ENCLOSURE 2

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT
UNITS 1 AND 2

ANP-3053(NP), Revision 4, Sequoyah HTP Fuel Transition –
NRC RAIs and Responses, June 2012
(Non-Proprietary Version)

ANP-3053(NP)
Revision 4

Sequoyah HTP Fuel Transition – NRC RAIs and Responses

June 2012

AREVA NP Inc.

AREVA

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**Sequoyah HTP Fuel Transition –
NRC RAIs and Responses**

Nature of Changes

Item	Page	Description and Justification
1.	All	Page numbers changed due to Table of Contents expansion from the addition of responses to the thermal-hydraulics RAIs.
2.	1 – 13, 103 - 109	Changes due to responses to the thermal-hydraulics RAIs.

Table of Contents

List of Tables

List of Figures

Nomenclature

AFD	Axial Flux Distribution
ANS	American Nuclear Society
ANSI	American National Standard Institute
AOO	Anticipated Operational Occurrence
AOR	Analysis of Record
AST	Alternate Source Term
ASI	Axial Shape Index
BOC	Beginning of Cycle
CAP	Corrective Action Program
CCP	Centrifugal Charging Pump
CE	Combustion Engineering
CFR	Code of Federal Regulations
CFM	Centerline Fuel Melt
CHF	Critical Heat Flux
COLR	Core Operating Limits Report
CSL	Core Safety Limit
DC	Downcomer
DBA	Design-basis Accident
DNB	Departure from Nucleate Boiling
DNBR	Departure from Nucleate Boiling Ratio
ECC	Emergency Core Cooling
ECCS	Emergency Core Cooling System
EFPD	Effective Full Power Days
EOC	End-of-cycle
EOP	Emergency Operation Procedures
EPRI	Electric Power Research Institute
FA	Fuel Assembly
FHA	Fuel Handling Accident
ft	Foot
F/R	Fuel Rod
GWd	Giga-watt-day
G/T	Guide Tube
HFP	Hot Full Power
HPSI	High Pressure Safety Injection
HTC	Heat Transfer Coefficient
ID	Inner Diameter
IFM	Intermediate Flow Mixing Grids
kW	kilowatt

LAR	License Amendment Request
LBLOCA	Large Break Loss of Coolant Accident
LCO	Limiting Conditions for Operation
LHGR	Linear Heat Generation Rate
LHR	Linear Heat Rate
LOCA	Loss of Coolant Accident
LPSI	Low Pressure Safety Injection
MAP	Maximum Axial Peaking
MPFR	Minimum Protected Flow Rate
MSMG	Midspan Mixing Grid
MTC	Moderator Temperature Coefficient
mtU	Metric Tons of Uranium
NAF	Neutron Absorber Fuel
NRC	Nuclear Regulatory Commission
OD	Outer Diameter
OPΔT	Overpower Delta-Temperature
OSG	Original Steam Generator
OTΔT	Overtemperature Delta-Temperature
PCT	Peak Clad Temperature or Peak Cladding Temperature
PLHGR	Planar Linear Heat Generation Rate
QD	Quick Disconnect
RAI	Request for Additional Information
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RFR	Required Flow Rate
RG	Regulatory Guide
RHR	Residual Heat Removal
RLBLOCA	Realistic Large Break Loss of Coolant Accident
RPS	Reactor Protection System
RSE	Reload Safety Evaluation
RSG	Replacement Steam Generator
RTP	Rated Thermal Power
RWST	Refueling Water Storage Tank
SBLOCA	Small Break Loss of Coolant Accident
SCD	Statistical Core Design
SDL	Statistical Design Limit
SER	Safety Evaluation Report
SG	Steam Generator
SIP	Safety Injection Pump
SIT	Safety Injection Tank
SI	Safety Injection

SQN	Sequoyah Nuclear Plant
SR	Surveillance Requirement
TCD	Thermal Conductivity Degradation
TDL	Thermal Design Limit
T-H	Thermal-Hydraulic
TR	Topical Report
TS	Technical Specification
TVA	Tennessee Valley Authority
UFSAR	Updated Final Safety Analysis Report
W	Westinghouse

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) (Reference 4). 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 in Round 1 (Reference 1), twenty-six RAIs in Round 2 (Reference 2), and one RAI in Round 3 (Reference 7).

Section 3.0 presents the Round 1 RAI responses that were prepared by AREVA in support of the fuel transition effort. These responses are also contained in Reference 3.

Section 4.0 presents the Round 2 RAI responses that were prepared by AREVA in support of the fuel transition effort. AREVA prepared responses for 23 of the 26 RAIs while TVA prepared responses for the remaining 3 RAIs. These responses are also contained in Reference 6.

Section 5.0 presents the thermal conductivity degradation RAI responses that were prepared by AREVA in support of the fuel transition effort.

Section 6.0 presents the thermal-hydraulics RAI responses that were prepared by AREVA in support of the fuel transition effort.

2.0 NRC RAIs

The NRC provided two Round 1 RAIs in Reference 1 regarding the Sequoyah Units 1 and 2 proposed Technical Specification (TS) changes to allow use of the AREVA Adv. W17 HTP Fuel (Reference 4). 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 the first round of RAIs are provided in Section 3.0 and are contained in Reference 3.

The NRC provided twenty-six Round 2 RAIs in Reference 2 regarding the Sequoyah Units 1 and 2 proposed TS changes to allow use of the AREVA Adv. W17 HTP Fuel (Reference 4). RAIs 1 through 10 address questions from the NRC's Reactor Systems Branch. RAIs 11 through 26 address questions from the NRC's Nuclear Performance and Code Review Branch.

AREVA prepared responses to 23 of the 26 Round 2 RAIs while TVA prepared responses to the remaining 3 RAIs; these responses are provided in Section 4.0 and in Reference 6.

The NRC provided one RAI in an email to TVA regarding the effects of thermal conductivity degradation (TCD) in Sequoyah (SQN). The responses are provided in Section 5.0.

The NRC provided two RAIs in an email to TVA regarding thermal-hydraulic changes due to the transition to AREVA Adv. W17 HTP Fuel in SQN. The responses are provided in Section 6.0.

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 Round 2 RAI Responses

This section presents responses to the second round of NRC RAIs received by TVA in Reference 2. 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:

Fuel Design – Mechanics (FDE-AL)
Fuel Design – Neutronics (FDN-AL)
Fuel Design – Thermal-Hydraulics (FDT-AL)
Fuel Design – Materials and Thermal-Mechanics (FDM-AL)
Engineering and Projects – Radiological and Environmental Analysis (PEPNE-A)
Engineering and Projects – LOCA, ECCS, and BWR Analysis (PEPNL-A)

The responses begin on the following page.

4.1 *RAI Question 1*

The AREVA Advanced W17 HTP fuel assembly design consists of standard uranium dioxide (UO_2) fuel pellets with gadolinium oxide (Gd_2O_3) burnable poison and M5™ cladding. Please identify those plants used [sic] similar fuel and provide description for the discrepancy, if any, in the fuel design from the proposed Advanced W17 HTP fuel.

4.1.1 AREVA Response – Question 1

UO2 fuel and UO2-Gd2O3 fuel inside an alloy M5® cladding is currently in use in the following US nuclear plants:

a. Sequoyah Units 1 & 2
b. Three Mile Island Unit 1
c. Davis-Besse Unit 1
d. Fort Calhoun Unit 1
e. Calvert Cliffs Unit 2
f. Palisades Unit 1
g. Crystal River Unit 3

UO2 fuel inside an alloy M5® cladding is currently in use in the following US nuclear plants:

a. Oconee Units 1, 2, & 3
b. Arkansas Nuclear Unit 1
c. North Anna Units 1 & 2

The introduction of an alloy M5® cladding design with both UO2 fuel and UO2-Gd2O3 fuel is planned for the following nuclear plants in 2011 – 2012:

a. Harris Unit 1
b. Robinson Unit 2
c. Calvert Unit 1
d. Oconee Units 1, 2, 3

The Advanced W17 HTP fuel assembly fuel rod design is nearly identical to the current fuel rod design which is in use at both of the Sequoyah plants. The only difference is the diameter of the tip of the fuel rod lower end cap which is being modified due to the change from the TRAPPER™ bottom nozzle to the FUELGUARD™ lower tie plate. Section 2.0 of ANP-2986(P) Revision 3 contains a discussion of the design features of the Adv. W17 HTP fuel assembly.

HTP based designs are operating successfully in the following plants. Differences in the fuel design as compared to the Advanced W17 HTP design are noted in Table 1.

Table 1: Summary of AREVA PWR Fuel Designs in US Nuclear Reactors

The mechanical design of the Advanced W17 HTP fuel assembly to be introduced into the Sequoyah Unit 2 plant in Cycle 19 is most similar to the fuel assembly currently operating successfully in Harris Unit 1. The differences in the fuel designs at these two plants are summarized in Table 2. For more details on the design features of the Advanced W17 HTP fuel assembly, see Sections 1.2 and 2.0 of ANP-2986(P) Revision 3.

Table 2: Comparison of Advanced W17 HTP and Harris Unit 1 W17 HTP Fuel Designs

The differences between the Advanced W17 HTP fuel design and the successfully operating fuel design in Harris Unit 1 are small and are within the current base of experience with HTP fuel designs.

4.2 *RAI Question 2*

SQN plans to refuel and operate with AREVA Advanced W17 HTP fuel beginning with the cycles following the refueling outage in the fall of 2012 for SQN, Unit 2, and in the fall of 2013 for SQN, Unit 1. Please provide: (a) the core loading pattern with clearly specifying fuel types and quantities including their fresh or resident fuel with once-burned, twice-burned or thrice-burned for SQN, Units 1 and 2; (b) the guidelines or procedures used to generate the final core loading pattern; and (c) a detailed description of the impact on the departure from nucleate boiling ratio (DNBR) limit calculation due to the final selected core loading pattern.

4.2.1 AREVA Response – Questions 2(a) and 2(b)

The final core loading pattern for a reload cycle is typically not generated until approximately 6 months prior to a refueling outage. This allows for finalization of reload cycle energy requirements and consideration of the latest operational history information in the core design process. Therefore, the Unit 2 Cycle 19 and Unit 1 Cycle 20 core designs are not yet complete. Instead, neutronics evaluations described in Section 3.0 of ANP-2986(P) Revision 3 include representative transition core designs, subject to design ground rules similar to those that will be used for future core designs. The core

design methods used are previously approved and are the same for representative and actual cycle designs. The discussion and data presentation in Section 3.0 of ANP-2986(P) Revision 3 indicates that these designs differ by a similar degree as standard cycle-to-cycle variations.

The core loading pattern is developed using the previously approved methods currently referenced in TS 6.9.1.14.a, and allow for normal expected cycle-to-cycle variations due to changes in cycle energy requirements and plant power operational history. Core design ground rules and design targets are generated by TVA on a cycle-by-cycle basis. Engineering guidelines used by AREVA core designers and analysts are available for review at AREVA facilities. These guidelines are designed to comply with the approved methodologies.

4.2.2 AREVA Response – Question 2(c)

The Departure from Nucleate Boiling Ratio (DNBR) analysis process used for the SQN units is based on modeling a core composed entirely of the new fuel design, in this case the Adv. W17 HTP fuel design, and a conservative reference core power distribution. This thermal-hydraulic modeling differs from the final selected core loading plan in two primary aspects: 1) fuel assembly hydraulics and 2) power distributions simulated in the maneuvering analysis. During transition cores, there may be core flow redistribution taking place as a result of hardware-driven hydraulic differences between the new and co-resident fuel designs that will impact the DNBR performance of the hot rod. In addition, the conservative reference core power distribution does not reflect all permissible core power distributions during core maneuvering. The conservative nature of the reference core power distribution (see Section 4.2.5 of the ANP-2986(P) Revision 3) is reflected not in absolute terms of just the hot rod peak, but rather in terms of creating a nearly flat pin power distribution at and around the hot pin to produce a conservative power generation environment for obtaining a predicted minimum DNBR.

The DNBR analysis process for the Adv. W17 HTP full core modeling for DNB predictions is subsequently supplemented with analyses to address the above noted two differences. A DNBR transition core penalty is determined that reflects the impact of the hydraulically–driven core flow redistribution in the mixed core condition. The mixed core thermal-hydraulic modeling may reflect the actual or a conservative representation of the core loading plan. The DNBR transition core penalty, discussed in Section 4.4.1 of the ANP-2986(P) Revision 3, is offset by DNBR margin within the Thermal Design Limit (TDL) which protects the critical heat flux (CHF) correlation DNBR limit.

The verification of the DNBR performance with respect to the possible maneuvering power distributions is a multi-step process where DNBR-based curves of Maximum Allowable Peaking (MAP) are generated for various axial peaks and axial peak locations at steady-state and transient conditions. These DNBR-based MAP curves possess the conservative power generation environment at and around the hot rod. The limiting core power distributions predicted for the actual core loading plan are compared to these DNB-based MAP curves to assure the DNB performance for the core loading plan will support the alarms and trip function setpoints. This action addresses the core power distribution impact of the actual core loading plan with respect to the DNBR limit.

4.3 *RAI Question 3*

The proposed TS changes for the DNBR limits for each fuel type are as follows:

For the Advanced W17 HTP fuel design
DNBR ≥ 1.132 for the BHTP correlation
DNBR ≥ 1.21 for the BWU-N correlation

For the Mark-BW fuel design
DNBR ≥ 1.21 for the BWCMV correlation
DNBR ≥ 1.21 for the BWU-N correlation

Please provide: (a) a detailed description with respect to the application of the DNBR correlations to the Advanced W17 HTP fuel and the Mark-BW fuel design; (b) rationale to apply two DNBR correlations to each fuel type; (c) operating procedures and method of any core monitoring system to assure that these two DNBR limits would not be violated.

4.3.1 AREVA Response – Question 3(a)

The critical heat flux (CHF) performance of the Adv. W17 HTP fuel design is represented by two CHF correlations (noted as DNBR correlations in the RAI) which are axial region specific as a result of the axial positioning of the HTP and HMP spacer grids.

The CHF performance of the Adv. W17 HTP fuel design, within the axial region beginning at the lowermost HTP spacer grid (shown in Figure 2-7 of ANP-2986P Revision 3) and extending upward to the top of the fuel stack, is represented by the NRC approved BHTP CHF correlation discussed in BAW-10241P-A, Revision 1. The grid hardware within this axial region includes 7 HTP spacer grids and 3 intermediate flow mixing (IFM) grids as shown in Figure 3. The BHTP CHF correlation was developed using the HTP CHF data base from EMF-92-153(P)(A), Supplement 1, and for the application with subchannel local conditions predicted with the LYNXT code (BAW-10156, Revision 1). The application of the BHTP correlation on the Adv. W17 HTP fuel design within the above stated axial region using the LYNXT code is in compliance to the NRC approved range of local coolant conditions, on page 3 of the SER of BAW-10241P-A, Revision 1, and the NRC approved range of fuel design parameters, in Table 2 of the Revision 0 SER within BAW-10241P-A, Revision 1.

The CHF performance of the Adv. W17 HTP fuel design within the axial region below the lowermost HTP spacer grid (shown in Figure 2-8 of ANP-2986P Revision 3) is dictated by the upstream HMP spacer grid (shown in Figure 2-9 of ANP-1986P Revision 3). The HMP grid, consistent in general geometry to the HTP spacer grid with the exception that the flow channels (castellations) are completely vertical along their length, is a non-mixing grid design, and is represented by the CHF performance of the BWU-N CHF correlation, in BAW-10199P-A, for non-mixing grids. The application of the BWU-N CHF correlation is in compliance to the NRC approved application limits as stated in Table 3, page xxii of the SER of BAW-10199P-A.

