ENCLOSURE 2

Sequoyah Nuclear Plant, Units 1 and 2

Response to Request for Additional Information Regarding Application to Modify Technical Specifications for Use of AREVA Advanced W17 HTP Fuel (TS-SQN-2011-07) (Non-Proprietary Version)

ANP-3053(NP)
Revision 0

Sequoyah HTP Fuel Transition - NRC RAIs and Responses

November 2011

AREVA NP Inc.

AREVA

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ANP-3053(NP)
Revision 0

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Nature of Changes

Item	Page	Description and Justification
1.	All	Initial release.

Table of Contents

Nomenclature

AST	Alternate Source Term
CFR	Code of Federal Regulations
DBA	Design-basis Accident
DNB	Departure from Nucleate Boiling
EFPD	Effective Full Power Days
EOC	End-of-cycle
FA	Fuel Assembly
FHA	Fuel Handling Accident
ft	Foot
GWd	Giga-watt-day
IFM	Intermediate Flow Mixing Grids
kW	kilowatt
LAR	License Amendment Request
LHGR	Linear Heat Generation Rate
mtU	Metric Tons of Uranium
NAF	Neutron Absorber Fuel
NRC	Nuclear Regulatory Commission
RAI	Request for Additional Information
RG	Regulatory Guide
TVA	Tennessee Valley Authority

1.0 Introduction

This document presents the Requests for Additional Information (RAIs) that were generated by the Nuclear Regulatory Commission (NRC) following their review of the Sequoyah Units 1 and 2 proposed HTP fuel transition License Amendment Request (LAR). The Sequoyah Units 1 and 2 are switching to the AREVA Advanced W17 HTP fuel assembly design.

As discussed in Section 2.0, the NRC provided two RAIs.

Section 3.0 presents the responses that were prepared by AREVA in support of the fuel transition effort.

2.0 NRC RAIs

The NRC provided two RAIs regarding the Sequoyah Units 1 and 2 proposed technical specifications changes to allow use of the AREVA Adv. W17 HTP Fuel. The first RAI addresses the impacts of the fuel type change on the source term for the radiological design-basis accident (DBA) analyses. The second RAI deals with the mechanical design differences between the proposed Adv. W17 HTP fuel assembly and the resident Mark-BW fuel assembly.

Responses to these RAIs are provided in Section 3.0.

3.0 RAI Responses

This section presents responses to the NRC RAIs received by Tennessee Valley Authority (TVA). The RAI responses are prepared by the engineering disciplines that are responsible for the subject matter.

The technical groups that have provided responses to the RAIs are shown below.

Engineering and Projects – Nuclear Analysis (EPNA)
Fuel Design –Mechanics (FDM)

The responses begin on the following page.

3.1 *RAI Question 1 - Radiological Design-basis Accident (DBA) Analyses*

Title 10 of the Code of Federal Regulations (10 CFR), Section 100.11, "Determination of exclusion area, low population zone, and population center distance," and 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 19, "Control room," set regulatory dose limits for offsite and control room after a design basis accident. Section 50.67, "Accident source term," of 10 CFR, states that the NRC may issue amendments only if the applicant's analysis demonstrates that certain onsite and control room dose limits are met.

Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," (ADAMS Accession No. ML003716792) provides guidance to licensees on performing evaluations and reanalyses in support of meeting the dose limits in 10 CFR 50.67. RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Nuclear Power Reactors," (ADAMS Accession No. ML031490640) provides guidance to licensees for performing evaluations and reanalysis in support of meeting 10 CFR 100.11.

Standard Review Plan 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," (ADAMS Accession No. ML003734190) states:

The reviewer should evaluate the AST [Alternative Source Term] proposed by the licensee against the guidance in RG-1.183. Differences between the licensee's proposal and the guidance should be resolved with the licensee. Although the licensee is allowed to propose alternatives to the guidance, large amounts of staff resources were expended in developing the revised source term (Ref. 5) from which the RG-1.183 source term was derived. Section 2.0 of RG-1.183 provides generic guidance on what would be expected before the staff would approve an AST with deviations from the AST in Section 3.0 of the guide.

