Calvert Cliffs Nuclear Power Plant

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657

CENG
a joint venture of
Constellation Energy · eDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

NRC 11-001

January 14, 2011

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

REFERENCES:

(a) Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated October 29, 2010, Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

(b) Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated November 19, 2010, Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

(c) Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated December 30, 2010, Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

(d) 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. Responses to the questions were provided in References (a) and (b). After review of the responses, the NRC staff requested additional information. This supplemental information was discussed during a followup audit conducted on December 8 and 9, 2010 and some of the requested supplemental information was provided in Reference (c). Additional responses are provided in

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Attachment (1). This supplemental information does not change the No Significant Hazards determination previously provided in Reference (d).

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 Attachment (1) is included (Attachment 3).

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

Very truly yours,

Thomas E. Trepanier
Plant General Manager

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

I, Thomas E. Trepanier, being duly sworn, state that I am Plant General Manager - 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 14 day of January, 2011.

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires:

7/8/14
Date

TET/PSF/bjd

Attachments:
(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
January 14, 2011

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 T.E. Trepanier (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," numbered NRC 11-001 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 13th

day of January, 2011.

Kathleen A. Bennett

Kathleen Ann Bennett
NOTARY PUBLIC, COMMONWEALTH OF VIRGINIA
MY COMMISSION EXPIRES: 8/31/11
Reg. # 110864

KATHLEEN ANN BENNETT
Notary Public
Commonwealth of Virginia
110864
My Commission Expires Aug 31, 2011

ATTACHMENT (3)

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

Calvert Cliffs Nuclear Power Plant, LLC
January 14, 2011

Based on a review of References 1 and 2, the Nuclear Regulatory Commission (NRC) staff has identified the need for additional information. The additional questions provided by the NRC staff and our responses are below.

Question 1:

In response to Question 1b, the licensee states that a reactor trip terminates the asymmetric steam generator transient (ASGT) event prior to the development of any asymmetry at the core inlet. Based on this position, the ASGT is modeled using a uniform core inlet flow and temperature distribution. This modeling assumption essentially removes the unique aspects of the ASGT including the asymmetric core inlet temperature distribution and resulting core power tilt. The existing CCNPP (Calvert Cliffs Nuclear Power Plant) licensing basis for the ASGT specifically captures the asymmetric core inlet flow distribution. Without new information, the staff is unable to accept this change to the CCNPP licensing basis. The staff requests that the ASGT be re-analyzed using a justified asymmetric core inlet temperature distribution.

CCNPP Response 1:

Asymmetric Steam Generator Event (UFSAR Event 14.12)

Description of Analyses and Evaluations

An Asymmetric Steam Generator (SG) event is defined as any initiator that affects only one of the two SGs. The Asymmetric Loss of Load event is one of the Asymmetric SG events. The Asymmetric Loss of Load event is initiated by the inadvertent closure of a single main steam isolation valve (MSIV). This causes the pressure and temperature in the affected SG (i.e., SG associated with closed MSIV) to rapidly increase, resulting in the heatup of its associated Reactor Coolant System (RCS) loop. The single MSIV closure also isolates the affected SG from the turbine header causing the turbine header pressure to decrease. The decreased turbine header pressure then causes the unaffected SG (i.e., SG associated with open MSIV) to experience an increase in steam flow to compensate for the lost load, with its associated pressure and temperature decrease, which causes a cooldown of the unaffected RCS loop. Current operating practice is to maintain the turbine control valve flow area constant, which lessens the severity of the event. The analysis conservatively assumed that the turbine demand remains constant and that the turbine control valve opens as needed to satisfy the demand which further increases the steam flow from the unaffected SG and cooldown of that loop. The result is asymmetry in the coolant temperatures entering the reactor vessel from the affected and unaffected RCS loops. The decreasing coolant temperature in the unaffected loop combined with a large negative end of cycle moderator temperature coefficient will have a more dominant effect on core power than the increasing temperature in the affected loop combined with the highest positive moderator temperature coefficient. Thus, the analysis uses the most negative end of cycle moderator temperature coefficient. The net effect is a small increase in core power and augmented radial peaking due to the asymmetric core inlet coolant temperatures. As the SG pressures continue to diverge, a reactor trip signal is generated by the Asymmetric Steam Generator Transient trip.