Each of the two CHF correlations for the Adv. W17 HTP fuel design has a corresponding CHF design limit as stated in the RAI. During the application of the two CHF

correlations in DNB analyses, each correlation is exclusively applied within its appropriate axial region of the fuel design. Each of these correlations has a respective Statistical Design Limit (SDL) and Thermal Design Limit (TDL) for analyses using the Statistical Core Design Methodology discussed in ANP-2986(P) Revision 3 (Section 4.2.4). Therefore, the predicted minimum DNBRs in both of the two axial regions are compared against the respective TDL to determine which axial region possesses the minimum DNB margin for a given core power distribution (in steady-state or transient mode). Since the BWU-N correlation axial region is only one grid span (distance between consecutive grids) in length at the bottom of the fuel stack where substantial subcooling exists, this axial region on the Adv. W17 HTP fuel design is not expected to result in limiting DNB margin predictions. The same conclusion is reached that the BWU-N correlation's axial region is non-limiting for any non-SCD analyses (performed for some transient events) where uncertainties would be deterministically applied.

The CHF performance of the resident Mark-BW fuel design is represented by two CHF correlations which are also axial region specific as a result of the combined axial positioning of the vaneless lower end grid and vaneless lowermost intermediate spacer grid and the downstream mixing vaned spacer grids as shown in Figure 3 below.

The CHF performance of the Mark-BW fuel design, within the axial region beginning at the lowermost mixing vaned spacer grid and extending upward to the top of the fuel stack, is represented by the NRC approved BWCMV CHF correlation discussed in BAW-10159P-A and BAW-10189P-A. (The CHF performance applied for the Mark-BW spacer grid design, using an effective grid spacing discussed in BAW-10189P-A, is sometimes noted as the BWCMV-A CHF correlation.) The grid hardware within this axial region includes 5 mixing vaned spacer grids as shown in Figure 3 below. The application of the BWCMV correlation on the Mark-BW fuel design within the above stated axial region using the LYNXT code is in compliance with the NRC approved limitations stated in Table 1, page iv of the SER of BAW-10189P-A for the Mark-BW spacer grid design.

The CHF performance of the Mark-BW fuel design within the axial region below the lowermost mixing vaned spacer grid is dictated by the presence of the vaneless lower end spacer grid and the downstream vaneless intermediate spacer grid. These two grids define the lowermost two grid spans of the Mark-BW fuel design. These vaneless grids are represented by the CHF performance of the BWU-N CHF correlation, in BAW-10199P-A, for non-mixing grids. The application of the BWU-N CHF correlation is in compliance with the NRC approved application limits as stated in Table 3, page xxii of the SER of BAW-10199P-A.

Each of the two CHF correlations for the resident Mark-BW fuel design has a corresponding CHF design limit as stated in the RAI. The DNB margin comparison process for both correlations in DNB analyses for the Mark-BW fuel design is identical to that stated above for the Adv. W17 HTP fuel design with two CHF correlations. During the application of the BWCMV and BWU-N correlations for the Mark-BW fuel design in DNB analyses, each correlation is exclusively applied within its appropriate axial region of the fuel design. Each of these correlations has a respective Statistical Design Limit (SDL) and Thermal Design Limit (TDL) for DNB analyses using the SCD methodology. Therefore, the predicted minimum DNBRs in both of the two axial regions are monitored against the respective TDL for the CHF correlation to determine which axial region

possesses the minimum DNB margin for a given core power distribution (in steady-state or transient mode) when using the SCD methodology. Since the BWU-N correlation axial region is two grid spans in length at the bottom of the fuel stack where substantial subcooling exists, this axial region on the Mark-BW fuel design is not expected to result in limiting DNB margin. The same conclusion is reached that the BWU-N correlation's axial region is non-limiting for any non-SCD analyses (performed for some transient events) where uncertainties would be deterministically applied.

4.3.2 AREVA Response – Question 3(b)

Since each fuel design contains grid hardware with a higher CHF performance capability in the central and upper axial regions as opposed to a lower CHF performance capability in the lower axial regions, it is appropriate to utilize CHF correlations (noted as DNBR correlations in the RAI) approved for the respective levels of CHF performance in each of the two axial regions.

4.3.3 AREVA Response – Question 3(c)

The means by which the violation of these two DNBR limits is avoided within reload design analyses, for each fuel design, is by the imposed constraints of the TS core safety limits, the verification of the OTΔT trip DNBR protection, verification of acceptable DNBR results for the Condition I/II transients, the establishment of allowable core power distributions for steady-state and transient conditions, and the verification of core power distribution DNBR performance for alarm setpoints and trip function setpoints. These DNB-based analyses are performed using the LYNXT subchannel code which is equipped to predict DNBR values for axial region-specific CHF correlations and, in this case, is needed for the Adv. W17 HTP and Mark-BW fuel designs.

Additional means by which the violation is avoided during core power operation are core monitoring procedures. The core power distribution is periodically measured using the moveable incore detector system during power operation to ensure that the core is operating as designed. These requirements are provided in the TSs specifically for the Limiting Conditions for Operation (LCO). Periodic surveillance of the core power distribution is designed to provide a continual check on power peaking factors and provides the reactor engineer with any indications of operational conditions that may cause higher peaking factors to exist. Measured peaking factors are compared to predicted peaking factors periodically, and the Surveillance Requirements for LCO 3.2.3 help ensure that appropriate actions are taken in instances where the core may deviate from designed behavior. The methods and procedures that are used to perform core power distribution monitoring as specified in the Technical Specifications are described in Sections 6 and 7 of approved topical report BAW-10163P-A.

Figure 3: Distribution of Grid Types and Applicable CHF Correlations for the Adv. W17 HTP and Mark-BW Fuel Designs

(not to scale)

4.4 *RAI Question 4*

The proposed Safety Limit 2.1.1.2 states, "The maximum local fuel pin centerline temperature shall be maintained ≤ 4901 °F [degrees Fahrenheit], decreasing by 13.7 °F per 10,000 MWD/MTU [megawatt-days/metric ton of uranium] of burnup for COPERNIC applications, and ≤ 4642 °F, decreasing by 58 °F per 10,000 MWD/MTU of burnup for TACO3 application." Please provide: (a) details of the reason why these two codes are needed to set the fuel pin centerline temperature limit; and (b) the means used to monitor the maximum local fuel pin centerline temperature.

4.4.1 AREVA Response – Question 4

(a) AREVA is currently transitioning from the approved TACO3-based computer code and related analyses to the more modern approved COPERNIC fuel rod design computer code and related analyses. COPERNIC is approved for analyses of fuel rods with M5® advanced alloy cladding, to accurately simulate the behavior of a fuel rod during irradiation and to verify that the specified fuel rod design meets all design and safety criteria. However, in order to provide the plant with the licensing flexibility to reinsert burned fuel assemblies fabricated with Zircaloy-4 clad fuel rods, TACO3 is being retained as an approved methodology in the core Safety Limits for this type of application. TACO3 is approved for the analyses of fuel rod designs with Zircaloy-4 cladding.

(b) The maximum local fuel rod centerline temperature is not directly observable; therefore, it cannot be directly monitored. Instead, the fuel centerline melt analysis provides local maximum allowable linear heat generation rate limits based upon the fuel rod centerline melt relationships prescribed in the TSs to ensure that fuel melt does not occur during normal operation and anticipated operational occurrences. The automatic reactor protection system trip function (specifically, the Overpower Delta-Temperature or OPΔT trip), ensures that the local linear heat rate will not exceed the centerline fuel melt limits when the appropriate trip settings are reached and the reactor is tripped.

4.5 *RAI Question 5*

In the proposed Figure 3.2-1, Flow versus Power for 4 Loops in Operation, there is [sic] a 3.5 percent measurement uncertainty for flow is included. Please provide the basis for a 3.5 percent flow measurement uncertainty and justify that this 3.5 percent is conservative.

4.5.1 TVA Response – Question 5

Sequoyah Nuclear Plant (SQN) Departure from Nucleate Boiling (DNB) analyses are based on a specific Minimum Protected Flow Rate (MPFR) for reactor coolant system (RCS) flow through the vessel. To ensure RCS flow is always greater than the MPFR, RCS flow is monitored to maintain a Required Flow Rate (RFR) that is approximately 3.5 percent (%) greater than the MPFR. The TSs, therefore, require that the RFR be approximately 3.5% greater than assumed in the DNB analyses to account for measurement uncertainty. For example, the MPFR at 100% Rated Thermal Power (RTP) is currently (4 x 87,000) = 348,000 gpm and the RFR is (1.035 x 348,000) ≈

360,100 gpm. The proposed TS change increases both the MPFR and the RFR by approximately 5%, to 365,600 gpm and 378,400 gpm, respectively.

The 3.5% margin for measurement uncertainty is conservative because a plant specific uncertainty analysis determined that RCS flow measurements are more accurate than ±3.5% of the MPFR. RCS flow is measured from RCS elbow flow meters that produce a differential pressure proportional to the square of the flow rate through each RCS loop. The constants of proportionality ("K values") for the RCS elbow flow meters were determined from heat balance measurements taken during the initial startup of each unit where the differential pressure being measured across each RCS elbow was correlated to its RCS loop flow rate. Therefore, the measurement uncertainty in RCS flow is composed of (1) the uncertainties associated with the heat balance measurements used in the determination of the K values for the RCS elbow flow meters and (2) the uncertainties in measuring the differential pressure across the RCS elbows. The uncertainties associated with the determination of the K values dominate the overall uncertainty in the measurement of RCS flow.

The plant specific uncertainty analysis determined an overall uncertainty of 2.44% of the RFR as indicated via the plant process computer. Because this analysis determined overall measurement uncertainty is less than 3.5% of the MPFR, it demonstrates that when indicated flow meets the RFR, actual flow meets the MPFR assumed in DNB analyses. For example, if indicated flow was equal to the proposed 100% RTP RFR (378,400 gpm), then actual flow would be no less than approximately (1 - 0.0244) x 378,400 gpm ≈ 369,165 gpm, which exceeds the 365,600 gpm MPFR.

RCS flow is verified once per 12 hours and once per 18 months as required by SQN TS Surveillance Requirements (SRs) 4.2.5.1 and 4.5.2.2, respectively. The 12-hour SR verifies that the 10-minute average of the differential pressures across the four RCS elbow flow meters meets or exceeds the differential pressure corresponding to the RFR as indicated by the Eagle 21 reactor protection system. The 18-month surveillance measures the differential pressure directly from the RCS elbow flow meter pressure transmitters' analog outputs when calculating the RCS loop flows using pressure transmitter specific K values. The calculated RCS flow is required to meet or exceed the RFR.

RCS flow measurement surveillance procedures require immediate notification of the Unit Senior Reactor Operator for evaluation of operability under TS LCO 3.2.5 if RCS flow is less than the RFR in TS Figure 3.2-1, and surveillance procedure deficiencies require reporting and evaluation under the Corrective Action Program (CAP). Calibration of RCS flow instrumentation is required once per 18 months. An out of tolerance finding on instrumentation used to verify TS compliance requires reporting and evaluation under the CAP.

It is expected that RCS flows on both units will significantly exceed the RFR value requested in this TS change. At full power during the most recent operating cycle on Unit 1, RCS flow has consistently measured over 103% of the (proposed) RFR. The Unit 2 RCS flow rate will increase and approximately match that of Unit 1 following Unit 2 steam generator replacement (scheduled for the next Unit 2 refueling outage commencing in the fall of 2012, coincident with the introduction of AREVA Advanced W17 HTP fuel).

In conclusion, as presented in the foregoing assessment, the 3.5% of the MPFR allowance for RCS flow measurement uncertainty is conservative because evaluations have shown that installed instrumentation and associated maintenance practices and surveillance procedures result in measurements that are more accurate than the 3.5% analytical margin between the RFR (per TS Figure 3.2-1) and the MPFR. Additionally, actual RCS flow is expected to exceed the requested TS flow requirements by approximately 3%.

4.6 *RAI Question 6*

Provide a flow chart or table to clearly demonstrate that all the approved methodologies listed in the proposed revision to TS 6.9.1.14.a are necessary to support the cycle-specific parameters listed in TS 6.9.1.14.

4.6.1 AREVA Response – Question 6

The topical report (TR) references proposed for TS 6.9.1.14.a were selected to provide the set of codes and methodology that, when used together, define the complete code and method set used to generate or to validate the COLR parameters. The references proposed for TS 6.9.1.14.a were also selected as an extension of the methodology described in BAW-10220P (Mark-BW Fuel Assembly Applications for Sequoyah Nuclear Units 1 and 2), which was approved with the original transition to Mark-BW fuel.

AREVA NP does not provide a single methodology topical report for SQN reload safety evaluation (RSE) methodology. However, the approved TRs proposed for reference in TS 6.9.1.14.a are used together to support analysis and generation of COLR parameters, and may be categorized as follows: TRs that specify material properties used as critical inputs by evaluation models for mechanical and ECCS performance; safety analysis TRs that describe models and/or generate limits that are directly used in reload safety evaluations; and TRs that describe calculation of core operating limit parameters. Table 3 below illustrates the relationships between the TRs; for simplicity, the revision numbers have been omitted from the table.

Table 3: Topical Report Categories

Methodology	TR Number	Application
Material properties	BAW-10186P-A BAW-10227P-A	Property inputs to evaluation models for safety, mechanical, thermal, and thermal-hydraulic analyses
Safety analysis (Non-LOCA & LOCA)	BAW-10169P-A EMF-2103P-A EMF-2328P-A	Transient analysis (non-LOCA) RLBLOCA analysis & limits SBLOCA analysis & limits
Fuel thermal & mechanical analysis	BAW-10231P-A	Fuel thermal & mechanical analysis & limits (reference for SL 2.1.1)
Thermal-hydraulic analysis	BAW-10159P-A BAW-10189P-A BAW-10199P-A BAW-10241(P)(A)	BWCMV CHF BWCMV CHF BWU CHF BHTP CHF (references for SL 2.1.1)
Reload analysis	BAW-10180-A BAW-10163P-A	Nuclear design code & core reactivity parameters Power distribution & peaking surveillance parameters

Generally, the material property TRs are inputs to the various safety, thermal, mechanical, and thermal-hydraulic evaluation models that are used to perform thermal, mechanical, and thermal-hydraulic evaluations and system response analyses. These

evaluations are performed using the codes and methods TRs identified as safety, thermal, mechanical, and thermal-hydraulic analysis methods. The outputs of these evaluations define the peaking limits that are used directly in cycle-specific reload safety evaluations to validate margin to specific operating limits, or to define the value of the operating limit at which a specified margin is preserved. Generation of individual operating limits for reactivity and power distribution control for a reload safety evaluation is discussed in the last row of Table 3. The results of the reload safety evaluation validate or define the core operating limits related to core reactivity, power distribution control, and power peaking surveillance specified in the COLR.

The Enclosure of the application to modify the Sequoyah Nuclear Plant Technical Specifications for use of AREVA Advanced W17 HTP fuel (Reference 4) contains a table to illustrate the changes proposed to TS 6.9.1.14.a (Core Operating Limits Report (COLR) Reference List). NRC Generic Letter 88-16 specifies the guidance that controls determination of values of the cycle-specific parameters listed in TS 6.9.1.14 and assures conformance to 10 CFR 50.36, by specifying the calculation methodology and acceptance criteria. The TRs proposed in the Enclosure noted above describe the previously approved codes and methods applicable to generation and validation of the core operating limits specified in the COLR.

