George H. Gellrich
Vice President

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CENG
a joint venture of
Constellation Energy · eDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

October 29, 2010

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit Nos. 1 & 2; Docket Nos. 50-317 & 50-318
Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

REFERENCE: (a) Letter from Mr. T. E. Trepanier (CCNPP) to Document Control Desk (NRC), dated November 23, 2009, License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

On August 23 and 24, 2010, the Nuclear Regulatory Commission (NRC) staff conducted an audit of analyses related to the proposed license amendment to support the transition from Westinghouse nuclear fuel to AREVA Advanced CE-14 High Thermal Performance fuel. A number of questions were raised by the NRC staff during the audit. The responses to some of the questions are contained in Attachment (1). Responses to the remaining questions will be submitted as they become available. This supplement does not change the No Significant Hazards determination previously provided in Reference (a).

Attachment (1) contains information that is proprietary to AREVA, therefore, it is accompanied by an affidavit signed by AREVA, owner of the information (Attachment 2). The affidavit sets forth the basis on which information may be withheld from public disclosure by the Commission, and addresses, with specificity, the considerations listed in 10 CFR 2.390(b)(4). Accordingly, it is requested that the information that is proprietary to AREVA be withheld from public disclosure. The non-proprietary version of the Attachment is included (Attachment 3).

A001
NRR

Should you have questions regarding this matter, please contact Mr. Douglas E. Lauver at (410) 495-5219.

Very truly yours,

STATE OF MARYLAND :
: **TO WIT:**
COUNTY OF CALVERT :

I, George H. Gellrich, being duly sworn, state that I am Vice President - Calvert Cliffs Nuclear Power Plant, LLC (CCNPP), and that I am duly authorized to execute and file this License Amendment Request on behalf of CCNPP. To the best of my knowledge and belief, the statements contained in this document are true and correct. To the extent that these statements are not based on my personal knowledge, they are based upon information provided by other CCNPP employees and/or consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 29 day of October, 2010.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires:

Wendy L. Hunter
NOTARY PUBLIC
Calvert County, Maryland
My Commission Expires 1/9/2014

1-9-14
Date

GHG/PSF/bjd

Attachment:
(1) Proprietary Supplement to License Amendment Request: Transition to AREVA Nuclear Fuel
(2) AREVA Proprietary Affidavit
(3) Non-Proprietary Supplement to License Amendment Request: Transition to AREVA Nuclear Fuel

cc: **[Without Attachment (1)]**
D. V. Pickett, NRC
W. M. Dean, NRC
Resident Inspector, NRC
S. Gray, DNR

ATTACHMENT (2)

AREVA PROPRIETARY AFFIDAVIT

Calvert Cliffs Nuclear Power Plant, LLC
October 29, 2010

AFFIDAVIT

COMMONWEALTH OF VIRGINIA)
) ss.
CITY OF LYNCHBURG)

1. My name is Gayle F. Elliott. I am Manager, Product Licensing, for AREVA NP Inc. 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 attachment to a letter from G.H. Gellrich (Calvert Cliffs Nuclear Power Plant) to Document Control Desk (NRC) entitled "Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel," dated October 29, 2010 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(b) and 6(c) above.

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

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

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

SUBSCRIBED before me this 29th

day of October, 2010.

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

SHERRY L. MCFADEN
Notary Public
Commonwealth of Virginia
7079129
My Commission Expires Oct 31, 2010

ATTACHMENT (3)

NON-PROPRIETARY SUPPLEMENT TO LICENSE AMENDMENT REQUEST: TRANSITION TO AREVA NUCLEAR FUEL

Calvert Cliffs Nuclear Power Plant, LLC
October 29, 2010

Contained below are responses to questions and concerns raised by the Nuclear Regulatory Commission (NRC) staff during their audit of analyses related to the transition from Westinghouse nuclear fuel to AREVA Advanced CE-14 High Thermal Performance fuel. All of the questions raised by the NRC staff are listed below. Note that only some responses have been provided at this time. The remainder of the responses will be provided as they become available.

General Comments on Non-LOCA Transient Analyses

Question 1:

Modeling assumptions for flow mixing in the lower plenum of the reactor vessel and non-uniform fuel assembly inlet flow distribution have a 1st order impact on calculated core parameters (e.g., power distribution, minimum DNBR) during anticipated operational occurrences (AOOs) and accidents.

a. *The current UFSAR methodology for calculating minimum DNBR consists of a detailed 3D open channel core thermal hydraulics model (i.e., TORC) which specifically models the core inlet flow distribution (mapping of fuel assembly flow factors). This current methodology accounts for flow mixing and non-uniform flow distribution in the lower plenum of the reactor vessel. Separate core inlet flow distributions exist for 4-pump and 3-pump configurations. Please identify and discuss differences in the treatment of core inlet flow distribution in all current and new UFSAR Chapter 15 analysis of records (AORs). Include a description of the basis of each model and whether empirical data (e.g., plant flow testing measurements, scale models) were used in their development.*
b. *The new Asymmetric Steam Generator Transient analysis does not model an asymmetric core inlet temperature distribution and its impact on power distribution. Please identify and discuss differences in the treatment of core inlet temperature distribution in all current and new UFSAR Chapter 15 AORs. Include a description of the basis of each model and whether empirical data (e.g., plant flow testing measurements, scale models) were used in their development. Provide information to justify that any analytic penalties are appropriately conservative.*

CCNPP Response 1:

Modeling assumptions for flow mixing in the lower plenum and non-uniform fuel assembly inlet flow distribution do not have a first order effect on the power distribution or minimum departure from nucleate boiling ratio (DNBR). The inlet flow distributions are washed out quickly in an open lattice pressurized water reactor core and inlet temperature differences are generally second order effects.

a. The current DNBR analyses for the Calvert Cliffs Updated Final Safety Analysis Report (UFSAR) Chapter 14 events are performed with CETOP. CETOP is "simplified TORC" and determines the hot channel core response (Reference 1). CETOP is benchmarked to TORC to ensure CETOP is conservative to TORC over the range of operating conditions (temperature, pressure, flow, and axial shape index). The CETOP model used to determine the minimum DNBR for the transient analyses includes the 4-pump inlet flow factor.

   Reference 2 provides 4-pump inlet flow distributions based upon a 1/5 scale, 3400 MWt Reactor Flow Model Test Program and Palisades flow model tests. The Palisades data was applied to the 1/5 scale, 3400 MWt Reactor Flow Model 4-pump data to generate 3-pump data.

   The most recent Reactor Coolant Pump (RCP) Seized Rotor event uses CETOP to calculate available margin at the initial conditions assuming 4-pump flow, and at the time of most adversity for departure from nucleate boiling (DNB). This includes an assumption of an instantaneous degradation to the 3-pump asymptotic flow and includes the 3-pump inlet flow factor. The required over power margin is determined based upon the margin change due to the instantaneous loss of flow while the other parameters remain unchanged.

Core inlet flow distribution is applicable in those Calvert Cliffs UFSAR Chapter 14 analyses that require calculation of DNBR. The events for which an explicit DNBR analyses is performed are:

UFSAR 14.2 Control Element Assembly Withdrawal Event
UFSAR 14.4 Excess Load Event
UFSAR 14.8 Reactor Coolant System Depressurization
UFSAR 14.9 Loss-of-Coolant Flow Event
UFSAR 14.11 Control Element Assembly Drop Event
UFSAR 14.12 Asymmetric Steam Generator Event
UFSAR 14.14 Steam Line Break Event
UFSAR 14.16 Seized Rotor Event

The minimum DNBR for the AOOs [Control Element Assembly (CEA) Withdrawal, Excess Load, Reactor Coolant System (RCS) Depressurization, Loss-of-Coolant Flow, CEA Drop, and Asymmetric Steam Generator Event] is determined by inputting transient analysis results (flow, temperature, pressure, and power) into CETOP along with limiting axial shapes to determine the required over power margin.

Currently, the AOR for the postulated accidents [Pre-Trip Main Steam Line Break (MSLB) and RCP Seized Rotor] also determine the required over power margin to prevent violation of the specified acceptable fuel design limits (SAFDLs).

