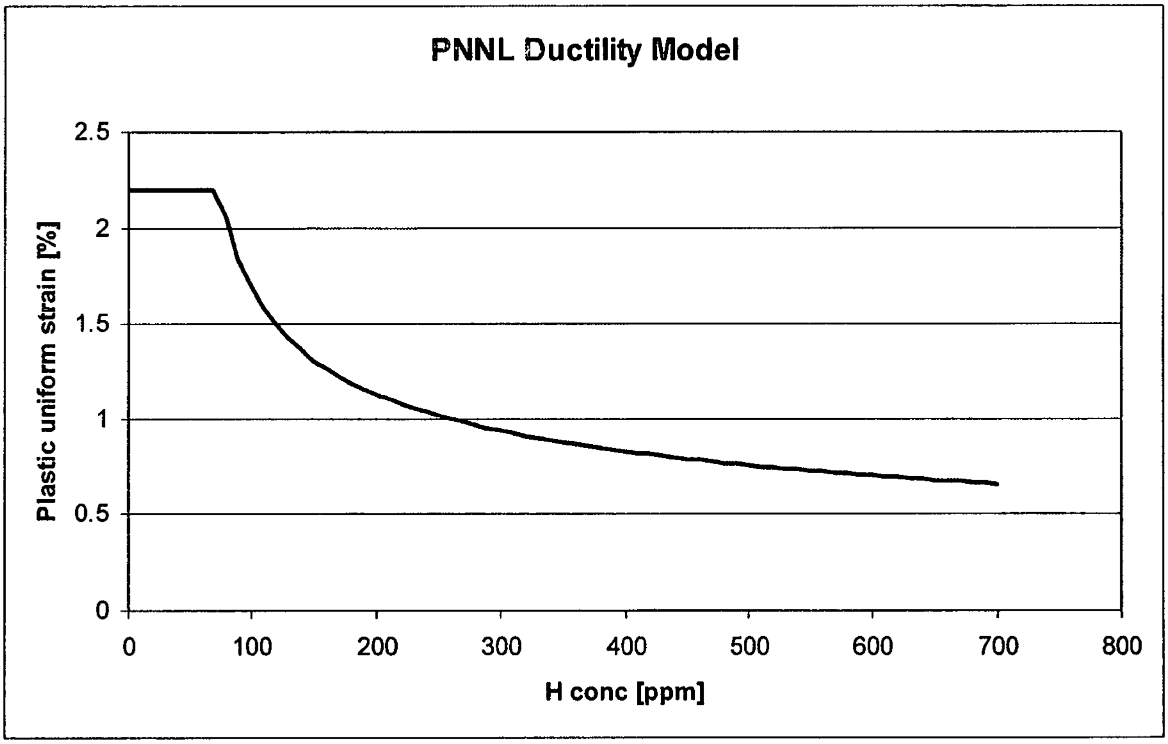

Figure 13 Results of PNNL ductility model

SRP 4.2 limits the calculated uniform (elastic plus plastic) strain of the cladding to no more than 1%. Therefore, the material ductility after irradiation with the associated H pick-up has to be better, i.e., the uniform (elastic plus plastic) strain must be larger than 1%. The lower bound of the material ductility can be estimated as follows.

A lower bound value of the irradiated yield strength is [ ] MPa at a corresponding elastic strain greater than [ ] %, based on AREVA NP test data obtained at ~300 °C. The uniform strain capability of the cladding (i.e., [ ] % elastic strain plus plastic strain from Figure 13) exceeds 1% with no hold time at high H concentrations. Even accounting for a hold time that would convert some of the elastic strain into plastic, the uniform strain capability of the cladding will exceed the SRP 4.2 criterion at a H pick-up of [ ] ppm.

2.2.2.3 Upper Bound of H Pick-up Limit based on the AREVA NP Database

The limit of [ ] ppm is justified by the mechanical tests and the associated ductility of irradiated and hydrided cladding, as described above in Section 2.2.2.2.

The existing AREVA NP H pick-up database that was presented in Reference 8 can be used to evaluate the 95/95 upper bound of the H pick-up at high exposure. The high exposure then defines a bounding exposure limit as associated with this H pick-up.