The analysis of this event verifies the Asymmetric Steam Generator Transient setpoint and delay time.

Input Parameters and Assumptions

The pre-trip Steam Line Break (SLB) model described in the approved methodology, Reference 10, was used for this analysis. Consistent with the approved methodology, [

]

In the plant, mixing between the parallel affected and unaffected sectors within the reactor pressure vessel will tend to occur in the lower plenum, the core, and the upper plenum—due to lateral momentum imbalances, turbulence or eddy mixing, and the relative angular positions of the cold legs to the hot legs. Some mixing may also occur in the downcomer. Mixing and/or crossflow acts to reduce the positive reactivity feedback effects—due to a reduced rate and magnitude of cooldown of the unaffected loop. [ ]

Power fractions [ ] producing a conservative overall core power response.

Neutronic calculations are performed to determine [ ]. The asymmetric [ ] are applied in the specified acceptable fuel design limit analyses.

The input parameters and biasing for this event are shown in Table 1-1. The input parameters and biasing were consistent with the approved methodology (Reference 10).

- Initial Conditions - This event was assumed to initiate from hot full power conditions with a maximum core inlet temperature and Technical Specification minimum RCS flow. This set of conditions minimizes the initial margin to departure from nucleate boiling (DNB).
- Reactivity Feedback - The reactivity feedback was biased according to the approved methodology. [ ]
- Reactor Protective System Trips and Delays - The Asymmetric Steam Generator Transient trip setpoint and response time were conservatively biased to delay the actuation of the trip function. In addition, a maximum control element assembly (CEA) holding coil delay was assumed.
- Steam Generator Tube Plugging - Both cases assume 0% SG tube plugging [

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- MSSV Setpoints – The lowest allowable MSSV opening setpoints were used. Earlier opening of the MSSVs on the affected SG (i.e., with the closed MSIV) results in a delay of the Asymmetric Steam Generator Transient since the unaffected SG must depressurize further to initiate the trip. Delaying the trip can worsen the asymmetric effects in the core.
- Radial Peaking Augmentation – The asymmetric core inlet temperatures cause an asymmetric core power distribution. Radial peaking augmentation factors [ ] were applied to the peak rod power for the DNB calculations.

Acceptance Criteria

This event is classified as an anticipated operational occurrence, which may occur during the life of the plant.

The event can challenge the primary and secondary overpressure criteria; however, the over-pressurization consequences of this event are bounded by the Loss of Load to both SGs event where the heat removal capacity from both SGs is simultaneously lost.

This event primarily challenges the specified acceptable fuel design limits for departure from nucleate boiling ratio (DNBR) and fuel centerline melt. Fuel cladding integrity must be maintained by ensuring that the DNBR and fuel centerline melt specified acceptable fuel design limits are not exceeded. This is demonstrated by assuring that the minimum DNBR is greater than the 95/95 DNB correlation limit and that the peak linear heat rate is less than the fuel centerline melt limit.

Results

Two cases were analyzed: one case with a nearly instantaneous MSIV closure time and a second case with a maximum MSIV closure time of 7 seconds.

Table 1-2 provides the sequence of events for both cases. Figures 1-1 to 1-9 show the transient responses of key parameters for the case with an instantaneous MSIV closure time [

]

[ ], the asymmetric core inlet temperature at the time of scram, and shortly thereafter, caused augmented radial peaking. PRISM calculations based on [

] The minimum DNBR was calculated to be 1.76, which is above the 95/95 critical heat flux correlation limit. The peak linear heat rate was calculated to be 18.61 kW/ft, which is below the limit for fuel centerline melt.