The minimum DNBR calculation associated with the Post-Trip MSLB analysis is calculated using the MacBeth core heat flux (CHF) correlation and a single closed channel model which does not explicitly model a core wide inlet flow distribution. For the Post-Trip MSLB 3-D reactivity feedback, the hot channel is analyzed using HERMITE/TORC methodology which determines inlet flow based on pointwise power; thus, inlet flow distribution is not explicitly modeled.

Although DNBR is an acceptance criterion for the following events, explicit DNB analysis was not performed.

UFSAR 14.3	Boron Dilution	Bounded by the reactivity addition associated with CEA Withdrawal.
UFSAR 14.5	Loss of Load	This is a peak pressure event. Minimum DNBR SAFDL is not challenged.
UFSAR 14.6	Loss of Feedwater Flow	Pressure increases concurrent with a small increase in power. Bounded by more limiting events.
UFSAR 14.7	Excess Feedwater Heat Removal	Bounded by the Excess Load event.
UFSAR 14.10	Loss-of-Non-Emergency AC Power	Loss-of-Coolant Flow event analyses ensures the SAFDLs are not exceeded.
UFSAR 14.15	Steam Generator Tube Rupture	Rate of depressurization is less than that associated with the RCS Depressurization. DNBR is bounded by that event.

Both the AREVA and Westinghouse methods are acceptable, as both have been reviewed and approved by the NRC. Deterministic models developed independently by different vendors will provide differing results. Both results are conservative based on the approved models and it is entirely likely that one could be more conservative than the other. The relative amount of conservatism between two models is not important from a safety perspective. The significant factor is that both are conservative.

The AREVA methodology applies a 5% flow penalty (Reference 3) to the limiting fuel assembly and the four face-adjacent fuel assemblies. This penalty is justified by the flow penalties that were derived for Calvert Cliffs specific testing as described below.

From Reference 4: "Tests were conducted with scale models for the Palisades (AEC Docket No. 50-255), Maine Yankee (AEC Docket No. 50-309), Fort Calhoun (AEC Docket No. 50-285), and Calvert Cliffs Unit 1 (AEC Docket No. 50-317) reactors to determine the hydraulic performance for normal and part-loop reactor configurations. The models were geometrically similar to each of the reactors except for the core regions, where each fuel assembly was simulated by cylindrical tubes with orifices to provide the proper axial flow resistance. Air was used as the test medium for the Palisades and Omaha flow models, while water was used for the Maine Yankee and Calvert Cliffs models. Core flow distribution measurements and reactor pressure loss coefficients were obtained for the various pumping configurations. The flow model programs are further discussed in Section 4.4.3. Taking into consideration the similarities between this reactor and other C-E reactors in conjunction with the experimental data from the flow model programs, the following flow distribution factors for the various pumping configurations were established.
1.05 – four pump operation
1.06 – three pump operation"

b. Response to be provided separately.

Question 2:

The strategy for addressing the presence of both Westinghouse TURBO fuel assemblies and AREVA CE14 HTP fuel assemblies relies on limiting the relative power in the TURBO fuel bundles. During transition cores, fuel management schemes will ensure that resident TURBO fuel assemblies operate at reduced power levels relative to the AREVA CE14 HTP fuel assemblies. It is the staff's understanding that peak fuel rod radial peaking factors (Fr) within any TURBO fuel assembly will remain 9% lower than the leading Fr within any AREVA CE14 HTP fuel assembly. In theory, this additional thermal margin will ensure that resident TURBO fuel assemblies will never be limiting during any AOO and accident condition. The staff requests further information to assess this strategy:

a. *For lower power events which do not rely upon initial HFP thermal margin (e.g., Post-Trip MSLB, CEA ejection, bank withdrawal, excess load), neither approach to DNBR or fuel centerline melt SAFDLs will be quantified for Westinghouse TURBO fuel rods. How do the transition core reload methods ensure that Westinghouse fuel does not violate its own SAFDLs during these events?*
b. *For CCNPP-2 Cycle 19 and future transition cores, will the 9% thermal margin be preserved under all rodded conditions allowed by the COLR PDIL?*
c. *For CCNPP-2 Cycle 19 nominal HFP conditions, provide the Fr, calculated DNBR, and overpower DNB margin for the limiting Westinghouse and AREVA fuel rods.*

d. *At different exposure levels, compare the calculated AOO and accident overpower required to achieve the Westinghouse and AREVA cladding strain SAFDL and compare to the predicted overpower for all Chapter 15 AOO and accidents.*

e. *NUREG-0800, SRP-4.2 requires that the number of failed fuel rods not be under predicted. How do the transition core reload methods quantify the number of Westinghouse fuel rods which violate any SAFDLs during any accident conditions?*

f. *For CCNPP-2 Cycle 19, provide a plot of minimum DNBR versus time (current UFSAR analysis (TURBO) versus new AREVA analysis (CE14 HTP)) for several AOO and accident analyses.*

CCNPP Response 2:

a. Response to be provided separately.

b. The 9% margin will be preserved under all rodded conditions allowed by the Core Operating Limits Report (COLR) power dependent insertion limits. The margin was explicitly verified for the following steady-state conditions:

   - Unit 2 Cycle 19 based on a short, nominal, and long Unit 2 Cycle 18
   - Beginning of cycle through end of cycle in burnup steps of 0.5 GWd/MTU or smaller
   - 5% power through 100% power in intervals of 10% power or less
   - Long-term insertion limits and power dependent insertion limits

   Also, the 9% margin was verified for the limiting cases of transient events that result in power redistribution, including MSLB, CEA Withdrawal, and CEA Drop.

c. Response to be provided separately.

d. Response to be provided separately.

e. Response to be provided separately.

f. The requested plots are shown below (Figures 2-1 through 2-4). Note that for the current Calvert Cliffs UFSAR analysis, the core is assumed to be loaded with Turbo fuel only. The AREVA analyses reflected in the Figures below assume a mixed core of AREVA and Westinghouse fuel (except for the Excess Load at hot full power figure which assumes a full core of AREVA fuel).