Table 4 (below) is a revised table to clarify which topical reports support each core operating limit specified in TS 6.9.1.14, and which also demonstrates the applicability of each topical report to the Advanced W17 HTP design and to the Mark-BW design. Since some Sequoyah core designs will operate with both Mark-BW and Advanced W17 HTP fuel resident in the core, the references to previously approved methodology in TS 6.9.1.14.a must be capable of supporting complete reload safety evaluations for both fuel designs.

Table 4: TS SECTION 6.9.1.14.a – CORE OPERATING LIMITS REPORT (COLR) Reference List

Current SQN TS 6.9.1.14.a COLR Reference List	Fuel Design Applicability	Proposed SQN TS 6.9.1.14.a COLR Reference List	Operating Limit(s) Directly Supported	Operating Limit(s) Indirectly Supported	Comments
1. BAW-10180P-A, NEMO – Nodal Expansion Method Optimized	Adv W17 HTP Mark-BW	1. BAW-10180-A Revision 1, NEMO – Nodal Expansion Method Optimized, March 1993	MTC RIL (Shutdown Bank) RIL (Control Bank)	AFD OTΔT $f_1(\Delta I)$ OPΔT $f_2(\Delta I)$	BAW-10180-A, Revision 1 is the approved nuclear design code for simulation of core reactivity and power distribution and supports generation of the cycle-specific reactivity and power distribution limits in the Sequoyah Nuclear Plant Units 1 & 2 COLRs.
2. BAW-10169P-A, RSG Plant Safety Analysis – B&W Safety Analysis Methodology for RSG Plants	Adv W17 HTP Mark-BW	2. BAW-10169P-A, Revision 0, RSG Plant Safety Analysis – B&W Safety Analysis Methodology for RSG Plants, October 1989		OTΔT $f_1(\Delta I)$ OPΔT $f_2(\Delta I)$	BAW-10220P describes changes to the RELAP5 model used for Sequoyah analysis
3. BAW-10163P-A, Core Operating Methodology for Westinghouse-Designed Reactors	Adv W17 HTP Mark-BW	3. BAW-10163P-A Revision 0, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June 1989	AFD OTΔT $f_1(\Delta I)$ OPΔT $f_2(\Delta I)$ Monitoring parameters in COLR		This TR also specifies the methods that determine parameters defined in the COLR used for power peaking surveillance
4. BAW-10168P-A, RSG LOCA-B&W LOCA Evaluation Model for RSG Plants	Adv W17 HTP Mark-BW	4. EMF-2328(P)(A), Revision 0, PWR Small Break LOCA Evaluation Model, March 2001	LOCA F_Q LOCA K(Z)		EMF-2328(P)(A) implements S-RELAP5 code methodology (Note 1)
5. WCAP-10054-P-A, Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code	N/A				Historical: Not applicable to current fuel in the core
6. WCAP-10266-P-A, The 1981 Revision of Westinghouse Evaluation Model Using BASH Code	N/A				Historical: Not applicable to current fuel in the core
7. BAW-10227P-A, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel	Adv W17 HTP Mark-BW	5. BAW-10227P-A, Revision 1, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, June 2003		OPΔT $f_2(\Delta I)$ LOCA F_Q LOCA K(Z)	This TR specifies material properties for the methods used to calculate centerline fuel melt and transient cladding strain limits for fuel designs that utilize M5® cladding and provides the licensing basis for M5® application

Table 4 (continued): TS SECTION 6.9.1.14.a – CORE OPERATING LIMITS REPORT (COLR) Reference List

8. BAW-10186-A, Extended Burnup Evaluation	Adv W17 HTP Mark-BW	6. BAW-10186P-A, Revision 2, Extended Burnup Evaluation, June 2003		OPΔT $f_2(\Delta I)$ LOCA F_Q LOCA K(Z)	This TR validates the methods used for calculation of burnup dependencies in centerline fuel melt and F_Q Limits for rod average burnups up to 62 GWd/mtU
9. EMF-2103P-A, Realistic Large Break LOCA Methodology for Pressurized Water Reactors	Adv W17 HTP Mark-BW	7. EMF-2103P-A, Revision 0, Realistic Large Break LOCA Methodology for Pressurized Water Reactors, April 2003	LOCA F_Q LOCA K(Z)	AFD RIL (Control Bank)	This TR specifies methods used for calculation of the LOCA F_Q and K(Z) limits
	Adv W17 HTP	8. BAW-10241(P)(A), Revision 1, BHTP DNB Correlation Applied with LYNXT, July 2005		MAP Limits OPΔT $f_1(\Delta I)$ AFD	This TR supports the CHF correlations specified in SL 2.1.1 (Note 2)
	Adv W17 HTP Mark-BW	9. BAW-10199P-A, Revision 0, The BWU Critical Heat Flux Correlations, August 1996		MAP Limits OPΔT $f_1(\Delta I)$ AFD	This TR supports the CHF correlations specified in SL 2.1.1 (Note 2)
	Mark-BW	10. BAW-10189P-A, Revision 0, CHF Testing and Analysis of the Mark-BW Fuel Assembly Design, January 1996		MAP Limits OPΔT $f_1(\Delta I)$ AFD	This TR supports the CHF correlations specified in SL 2.1.1 (Note 2)
	Mark-BW	11. BAW-10159P-A, Revision 0, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, August 1990		MAP Limits OPΔT $f_1(\Delta I)$ AFD	This TR supports the CHF correlations specified in SL 2.1.1 (Note 2)
	Adv W17 HTP Mark-BW	12. BAW-10231P-A, Revision 1, COPERNIC Fuel Rod Design Computer Code, January 2004		OPΔT $f_2(\Delta I)$	This TR specifies methods used for calculation of centerline fuel melt and transient cladding strain limits (Note 3)

Notes to COLR Reference List Table:

Note 1: The added topical report replaces COLR reference 4 (BAW-10168P-A Rev. 3, RSG LOCA – BWNT LOCA EM for Recirculating Water Steam Generator Plants) with EMF-2328(P)(A), PWR Small Break LOCA Evaluation Model. The S-RELAP5 implementation is consistent with code strategy for non-Babcock & Wilcox plants, and continues the use of approved methodology including RODEX2A.

Note 2: The applicable and previously approved Critical Heat Flux reports are added for consistency with the information provided in BAW-10220P and provide a complete COLR reference list consistent with the CHF correlation safety limits specified in SL 2.1.1.

Note 3: The added topical report presents the approved modern fuel performance code and method for thermal-mechanical analyses and provides a COLR reference list consistent with the CFM temperature safety limit specified in SL 2.1.1.

It should be noted that BAW-10220P (Mark-BW Fuel Assembly Application for Sequoyah Nuclear Units 1 and 2, March 1996) was written to support the licensing submittal to modify the Sequoyah Nuclear Plant Technical Specifications for use of AREVA Mark-BW fuel. That report presented the complete LOCA, non-LOCA, mechanical, nuclear, and thermal-hydraulic codes and methods that supported the Mark-BW fuel design and transition. BAW-10220P remains applicable to evaluation of Mark-BW fuel in Sequoyah Nuclear Plant reload cores, except for the large break LOCA (LBLOCA) and small break LOCA analyses. The original LBLOCA analysis has been replaced by an analysis based upon approved realistic large break LOCA (RLBLOCA) methodology, described in EMF-2103(P)(A). The original SBLOCA analysis will be replaced with the approved methodology described in EMF-2328(P)(A) when the Adv. W17 HTP fuel is loaded. In addition, the methodology for fuel rod thermal and mechanical performance will be replaced with the approved methodology described in BAW-10231P-A, Revision 1, when the Adv. W17 HTP fuel is loaded. Where NRC-approved TR references were appropriate, a brief discussion was presented in BAW-10220P with a reference to the individual approved TRs for details. BAW-10220P was approved in a Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment 223 to Facility Operating License No. DPR-77 and Amendment 214 to Facility Operating License No. DPR 79, dated April 21, 1997 (ADAMS Accession No. ML013320456). The NRC approved the application of M5® for Sequoyah in a Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment 258 to Facility Operating License No. DPR-77 and Amendment 249 to Facility Operating License No. DPR 79, dated July 31, 2000 (ADAMS Accession No. ML003737235).

Further details that illustrate the application of the TRs proposed for reference by TS 6.9.1.14.a are discussed in Attachment 5 (ANP-2986(P), Revision 3, Sequoyah HTP Fuel Transition, June 2011) attached to the submitted Application to modify the Technical Specifications for use of AREVA Adv. W17 HTP fuel, dated June 17, 2011 (Reference 4). In addition to providing a description of the mechanical features of the AREVA Adv. W17 HTP fuel design, ANP-2986(P) Revision 3 addresses the application of these TRs to each type of evaluation (neutronic, mechanical, thermal, thermal-hydraulic, LOCA, and non-LOCA safety analysis) required to support the modification of the technical specifications and provides the detailed technical justification for operation of the fuel within the applicable safety acceptance criteria.

4.7 *RAI Question 7*

It appears that TACO3 code is applied to calculation for limitation of local fuel pin centerline temperature. Please justify that TACO3 code does not support the parameter listed in TS 6.9.1.14.1.

4.7.1 AREVA Response – Question 7

AREVA will use the modern COPERNIC fuel rod design computer code, which is approved for use with M5® advanced alloy cladding, to accurately simulate the behavior of a fuel rod during irradiation and to verify that the specified fuel rod design meets all design and safety criteria. However, in order to provide the plant with the licensing

flexibility to reinsert burned fuel assemblies fabricated with Zircaloy-4 clad fuel rods, TACO3 is being retained as an approved methodology in TS 6.9.1.14 for this type of application. The TACO3 code was approved for use at Sequoyah Nuclear Plant in the SER for BAW-10220P at the time of the original fuel transition to AREVA fuel.

4.8 *RAI Question 8*

Provide a detailed description with respect to the applicability of all the approved methodologies listed in the proposed TS 6.9.1.14.a to the AREVA Advanced W17 HTP fuel design.

4.8.1 AREVA Response – Question 8

ANP-2986(P), Revision 3 was prepared to describe the codes and methods that will be applied in reload safety evaluations for reload cores of Sequoyah Nuclear Plant Units 1 and 2 with Adv. W17 HTP fuel. In all analyses that involve evaluation of the fuel rod performance (including neutronic, thermal, mechanical, and safety analyses), the codes and methods currently applied for Mark-BW fuel remain applicable for evaluation of the Adv. W17 HTP fuel because of the similarity in the fuel rod design. Section 2.2.2 of ANP-2986(P), Revision 3 provides more details of the fuel rod mechanical design. As described in Section 2.4 of ANP-2986(P) Revision 3, the fuel rod design for the Adv. W17 HTP fuel assembly is the same as the Mark-BW fuel rod design that comprises the current resident fuel in Sequoyah Units 1 and 2. The topical reports that are referenced in TS 6.9.1.14.a are reviewed in ANP-2986(P), Revision 3 with respect to their purpose and SER restrictions to address their applicability for analysis of Adv. W17 HTP fuel. Table 5 (below) provides the fuel design applicability of each topical report referenced in TS 6.9.1.14.a and provides a summary of the justification for their use in reload safety evaluations (RSEs) for Sequoyah reload cores.

Table 5: Applicability of Approved Codes & Methods in TS 6.9.1.14.a

Approved Topical Report	Applicability	Remarks	ANP-2986(P) Rev. 3 Supporting Section
1. BAW-10180-A Revision 1, NEMO – Nodal Expansion Method Optimized, March 1993	Adv. W17 HTP Mark-BW	Benchmarking of the NEMO code has been performed and demonstrated acceptable for Sequoyah Nuclear Station Units 1 & 2 for the past six cycles of operation at each Unit, including startup testing. These confirm accurate predictions by the NEMO code package.	3.0
2. BAW-10169P-A, Revision 0, RSG Plant Safety Analysis – B&W Safety Analysis Methodology for RSG Plants, October 1989	Adv. W17 HTP Mark-BW	The RCS system transient response is not primarily affected by the fuel rod or fuel assembly design. Since the transition to Adv. W17 HTP fuel does not change the RCS design or function, this TR remains applicable for non-LOCA safety analysis system transient response analysis.	5.0
3. BAW-10163P-A Revision 0, Core Operating Limit Methodology for Westinghouse-Designed PWRs, June 1989	Adv. W17 HTP Mark-BW	Since the evaluation of core power distribution parameters is primarily dependent upon the fuel rod design and model, the similarity in Mark-BW and Adv. W17 HTP fuel rod designs establishes the applicability of this TR for application in RSEs.	3.0
4. EMF-2328(P)(A), Revision 0, PWR Small Break LOCA Evaluation Model, March 2001	Adv. W17 HTP Mark-BW	The SBLOCA TR is applicable to W and CE 2x4 plants and the fuel designs utilized by these plants. The TR sample problem fuel design was an HTP assembly design. The SBLOCA TR application models an entire HTP core as there is no PCT sensitivity for mixed-core considerations. This TR has been applied to other operating plants for HTP fuel designs and approved by the NRC. This establishes the applicability of this TR to the Adv. W17 HTP fuel design.	5.0
5. BAW-10227P-A, Revision 1, Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel, June 2003	Adv. W17 HTP Mark-BW	This TR supports development and validation of fuel rod thermal-mechanical and safety analysis limits used in RSEs for both Mark-BW and Adv. W17 HTP fuel rod designs that use M5® cladding. Both the Mark-BW and Adv. W17 HTP fuel rods feature M5® cladding. Since the cladding material is identical for the two designs, the similarity in Mark-BW and Adv. W17 HTP fuel rod designs and cladding material establishes the applicability of this TR for application in RSEs.	2.0
6. BAW-10186P-A, Revision 2, Extended Burnup Evaluation, June 2003	Adv. W17 HTP Mark-BW	This TR discusses the issues related to fuel system damage, fuel rod failure, fuel coolability and fuel surveillance that are applicable to both the Mark-BW and Adv. W17 HTP fuel designs. The TR supports development and validation of fuel rod thermal-mechanical and safety analysis limits used in reload safety evaluations. Since these evaluations are dependent upon the fuel rod design and model, the similarity in Mark-BW and Adv. W17 HTP fuel rod designs establishes the applicability of this TR for application in RSEs.	2.0