Table 1-1, Asymmetric Loss of Load: Initial Conditions and Biasing

Parameter	Analysis Value
Initial Reactor Power	2754 MWt
Initial Core Inlet Temperature	548°F
Initial Reactor Coolant Flow Rate	370,000 gpm
Initial Pressurizer Pressure	2250 psia
Initial Pressurizer Level	60% span
Initial SG Pressures	890.96 psia (SG-1) 893.30 psia (SG-2)
MSIV Closure Time	≤ 7.0 sec.
MSSV Opening Setpoint	949.7 psia
Trip Reactivity	5740.8 pcm
Trip Delay Time	0.5 sec
SG Tube Plugging Level	0%
Asymmetric Steam Generator Transient Setpoint	186.0 psid
Asymmetric Steam Generator Transient delay time	0.9 sec
Moderator reactivity vs. moderator density	End of cycle (based on hot full power TS MTC limit of -33 pcm/°F)
Pressurizer Heaters	Disabled
Pressurizer Spray	Available
Pressurizer PORVs	Available
Steam Bypass Valves	Disabled
Steam Dump Valves	Disabled
Reactor Coolant Pumps	Operating
Main Feedwater	Auto Mode
Auxiliary Feedwater	Available
Charging and letdown system	Not modeled

Table 1-2, Asymmetric Loss of Load: Sequences of Events

Event	MSIV Closure	
	Instantaneous	7 sec.
	Time (sec)	Time (sec)
Initiation of event (initiation of closure of MSIV on SG-1)	0.0	[ ]
SG-1 MSIV fully closed	0.01	[ ]
MSSV flow begins for SG-1	0.10	[ ]
Asymmetric Steam Generator Transient setpoint reached	6.04	[ ]
Asymmetric Steam Generator Transient trip occurs (after 0.9 sec. delay)	6.94	[ ]
Minimum DNBR occurs	7.4	[ ]
CEA insertion begins (after 0.5 sec. delay)	7.44	[ ]