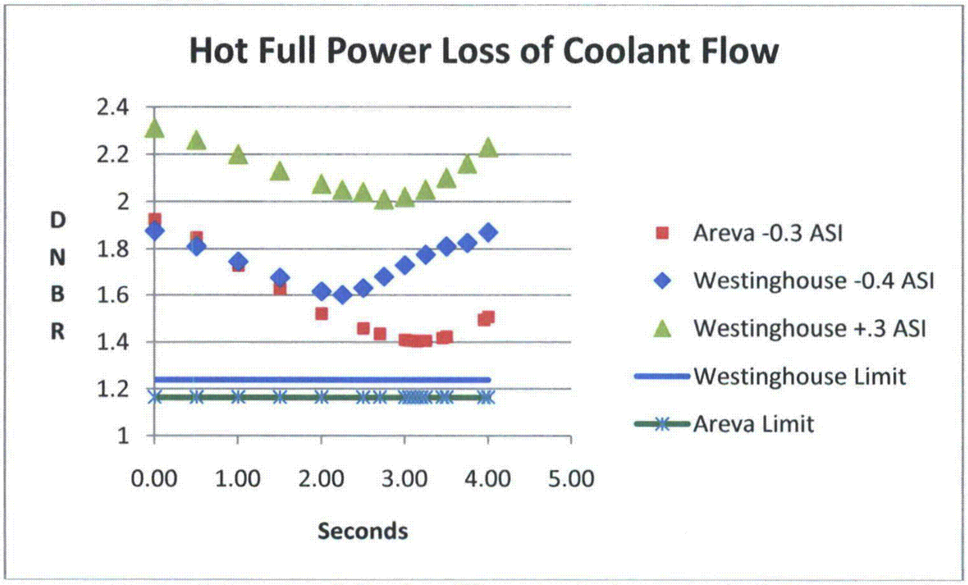

Figure 2-1, Loss-of-Coolant Flow Analysis at Hot Full Power

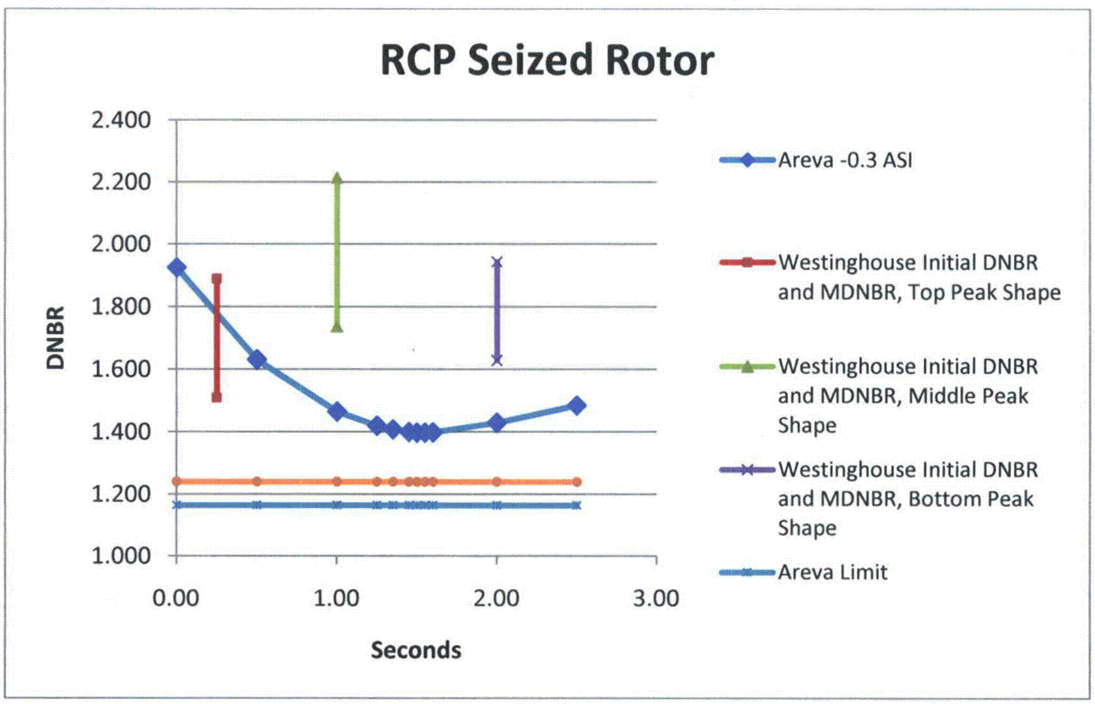

Figure 2-2, RCP Seized Rotor Analysis

Note: Westinghouse calculates the minimum DNBR at the initial 4-pump flow conditions and at 3-pump flow conditions. Therefore, the Westinghouse DNBR data does not correlate to time.

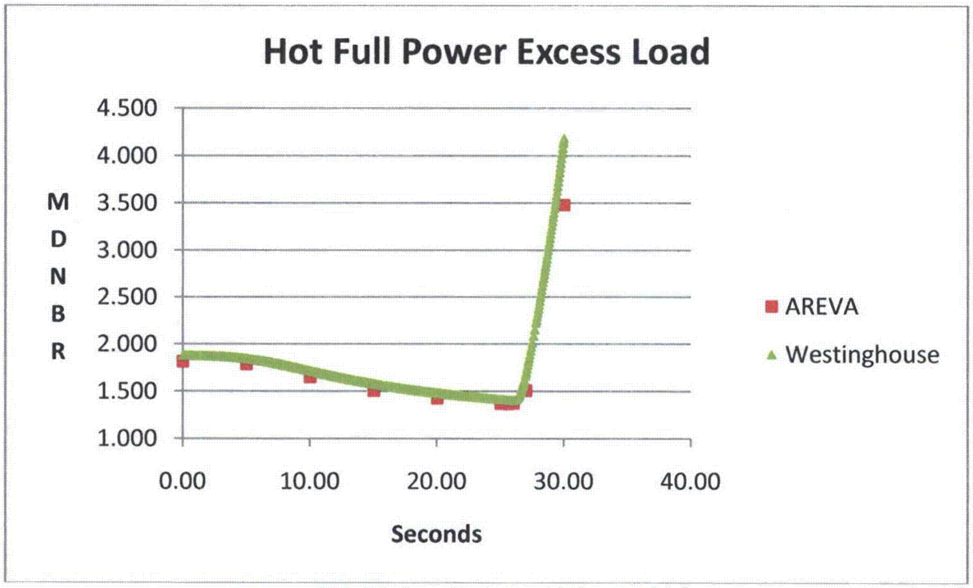

Figure 2-3, Excess Load at Hot Full Power

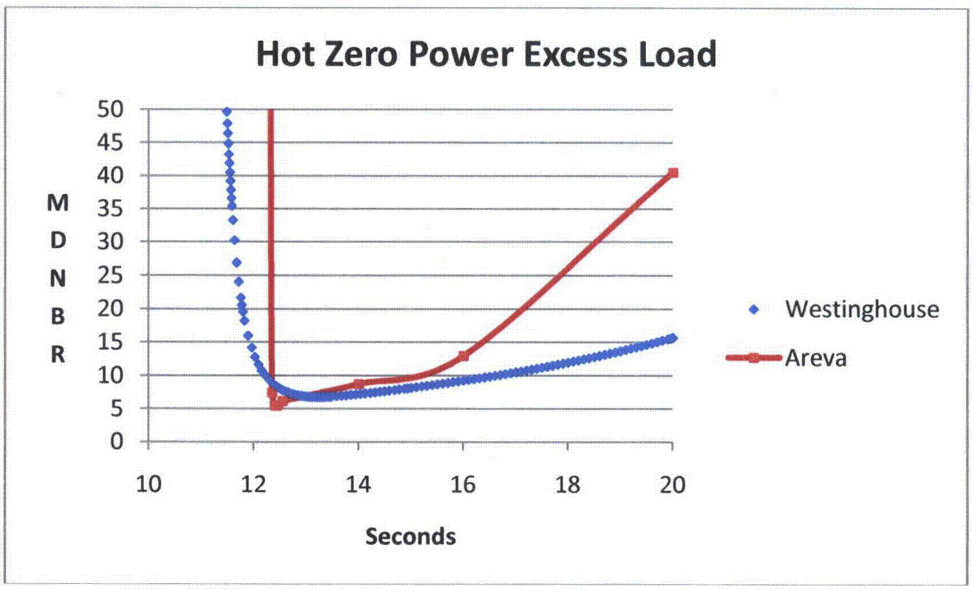

Figure 2-4, Excess Load at Hot Zero Power

Question 3:

Some of the postulated accidents and transients that are analyzed and described in the Calvert Cliffs Updated Final Safety Analysis Report are sensitive to the initial power level. This is of concern, but may not be limited to, reactivity and power distribution anomalies.

While the current licensing basis and the proposed safety analysis methodology include prospects for analyzing these events at zero- and full-power conditions, the NRC staff has not located documentation describing further analyses, data, and/or sensitivity studies to indicate that the consequences of these events, if initiated at a power level between zero- and full-power, would be less severe than the two power levels analyzed. Further, allowable operating ranges in the COLR LCOs often vary as a function of *power level (e.g., ASI, peaking factor, control rod insertion). The basis for these power-dependent breakpoints must be grounded in safety analysis.*

Please identify the limiting set of initial conditions for those transients that are sensitive to the initial core power level and demonstrate that, when initiated at those initial conditions, the analytic results remain within the applicable acceptance criteria.

In particular, provide information to demonstrate appropriate consideration of the following:

a. *Combinations of initial power level and instrument uncertainty that will provide for a) the greatest challenge to reactor protection system effectiveness and b) the greatest rise in power between event initiation and trip completion*
b. *The basis for allowable control rod insertion as a function of core power, and the CEA worth and core design parameters that correspond to those limits*
c. *Initial thermal margin available at the transient onset and the reduction in that margin throughout the transient*
d. *Core conditions at varying exposures, including mid-cycle cases*
e. *Assumption of more severe axial power shapes and radial power distributions reflective of operation at lower power levels*

CCNPP Response 3:

Response to be provided separately.