Table 5 (continued): Applicability of Approved Codes & Methods in TS 6.9.1.14.a

Approved Topical Report	Applicability	Remarks	ANP-2986(P) Rev. 3 Supporting Section
7. EMF-2103P-A, Revision 0, Realistic Large Break LOCA Methodology for Pressurized Water Reactors, April 2003	Adv. W17 HTP Mark-BW	The RLBLOCA TR is applicable to W and CE 2x4 plants and the fuel designs utilized by these plants. The TR sample problem fuel design was an HTP assembly design. This TR was applied to Sequoyah Units 1 and 2 in ANP-2695P Rev. 0 and ANP-2655P Rev. 1 for Mark-BW fuel design and approved by the NRC. The TR models a mixed core when applicable, thus the Adv. W17 HTP and Mark-BW fuel are included in the current application of the TR. The TR has been applied to other operating plants for HTP fuel designs and approved by the NRC. This establishes the applicability of this TR to the Adv. W17 HTP fuel design.	5.0
8. BAW-10241(P)(A), Revision 1, BHTP DNB Correlation Applied with LYNXT, July 2005	Adv. W17 HTP	The Adv. W17 HTP fuel design utilizes HTP and IFM grids. The application of the BHTP CHF correlation, with the LYNXT T-H code, is in compliance with the NRC approved range of local coolant conditions, on page 3 of the SER of BAW-10241(P)(A), Revision 1, and the NRC approved range of fuel design parameters, in Table 2 of the Revision 0 SER within BAW-10241(P)(A), Revision 1. Also, refer to the supplied response to RAI Question 3.	4.0
9. BAW-10199P-A, Revision 0, The BWU Critical Heat Flux Correlations, August 1996	Adv. W17 HTP Mark-BW	The BWU-N CHF correlation, for non-mixing grids, is applicable to the HMP grid and Mark-BW grids that do not have mixing vanes. Also, refer to the supplied response to RAI Question 3.	4.0
10. BAW-10189P-A, Revision 0, CHF Testing and Analysis of the Mark-BW Fuel Assembly Design, January 1996	Mark-BW	This TR supports the Mark-BW fuel design. Since the CHF performance of the Mark-BW spacer grid and mid-span mixing grid (MSMG) designs exceeded the BWCMV performance level (in BAW-10159P-A), this higher CHF performance was incorporated into reload licensing by the use of an effective grid spacing factor in BAW-10189P-A. The use of this elevated CHF performance is reflected in the application of an –A designation to the BWCMV CHF correlation to produce BWCMV-A.	Supports Mark-BW fuel design
11. BAW-10159P-A, Revision 0, BWCMV Correlation of Critical Heat Flux in Mixing Vane Grid Fuel Assemblies, August 1990	Mark-BW	This TR supports development and validation of the original BWCMV CHF correlation for application to the Mark-BW fuel design.	Supports Mark-BW fuel design
12. BAW-10231P-A, Revision 1, COPERNIC Fuel Rod Design Computer Code, January 2004	Adv. W17 HTP Mark-BW	This TR supports development and validation of fuel rod thermal-mechanical limits used in reload safety evaluations. Since the evaluation of fuel rod thermal-mechanical parameters is primarily dependent upon the fuel rod design and model, the similarity in Mark-BW and Adv. W17 HTP fuel rod designs establishes the applicability of this TR for application in RSEs.	2.0

4.8.2 Modifications Required to Core Safety Limits for Compliance with Approved Methods

With respect to application of the approved methodologies and proper implementation of the Adv. W17 HTP fuel design, two changes to the previously submitted LAR for use of the HTP fuel design (Reference 4) are required. The first change is related to a revision of the core safety limits and the second change is related to accident analysis validation (specifically, the steam line break coincident with rod withdrawal at power). For consistency with the described changes, the Bases for Reactor Core Safety Limit 2.1 in the SQN Nuclear Plant Technical Specifications was also modified. The modified Bases are provided at the end of this section.

Core Safety Limits

Section 4.2.10 of ANP-2986(P), Revision 3, discusses the evaluation of core DNB performance and references BAW-10220P as the source of the general description of the processes used to develop or validate core safety limit (CSL) lines, to perform transient DNB analyses, and to develop DNB maximum allowable peaking (MAP) limits.

The purpose of the thermal overpower and overtemperature protection features in the reactor protection system (RPS) is to define a region of permissible thermal power, reactor coolant temperature, reactor coolant system (RCS) pressure, and axial power distribution (as determined by axial power imbalance, ΔI) within which the reactor will operate with positive margin to the centerline fuel melt (CFM) and DNB safety limit criteria. The allowable operational region is bounded by (1) the thermal overpower limit, which protects against CFM; (2) the thermal overtemperature limit, which protects against DNB and hot leg boiling within pressure bounds defined by the high and low pressurizer pressure limits; and (3) the locus of conditions at which the steam generator safety valves actuate. Based upon these constraints, the Overtemperature Delta-Temperature (OTΔT) reactor trip function is defined as a function of allowable thermal power (converted to delta-temperature, ΔT), average RCS temperature, and pressure in order to ensure acceptable DNB performance is maintained by operation within trip setpoints given in the TSs.

Section 7.3 of BAW-10220P describes how the CSL lines are generated to provide the thermal overpower and thermal overtemperature protection. When the CSL lines are plotted as ΔT versus Tavg coordinates, they form a family of lines that are the basis for a protected OTΔT trip function. As discussed in Section 6.1.3 of BAW-10220P, the system transient response simulations model the limiting trip point assumed for accident analyses; however, the nominal trip setpoints are specified in the plant TSs. The difference between the limiting trip setpoint assumed for the accident analyses (the protected OTΔT limit) and the nominal trip setpoint specified in the TSs represents the allowance for instrumentation channel and setpoint uncertainties. As discussed in Section 7.3.3 of BAW-10220P, the reactor core safety limits establish the bounds of the OTΔT trip function; therefore, the CSL limits must bound the protected OTΔT limit (that is, the CSL must not be below or to the left of the protected OTΔT limit line when viewed on a figure of ΔT versus Tavg).

To accommodate changes in the core axial power distribution during power operation and transients, a trip reset function, $f_1(\Delta I)$ is applied to the OTΔT trip limit to reduce the allowable ΔT as the core ΔI increases beyond limits specified in the Core Operating

Limits Report (COLR). The determination of the ΔI breakpoints and rate of reduction of ΔT required as core ΔI increases is determined in a cycle-specific maneuvering analysis that simulates potential three-dimensional power distributions that could occur during normal operation and anticipated operational occurrences. The values of the $f_1(\Delta I)$ trip reset function are determined by evaluation of margin to DNB maximum allowable peaking (MAP) limits, as described in Section 7.5 of BAW-10220P and in Sections 3 and 4 of approved topical report BAW-10163P-A.

[

]

The initial conditions for the DNB-limiting Anticipated Operational Occurrence (AOO) events that do not rely on the OTΔT trip for the event termination utilize a reference design peak (radial peak, axial peak, and axial peak location) to determine DNBR performance that conservatively bounds the DNBR performance of actual power distributions that may occur during normal operation. Section 4.2.5 of ANP-2986(P), Revision 3 describes the use of a hot pin design radial peak of 1.64, in combination with an axial peak and axial peak location, to ensure bounding DNBR performance.

The appropriate values of the reference design hot pin power distributions will be provided in the UFSAR markups.

As a result of these changes, the core safety limit lines shown in Figure 2.1-1 of the SQN Units 1 and 2 Technical Specifications will be modified to represent the protected OTΔT trip limit lines and are shown in Figure 4 below.

In addition, Figure 4-1 of ANP-2986(P) Revision 3 is replaced by Figure 5, below, which provides a comparison between the CSL lines consistent with the protected OTΔT limit lines, with a flow rate of 378,400 gpm, and the current existing Sequoyah TS CSL lines. It is noted that the pressures are not identical for the comparison between the Adv. W17 HTP line (at 1985 psia) and the Sequoyah TS CSL line (at 2000 psia).

Similarly, Figure 1 of the LAR Enclosure should be replaced with Figure 5 below (it should be noted that Figure 1 of the Enclosure is mislabeled, as it refers to Current

Mark-BW versus Proposed Adv. W17 HTP Fuel; the figure is actually a comparison of the existing Technical Specification Figure 2.1-1 to the Proposed Adv. W17 HTP Fuel and does not present the Mark-BW-specific core safety limit lines).

The revised CSL lines generated for the Adv. W17 HTP fuel design at the higher reactor coolant flow rate were evaluated against the Mark-BW CSL lines at the current reactor coolant flow rate, and it was shown that the CSL lines shown in Figure 4 are applicable to both fuel designs resident in the SQN Units 1 and 2 cores. Therefore, Figure 4 is applicable for use in the SQN Units 1 and 2 Technical Specifications.

Accident Analysis Verification

Accident simulations are performed to verify that RPS response to the time dependence of events results in acceptable margins to CFM and DNB criteria and to verify the adequacy of the steady-state assumptions used to derive the values of the RPS trip equation coefficients. The licensing basis events that require credit for the OTΔT or OPΔT trip function are specified in the UFSAR; each event is evaluated to validate that the consequences of the event are acceptable and meet acceptance criteria. The impact of the system transient response on DNB is found by application of an approved LYNXT model.

The steam line break with coincident rod withdrawal at power is the limiting event. Section 5.2.2.20 of ANP-2986(P), Revision 3 describes the evaluation of this event. The event description, key parameters, and acceptance criteria specified in Section 5.2.2.20 are correct. However, implementation of the Adv. W17 HTP fuel design was found to reduce the margin for this event. The DNBR response will be calculated for the cycle-specific reload and compared with the results in the analysis of record. If the DNB performance is not bounded by the analysis of record, then the transient will be re-analyzed or re-evaluated on a cycle-specific basis using the approved methods as part of the reload safety evaluation, and the results will be reported in the Safety Analysis Report for that cycle.

Figure 4: Updated Core Safety Limit Lines for Sequoyah Technical Specifications

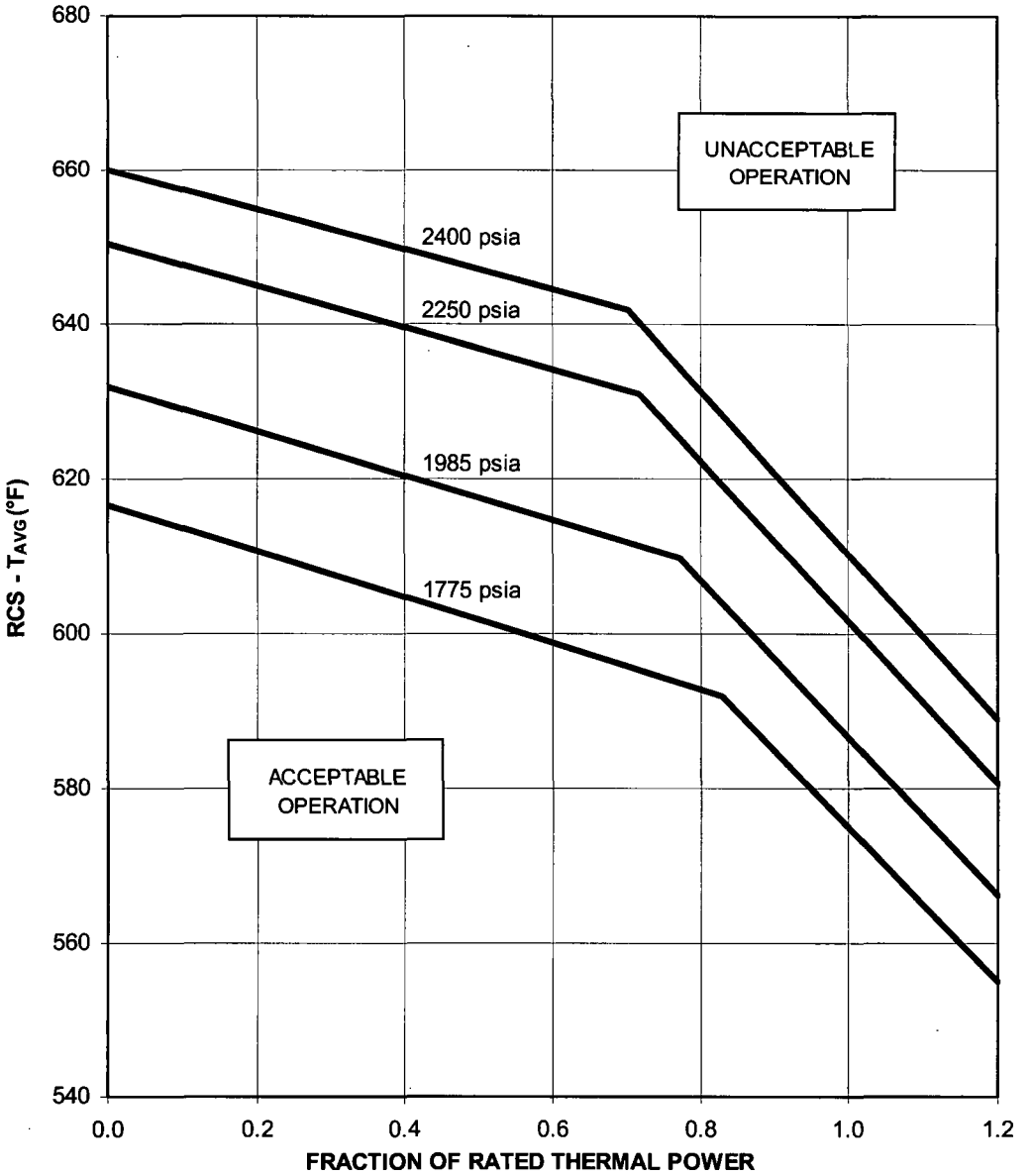
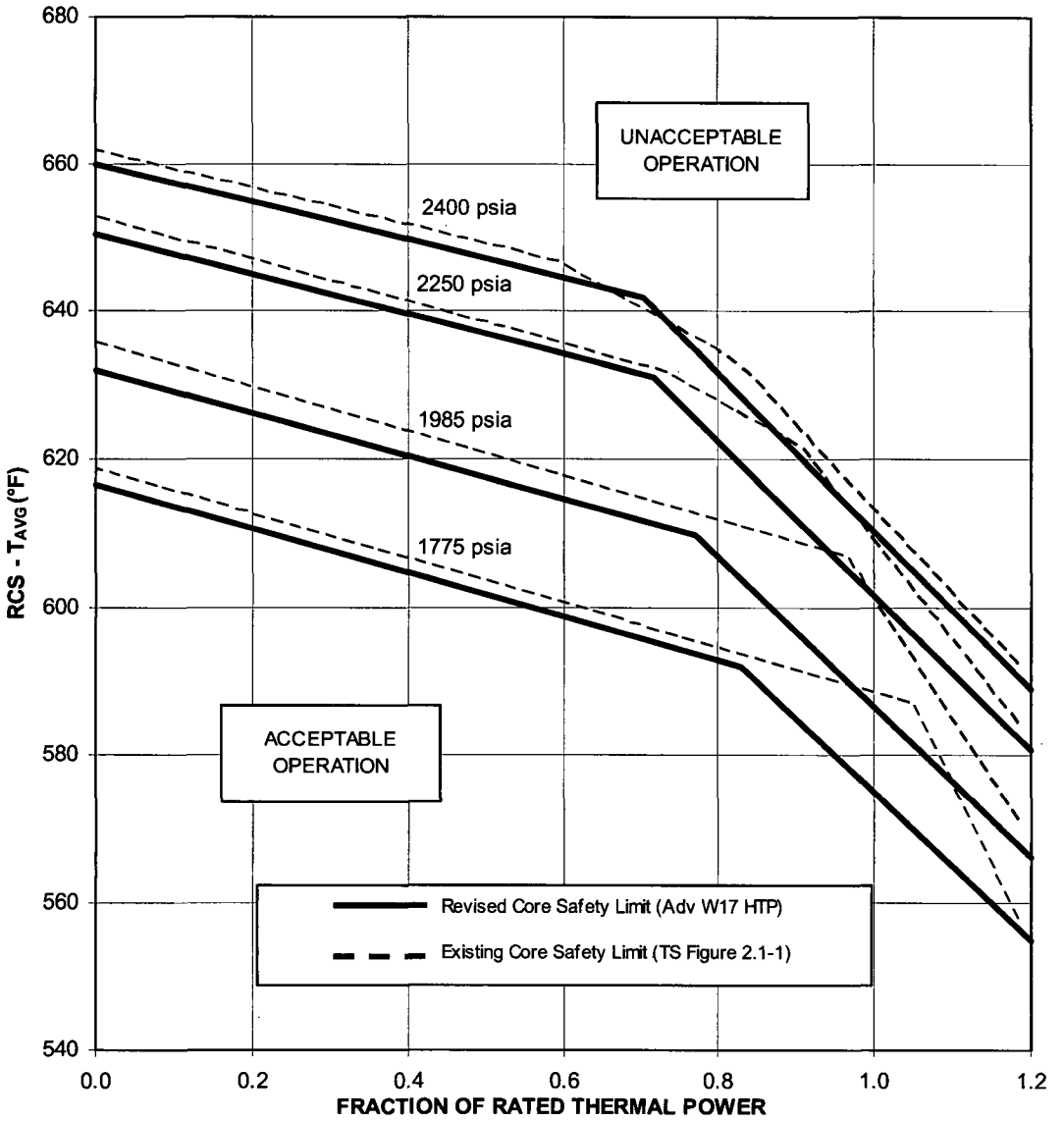


Figure 2.1-1 Reactor Core Safety Limit - Four Loops in Operation

Figure 5: Sequoyah Core Safety Limit Comparison for Advanced W17 HTP Transition



***Note**: This figure replaces Figure 4-1 of ANP-2986(P), Revision 3*
***Note**: This figure replaces Figure 1 of the LAR Enclosure*

4.8.3 TS Bases 2.1.1 (Revised)

The Bases for the Reactor Core Safety Limits specified in Section 2.1.1 of the Sequoyah Nuclear Plant Technical Specifications is modified as provided in this section. These modifications are being implemented to describe the changes in the Core Safety Limits, which more closely conform to the Improved Standard Technical Specifications content and to remove unnecessary detail and duplication within the Bases for Section 2.1.1. The presentation of the Bases for Section 2.1.1 is being changed; however, the underlying DNBR and fuel centerline temperature safety criteria have not changed. The Bases provided below clarify the AREVA methodology for assuring that plant operating conditions remain within the established safety limits.