Standard Review Plan 15.0.1 also states:

The analysis methods and assumptions used by the licensee in determining the core inventory should be reviewed to ensure that they are based on current licensing basis rated thermal power, enrichment, and burnup.

A modification to the licensing basis fuel type can have the potential to change the core isotopic distribution and inventory assumed in post accident conditions. The impacts regarding the core inventory and enrichment are not discussed in the proposed amendment. The amendment states that burnup limits are "similar" to the current fuel, but does not provide the new burnup limits.

Please verify that the core inventory, enrichment, and burnup are not changed or are bounded by the proposed amendment. If these parameters are bounded, please provide a justification for the statement that they are bounded. Please confirm that the burnup and linear heat generation rates in footnote 11, Regulatory Position 3.2 of RG 1.183 are met, or justify the source terms used.

3.1.1 AREVA Response

The Alternative Source Term methodology was implemented for the Fuel Handling Accident (FHA) scenario only. In accordance with SQN UFSAR Section 15.1.7.1 and Section 15.5.6 and Table 15.5.6-1, the fission product inventory used for the Fuel Handling Accident is based on (a) a rated thermal power of 3479 MWth (3455 + 0.7% uncertainty), the highest powered fuel assembly at End of Cycle (EOC) conditions for 1500 EFPD (i.e., maximum assembly burnup of 60 GWd/mtU), (b) a maximum enrichment of 5% U-235, (c) a radial peaking factor of 1.70, (d) 100 hours decay after shutdown prior to fuel movement, and maximum heavy metal loading of 470 kgU/FA. The core inventory was calculated using ORIGEN-S computer code. None of these parameters have been changed and are all still applicable and bounding.

Table 2-1 of AREVA Report ANP-2986P Revision 003 compares HTP and the Mark-BW fuel assembly designs. As noted in Table 2-1, the fuel rod pitch and fuel rod length are identical. Likewise, Table 2-2 of AREVA Report ANP-2986P Revision 003 compares the HTP and Mark-BW fuel rod design parameters. From Table 2-2 the fuel rod length, active stack height, plenum volume, and pellet diameters are identical. Table 2-2 indicates that the fuel rod cladding material of M5™ for HTP and Mark-BW designs is identical. Only the guide tube material and instrument tube change from M5™ to recrystallized Zircaloy-4 and the addition of three Intermediate Flow Mixing Grids (IFMs) composed of recrystallized Zircaloy-4 (shown in Table 2-3 and Table 2-6 of AREVA Report ANP-2986P Revision 003) could impact the fission product inventory. The impact of this change on the neutron spectra and resulting fission product inventory is insignificant.

In Figure 3-8 and Figure 3-9 of AREVA Report ANP-2986P Revision 003 the transition core maximum EOC fuel assembly burnup is 52.189 GWd/mtU at location A11 and 52.772 GWd/mtU at location C14, for the 1st and 2nd HTP transition cores respectively. In Figure 3-10 of AREVA Report ANP-2986P Revision 003, the maximum EOC fuel assembly burnup for a full HTP core is 46.842 GWd/mtU at locations G8 and H9. These burnups are well within the fuel assembly burnup limit of 60 GWd/mtU assumed in the analysis of record for the fission product inventory for the FHA.

Thus, design parameters important to fission product inventory of the maximum fuel enrichment of 5% U-235, rated thermal power, maximum fuel assembly and fuel rod burnup are not impacted by the transition to HTP fuel. Therefore, the fission product inventory used for the Fuel Handling Accident remains bounded for the HTP fuel.

In addition, as part of the normal reload analyses, cycle-specific input parameters important to source term and dose are compared to the analysis of record inputs as provided by the licensee. These include rated thermal power, maximum assembly burnup, maximum assembly heavy metal loading, maximum assembly enrichment, radial peaking factors, % DNB for locked rotor event, % DNB for rod cluster control assembly ejection accident, and spent fuel pool rod internal pressure. If maximum assembly average burnups are 54 GWd/mtU or greater, then a pin census at EOC conditions will be performed to ensure that Regulatory Guide 1.183, Table 3, Footnote 11 criterion is not exceeded. That is, the maximum linear heat generation rate (LHGR) is less than 6.3 kW/ft peak rod average power for burnups exceeding 54 GWd/mtU. If fuel rod burnups were to exceed 54 GWd/mtU and any pins exceed the LHGR of 6.3 kW/ft, then the gap release fraction for the non-LOCA events would be conservatively doubled or evaluated using the ANSI/ANS-5.4 methodology based on the maximum burnup. Alternatively, the core design would be modified to ensure that the LHGR criterion is met.