<u>Question 4</u>:

Per the EMF-2310 methodology, the S-RELAP5 analysis for any given transient typically assumes a significant number of initial conditions are taken at nominal values. The licensing basis transient analysis, however, must demonstrate acceptable results with respect to both specified acceptable fuel design limits and reactor coolant pressure boundary integrity. The analytic assumptions that deliver a conservative result for one will, at times, deliver a non-conservative result with respect to the other.

While the EMF-2310 methodology describes detailed thermal-hydraulic analysis, which relies on parametric biasing to provide conservative results with respect to fuel thermal margin, similar parametric biasing to provide conservative results with respect to peak RCS pressure is not always performed.

For transients and accidents that challenge both fuel thermal and RCS pressure margins, provide plant analyses to demonstrate the effects of initiating the selected transients at pressure-limiting initial conditions, including, for example, RCS pressure, main steam system initial pressure, and steam generator initial level.

CCNPP Response 4:

Response to be provided separately.

<u>Question 5</u>:

Provide a detailed summary describing the process for transient-specific verification of analog instrument setpoints, delays, and uncertainties, and the evaluation of the resultant impact on transient and accident analysis results.

CCNPP Response 5:

The Thermal Margin/Low Pressure (TM/LP) limiting safety system settings and linear power density limiting safety system setting functions in analog Combustion Engineering plants are uncompensated functions. Thus, the trip setpoint must be adjusted to accommodate the potential for overshoots or undershoots. For Calvert Cliffs Units 1 and 2, these allowances are already incorporated in the plant setpoints, and the problem reduces to deriving trip process variable overshoots/undershoots in transient analysis, then penalizing the setpoint analysis with the results. These changes in trip process input parameters during the trip delay are known as "transient biases" or "transient shifts." Transient shifts are defined both for the linear power density limiting safety system setting and the TM/LP limiting safety system setting, and used in the setpoint calculations for these two trip functions.

Figure 5-1, shown below, graphically depicts the concept of a transient shift. Consider a time-dependent trace of an uncompensated trip input parameter, V, near the point where the setpoint of that trip is reached in a transient. This process parameter continues to change between the point at which the setpoint is reached, and the point of peak DNBR or fuel centerline melt challenge. If dynamic compensation on the signal is not used to reduce the overshoot ΔV, then the overshoot bias ΔV must be accommodated in the setpoint calculation. This adjustment is performed in the Combustion Engineering statistical setpoint analysis calculation.

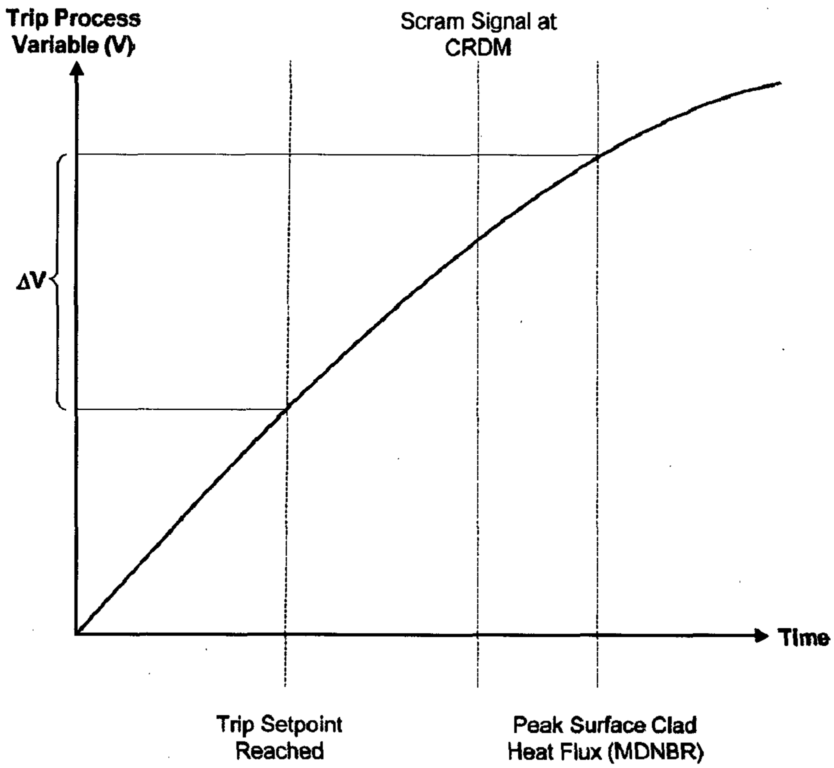

Figure 5-1, Graphical Depiction of Transient Shift

The generation of these transient shifts is performed in S-RELAP5 which can model the dynamics of the TM/LP and linear power density trips, as discussed below.

S-RELAP5 Support Control Variables

TM/LP Transient Power Shift

Figure 5-2, TM/LP Power Shift Calculation Flow

TM/LP Transient Hot Leg Temperature Shift

Figure 5-3, TM/LP Hot Leg Temperature Shift Calculation Flow

TM/LP Transient Cold Leg Temperature Shift

Figure 5-4, TM/LP Cold Leg Temperature Shift Calculation Flow

TM/LP Transient Pressure Shift

Figure 5-5, TM/LP Pressure Shift Calculation Flow

Linear Power Density Limiting Safety System Setting Transient Power Shift

Figure 5-6, Linear Power Density Limiting Safety System Setting Shift Calculation

Question 6:

Provide recent data concerning fuel rod bowing to demonstrate that (1) legacy analyses for fuel rod bowing remain applicable to modern fuel designs and operating strategies, (2) that thermal-hydraulic testing accounts for fuel rod bowing, and (3) thermal-hydraulic analysis includes appropriate treatment of fuel rod bowing in light of recently observed data.

CCNPP Response 6:

Fuel rod bow measurements of modern AREVA fuel designs and operating strategies are well bounded by the legacy analyses (see Figure 6-1). Measurements recently taken on Advanced CE-14 High Thermal Performance fuel at Ft. Calhoun show fuel rod bow performance that is lower than the 95/95 upper tolerance limit prediction (Reference 5, Supplement 1, Equation 3.2). Further evidence of acceptable fuel rod bow performance is provided by recent poolside evaluations at Calvert Cliffs Units 1 and 2, Millstone Unit 2, and St. Lucie Unit 1. Based on visual observations, these inspections show the fuel rod to fuel rod gaps on discharged fuel are not significantly changed due to fuel rod bow. The Calvert Cliffs fuel assemblies showed no significant fuel rod bowing at a burnup of greater than [ ].

Figure 6-1, Water Channel Standard Deviation

Fuel rod bowing is not purposely introduced into thermal-hydraulic testing unless the goal of the test is to quantify the affect of fuel rod bowing on a specific performance parameter, e.g., CHF impact. Reference 5 is AREVA's fuel rod bow topical report applicable for the Advanced CE-14 High Thermal Performance fuel design. This topical report utilizes CHF test results, discussed in References 6 and 7, where a specific magnitude of fuel rod bow was intentionally constructed into the small test bundle arrays to allow the measurement of the impact of various severities of bowed rod conditions.

The gap closure due to fuel rod bow is based on the fuel rod bow model in Reference 5. The CHF testing conservatively shows that DNB results are not adversely impacted for fuel rod to fuel rod gap closure less than 50%. Therefore thermal-hydraulic analyses include appropriate and conservative treatment of fuel rod bowing effects.

Locked Rotor Transient Analysis

Question 7:

Section 5.1, Assumption #1, [

] In light of this assumption, describe the assembly inlet flow factors and flow coast down characteristics in each region of the core. Provide a justification for this assumption. As part of this justification, identify any differences between the new core inlet flow distribution and the current UFSAR AOR.

{In the event that the transient simulation is re-run, consider delaying the turbine trip such that primary pressure does not increase as a result of the loss of secondary heat removal prior to minimum DNBR.}

CCNPP Response 7:

Response to be provided separately.