Each Unit at SQN has individual Technical Specifications, and the text provided in this section applies to both Units. The modifications to Section 2.1.1 of the Bases maintain consistency between both Units and eliminate the need to amend the Unit 1 Technical Specification Bases when the proposed Adv. W17 HTP fuel is implemented on Unit 1.

The previous Bases discussed an increase in the allowable enthalpy rise hot channel factor, or reference design radial peak $F^N_{\Delta H}$, for operation at thermal powers less than rated thermal power for the Unit. The response to RAI Question 8 notes that a slightly different reference design radial peak relationship versus core power is being determined. The numeric expression for the relationship for reduced power flexibility is unavailable at the time of this response to the RAI; therefore, the numeric expression and appropriate values will be provided in the UFSAR markups.

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding (due to departure from nucleate boiling) and overheating of the fuel pellet (centerline fuel melt), either of which could result in cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs.

Operation above the upper boundary of the nucleate boiling regime could result in excessive temperature because of the onset of departure from nucleate boiling (DNB) and the corresponding significant reduction in heat transfer coefficient from the outer surface of the cladding to the reactor coolant water. Inside the steam film, high cladding temperatures are reached, and a cladding water (zirconium water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding to a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. The DNB correlations have been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local

DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is that there must be at least a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the design DNBR limit.

To meet the DNB Design Basis, a statistical core design (SCD) process has been used to develop an appropriate statistical DNBR design limit. Uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability at a 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. This DNBR uncertainty derived from the SCD analysis, combined with the applicable DNB critical heat flux correlation limit, establishes the statistical DNBR design limit which must be met in the plant safety analysis using values of input parameters without adjustment for uncertainty.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the safety analysis DNBR limit, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid. These lines are bounding for all fuel types. The curves in Figure 2.1-1 are based upon enthalpy rise hot channel factors that result in acceptable DNBR performance of each fuel type. Acceptable DNBR performance is assured by operation within the DNB-based Limiting Safety System Settings (RPS trip limits). The plant trip setpoints are verified to be less than the limits defined by the safety limit lines in Figure 2.1-1 converted from power to delta-temperature and adjusted for uncertainty.

Operation above the maximum local linear heat generation rate for fuel melting could result in excessive fuel pellet temperature and cause melting of the fuel at its centerline. Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant. The melting point of uranium dioxide varies slightly with burnup. As uranium is depleted and fission products are produced, the net effect is a decrease in the melting point.

Fuel centerline temperature is not a directly measurable parameter during operation. The maximum local fuel pin centerline temperature is maintained by limiting the local linear heat generation rate in the fuel. The local linear heat generation rate in the fuel is limited so that the maximum fuel centerline temperature will not exceed the acceptance criteria in the safety analysis.

The limiting heat flux conditions for DNB are higher than those calculated for the range of all control rods from fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the $f_1(\Delta I)$ function of the Overtemperature Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_1(\Delta I)$ trip reset function, the

Overtemperature Delta Temperature trip setpoint is reduced by the values in the CORE OPERATING LIMITS REPORT to provide protection required by the core safety limits.

Similarly, the limiting linear heat generation rate conditions for CFM are higher than those calculated for the range of all control rods from fully withdrawn to the maximum allowable control rod insertion assuming the axial power imbalance or Delta-I (ΔI) is within the limits of the $f_2(\Delta I)$ function of the Overpower-Delta Temperature trip. When the axial power imbalance exceeds the tolerance (or deadband) of the $f_2(\Delta I)$ trip reset function, the Overpower-Delta Temperature trip setpoint is reduced by the values specified in the CORE OPERATING LIMITS REPORT to provide protection required by the core safety limits.

4.9 *RAI Question 9*

Please clarify that the analytical methods used to determine the reactor coolant pressure [sic] (RCS) pressure and temperature limits listed in TS 6.9.1.15.a is still applicable to AREVA Advanced W17 HTP fuel design.

4.9.1 TVA Response – Question 9

The analytical methods used to determine the RCS pressure and temperature limits are as described in the following documents:

Units 1 & 2:

1. Westinghouse Topical Report WCAP-14040-NP-A, "Methodology used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

2. Westinghouse Topical Report WCAP-15984, "Reactor Vessel Closure Head/Vessel Flange Requirements Evaluation for Sequoyah Units 1 and 2."

Unit 1 only:

3. Westinghouse Topical Report WCAP-15293, "Sequoyah Unit 1 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation."

Unit 2 only:

4. Westinghouse Topical Report WCAP-15321, "Sequoyah Unit 2 Heatup and Cooldown Limit Curves for Normal Operation and PTLR Support Documentation."

These analytical methods have been previously reviewed and approved by the NRC and applied to SQN for cores utilizing Mark-BW fuel assemblies.

AREVA report ANP-2986(P) (Attachment 5 of the license amendment request), establishes that the Adv. W17 HTP fuel design is mechanically compatible with the SQN core internals, handling equipment, storage racks, and the co-resident Mark-BW fuel. Thus, the sole fuel-related impact on the RCS pressure and temperature limits is tied to the radiation induced embrittlement of the reactor vessel and internals. This radiation induced embrittlement is directly correlated with the fast neutron fluence to which the vessel is exposed.

Fast neutron fluence is a function of the fuel assembly design, as well as the cycle-specific core loading and operational profile. Following the implementation of Adv. W17 HTP fuel, SQN will continue to utilize low leakage core loading patterns. Other operating parameters, such as core thermal power, core flow, and RCS temperature remain unchanged. Therefore, the impact on the fluence analyses will be tied only to the fuel assembly design.

The main aspects of fuel assembly design that could impact the fast fluence analyses are the lattice geometry and fuel rod design. ANP-2986(P), provides a comparison of the Mark-BW and Adv. W17 HTP fuel designs. Table 2-1 of ANP-2986(P) shows that the lattice geometry of the Adv. W17 HTP fuel design is essentially unchanged from that of the Mark-BW fuel design. The number of fuel rods and guide tubes is unchanged in the HTP fuel design, and the fuel rod pitch is identical to that of the Mark-BW fuel design. Thus, the lattice geometry of the HTP fuel design will not impact the applicability of the referenced methods.

Per Table 2-2 of ANP-2986(P), the fuel rod design for Adv. W17 HTP fuel is the same as that for Mark-BW fuel. The fuel rod length, fuel column length, plenum volume and spring locations, fill gas type and pressure, clad material, clad thickness and diameter, clad-to-pellet gap, and fuel pellet diameter are all unchanged. Therefore, the HTP fuel rod design will have no impact on the neutronic performance of the fuel, nor will it impact the applicability of the referenced methods.

As previously described, the other changes to the fuel assembly design (e.g., grid design) will not impact on the neutronic performance of the fuel; and therefore, will not impact the vessel fluence calculations or the applicability of the referenced methods.

4.10 *RAI Question 10*

Please describe in details [sic] the application of both proposed TS 6.9.1.14.a.4, "PWR [pressurized water reactor] Small Break LOCA [loss-of-coolant accident] Evaluation Model, March 2001," and TS 6.9.1.14.a.7, "Realistic Large Break LOCA Methodology for Pressurized Water Reactors, April 2003," to the proposed fuel transition analyses.

4.10.1 AREVA Response – Question 10

Summaries of the LOCA applications (AREVA documents ANP-2971 Revision 1 and ANP-2970 Revision 0) were attached to the original TS modification request (Reference 4). The summaries describe in detail analyses performed in support of the proposed HTP fuel implementation.

4.11 *RAI Question 11*

The references of Mark-BW fuel design leading to the Advanced W17 HTP fuel design in Sections 1 and 2 of ANP-2986(P), Rev. 2 are unclear. Please provide a complete list of references and/or the NRC staff safety evaluations showing how the Mark-BW fuel design evolves to the Advanced W17 HTP fuel design.

4.11.1 AREVA Response – Question 11

AREVA NP was formed as a joint venture of Framatome and Siemens. Both companies had US subsidiaries with competing reload fuel designs approved by the NRC for use at W 3 & 4 loop reactors. Additionally, both had independent NRC approved reload

methods. To maintain their then current licensing basis, customers continued to use the fuel product and reload methods in place prior to the joint venture. A brief discussion of these fuel designs is presented below followed by a description of the design change process being used for the Advanced W17 HTP implementation at SQN.

The Mark-BW fuel assembly, supplied by Framatome, has been in service at McGuire Units 1 and 2, Catawba Units 1 & 2, Trojan, and Sequoyah Units 1 & 2. The Mark-BW Design Topical Report is BAW-10172P-A. Improvements to the Mark-BW were incorporated by Framatome in the Advanced Mark-BW design which has been in service at North Anna Units 1 & 2. The Advanced Mark-BW Design TR is BAW-10239P-A. M5® fuel rod cladding and structural materials, approved in BAW-10227P-A, have been implemented in both the Mark-BW and Advanced Mark-BW designs.

The W17 HTP fuel assembly, supplied by Siemens, has been in service at the Harris Nuclear Plant. The Design TR for the W17 HTP is EMF-93-074(P)(A). The mechanical design criteria are from EMF-92-116(P)(A) which also describes the Siemens design change process. After the joint venture, M5® clad and structural materials were incorporated into the Siemens methods in BAW-10240P-A.

As part of the NRC review and approval of the Advanced Mark-BW Design TR, AREVA NP proposed to incorporate the Siemens design change process, from EMF-92-116(P)(A), into future design changes associated with the Advanced Mark-BW design. This process was approved in BAW-10239P-A and allows AREVA NP to make changes to the Advanced Mark-BW fuel design without seeking prior NRC review and approval. As noted in BAW-10239P-A, licensees would continue to perform reviews under the requirements of 10 CFR 50.59 and submit a LAR if necessary.

The design features of the Advanced W17 HTP are compared in the following table to those of the current SQN fuel design (Mark-BW) and the Advanced Mark-BW. Three categories of permissible design changes described in BAW-10239P-A are identified in the table – fuel assembly length change to accommodate reactor specific dimensions, substitution of components that have been separately approved by the NRC, and the first use of an assembly design feature previously irradiated in conjunction with one lattice (i.e. 14x14) in a different lattice (i.e. 17x17). These design change categories correspond with Notes 1, 2, and 3, respectively, in Table 6 below.

Table 6: Design Features and Comparison of Advanced W17 HTP, Mark-BW, and Advanced Mark-BW Fuel Designs

Table 6 (continued): Design Features and Comparison of Advanced W17 HTP, Mark-BW, and Advanced Mark-BW Fuel Designs

4.12 *RAI Question 12*

In responding to the NRC staffs request for additional information, the licensee stated (ADAMS Accession No. ML 113200023, page 11):

> If fuel rod burnups were to exceed 54 GWd/mtU [gigawatt-days/metric ton of uranium] and any pins exceed the LHGR [linear heat generation rate] of 6.3 kW/ft [kilowatts per foot], then the gap release fraction for the non-LOCA events would be conservatively doubled or evaluated using the ANSI/ANS-5.4 [American National Standards Institute/American Nuclear Society] methodology based on the maximum burnup.

Please provide detailed description in (a) the criteria of selecting either analytical methodology in determining the gap release fraction, and (b) how the ANSI/ANS-5.4 methodology will be evaluated.

4.12.1 AREVA Response – Question 12

(a) A detailed description of the criteria for selecting the gap fractions used for AREVA Advance W17 HTP fuel begins with the pin census. Once a pin census has identified that there are pins that will exceed Footnote 11 of Table 3 of the RG 1.183 criteria for a given cycle, then the conservative "doubling" factor method is applied to the gap fractions for non-LOCA accidents specified in RG 1.183, Table 3. The factors that would be applied are shown in Table 7 below.

Table 7: Gap Fractions for Non-LOCA Accidents Exceeding Footnote 11 Criteria

Radionuclide	Adjustment Factor	Adjusted Gap Fraction
I-131	2	16%
I-133	3 [(a)]	15%
Other Halogens	2	10%
Kr-85	2	20%
Xe-133m & Xe-133	3 [(a)]	15%
Other Noble Gases	2	10%
Alkalis Metals (Cs, Rb)	2	24%

(a) These isotopes are precursors and would be conservatively tripled.

For non-LOCA events which experience fuel damage (e.g., fuel handling accident), increasing the RG 1.183 fuel/clad gap fission product inventory by a factor of two is conservative because on a core wide basis, only a small fraction of the fuel rods exceed the applicability criteria specified in footnote 11 of RG 1.183 Table 3. This would be the primary method to address this issue.

Doubling non-LOCA gap fractions has been conservatively applied by other licensees (e.g., Calvert Cliffs' amendment to implement Alternative Source Term (Accession # ML072210207) and Fort Calhoun (Accession # ML013030027)). These licensees calculated gap fractions using ANSI/ANS-5.4-1982 methodology and showed that

doubling the gap release fractions in RG 1.183, Table 3 was bounding. Therefore, precedence exists for increasing the gap fractions by a factor of two in the event the footnote 11 of RG 1.183 Table 3 criteria are exceeded on cycle specific basis. The doubling factor would be applied in lieu of calculating the gap fractions using the ANSI/ANS-5.4-1982 methodology.

If dose consequences from the application of the doubling factor method described above for increasing gap fractions (Table 12-1) exceed dose criteria, then the ANSI/ANS-5.4 based method for determining Gap Fractions for Non-LOCA accidents with fuel damage will be utilized.

(b) Application of an industry applied and NRC reviewed methodology (e.g., Calvert Cliffs amendment to implement AST, ADAMS Accession # ML072210207)) presented in ANSI/ANS-5.4-1982 for high and low temperature releases will be used to calculate the gap release fractions for noble gases and halogens. The application of the ANSI/ANS-5.4 based methodology for determining Non-LOCA gap release fractions will consist of four steps. First, determine a bounding pin power history based on cycle specific parameters. Second, determine the total gap composition, using a fuel performance code for the fuel. Third, use the bounding pin and total gap release to calculate the isotopic gap release fractions using the ANSI/ANS-5.4 equations. Fourth, compare the calculated isotopic gap release fractions to non-LOCA gap release fractions in Table 3 of RG 1.183. The higher of the gap release fractions would then be used for the fuel handling accident dose consequence analysis.