Therefore, the core inventory, enrichment, burnup, and linear heat generation rates for the proposed HTP Fuel amendment will be verified to bound the license basis for radiological design for each reload.

3.2 *RAI Question 2 - Mechanical Design Comparison - Advanced W17 HTP vs. Mark-BW Fuel*

Regulatory Position 1.3.2, "Re-Analysis Guidance," of RG 1.183 states:

Any implementation of an AST, full or selective, and any associated facility modification should be supported by evaluations of all significant radiological and nonradiological impacts of the proposed actions. This evaluation should consider the impact of the proposed changes on the facility's compliance with the regulations and commitments listed above as well as any other facility-specific requirements. These impacts may be due to (1) the associated facility modifications or (2) the differences in the AST characteristics. The scope and extent of the re-evaluation will necessarily be a function of the specific proposed facility modification[6] and whether a full or selective implementation is being pursued. The NRC staff does not expect a complete recalculation of all facility radiological analyses, but does expect licensees to evaluate all impacts of the proposed changes and to update the affected analyses and the design bases appropriately. An analysis is considered to be affected if the proposed modification changes one or more assumptions or inputs used in that analysis such that the results, or the conclusions drawn on those results, are no longer valid.

The proposed amendment states that the fuel rod and fuel pellet materials and design are "similar" to the current fuel, but does not provide any specific details regarding the impact of the proposed change on the analysis of record. For example, changes to the structure of the fuel rod could impact the assumed amount of fuel damage as a result of a fuel handling accident. In order for the NRC staff to evaluate the impact of the proposed change, please provide any changes to the parameters, assumptions, or methodologies in the radiological design-basis accident (DBA) analyses and a justification for those changes. If there are changes to the radiological DBA analyses, please provide the resulting change to the calculated radiological consequence of the DBA.

3.2.1 AREVA Response – Mechanical Design

Fuel Assembly Design

The Adv. W17 HTP and Mark-BW fuel assembly weights are [ ] lbs and [ ] lbs, respectively. There is only [ ]% ([ ]%) increase in weight for Adv. W17 HTP fuel with respect to the Mark-BW fuel design. This increase is an insignificant change for fuel assembly handling accident condition analysis.

Fuel Rod Design

The Adv. W17 fuel rod and fuel pellet materials are identical to the current fuel rod design of the Mark-BW fuel assembly. The cross sectional properties of the cladding, fuel column length, rod fill gas pressure (nominal initial fuel rod pressurization of [ ] psig for UO_2 and [ ] psig for NAF rod), and upper end cap are identical (See Table 2-2 of AREVA Report ANP-2986P Revision 003). It should be noted that the Adv. W17 HTP fuel rod overall length is only [ ] inch longer than the Mark-BW fuel rod design. This difference is insignificant.

There are no design changes in the fuel rod between the Adv. W17 HTP and the Mark-BW fuel rod that would impact the fuel handling accident. The only design difference between the Adv. W17 fuel rod and the Mark-BW fuel rod is between the lower end caps of the fuel rod. Because of the change in the bottom nozzle; from the current Trapper design to the Fuel Guard design, the tip diameter of the lower end cap is modified. This is due to the difference in the blocked area of the debris filtering feature.

The current design of the Mark-BW fuel rod lower end cap is shown in Figure 1.

[ ]

Figure 1 – Mark-BW Fuel Rod Lower End Cap

The lower end cap for the Adv. W17 HTP fuel rod is shown in Figure 2.

[ ]

Figure 2 Adv. W17 HTP Fuel Rod Lower End Cap

The difference in the two lower end cap designs is the diameter of the tip end flat, [ ] inch (Mark-BW) vs. [ ] inch (Adv. W17 HTP). By keeping the same [ ] angle, the barrel section of the two lower end caps is slightly different. The weight of the end cap is insignificantly affected. All other design attributes of the lower end caps are identical. This design change is insignificant.