Pre-Trip MSLB Transient Analysis

Question 8:

Item 11 on Page 26 indicates that RCP coastdown begins at reactor scram and not concurrent with reactor trip signal. Justify this change relative to UFSAR.

CCNPP Response 8:

[

.] Nevertheless, the analysis provides a valid comparison of the various cases analyzed in the break size, location and moderator temperature coefficient spectrums, and adequately identifies the limiting break cases regarding minimum DNBR and linear heat generation rate. These two limiting cases were re-run with the RCP trip time changed to reactor trip and the limiting minimum DNBR and linear heat generation rate were found to be negligibly impacted.

Question 9:

The S-RELAP5 scenarios describe symmetric and asymmetric cases. Prior to MSIV closure, steam flow should increase from both SGs. Describe the asymmetric steam flow cases. Include in your description plots of steam flow versus time for all of the cases. In addition, discuss the scenarios which credit the asymmetric SG trip.

CCNPP Response 9:

Response to be provided separately.

Question 10:

New reactor trips are credited (i.e., Thermal Margin/Low Power, Low Steam Generator Pressure, SGdP) relative to trip functions cited in the UFSAR for the pre-trip scenario (i.e., HCPT and Variable High Power Trip). Describe how initial conditions and assumptions were manipulated to delay these trips.

CCNPP Response 10:

The Calvert Cliffs UFSAR includes only the limiting case and it was terminated with a Variable High Power trip. The Low Steam Generator Pressure trip was credited in the Calvert Cliffs UFSAR analysis but the Calvert Cliffs UFSAR acknowledges that other trips would be actuated before this trip.

The analysis was performed based on the AREVA methodology. Below is a discussion of conservative assumptions employed in simulating the trips used in the analysis and uncertainty biasing to delay the trips:

Variable High Power Trip: The analysis value of the reactor power is 2754 MWt (Rated Thermal Power is 2737 MWt, plus 17 MWt measurement uncertainty). The trip setpoint is set at 112.0% of 2754 for both the Variable High Power-Delta Temperature and Variable High Power-Nuclear Instrument. Once this setpoint is reached, there is a delay of 0.4 seconds prior to reactor trip signal.

In addition to biasing power, conservative decalibration of both the nuclear instrument and delta temperature power input to this trip function is modeled. The indicated nuclear instrument power is decalibrated by the shadowing effect of the cooler fluid in the reactor vessel downcomer which is conservatively specified to be [ ]. The delta temperature power is decalibrated by conservatively adjusting the assumed hot and cold leg resistance temperature detector response times. These decalibrations result in a conservative delay of the Variable High Power Trip.

Steam Generator Pressure - Low: The trip setpoint for a harsh environment condition is set at 600 psia. The Technical Specification setpoint is 685 psia, and an uncertainty of -81.53 psi for High Energy Line Break-Inside Containment, and additional margin of ~ -3.5 psi for additional conservatism, is considered. The trip delay of 0.9 seconds is assumed. For breaks outside Containment, the trip setpoint is conservatively assumed to be 650 psia, which is lower than the field setpoint minus the uncertainty for High Energy Line Break-Outside Containment. The field setpoint is higher than the Technical Specification setpoint.

The analysis also conservatively assumes no steam generator tube plugging, which results in the highest initial steam generator pressure for the condition analyzed, due to the undegraded heat transfer. High initial steam generator pressure delays the Steam Generator Pressure-Low trip.

Asymmetric Steam Generator ΔP: The trip setpoint was set to an analysis setpoint of 186.0 psid. This is more conservative than the Technical Specification setpoint of 135 psid. A trip delay of 0.9 seconds is assumed.

Containment Pressure-High: The trip setpoint is set at 19.45 psia. The Technical Specification setpoint is 18.7 psia, with an uncertainty added. A trip delay of 0.9 seconds is assumed.

The containment response is conservatively simulated by a simple single node S-RELAP5 volume which underpredicts the pressure rise relative to a more detailed model. The initial condition was the minimum containment pressure including operational variation (13.7 psia) and the maximum net free containment volume was assumed.

Thermal Margin/Low Pressure (TM/LP): The TM/LP modeling conservatively assumes the COLR A_1 function value remains continuously at 1.0, minimizing the trip setpoint. While this trip was simulated in the analysis, no cases analyzed challenged this trip, and therefore the analysis does not credit this trip.

Post-Trip MSLB Transient Analysis

Question 11:

[

] Describe whether the inclusion of these wider ranges would influence the timing of the transient scenario. Specifically discuss:

a. Higher initial pressurizer pressure may delay timing of LPP SIAS.

b. Higher initial pressurizer pressure may delay delivery of HPSI.

CCNPP Response 11:

The initial pressurizer pressure is not a key parameter for a post-trip MSLB, it is a second order effect. The transient responses are less influenced by initial pressurizer pressure than they are influenced by break size, location, steam generator inventory, etc. The safety injection actuation signal setpoint and delay times have the necessary uncertainty included for conservatism. Boron injection is precluded until the sweepout time of the high pressure safety injection lines has been satisfied. [

]

During the transient, depressurization of the RCS is fairly rapid. For the limiting hot full power - no loss of offsite power case, pressure drops to the Pressurizer Pressure-Low safety injection actuation signal setpoint in 15.9 seconds. A small change in the initial pressurizer pressure to accommodate the pressure measurement uncertainty of 36 psi will not significantly change the timing of the Pressurizer Pressure-Low safety injection actuation signal or affect the timing of high pressure safety injection flow and will not significantly affect the peak return to power. The additional delay in these parameters would be about 0.8 seconds. Power at the point of peak return to power for this case when boron reaches the core is changing at a rate of only 0.01% rated thermal power/second.

[

] A small increase in initial core inlet temperature would have negligible effect on the rate of initial RCS depressurization and therefore would not influence the timing of the safety injection actuation signal or high pressure safety injection flow.

A closer review of the analysis provides additional details about the interaction of the safety injection actuation signal and the effect imposed upon the overall transient behavior.

For the hot full power case, the RCS pressure drops to saturation conditions (just under 1500 psia) by 15 seconds, then decreases more slowly as dictated by the upper head depressurization rate. High pressure safety injection flow begins at 62 seconds, after reaching the high pressure safety injection shutoff head (safety injection actuation signal and delay times have been satisfied). In this case, the initial bias on pressurizer pressure would have no effect since the RCS drops to saturation conditions so quickly and well ahead of the beginning of high pressure safety injection flow. Also, a further bias on initial T_{cold} would have no significant effect on timing of high pressure safety injection flow because the change in saturation pressure would be small and the delay in high pressure safety injection flow would be minor.

For the hot zero power case, the RCS pressure rapidly reaches saturation conditions (about 873 psia) by 40 seconds. In this case, the high pressure safety injection shutoff head has been reached by 34 seconds. High pressure safety injection flow doesn't start until after the safety injection actuation signal plus a delay time at about 53 seconds. In this case, an initial bias on pressurizer pressure would have no significant effect since the RCS pressure drops to saturation conditions fast enough so that there would be a very minor increase in timing of the safety injection actuation signal. Also, further bias on initial T_{cold} would have no significant effect because the saturation pressure would still be significantly below the high pressure safety injection shutoff head.

Question 12:

The MSLB analysis supporting the migration to AREVA fuel and methods does not include scenarios initiated from lower plant operating modes (as defined in the plant Tech Specs). In lower modes, certain trip functions and ESFAS equipment important in the mitigation of the event may be unavailable. Please discuss the availability of safety related equipment and demonstrate that the HZP case bounds scenarios initiated from lower modes.

CCNPP Response 12:

Response to be provided separately.

Question 13:

Discuss differences in the moderator reactivity versus moderator density curve used in the current S-RELAP5 calculations relative to the current UFSAR AOR. Include a discussion of the effects of stuck rod core location and how cycle-specific differences will be addressed for future reloads.