The first step in the method outlined above is to develop a bounding pin power envelope based on end-of cycle (EOC) conditions for two cycle operation of a given fuel assembly. Note that temperature increases with increasing burnup; however, pin power decreases radially in the core. The bounding pin power history for specific fuel loads can be obtained from fuel performance codes such as TACO3 for Zr-4 clad fuel or COPERNIC for Adv. W17 HTP fuel. A bounding pin power history would be developed based on sensitivity cases such as (1) holding the radial peaking factor constant at [ ] and (2) allowing the radial peaking decrease (radial falloff curves) with increasing burnup. These cases would be evaluated with the application of a conservative axial power factor (flatter axial power distribution (low value Fz) resulting in higher fission gas release along fuel stack).

The second step in the method outlined above is to determine the total gap composition, using a fuel performance code for the fuel. The fuel performance code used for AREVA Advance W17 HTP fuel in SQN is COPERNIC. COPERNIC is used to generate internal rod pressures based on the starting backfill pressure of the fill gas and the release of noble gases during irradiation of fuel up to rod-average burnups of [ ] as a function of burnup and temperature. COPERNIC utilizes pin power envelopes, burnup values, and temperature profiles to model the diffusion of fission product gases from the fuel matrix to the gap due to increasing saturation of the grain boundaries due to thermal and athermal contributions. The fission gases are used to calculate rod internal pressure based on increasing burnup and temperature profiles. The fuel rod internal pressures are for safety related functions and are extensively benchmarked. COPERNIC must accurately determine the fission gas release in order to accurately calculate the internal pressure as a function of

burnup and temperature. Therefore, the application of the NRC approved COPERNIC computer code to project the fission product gap release fractions is appropriate based on the SER restrictions for the COPERNIC fuel performance code (BAW-10231P-A).

The third step in the method outlined above is to determine the isotopic gap release fractions using ANSI/ANS-5.4 equations. The noble gas gap release fractions from COPERNIC would be used to develop a limiting gas gap fraction for long lived and short lived radionuclides based on ANSI/ANS-5.4 methodology. The noble gas gap release fractions obtained from COPERNIC and diffusion rate ratios specified in Section 4.1 of ANSI/ANS-5.4-1982 would be used in the ANSI/ANS-5.4-1982 equations in Sections 3.1 to calculate the gap fractions for other nuclides. The ANSI/ANS-5.4-1982 methodology for calculating gap release fractions is detailed below:

1. Low Temperature and long-lived calculations:

   ANSI/ANS-5.4-1982 methodology calculates the cumulative release (F) for long-lived nuclides (i.e., half-lives greater 1 year) as a function of burnup for low temperature calculations:

   $F = 7E-08 * BU$ (Equation 6, ANSI/ANS-5.4-1982)

   Where:

   BU = rod averaged accumulated burnup (MWd/mtU)

2. Low Temperature and short-lived calculations:

   Next cumulative release (F) short-lived nuclides (i.e., half-lives less than 1 year) for low temperatures is calculated as follows:

   $F = \left(\frac{1}{\lambda}\right) * \left[1E-7 * \sqrt{\lambda} + 1.6E-12 * P\right]$ (Equation 7, ANSI/ANS-5.4-1982)

   Where:

   P = Specific Power (MW/mtU)

   P is conservatively assumed to be maximum power level during last two half-lives of operation.

   λ = decay constant (1/sec)

   $\lambda = \ln 2 / \tau_{1/2}$

   $\tau_{1/2} = half - life$ (sec)

3. High Temperature and long-lived nuclide calculations:

For long-lived nuclides (excluding noble gases such as Kr-85), the gap release fraction for these nuclides will be calculated using the equations and the diffusion rate ratios specified in Section 3.1.1 and Section 4.1 of ANSI/ANS-5.4-1982, respectively. From Section 3.1.1 the fractional release (F at the end of the burnup increment "k" is calculated as follows:

$$F = 1 - g(\tau) \qquad \text{(Equation 1, ANSI/ANS-5.4-1982)}$$

$$F_k = 1 - \frac{\left\{ \sum_{i=1}^{k-1} \left[\frac{B_i(\tau_{i+1} g_{i+1})}{D'_i} \right] + B_k \Delta t_k g_k \right\}}{\sum_{i=1}^{k} B_i \Delta t_i} \qquad \text{(Equation 2)}$$

Where:

$$gi = g(\tau(i) = 1 - 4 * \sqrt{\frac{\tau(i)}{\pi}} + 1.5 * \tau(i)$$

B_i = Fission product production rate (birth rate) during the ith step

Δt_i = Length of the ith time step (sec)

$$\tau_1 = \sum_{i=1}^{k} D'_i \, \Delta t_i, \quad \tau_2 = \sum_{i=2}^{k} D'_i \, \Delta t_i, \ldots, \tau_k = D'_k \, \Delta t_k$$

$$g_i = g(\tau_i) = 1 - 4\sqrt{\frac{\tau_i}{\pi}} + \frac{3\tau_i}{2}; for\ \tau \bar{<} 0.1$$

$$g_i = g(\tau_i) = \frac{1}{15\tau_i} - \frac{6}{\tau_i} \sum_{n=1}^{3} \frac{\exp\left(-n^2 \pi^2 \tau_i\right)}{n^4 \pi^4}; for\ \tau > 0.1$$

$$D'_i = f_{isotope} * \frac{D_o}{a^2} * \exp\left(\frac{-Q}{R\,T_{fi})}\right) * 100^{(BU_i / 280000)}$$

Where:

R	=	*1.987 cal/mol-K* (ANSI/ANS-5.4-1982)
Q	=	*72300 cal/mol* (ANSI/ANS-5.4-1982)

D_o/a^2 = 0.61 /sec (ANSI/ANS-5.4-1982)

$$\frac{D'noble}{D'noble} = 1 = f_{tellurium}$$

$$\frac{D'iodine}{D'noble} = 7 = f_{iodine}$$

$$\frac{D'cesium}{D'noble} = 2 = f_{cesium}$$

$$\frac{D'tellurium}{D'noble} = 30 = f_{tellurium}$$

BU_i = *Burnup (MWd/mtU)* for i^{th} step (sec)

T_{fi} = Fuel temperature in degrees Kelvin from COPERNIC

p_f = Radial peaking factor from pin census

a_f = Axial power peaking factor (low Fz assumed)

t(i,k) = *EFPD(i,k) * 24 * 3600 = Effective Full Power Seconds*

$$EFPD(i,k) = EFPD(i,k-1) + \frac{(BU(i,k) - BU(i,k-1))}{P} = Effective\ Full\ Power\ Days$$

$$P = Specific\ Power(MW / mtU)$$

Where:

$$P = pf * af * \frac{Core\ Power(MWth)}{\#Fuel\ Assemblies / core} \div M_p * 1E+6$$

$$Mp = Dp * Vp * \frac{U - 238\ Mass}{UO2\ Mass}$$

Where:

R_p = fuel rod radius (cm)

A_p = fuel rod height (cm)

V_p = fuel rod volume (cc)

D_p = UO2 fuel rod density = UO2 density * TD (theoretical density)

$$\tau(i,k) = \tau(i,k,-1) + D'(i,k) * \left(t(i,k) - t(i,k-1)\right)$$

The gap fractions would be calculated for several radial positions (i) for each burnup step (k) and volumetrically weighted to obtain an overall gas gap fraction.

4. High Temperature and short-lived calculations:

Likewise, for high temperature calculations for short-lived nuclides (excluding noble gases), the cumulative fractional release F(i,j) is calculated at the end of burnup increment (k) at radial position (i) at constant temperature and power as follows based on equations from Section 3.1.2 of ANSI/ANS-5.4-1982:

$$F = \frac{3}{1-\exp(-\mu\tau)} - \left[\frac{1}{\sqrt{\mu}}\left[erf\left(\sqrt{\mu\tau}\right) - 2\sqrt{\frac{\mu\tau}{\pi}}\exp(-\mu\tau)\right] - \frac{1-(1+\mu\tau)\exp(-\mu\tau)}{\mu}\right]; \tau < 0.1 \quad \textit{Equation } 3$$

$$F = 3\left[\frac{1}{\sqrt{\mu}}\coth\left(\sqrt{\mu}\right) - \frac{1}{\mu}\right] - \frac{6\mu}{\exp(\mu\tau)-1} \times \sum_{n=1}^{3}\left[\frac{1-\exp\left(-n^2\pi^2\tau\right)}{n^2\pi^2\left(n^2\pi^2+\mu\right)}\right]; \tau > 0.1 \quad \textit{Equation } 4$$

Where:

$$\mu = \frac{\lambda}{D'}$$

$$\tau = D't$$

λ = decay constant (sec^{-1})

t = time (sec) during constant temperature and constant power irradiation time

$$\frac{D'noble}{D'noble} = 1 = f_{tellurium}$$

$$\frac{D'_{iodine}}{D'_{noble}} = 7 = f_{iodine}$$

$$\frac{D'_{cesium}}{D'_{noble}} = 2 \quad = f_{cesium}$$

$$\frac{D'_{tellurium}}{D'_{noble}} = 30 = f_{tellurium}$$

$$D'_i = f_{isotope} * \frac{D_o}{a^2} * \exp\left(\frac{-Q}{R\,T_{fi})}\right) * 100^{(BU_i / 280000)}$$

Where:

R = *1.987 cal/mol-K*

Q = *72300 cal/mol*

D_o/a_2 = *0.61 /sec*

λ = Decay Constant (1/sec)

BU_i= Burnup (MWd/mtU) (total accumulated burnup)

T_{fi} = Fuel temperature in degrees Kelvin from COPERNIC

$$P = Specific\ Power(MW / mtU)$$

Where:

$$P = p_f * a_f * \frac{Core\ Power(MWth)}{\#Fuel\ Assemblies / core} \div M_p * 1E+6$$

p_f = Radial peaking factor from pin census

a_f = Axial power peaking factor (low Fz assumed)

t(i,k) = *EFPD(i,k) * 24 * 3600 = Effective Full Power Days*

Where:

$$EFPD(i,k) = EFPD(i,k-1) + \frac{(BU(i,k) - BU(i,k-1))}{P} = Effective\ Full\ Power\ Days$$

$$P = Specific\ Power(MW / mtU)$$

$$Where:$$

$$P = p_f * a_f * \frac{Core\ Power(MWth)}{\#Fuel\ Assemblies / core} \div M_p * 1E+6$$

$$Mp = Dp * Vp * \frac{U-238\ Mass}{UO2\ Mass}$$

Where:

R_p = fuel rod radius (cm)

A_p = fuel rod height (cm)

V_p = fuel rod volume (cc)

D_p = UO2 fuel rod density = UO2 density * TD (theoretical density)

$$\tau(i,k) = \tau(i,k,-1) + D'(i,k) * (t(i,k) - t(i,k-1))$$

$$g(i,k) = 1 - 4 * \sqrt{\frac{\tau(i,k)}{\pi}} + 1.5 * \tau(i,k)$$

In accordance with ANSI/ANS-5.4-1982 requirements, the gap fractions would be calculated for six or more radial nodes (j), ten or more axial nodes unless otherwise justified and the irradiation period will be divided into a series of each burnup steps (k) not to exceed 2000 MWd/mtU. The axial and radial temperature profiles will be taken from COPERNIC fuel performance code. From this, the gap fractions will volumetrically weighted to obtain an overall gas gap fraction.

The fourth step in the method outlined above is the comparison of the calculated isotopic gap fractions. The isotopic gap fractions determined using ANSI/ANS-5.4 methodology would be compared to the isotopic gap fractions in Table 3 of RG 1.183. The higher of the isotopic gap fractions would be applied to all the fuel for which fuel damage is

postulated due to non-LOCA steady state design basis accidents, excluding reactivity insertion accidents such as the control rod ejection accident.

It is recognized that a draft Revision 1 of RG 1.183 (DG-1199) is currently being evaluated. When RG 1.183, Revision 1, is approved and adopted, it will be considered for replacing the two methods described in (a) and (b) above for determining gap fractions.

4.13 *RAI Question 13*

Is the moderator density reactivity curve in Table 3-3 based on the most positive moderator temperature coefficient (MTC)? If not, please show the effect of the use of the most positive MTC for moderator density feedback. Please explain.

4.13.1 AREVA Response – Question 13

Table 3-3 of the Summary Report (ANP-2971(P) Revision 1) provides the moderator density reactivity table which was used in the SBLOCA analysis. The reactivity defects are biased to be representative of a core with a HFP MTC that is at the Technical Specification (TS) limit. SQN TS 3.1.1.3 limit the moderator temperature coefficient to values less than 0 Δk/k/°F. The moderator reactivity is based on BOC conditions because it is the least negative reactivity.

4.14 *RAI Question 14*

Please explain what is meant by the 166 seconds (sec) trip time for reactor coolant pumps (RCPs). Is this the delay time to RCP trip once the pressure set point for tripping RCPs has been reached? Please explain.

4.14.1 AREVA Response – Question 14

The standard SBLOCA break spectrum analyses are performed with the assumption that the Reactor Coolant Pumps (RCPs) trip on a reactor trip. TVA Emergency Operating Procedures require that the RCPs be tripped when either:

- primary system pressure is less than 1250 psig and at least one charging or one safety injection pump is running, or
- a Phase B isolation signal has been generated on high-high containment pressure (i.e. containment pressure ≥ 2.9 psig).

The high-high containment pressure setpoint is conservatively neglected and the primary pressure is used to initiate the action to trip the RCPs. TVA indicated that it is reasonable to expect the RCPs to be tripped within 2 minutes of the RCP EOP trip condition. The primary pressure setpoint is degraded for uncertainty to extend the RCP trip time. Therefore, the time of 166 seconds is equal to 46 seconds, the time at which the pressure is below the setpoint and one charging pump has started, plus two minutes.

4.15 *RAI Question 15*

Does the high-pressure safety injection (HPSI) curves include allowance for pressure and flow measurement from the surveillance requirement on HPSI flow testing? Please explain.

4.15.1 TVA Response – Question 15

Yes. The surveillance requirement (SR) on HPSI flow testing is set well above the HPSI flow performance used in the SBLOCA analyses so there is an allowance that can be used to account for HPSI pump pressure and flow measurement errors. Should the HPSI pumps be performing at the minimum SR acceptance criteria and all measurement and test equipment be at their maximum adverse error, the HPSI injection curves used in the SBLOCA analysis would still be bounded by actual pump / system performance.

For example, the centrifugal charging pump 2A (CCP-2A) SR flow testing requires pump differential pressure to be greater than 2234.9 psid at a flow rate of 138 gpm. The instrumentation used to measure the CCP flow rate has a maximum error of 6 gpm, so CCP flow could be as low as 132 gpm. The instrumentation used to measure the CCP differential pressure has a maximum error of 39.3 psid, so the measured CCP differential pressure could be as low as: 2234.9 – 39.3 = 2195.6 psid at the SR flow testing acceptance criteria. The CCP differential pressure credited in the SBLOCA analysis at 132 gpm is 2129.2 psid, which is 2195.6 - 2129.2 = 66.4 psid less than the minimum acceptable CCP differential pressure from the SR flow testing. This means that the CCP, operating at the minimum performance allowed by the SR, will deliver more flow at a higher discharge pressure than is credited in the SBLOCA analysis even if all the instrumentation used in the flow test has the maximum allowable error.