The results of this statistical analysis are presented in Figure 14, below, where the average value is the full line, while the dashed line represents the 95/95 upper bound. The black diamonds are the sub-set of five datapoints at very high burnup (with an average fuel rod exposure of [ ] GWd/tU), while the pink squares are the whole set above [ ] GWd/tU exposure. The upper 95/95 of the very high burnup sub-set is ~ [ ] ppm ([ ] ppm calculated value).

[

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Figure 14 Upper 95/95 bounds of H pick-up at high exposure

The fuel exposure range of applicability for the NRC-approved RODEX4 methodology is up to [ ] rod average exposure. As shown by Figure 14, the 95/95 upper bound H pick-up below this existing exposure limit is [ ] ppm. Therefore, no additional limit beyond the existing [ ] RODEX4 range of applicability for rod average exposure is required to monitor a H pick-up limit of [ ] ppm for AOO analyses.

3.0 References

1. 38-9157410-000, "RODEX4 RAI Issued by NRC," AREVA NP Inc., March 2011.

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

3. R. O. Meyer, et al., "A Regulatory Assessment of Test Data for Reactivity Accidents," *International Topical Meeting on Light Water Reactor Fuel Performance*, Portland OR, March 2-6, 1997, Proceedings, p. 729.

4. R. Montgomery, et al., "The Mechanical Response of Cladding with a Hydride Lens under PCMI Loading Conditions," *International Seminar on Pellet-Clad Interaction in Water Reactor Fuels*, Aix-en-Provence, France, 9-11 March 2004.

5. ANP-2978P Revision 0, "AREVA Responses to RAIs on the ATRIUM 10XM Compliance Audit and Brunswick LARs," AREVA NP Inc., December 2010.

6. L. F. P. Van Swam and S. H. Shann, "The Corrosion of Zircaloy-4 Fuel Cladding in Pressurized Water Reactors," *Zirconium in the Nuclear Industry: Ninth International Symposium,* ASTM STP 1132, pp 758-781.

7. A. M. Garde, G. P. Smith, and R. Pirek, "Effects of Hydride Precipitate Localization and Neutron Fluence on the Ductility of Irradiated Zircaloy-4," *Zirconium in the Nuclear Industry: Eleventh International Symposium*, ASTM STP 1295, pp 407-430.

8. ANP-2899P Revision 0, "Fuel Design Evaluation for ATRIUM™ 10XM BWR Reload Fuel," AREVA NP Inc., April 2010.

9. S. K. Yagnik, et al., "Effect of Hydrides on the Mechanical Properties of Zircaloy-4," *Proceedings of the 2004 International Meeting on LWR Performance*, Orlando, Florida, September 19-22, 2004, paper 1089.

10. A. M. Garde, et al., "Hydrogen Pick-Up Fraction for ZIRLO™ Cladding Corrosion and Resulting Impact on Cladding Integrity," *Proceedings of Top Fuel 2009*, Paris, France, September 6-10, 2009, paper 2136.

11. K. J. Geelhood, W. G. Luscher and C. E. Beyer, "PNNL Stress/Strain Correlation for Zircaloy," PNNL-17700, Pacific Northwest National Laboratory, July 2008.

List of Regulatory Commitments

The following table identifies the actions in this document to which the Brunswick Steam Electric Plant has committed. Statements in this submittal, with the exception of those in the table below, are provided for information purposes and are not considered commitments. Please direct questions regarding these commitments to Mr. Lee Grzeck, Acting Supervisor - Licensing/Regulatory Programs, at (910) 457-2487.

Commitment	Completion Date
When using AREVA Topical Report BAW-10247PA, *Realistic Thermal-Mechanical Fuel Rod Methodology for Boiling Water Reactors*, Revision 0, April 2008, to determine core operating limits, the fuel cladding peak oxide thickness calculated by the RODEX4 corrosion model will be limited to less than the proprietary value defined in Section 2.2 of AREVA Report ANP-2992P, Revision 0, *AREVA Response to Additional RAI on the Brunswick RODEX4 LAR*.	Upon implementation of the Unit 1 and Unit 2 license amendments authorizing the incorporation of AREVA Topical Report BAW-10247PA into Technical Specification 5.6.5.b.