CCNPP Response 13:

The moderator cooldown curve and the trip curve used in the current Calvert Cliffs UFSAR AOR are calculated assuming a stuck CEA, which reduces the scram worth and increases the magnitude of moderator reactivity feedbacks. The curves shown in Figure 13-1 assume an end of cycle-hot full power moderator temperature coefficient of -3.3 x10^{-4} $\Delta\rho$/°F. Calvert Cliffs UFSAR Figures 14.14-1A and 14.14-1B display the Unit 1 and Unit 2 moderator density vs. reactivity curves. The curves shown in the Calvert Cliffs UFSAR figures correlate to the worst stuck CEA; the reactivity inserted with lesser worth CEAs tends to flatten at increased density.

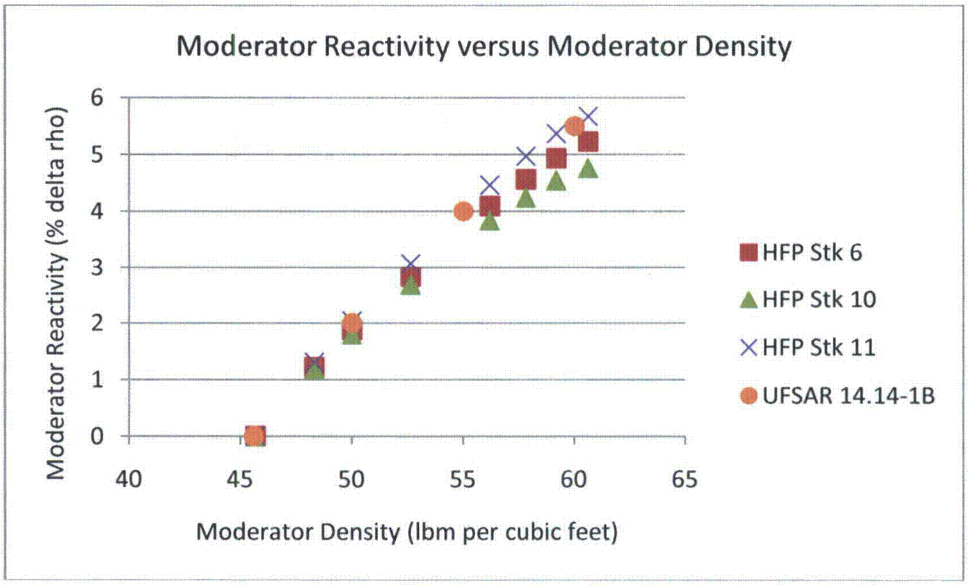

Figure 13-1, Moderator Reactivity vs Moderator Density

For the AREVA analysis the worst stuck CEA location in PRISM has a significant impact on the moderator density vs. reactivity results. The end of cycle worst stuck CEA in the Unit 2 Cycle19 evaluation is located near the center of the core. If the stuck CEA in the PRISM input deck is changed to a peripheral Shutdown Bank A location, the moderator density vs. reactivity results closely match the UFSAR Figure 14.14-1B.

The neutronics calculations for this event are performed for every reload. If the worst stuck CEA location changes in a future cycle, the safety analysis will be re-analyzed using the new worst stuck CEA location.

CEA Ejection Transient Analysis

Question 14:

The current UFSAR AOR includes a single case, bounding each input parameter based on conservative selection throughout burnup (BOC to EOC). The new analysis documents a single BOC and EOC case based on predicted physics parameters at the 2 exposure points. More cases may be necessary to ensure that the limiting combination of burnup-dependent parameters has been identified. Demonstrate that the limiting combination of initial conditions and core physics parameters has been captured by these 2 exposure points.

CCNPP Response 14:

Response to be provided separately.

Question 15:

The current UFSAR AOR cites a 0.05 sec time for the control rod to fully eject from the core. The new analysis assumes an ejection time of [ ]. As a result of this change, the reactor trip signal setpoint is reached prior to full withdrawal. Please justify the change in assumed CEA ejection time.

CCNPP Response 15:

The current Calvert Cliffs UFSAR analysis assumes a CEA ejection time of 0.05 seconds. [

]

In the AREVA analyses, the reactor trip setpoint is reached within 0.10 seconds, but with the trip delay time, the event is arrested by Doppler feedback before the CEAs take effect. This is an artifact of the rapid power excursion, and not uncommon for this event. Since the AREVA analyses are designed to be arrested by Doppler feedback, the thermal time constant of the fuel is large in comparison to the time scale of the event. As a result, the impact on the peak rod surface heat flux is small (<1%) even though the trip signal may precede the full withdrawal of the ejected CEA.

Question 16:

In the new analysis, Table 6.1 scram reactivity refers to N-1 values. CCNPP has traditionally used an N-2 scram curve (1 control rod sticks and 1 control rod ejected). Please justify the use of an N-1 scram curve.

CCNPP Response 16:

Response to be provided separately.

Question 17:

Calvert Cliffs reactor and protection system design criteria (UFSAR Chapter 1) dictate that the RPS be capable of performing its function in the event of a single failure. In addition, CCNPP Technical Specifications allow a single excore safety channel to be inoperable. The CEA ejection event exhibits a rapid, localized power excursion. The neutron flux levels measured and timing to reach the VHPT analytical setpoint at each of the four excore safety channels will be influences by their proximity to the ejected rod (as well as other factors including initial control rod configuration). Furthermore, a harsh environment may exist in containment and must be considered in the instrument response. Please describe how these factors were accounted for in the new analysis.

CCNPP Response 17:

Per Technical Specification 3.3.1, if a Reactor Protective System channel is inoperable, it is to be restored within 48 hours or the bistable unit is placed in trip. With a 2-out-of-4 trip logic scheme and one bistable unit placed in trip, the system provides a similar trip response in comparison to normal operations, even under the circumstance of an ejected CEA located cross-core from an operable detector. Also, since the time scale of the CEA Ejection event is so short, a harsh environment caused by a postulated rupture of a control rod drive mechanism housing is unlikely to degrade detector performance prior to actuation.

Ultimately, the power rise for the CEA Ejection event is arrested by Doppler reactivity feedback well before the Variable High Power Trip setpoint is reached. Any adjustment to the Variable High Power Trip setpoint due to inoperable excore detectors or harsh conditions would not significantly affect the DNBR or fuel centerline melt results for this event due to Doppler reactivity feedback.

Question 18:

Please discuss differences in analytical methodology and assumptions which prompted the significant change in predicted ejected rod worth in the new analysis relative to the current UFSAR AOR.

CCNPP Response 18:

Calvert Cliffs UFSAR Table 14.13-1 presents an ejected CEA worth of 0.31% $\Delta\rho$ at full power and a worth of 0.87% $\Delta\rho$ at zero power. The values presented are generic bounding values that are not recalculated each cycle.

For the AREVA analysis, calculations are performed for the maximum ejected CEA worths for hot zero power-beginning of cycle, hot full power-beginning of cycle, hot zero power-end of cycle, and hot full power-end of cycle conditions every reload. The results are compared with the safety AOR.

[

] Thus, a conservatively high value for ejected CEA worth is used in the analysis. Higher ejected CEA worths are used to bound cycle-to-cycle variations.

Question 19:

Please discuss the selection of the initial and final AXPD for each case. For example, the DNBR calculation for the BOC HZP case used a bottom peaked AXPD with a peak Fz of 1.3858. This benign AXPD does not appear to be limiting with respect to DNBR.

CCNPP Response 19:

The PRISM core neutronics computer code is used to generate the axial power shapes used in the DNB analysis, as well as the ejected CEA worths used in the transient system analysis. The DNB analyses, evaluated with XCOBRA-IIIC, are pseudo-steady-state evaluations performed at a "fixed" axial power shape.

For the beginning of cycle-hot zero power condition, the axial power shape is calculated [

] In this manner, both top and bottom, axially skewed power shapes are evaluated to bound possible operating ranges.

For the hot full power condition, a design axial power shape is conservatively constructed that will bound the limiting condition for operation and limiting safety system setting ASI limits. This shape ASI is approximately -30%. [

]

Also note, the ejected CEA worths may be conservatively increased during the S-RELAP5 simulations to ensure the core power rise is arrested by the Doppler feedback. This was the case for the AREVA analyses discussed above.