The results for the safety injection pump (SIP) SR flow testing are similar. For example, for SIP-2A, SR flow testing requires pump differential pressure to be greater than 1438.8 psid at a flow rate of 26.3 gpm. The instrumentation used to measure the SIP flow rate has a maximum error of 0.9 gpm, so SIP flow could be as low as 25.4 gpm. The instrumentation used to measure the SIP differential pressure has a maximum error of 22.6 psid, so the measured SIP differential pressure could be as low as: 1438.8 – 22.6 = 1416.2 psid at the SR flow testing acceptance criteria. The SIP differential pressure credited in the SBLOCA analysis at 25.4 gpm is 1363.5 psid, which is 1416.2 - 1363.5 = 52.6 psid less than the minimum acceptable SIP differential pressure from the SR flow testing. This means that the SIP, operating at the minimum performance allowed by the SR, will deliver more flow at a higher discharge pressure than is credited in the SBLOCA analysis even if all the instrumentation used in the flow test has the maximum allowable error.

Table 3-2 of the SBLOCA summary report, ANP-2971P, provides the combined flow from the ECCS pumps as a function of RCS pressure. The flow rates in Table 3-2 account for pressure loss in the injection lines and flow that is diverted through the pump minimum flow lines. These flow losses reduce the injection flow rate from that predicted based solely on pump differential pressure and flow as calculated above. Therefore, the

flow rates in Table 3-2 are considerably lower than the flow rates given in the above examples.

4.16 *RAI Question 16*

Fig 3-3 shows the nodalization for the vessel with the core barrel leakage paths opened. Was this [sic] the leakage path at the top of the downcomer open during the SBLOCA analyses? Please explain. If so, please show the impact of the leakage path closed for the limiting breaks.

4.16.1 AREVA Response – Question 16

As described in Section 3.2 of ANP-2971, the spray nozzles located between the downcomer and upper head are modeled as a leakage path (Junction 007 on Figure 3-3) since these are geometric features of the SQN units. These spray nozzles are small cylinders embedded in the core barrel flange that extend vertically through the Upper Support Plate flange, just outside the radius of the holddown spring. Figure 6 below shows the location of these spray nozzles in the core barrel flange. Figure 7 below shows a side view of a single spray nozzle. [

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Figure 6: Spray Nozzle Locations, Top View

Figure 7: Spray Nozzle, Side View

4.17 *RAI Question 17*

Please justify the accumulator temperature of 105 °F in the SBLOCA analyses. Does this temperature bound the highest accumulator temperature during the cycle? Please explain.

4.17.1 AREVA Response – Question 17

The accumulator temperature of 105°F is the nominal accumulator temperature. The accumulator temperature is limited by the containment temperature, for which the TS 3.6.1.5 limit is 125°F. A sensitivity study was performed on the limiting break using 130°F to bound the TS limit with measurement uncertainty. Table 8 displays the results from that study. As can be seen, the PCT and oxidation results using the higher accumulator temperature were bounded by those with the nominal accumulator temperature.

Table 8: Summary of Results for the Accumulator Temperature Sensitivity Study (9.76 inch Break)

Accumulator Temperature (°F)	105	130
Peak Clad Temperature (°F)	1469.3	[ ]
Core Wide Oxidation (%)	0.0013	[ ]
Local Maximum Oxidation (%)	0.1659	[ ]

4.18 *RAI Question 18*

Does the model also account for residual water remaining in the horizontal section of the suction leg piping after the vertical section clears? Please explain. Please show the amount of residual water remaining in the loop seals for the 2.75-, 3.0-, and 9.76-inch diameter breaks.

4.18.1 AREVA Response – Question 18

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Figure 8: Loop 2, Loop Seal Void Fractions for the 2.75 inch Break

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Figure 9: Loop 2, Loop Seal Void Fractions for the 3.00 inch Break

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Figure 10: Loop 2, Loop Seal Void Fractions for the 9.76 inch Break

4.19 *RAI Question 19*

Please show the heat transfer coefficient from 50 to 200 sec on an expanded scale for the 9.76-inch break in Fig 4-23.

4.19.1 AREVA Response – Question 19

Figure 4-23 in ANP-2791 used an S-RELAP5 output parameter that sums several heat transfer coefficients. However, each of the summed coefficients actually uses a different temperature difference to calculate the heat transfer. The effective heat transfer coefficient is defined as the heat flux divided by the temperature difference between the cladding temperature and the saturation temperature. Figure 11 shows the effective heat transfer coefficient at the PCT location for the full transient of the 9.76 inch break. Figure 12 shows the effective heat transfer coefficient at the PCT location for the 50 - 200 second time frame of the 9.76 inch break.

Figure 11: Effective Heat Transfer Coefficient for the 9.76 inch Break

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Figure 12: Effective Heat Transfer Coefficient for the 9.76 inch Break, 50 – 200 seconds

4.20 *RAI Question 20*

Please show the sink temperature for the 9.76-inch break versus time. What causes the temperature to decrease at about 75 sec during the steam cooling phase of core uncovery for the 9.76-inch break in Fig. 4-24? Are there entrained water droplets in the upward flowing vapor? Please explain. Is liquid allowed to downflow into the hot bundle from the upper plenum during periods of steam cooling? Please explain.

4.20.1 AREVA Response – Question 20

The S-RELAP5 evaluation model does not include a coupled containment model calculation (EMF-2328(P)(A), Revision 0). Instead the sink for the break is modeled as a time-dependent volume with pure steam at atmospheric pressure, 14.7 psia, conditions.

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Figure 13: Base 9.76 Inch Break – Core Exit Mass Flow Rates

Figure 14: Base 9.76 Inch Break – Hot Assembly Axial Mass Flow Rates

Figure 15: Base 9.76 Inch Break – Hot Assembly Radial Mass Flow Rates

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Figure 16: High Radial K, 9.76 Inch Break – Hot Assembly Radial Mass Flow Rates

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Figure 17: Base and High Radial K – 9.76 Inch Break – Peak Clad Temperature Comparison

4.21 *RAI Question 21*

Please describe the junctions shown connecting the downcomer to the top of the baffle region in Fig. 3-3.

4.21.1 AREVA Response – Question 21

SQN is a baffle downflow plant by design (See Figure 18). [

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Figure 18: Downcomer to Baffle Junction

4.22 *RAI Question 22*

Please present the decay heat multiplier chosen for the breaks shown in Fig. 3-8.

4.22.1 AREVA Response – Question 22

The decay heat multiplier chosen for the limiting transient, including the breaks shown in Figure 3-8 of ANP-2970, is set to 1.0. See Table 2-1, page 2-2 of ANP-2970.

The ANS/ANSI 5.1-1979 standard for decay heat is used to calculate the nominal decay heat curve as well as its uncertainty. The use of the ANS/ANSI 5.1-1979 standard for the RLBLOCA calculation is in accordance with RG 1.157. The ANS/ANSI 5.1-1979 standard was developed using experimental decay heat test data as well as mathematical operations to extend the trends to account for time periods beyond which experimental testing could reasonably cover.

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4.23 *RAI Question 23*

Please identify the values of the parameters in the sampled breaks in Fig. 3-9. Also provide the containment pressure at the time of peak cladding temperature (PCT) for these cases. The parameters are:

Case number, PCT (°F), PCT time (sec), case end time (sec), PCT elevation (ft), assembly burnup (GWd/mtU), core power (MWt), planar linear heat generation rate, PLHGR (kW/ft), axial skew (top, bottom), axial shape index, ASI, break type (guillotine, split), one sided break size (ft^2), Tmin (°F), initial hot rod fuel stored energy (°F), decay heat multiplier, film boiling heat transfer coefficient (HTC), dispersed flow film boiling HTC (Btu/hr-ft^2-°F), condensation interphase HTC (Btu/hr-ft^2-°F), initial reactor coolant system (RCS) flow rate (M lb/hr), initial operating temperature (Tcold, °F), pressurizer pressure (psia), pressurizer level (%), containment volume (ft^3), containment temperature (°F), containment pressure at time of PCT (psia), safety injection tank (SIT) temperature (°F), SIT pressure (psia), SIT volume (ft^3), start of broken loop SIT injection (sec), start of intact loop SIT injection (sec), broken loop SIT empty time (sec), intact loop SIT empty time (sec), start of HPSI (sec), low-pressure safety injection (LPSI) available (sec).

4.23.1 AREVA Response – Question 23

The plant-specific terminology is used to refer to parameters. For example, the SIT is indicated in the table as accumulators and the LPSI is listed as RHR. A numbered list of the parameters listed above is as follows:

1. Case number
2. PCT (F)
3. PCT time (sec)
4. Case end time (sec)
5. PCT elevation (ft)
6. Assembly burnup (GWd/MTU)
7. Core power (Mwt)
8. Fq
9. Axial skew (top, bottom)
10. ASI
11. Break type (guillotine, split)
12. One sided break size (ft^2)
13. Tmin (F)
14. Initial hot rod fuel stored energy (F)
15. Decay heat multiplier

16. Film boiling HTC (Btu/hr-ft^2-F)
17. Dispersed flow film boiling HTC (Btu/hr-ft^2-F)
18. Condensation interphase HTC (Btu/hr-ft^2-F)
19. Initial RCS flow rate (M lb/hr)
20. Initial operating temperature (Tcold, F)
21. Pressurizer pressure (psia)
22. Pressurizer level %
23. Containment volume (ft^3)
24. Containment temperature (F)
25. Accumulator temperature (F)
26. Accumulator pressure psia
27. Accumulator volume (ft^3)
28. Start of broken loop Accumulator injection (sec)
29. Start of intact loop Accumulator injection (sec)
30. Broken loop Accumulator empty time (sec)
31. Intact loop Accumulator empty time (sec)
32. Start of HPSI (sec)
33. RHR available (sec)

The values of the containment pressure at the PCT time for all breaks are listed in Table 9. Table 10 shows parameters 1 through 18, Table 11 shows parameters 19 through 27, and Table 12 shows parameters 28 through 33. The tables follow.

Table 9: Containment Pressure at PCT Time

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Table 9 (continued): Containment Pressure at PCT Time

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Table 9 (continued): Containment Pressure at PCT Time

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Table 10 (continued): Additional Parameters 1 to 18

Table 10 (continued): Additional Parameters 1 to 18

Table 10 (continued): Additional Parameters 1 to 18

Table 10 (continued): Additional Parameters 1 to 18

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Table 11: Additional Parameters 19 to 27

Table 11 (continued): Additional Parameters 19 to 27

Table 11 (continued): Additional Parameters 19 to 27

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Table 11 (continued): Additional Parameters 19 to 27

Table 11 (continued): Additional Parameters 19 to 27

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Table 12: Additional Parameters 28 to 33

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Table 12 (continued): Additional Parameters 28 to 33

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Table 12 (continued): Additional Parameters 28 to 33

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Table 12 (continued): Additional Parameters 28 to 33

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Table 12 (continued): Additional Parameters 28 to 33

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4.24 *RAI Question 24*

Please show the lateral k-factors used in the downcomer model for downcomer boiling analyses. How are the k-factors computed? Please explain. What is the worst single failure for the limiting downcomer boiling case? Please explain? What is the maximum refueling water storage tank (RWST) temperature assumed for the limiting downcomer boiling case? How is condensation of emergency core cooling (ECC) in the cold legs and upper downcomer modeled and what is the sensitivity of downcomer boiling to the condensation coefficient? Please also show the downcomer fluid temperatures versus time compared to saturation for the limiting downcomer boiling case.

4.24.1 AREVA Response – Question 24

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The DC fluid temperature vs time compared to saturation for the limiting DC boiling case is shown in Figure 19.

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Figure 19: DC Fluid Temperature

4.25 *RAI Question 25*

Please provide the decay heat multipliers for the breaks in Fig. 3-9.

4.25.1 AREVA Response – Question 25

The decay heat multiplier chosen for the limiting transient, including the breaks shown in Figure 3-9 of ANP-2970, is set to [ ]. See Table 2-1, page 2-2 of ANP-2970.

Additional related information is provided in response to RAI Question 22 above.

4.26 *RAI Question 26*

Does the LBLOCA methods account for loop seal refilling? Please explain.

4.26.1 AREVA Response – Question 26

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5.0 Thermal Conductivity Degradation RAI Responses

This section presents responses to the TCD RAI received by 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 RAI are shown below:

Fuel Design – Materials and Thermal-Mechanics (FDM-AL)
Engineering and Projects – LOCA, ECCS, and BWR Analysis (PEPNL-A)

The responses begin on the following page.

5.1 *RAI Question 1*

References 1 and 2 describe the thermal conductivity degradation (TCD) issue. TCD will affect fuel performance. Recently, the staff realizes that the peak cladding temperature (PCT) calculations in ECCS evaluation model could become nonconservative due to the TCD effect. Please provide explanation or analysis to confirm that the currently existing analytical results in the following are still bounding considering the incorporation of the TCD effect using the approved fuel performance code (COPERNIC or TACO3): (1) clad strain, (2) strain fatigue, (3) power to melt (UO2 and Gd2O3), and (4) PCT.

1. Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," October 8, 2009, ADAMS Accession No. ML091550527.

2. Information Notice 2011-21, "Realistic Emergency Core Cooling System Evaluation Model Effects Resulting from Nuclear Fuel Thermal Conductivity Degradation," December 13, 2011, ADAMS Accession No. ML113430785.

5.1.1 AREVA Response – Question 1 (Parts 1, 2 and 3)

As mentioned in Section 2.4 of the Sequoyah HTP fuel transition LAR inputs document, ANP-2986(P) Revision 3, AREVA will utilize the modern NRC-approved COPERNIC fuel performance code (Reference 7), which includes degradation of fuel thermal conductivity with burnup (Section 4.3, Reference 7), in conjunction with the implementation of the HTP fuel design at Sequoyah to perform the thermal-mechanical evaluations of the Adv. W17 HTP fuel rod. Use of the COPERNIC code will include evaluations of clad strain, strain fatigue, and power to melt. Note that the cladding fatigue criterion is not a limiting criterion, and previous evaluations of this criterion have typically shown the existence of significant margins to the criterion limit.

5.1.2 AREVA Response – Question 1 (Part 4)

The LBLOCA and SBLOCA analyses were provided to the NRC as part of TVA's LAR package (Reference 4), ANP-2970P Revision 0 (Reference 8) and ANP-2971P Revision 1 (Reference 9) respectively. The LBLOCA analysis specifically addressed thermal conductivity degradation (TCD) on pages 6-1 through 6-13 in ANP-2970P Revision 0 (Reference 8). The SBLOCA analysis (Reference 9) did not specifically address TCD in the report, but AREVA's response to IN 2009-23 is applicable for the Sequoyah HTP SBLOCA analysis (Reference 10 Attachment B, see below).

"The RODEX2 code is used to determine the initial core and hot pin stored energy for small break LOCA evaluations. Small breaks evolve through a pump coastdown and natural circulation phase to a loop draining phase followed by a boil-down and refill phase. The pump coastdown phase lasts approximately 100 seconds. For most of this phase a single or two phase forced circulation exists within the RCS which prevents a cladding temperature excursion and acts to remove the initial energy of the fuel and deposit it in the steam generators or the containment. In either case the energy content of the fuel has been reduced to that required to transport decay heat out of the fuel by the end of the coastdown phase. Thus, the peak cladding temperatures, which occur later in the transient depend on decay heat versus heat transfer and have no relationship

to the initial stored energy within the fuel. This was demonstrated in a recent sensitivity study performed for the U.S. EPR. In this study the centerline fuel temperature of a reference case was raised by 600°F with a negligible impact on the PCT which occurred at an extended time. Thus, whatever the adjustments (RODEX2 Corrections) made to the initial fuel temperature there will be no significant effect on the SBLOCA cladding temperatures or the local oxidation."