Cage Structure Design

The Adv. W17 HTP and Mark-BW fuel designs use the guide tube feature connected to spacer grids and tie plates to build the cage structure. The structural design of the Adv. W17 HTP fuel cage is relatively stronger than the Mark-BW fuel cage (See Table 2-1, 2-3, and 2-6 of ANP-2968P Revision 003) for the following reasons:

- Adv. W17 HTP fuel uses the [ ] design at the lower section. Both cage designs have the same guide tube section properties at the upper section of the cage.
- Guide tubes are welded to the spacer grid for the Adv. W17 HTP fuel design whereas Mark-BW fuel cage design allows some flexibility to displace spacer grids instead of somewhat rigid connection between guide tube and spacer grid. This provides additional cage structural strength for the Adv. W17 HTP fuel design.

- Adv. W17 HTP fuel design uses total of [ ] in a cage design whereas Mark-BW fuel design uses total of [ ]. This makes the Adv. W17 HTP fuel cage design at least as strong as Mark-BW fuel cage design.

It can be concluded that the changes to the design parameters result in an Adv. W17 HTP fuel structure that is at least as strong as the Mark-BW fuel design. Therefore, the number of fuel rods damaged from a fuel assembly drop for an Adv. W17 HTP fuel assembly is expected to be bounded by the Mark-BW design.

3.2.2 AREVA Response - Radiological

The Alternative Source Term methodology was implemented for the Fuel Handling Accident scenario only. The current licensing basis described in SQN UFSAR Section 15.5.6 and Table 15.5.6-1 for the fuel handling accident assumes a fuel assembly drop onto the fuel transfer pit floor resulting in the failure of 100% of the fuel rods in the fuel assembly dropped and a rod internal pressure of 1200 psi at spent fuel pool conditions after 100 hours of decay.

The HTP fuel assembly is designed with a welded cage and contains Intermediate Flow Mixing Grids (IFMs) constituting a more robust structural design than the Mark-BW fuel assembly design. Therefore, for the HTP fuel assembly design fewer fuel rods are expected to fail due to a fuel assembly drop within containment in the fuel transfer pit than the analysis of record. However, for conservatism 100% of the fuel rods are assumed to fail due to the fuel assembly drop inside containment.

As stated in Section 3.2.1 of this report, the HTP fuel rod design is identical to the Mark-BW fuel rod. Section 2.3, Table 2-2 of AREVA Report ANP-2986P Revision 003 shows that the back fill pressure is also identical, with the fuel rod internal pressure limits met in accordance with BAW-10183PA per Table 2-8 of AREVA Report ANP-2986P Revision 003. Therefore, rod internal pressure for the HTP at spent fuel pool conditions would be identical to that of the Mark-BW fuel. Cycle-specific affects on rod internal pressure such as burnup and enrichment are evaluated as part of the reload analyses to ensure that the rod internal pressure is less than 1200 psig.

Therefore, there are no changes to the radiological DBA analyses or radiological consequence of the DBA as a result of the proposed HTP Fuel amendment.

4.0 References

None.

ENCLOSURE 3

Sequoyah Nuclear Plant, Units 1 and 2

AREVA NP Affidavit

Attached is the affidavit supporting the request to withhold the proprietary information included in Enclosure 1 from the public.

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

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

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

3. I am familiar with the AREVA NP information contained in the report ANP-3053(P), Revision 0, entitled "Sequoyah HTP Fuel Transition – NRC RAIs and Responses," dated November 2011 and referred to herein as "Document." Information contained in this Document has been classified by AREVA NP as proprietary in accordance with the policies established by AREVA NP 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 NP 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 secret and commercial or financial information."

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

(a) The information reveals details of AREVA NP'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 NP.

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

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

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

7. In accordance with AREVA NP'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 NP only as required and under suitable agreement providing for nondisclosure and limited use of the information.

8. AREVA NP 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.

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SUBSCRIBED before me this 4th
day of November 2011.

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Sherry L. McFaden
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 10/31/14
Reg. # 7079129

SHERRY L. MCFADEN
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My Commission Expires Oct 31, 2014