Excess Load Transient Analysis

Question 20:

Please provide a plot of the AXPDs (current UFSAR AOR versus new analysis) used in all of the lower power AOO and accident calculations. Discuss any significant differences.

CCNPP Response 20:

The axial power distributions for the following current Calvert Cliffs UFSAR events are discussed below because these events are lower power events, and impacted by the axial power distribution.

UFSAR 14.2 Control Element Assembly Withdrawal
UFSAR 14.4 Excess Load
UFSAR 14.13 Control Element Assembly Ejection
UFSAR 14.14 Steam Line Break

The CEA Drop (UFSAR 14.11), is currently analyzed at lower power to provide input to the limiting conditions for operation. The lower power scenarios are not presented in Calvert Cliffs UFSAR. AREVA analysis also verifies the DNB limiting conditions for operation setpoint margin for a CEA Drop.

CEA Withdrawal

The CEA Withdrawal analysis includes linear heat rate and DNBR evaluations. The linear heat rate calculation uses results from the CESEC transient code, which uses a point kinetics model. The ASI is mimicked by incorporating a varying fraction of CEA worth into the trip reactivity insertion curve. The ASI [(Lower – Upper)/(Lower + Upper)] for the trip portion of the lower power cases is shown below.

Linear Heat Rate Case	Hot Zero Power	20%, 50%, 70% Power	90% Power
Axial Shape Index	-0.4	+0.6	+0.4

Departure from nucleate boiling calculations are normally "no-trip" cases, therefore the ASI value used in CESEC is not relevant. The DNB calculations determine the required over power margin that must be reserved in the Limiting Conditions for Operation. The required over power margin calculations are performed using CETOP, and a range of ASIs that are within the power band being analyzed is assumed. Thus, a required over power margin calculation at hot zero power or 20% power will use the axial shapes that encompass the range of +/- 0.60, consistent with Figure 3.3.1-1 of the COLR. As power is increased, the range of ASI values used in the required over power margin calculations that are needed to bound Figure 3.3.1-1 of the COLR becomes smaller.

The current analysis determines linear heat rate and DNB at intermediate power levels, and the axial shapes are those allowed by the COLR figures. The AREVA analysis is performed at hot zero power. The analysis includes a "hot spot" calculation to determine fuel centerline temperature. The power (F_q) used in the hot spot calculation is calculated in PRISM. The limiting axial shape described in Figure 20-1 is used in the DNB analysis.

Excess Load

The excess load analysis is performed at hot zero power conditions. A single CESEC case is run for the evaluation of peak linear heat rate, and the CESEC results are input to the DNBR calculation. The trip insertion used in CESEC corresponds to an ASI of -0.4. The DNBR calculation uses a core average ASI of -0.652.

The excess load analysis is performed at hot zero power conditions. The AREVA analysis includes a "hot spot" calculation to determine linear heat rate. The power (F_q) used in the hot spot calculation is calculated in PRISM. The limiting axial shape described in Figure 20-1 is used in the DNB analysis.

CEA Ejection

The CEA Ejection analysis is performed at hot zero power conditions using the STRIKIN-II computer code. STRIKIN-II calculates hot channel heat up, coolant enthalpy, and fuel temperatures. An axial shape with an ASI of -0.562 is used in STRIKIN-II to determine the fuel enthalpies (cal/gm). The trip reactivity insertion used in STRIKIN-II corresponds to an ASI of +0.6. The current methodology does not include a DNBR analysis.

The current CEA Ejection analysis performed at hot zero power conditions, inputs an axial shape with an ASI of -0.562 into STRIKIN-II to determine the fuel enthalpies (cal/gm). The AREVA calculation of fuel enthalpy does not depend upon axial shape. AREVA calculates fuel centerline temperature using a hot spot model which includes factors calculated by PRISM.

Steam Line Break

Post-trip Steam Line Break is analyzed to determine the return to power in the vicinity of an assumed stuck CEA. The event is analyzed to find the limiting cases with respect to DNB and linear heat rate SAFDLs. For hot zero power, DNBR is determined by inputting the output from the CESEC transient analysis code into HRISE for hot channel analysis using the MacBeth CHF correlation. Peak linear heat rate is assessed using the power from the CESEC transient analysis.

The axial shape used in CESEC input via the trip curve corresponds to an ASI of +0.6, but the CESEC axial shape is not of primary importance for the return-to-power analysis. For DNB analysis, the peaking factor input to HRISE consists of two factors. One peaking factor is due to the fission occurring during the approach to criticality. This peaking factor is high due to the absence of a CEA in one area of the core. This peaking factor has an axial component of 3.7456 and an ASI of +0.751. The second peaking factor is due to the production of decay power in the core. The ASI of the decay shape is about -0.129 and has an axial component of 1.235.

The trip insertion curve used in the AREVA S-RELAP5 transient calculations also mimics an insertion into a highly skewed, bottom peaked power shape.

The AOR and the AREVA post-trip Steam Line Break DNB analysis use peaking factors that have an axial component that reflects increased peaking factors due to the absence of a CEA in one area of the core.

For completeness, Calvert Cliffs UFSAR events that are either not lower power, or not impacted by the axial power distribution are listed below.

UFSAR 14.3	Boron Dilution	Core wide response. Local power distribution does not impact the event.
UFSAR 14.5	Loss of Load	Peak pressure event. Not analyzed for the transition.
UFSAR 14.6	Loss of Feedwater Flow	Pressure increases concurrent with a small increase in power. Not analyzed for the transition.
UFSAR 14.7	Excess Feedwater Heat Removal	Not analyzed at lower power.
UFSAR 14.8	Reactor Coolant System Depressurization	Not analyzed at lower power.
UFSAR 14.9	Loss-of-Coolant Flow	Not analyzed at lower power.
UFSAR 14.10	Loss-of-Non-Emergency AC Power	Not analyzed at lower power.
UFSAR 14.12	Asymmetric Steam Generator	Not analyzed at lower power.
UFSAR 14.15	Steam Generator Tube Rupture	Not analyzed for the transition.
UFSAR 14.16	Seized Rotor	Not analyzed at lower power.
UFSAR 14.25	Excessive Charging	Pressurizer fill. Not impacted by power distribution.
UFSAR 14.26	Feedline Break	Peak pressure event. Not analyzed for the transition.

Figure 20-1 shows the limiting design axial power profile that was used in the XCOBRA-IIIC calculation for all power levels. The limiting design axial power profile corresponds to an ASI of -0.30 asiu which is the local power density limiting safety system setting ASI limit at 90% rated thermal power. The limiting design axial power profile provides conservative minimum DNBR results compared to the cycle-specific limiting axial profile.