6.0 Thermal-Hydraulics RAI Responses

This section presents responses to the T-H RAIs received by 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 RAI are shown below:

Fuel Design – Thermal-Hydraulics (FDT-AL)

The responses begin on the following page.

6.1 *RAI Question 1*

The June 17, 2011, submittal stated that the thermal hydraulic analysis indicates that the transition from a full core of Mark-BW fuel to a full core of Advanced W17 HTP fuel will result in a small increase in bypass flow and a small decrease in the RCS loop flow due to the higher pressure drop of the Advanced W17 HTP fuel. It also stated that the combined effect of the fuel transition and the steam generator replacement is a small net increase in RCS loop flow. Based on the reason above, TVA requested to revise the design flow rate shown in Table 2.2-1 and Figure 3.2-1. Please provide: (1) the rationale for a small net increase in RCS loop flow due to the combined effect of the fuel transition and the steam generator replacement; (2) how to obtain or to generate the data for design flow rate of 91,400 GPM in Table 2.2-1 and total RCS flow rate of 378,400 GPM at 100% Thermal Power Fraction (TPF) and 359,480 GPM at 90% TPF in Figure 3.2-1, respectively; and (3) justification for proposed 5% RCS flow rate increase.

6.1.1 AREVA Response – Question 1 (Part 1)

In order to calculate the impact of both the steam generator replacement and the transition to Advanced W17 HTP fuel on the reactor coolant system (RCS) loop flow, a steady state hydraulic analysis of the RCS was performed. This analysis considered the change in flow resistance within the fuel and the change in flow resistance within the steam generators.

The analysis considered four combinations of fuel type and generator:

- Mark-BW fuel with the original steam generators (OSGs)
- Mark-BW fuel with the replacement steam generators (RSGs)
- Advanced W17 HTP fuel with the OSGs
- Advanced W17 HTP fuel with the RSGs

The results support the conclusion that the increase in the total RCS flow that results from the OSG replacement is larger in magnitude than the decrease in total RCS flow that results from the fuel transition, and the net impact of both is an increase in total RCS flow. Specifically, transitioning to Advanced W17 HTP fuel without replacing the OSGs causes RCS flow to decrease by approximately [ ], replacing the OSGs without transitioning to Advanced W17 HTP fuel causes the RCS flow to increase by approximately [ ], and performing both the Advanced W17 HTP fuel transition and the OSG replacement causes the RCS flow to increase by approximately [ ]. Analyses therefore predict a small net increase of [ ] for the transition to Advanced W17 HTP fuel with the RSGs.

Note that when the Unit 1 OSGs were replaced a significantly larger increase in flow was observed. The difference in results is attributable to conservatisms that were used in the determination of the [ ] increase in flow. The increase in Unit 1 RCS flow is discussed further in Parts 2 and 3, below.

6.1.2 AREVA Response – Question 1 (Part 2)

The flow rate of 91,400 gallons per minute (gpm) in Table 2.2-1 is derived by dividing the thermal design flow rate of 378,400 gpm by the number of primary system coolant loops (four) and then accounting for the 3.5% flow measurement uncertainty. These flow rates are utilized within departure from nucleate boiling (DNB) calculations as described in Section 4.2.9 of ANP-2986(P).

Technical Specifications (TS) Figure 3.2-1 relates to the Limiting Condition for Operation (LCO) for Section 3.2.5 of the SQN TS, "DNB Parameters." The acceptable operation region within the figure represents a conservative reduction in thermal power fraction (% of rated thermal power (RTP)) for a reduced measured flow rate for flow deficits of up to 5%. The value of 359,480 gpm at 90% RTP in the figure was developed for a conservative relationship of 2% reduction in RTP for every 1% of flow deficit from 100% RTP, 100% nominal flow to 90% RTP, and 95% of nominal flow (378,400 gpm x 0.95 = 359,480 gpm). A DNB analysis for conditions at 90% RTP and 95% flow confirmed that this relationship is conservative for Advanced W17 HTP fuel. The discussion of how the total RCS flow rate of 378,400 gpm was generated is provided in Parts 1 and 3.

6.1.3 AREVA Response – Question 1 (Part 3)

The 5% increase in minimum RCS flow rates in Table 2.2-1 and Figure 3.2-1 following the replacement of the OSGs with the RSGs is supported by experience gained in replacement of the Unit 1 OSGs. When the Unit 1 OSGs were replaced in the spring of 2003, measured RCS flow increased and consistently exceeds the proposed 378,400 gpm limit by more than 3%. A similar outcome is expected when the Unit 2 OSGs are replaced concurrently with the transition to the Advanced W17 HTP fuel type in the fall of 2012.

Unit 1 and Unit 2 TS were amended in Cycle 9 to reduce minimum required RCS flow by approximately 5% to its current Figure 3.2-1 value of 360,100 gpm at 100% of RTP. This change was made to recover operating margin in RCS flow after significant numbers of OSG tubes had been plugged in earlier cycles. Because the OSGs will have been replaced on both units, RCS flows will be restored and there will no longer be a need for a reduction in required RCS flow. The proposed increase in minimum required RCS flow rate restores required flow to approximately the original design value.

Based on the increase in RCS flow observed as a result of Unit 1 OSG replacement, sufficient RCS flow will exist in Unit 2 to support an increase in required RCS flows to the total value of 378,400 gpm specified in Figure 3.2-1 and the 94,600 gpm per loop flow as stated in Table 2.2-1, both of which are 3.5% more than the design value to allow for measurement uncertainty. The higher flow rates will be confirmed by periodic measurements taken in accordance with TS Surveillance Requirements.

6.2 *RAI Question 2*

In response to the staff RAI 2(c) in ANP-3053(P), Rev. 3 dated May 2012, the DNBR transition core penalty, discussed in Section 4.4.1 of the ANP-2986 (P) Rev. 3, is offset by DNBR margin within the Thermal Design Limit (TDL) which protects the critical heat flux (CHF) correlation DNBR limit. Please provide; (1) description of the Sequoyah DNBR transition core penalty and the DNBR margin available under all the operating conditions; (2) justification how the DNBR transition core penalty is offset by DNBR margin within TDL quantitatively; and (3) approved methodologies to support the calculation for the DNBR transition core penalty and DNBR margin.

6.2.1 AREVA Response – Question 2 (Part 1)

The departure from nucleate boiling ratio (DNBR) transition core penalty is a quantification of the impact that a mixed core of hydraulically dissimilar fuel types has on the minimum DNBRs calculated during generation of the DNB-based safety limit (SL) and operating limit (OL) maximum allowable peaking (MAP) limits. The SL and OL MAP limits are calculated at various combinations of power distributions and boundary conditions which represent the range of operational conditions for the plant. Section 7.5 of BAW-10220P (Mark-BW Fuel Assembly Application for Sequoyah Nuclear Units 1 and 2, March 1996) describes how SL and OL MAP limits are calculated for a particular fuel type based on full-core of that fuel. A description of the peaking limits is given in Section 7.5 of BAW-10220P. To account for the impact that a mixed core has on these calculations, a DNBR transition core penalty must be calculated and applied during each Advanced W17 HTP fuel transition cycle. A separate transition core penalty is calculated for each fuel type present in the core. A detailed description of this process is given in the response to part 2 of this RAI.

The retained thermal margin (RTM) is an amount of DNBR margin that is built into the thermal design limit. Following the methodology outlined in Section 7.2 of BAW-10220P and Section 5 of BAW-10170P-A (Statistical Core Design For Mixing Vane Cores, December 1998), a statistical design limit (SDL) is determined for each fuel type. This limit is the minimum DNBR which must be maintained in order to provide adequate protection against DNB in statistical analyses. A thermal design limit (TDL) is then selected which is higher than the SDL. For all statistical analyses, the TDL is assumed to be the minimum DNBR which provides adequate protection against DNB. The retained thermal margin (RTM) is the difference between the TDL and the SDL, and it can be expressed as the absolute difference between the two in DNB points where 1 DNB point is equal to 0.01 absolute DNBR (as shown in Section 4.2.4.2 of ANP-2986P Revision 3) or as a percentage of the TDL (as shown in Section 7.2.2 of BAW-10220P).

The purpose of the RTM is to provide the flexibility to accommodate cycle specific impacts on the DNBR analyses. Whenever any change is made to the fuel assembly or cycle design, if there is a transition core penalty, or if some non-conservatism is discovered, the impact of the change can be quantified and assessed against the RTM. The RTM can account for many different penalties which account for changes in DNB-based calculations which cover all operating conditions. If the remaining RTM is greater than or equal to zero after all penalties have been considered, then the existing DNB calculations and MAP limits are considered acceptable. The remaining RTM is

calculated each cycle for each fuel type present within the core. A representative diagram illustrating RTM (including a transition core penalty) is given below in Figure 20.

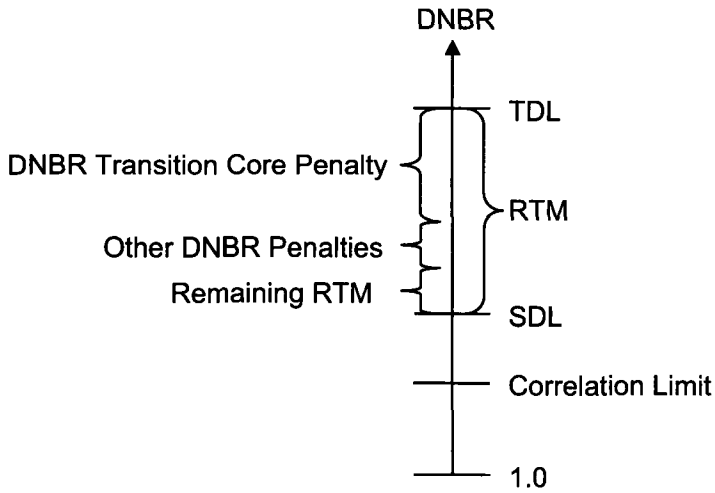

Figure 20: Representative Illustration of RTM and DNBR Penalties Applied Against the RTM

6.2.2 AREVA Response – Question 2 (Part 2)

This response describes how the DNBR transition core penalty (TCP) was derived for Advanced W17 HTP fuel. As mentioned below, no DNBR TCP is necessary for the resident Mark-BW fuel.

[ ]

[

]

Figure 21: Expected Advanced W17 HTP Transition Core Penalties (Maximum Calculated Penalties)

For each Advanced W17 HTP transition cycle, the appropriate DNBR TCP is determined based on the number of Advanced W17 HTP assemblies in the core. Alternatively, the penalty may be calculated for a specific cycle by repeating the analysis described above with the actual loading pattern for that Advanced W17 HTP transition cycle.

The DNBR TCP is then assessed against the RTM for the Advanced W17 HTP fuel. If the RTM is not negative after this penalty and any other penalties are assessed, then the use of peaking limits based on a full-core of Advanced W17 HTP fuel is valid for that

particular transition cycle. The amount of RTM expected for the transition cycles is approximately [ ] of the TDL. As can be seen in Figure 21, the expected DNBR TCP for a core with 89 Advanced W17 HTP assemblies (typical for a first transition cycle) is approximately [ ]. This indicates that there is sufficient RTM to account for mixed core effects on DNB. It will be verified on a cycle-specific basis that the RTM can accommodate the DNBR TCP and any other DNBR penalties required for DNBR-related concerns.

Similarly, a DNBR TCP analysis was performed to quantify the impact of a mixed core on the resident Mark-BW fuel. That analysis determined that the peaking limits improved as a result of flow diversion into the Mark-BW fuel which results from the lower pressure drop in the Mark-BW fuel. Therefore, no DNBR TCP will have to be assessed against the Mark-BW fuel's RTM during the Advanced W17 HTP transition cycles.

6.2.3 AREVA Response – Question 2 (Part 3)

The DNBR TCP calculation, as described in Section 4.4 of ANP-2986P Revision 3, is modeled after the TCP described in both Section 7.6.2 of BAW-10220P and the response to RAI 27 in BAW-10220P. The approved subchannel analysis code used for the calculation of TCPs, LYNXT, is described in BAW-10156-A. The RTM methodology is described in Section 7.2 of BAW-10220P and Section 5 of BAW-10170P-A.

7.0 References

1. Letter from NRC to TVA, "Sequoyah Nuclear Plant, Units 1 and 2 - Request for Additional Information Regarding the Propopsed [sic] Technical Specification Changes to Allow Use of AREVA Advanced W17 High Thermal Performance Fuel (TAC Nos. ME6538 and ME6539)", dated October 14, 2011 (ADAMS Accession No. ML11269A053).

2. Letter from NRC to TVA, "Sequoyah Nuclear Plant, Units 1 and 2 - Request for Additional Information Regarding the Propopsed [sic] Technical Specification Changes to Allow Use of AREVA Advanced W17 High Thermal Performance Fuel (TAC Nos. ME6538 and ME6539)", dated February 8, 2012 (ADAMS Accession No. ML12025A027).

3. Letter from TVA to NRC, "Response to NRC Request for Additional Information Regarding Application to Modify Technical Specifications for Use of AREVA Advanced W17 HTP Fuel (TS-SQN-2011-07)", dated November 14, 2011 (ADAMS Accession No. ML113200023).

4. Letter from TVA to NRC, "Application to Modify Technical Specifications for Use of AREVA Advanced W17 HTP Fuel (TS-SQN-2011-07)", dated June 17, 2011 (ADAMS Accession No. ML11172A071).

5. Letter from TVA to NRC, "Response to NRC Request for Supplemental Information Regarding Application to Modify Technical Specifications for Use of AREVA Advanced W17 HTP Fuel (TS-SQN-2011-07)", dated July 27, 2011 (ADAMS Accession No. ML112101798).

6. Letter from TVA to NRC, "Response to NRC Request for Additional Information Regarding Application to Modify Technical Specifications for Use of AREVA Advanced W17 HTP Fuel (TS-SQN-2011-07)", dated March 23, 2012 (ADAMS Accession No. ML12088A170).

7. BAW-10231P-A Revision 1, "COPERNIC Fuel Rod Design Computer Code", Framatome ANP, January 2004, (ADAMS Accession No. ML040150701).

8. ANP-2970(P) Revision 0, "Sequoyah Units 1 and 2 HTP Fuel Realistic Large Break LOCA Analysis", April 2011, (ADAMS Accession No. ML11172A064).

9. ANP-2971(P) Revision 1, "Sequoyah Units 1 and 2 HTP Fuel S-RELAP5 SBLOCA Analysis", May 2011, (ADAMS Accession No. ML11172A072).

10. AREVA Letter NRC:09:069, "Informational Transmittal Regarding Requested White Papers on the Treatment of Exposure Dependent Fuel Thermal Conductivity Degradation in RODEX Fuel Performance Codes and Methods", July 14, 2009, (ADAMS Accession No. ML092010157).

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
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 public disclosure in accordance with 10 CFR 2.390, "Public inspections, exemptions, requests for withholding," paragraph (a)(4).

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
COUNTY OF CAMPBELL)

1. My name is Russell Wells. I am U.S. EPR COLA Licensing Manager, 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 document titled "Sequoyah HTP Fuel Transition – NRC RAIs and Responses, ANP-3053, Revision 4," 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 secrets 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(c) and 6(d) above.

7. In accordance with AREVA NP's policies governing the protection and control of information, proprietary information contained in this Document has 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.

SUBSCRIBED before me this 19th

day of June 2012.

Kathleen A. Bennett

Kathleen A. Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/2015
Reg. #110864

KATHLEEN ANN BENNETT
Notary Public
Commonwealth of Virginia
110864
My Commission Expires Aug 31, 2015