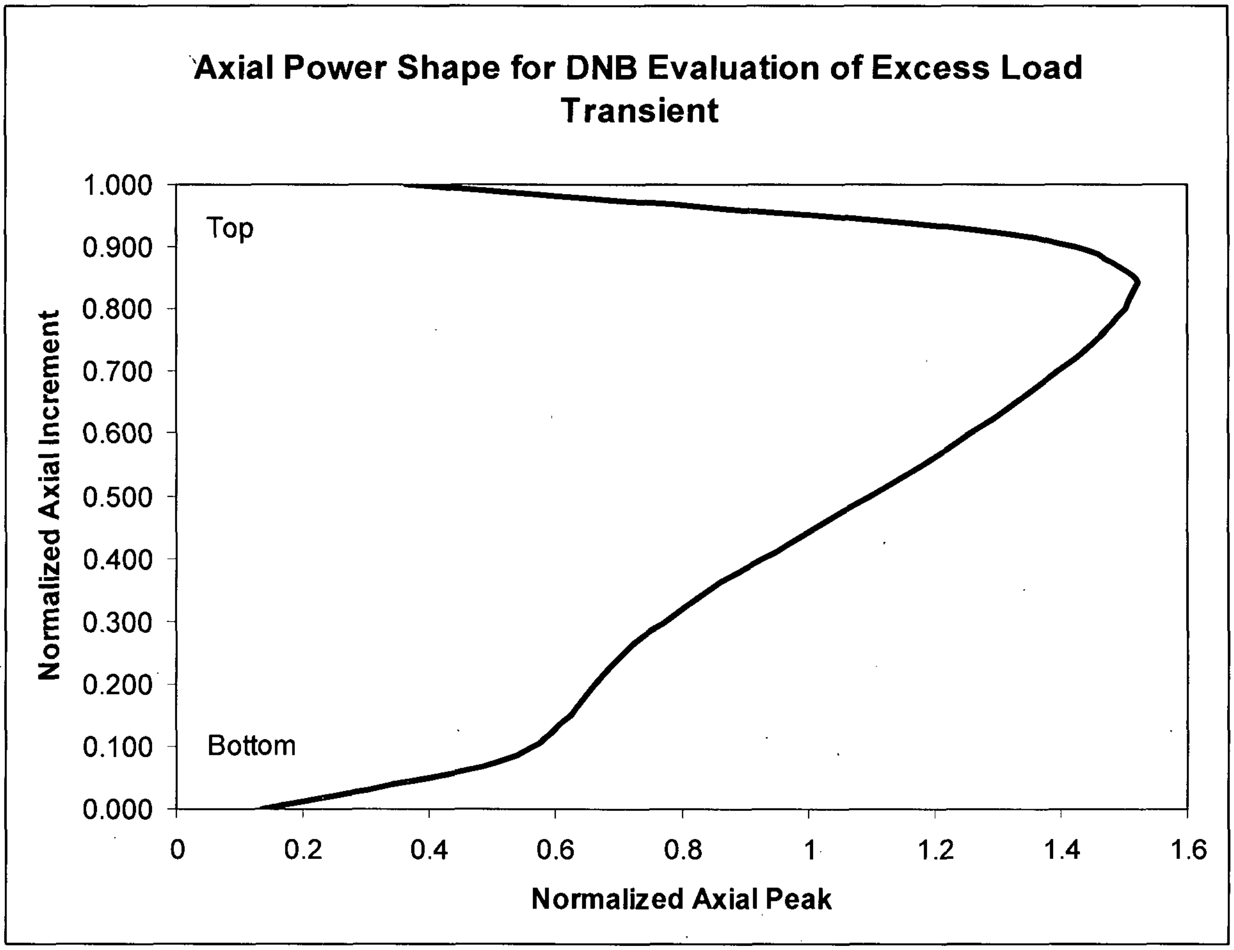

Figure 20-1, Axial Power Shape for Excess Load Event

Fuel Thermal Mechanical Design

Question 21:

Section 7.0 identifies penalties used to compensate for potential non-conservative impacts related to the lack of a fuel thermal conductivity model which accurately captures its degradation at higher exposures. The application of these penalties is outside the approved methodology listed in the proposed CCNPP Technical Specifications. Please provide a detailed description of the augmented methodology. In your description, identify the applicability of these penalties up to a peak rod power of 15 KW/ft.

CCNPP Response 21:

Response to be provided separately.

Question 22:

Additionally, because the augmented methodology is not described in documents listed in the TS COLR – References section, please ensure that the augmentation is summarized or described in an NRC-tracked manner (i.e., the NRC staff recommends adding a reference to TS COLR-References section or documenting the methodology augmentation in a Regulatory Commitment).

CCNPP Response 22:

Response to be provided separately.

Small-Break LOCA

Question 23:

[

] As such, the staff requests the following information:

a. Axial power shapes for the hot rod, hot bundle, and average core regions.
b. Were the upper core barrel and hot leg nozzle gap leakage paths included in the RELAP5 model?
c. Moderator temperature coefficient.
d. Moderator reactivity curve of reactivity vs moderator density.
e. Were the charging pumps credited in the analysis?
f. Decay heat power (fraction) vs time curve.
g. Void distribution in the hot bundle at time of PCT for the 0.09 ft^2 and 0.15 ft^2 CLBs
h. Please also explain the reasons for [

].
i. [

] Assure that the break spectrum identifies the largest break that results in RCS pressure hangup just above the SIT actuation pressure.
j. The axial power shape for the hot rod appears to be a mid peaked shape with the peak axial power just above the 6 foot elevation (node 13) vs node 20 for the remainder of the core (see Doc no. 32 – 9106667 – 001 RELAP5 SB-LOCA Base Deck Input Development. Please verify that the most top peaked axial distribution was used in the analysis, if not, please correct the shape in the re analysis for the hot rod.
k. Please also include moderator reactivity feedback effects in the SBLOCA analyses (moderator reactivity vs core density) basing the feedback curve on the most positive MTC.
l. The HPSI delivery flow rates are higher than those used in the last CE analysis, please explain the differences and verify the HPSI flow curves used in the re-analysis.
m. Please explain and justify the SIT maximum temperature of 90°F compared to 120°F used in the CE analysis. Please also explain and justify the 100°F RWST maximum temperature assumed in the analysis.
n. Please provide the results of a severed injection line and provide the values of the degrated HPSI flows to each cold leg.
o. Provide additional information regarding the 7-minute operator action that is credited to secure the reactor coolant pumps:
    a. How does operator training assure that this 7-minute action time will be executed successfully?
    b. Provide EOP revisions that incorporate this action.

CCNPP Response 23:

a. See Figure 23-1 for the power distributions in the four AREVA small break loss-of-coolant accident (LOCA) core model regions.

Figure 23-1, Core Power Distribution

b.

c.

d.

e. Charging pumps were not credited in this analysis.

f.

Figure 23-2, Decay Heat Power Fraction vs Time

Figure 23-3, Decay Heat Power Fraction vs Time

g.

Figure 23-4, Hot Bundle Void Fraction

h. Response to be provided separately.

i. Response to be provided separately.

j. [ ] Node 13 is twice the length (6 inches) of the upper core nodes (3 inches). The power fraction for node 13 is less than twice the power fraction of the peak power node (node 25: 3 inches).

k. Response to be provided separately.

l. The AREVA analysis assumed the same high pressure safety injection flow to the three intact loops as Calvert Cliffs UFSAR Table 14.17-11 shows for the AOR. [

]

m.

n. Response to be provided separately.

o. Response to be provided separately.

Fxy Surveillance Technical Specification

Question 24:

The proposed removal of TS 3.2.2, Total Planar Radial Peaking Factor (Fxy), appears to adversely affect the surveillance requirements for TS 3.2.1, Linear Heat Rate (LHR). Specifically, SR 3.2.1.1 stipulates that when monitoring the COLR LHR limit using the excore detector monitoring system, Fxy must be verified to be within specified limits every 72 hours. In addition to removing TS 3.2.2, the proposed TS change package includes the elimination of SR 3.2.1.1. No alternate means of surveillance for the LHR limit is proposed (when using the excore detector monitoring system). Please discuss the impact of removing this surveillance requirement or propose an alternative.

CCNPP Response 24:

Response to be provided separately.

REFERENCES

1. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981

2. CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods," CENPD-206-P-A, June, 1981 and SER, Acceptance for Referencing of Topical Report CENPD-206(P), "TORC Code, Verification and Simplified Modeling Methods," December 11, 1980

3. XN-NF-82-21(P)(A), Revision 1, "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations," Exxon Nuclear Company Inc., September 1983

4. Florida Power and Light, St Lucie Unit 1 Updated Final Safety Analysis Report, Section 4.4.2.4.5

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

6. Y. Nagino, et al., "Rod Bowed to Contact Departure from Nucleate Boiling Tests in Coldwall Thimble Cell Geometry," Journal of Nuclear Science and Technology, 15(8), pp. 568-573, August 1978

7. E. S. Markowski, et al., "Effect of Rod Bowing on CHF in PWR Fuel Assemblies," 77-HT-91, AICHE-ASME Heat Transfer Conference, Salt Lake City, Utah, August 15-17, 1977

8. EMF-2310(P)(A), Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," Framatome ANP, May 2004

9. EMF-2328 (P)(A), Revision 0, "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based," March 2001