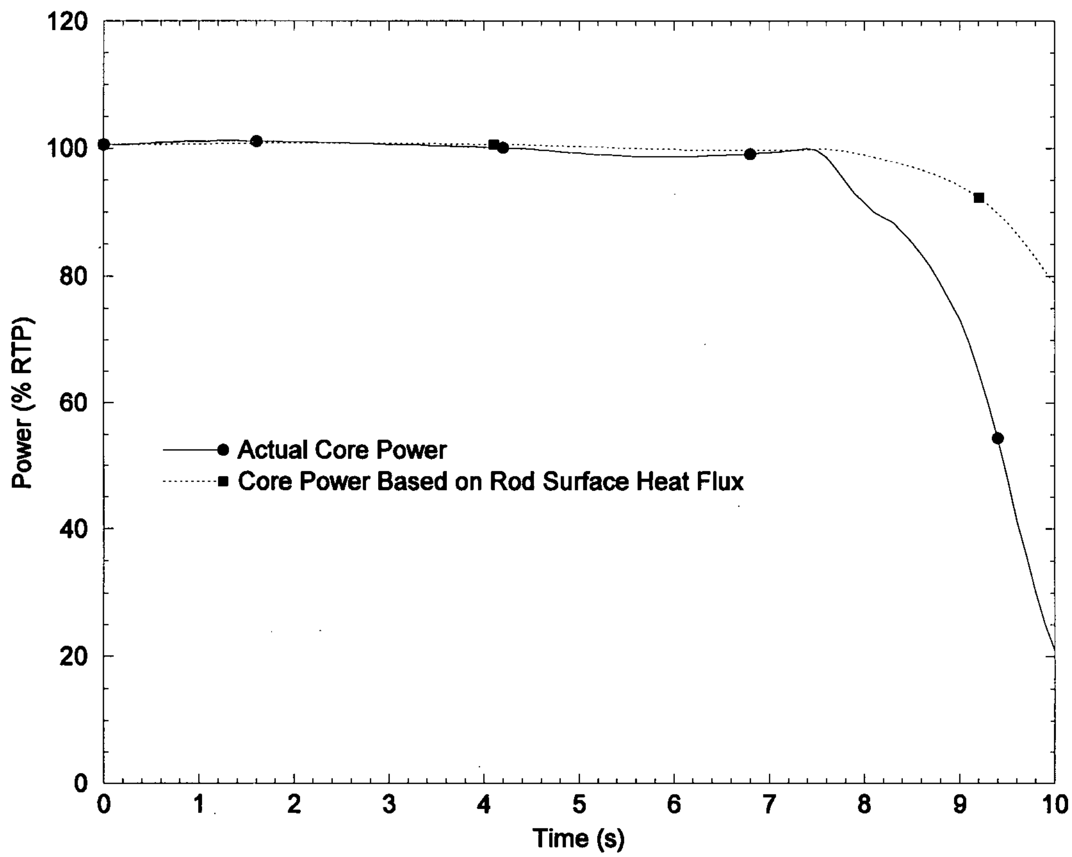

Figure 1-1, Reactor Power (Instantaneous MSIV Closure)

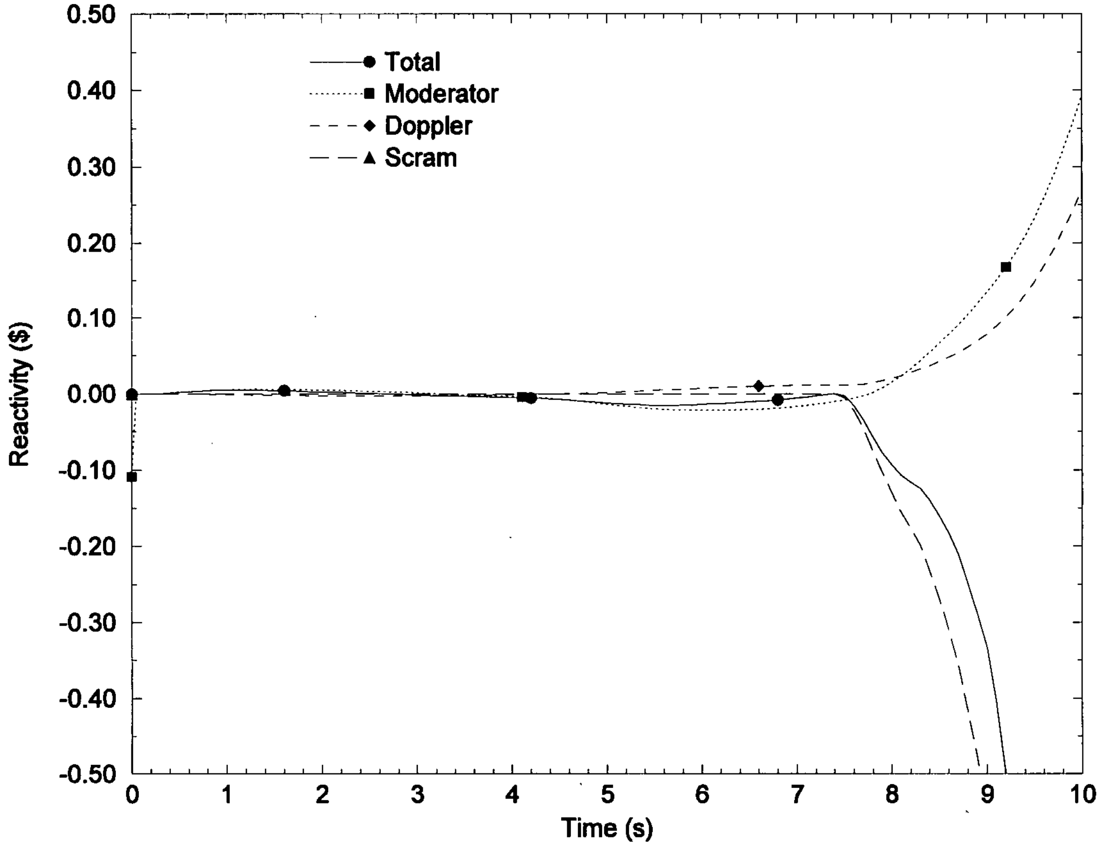

Figure 1-2, Reactivity Feedback (Instantaneous MSIV Closure)

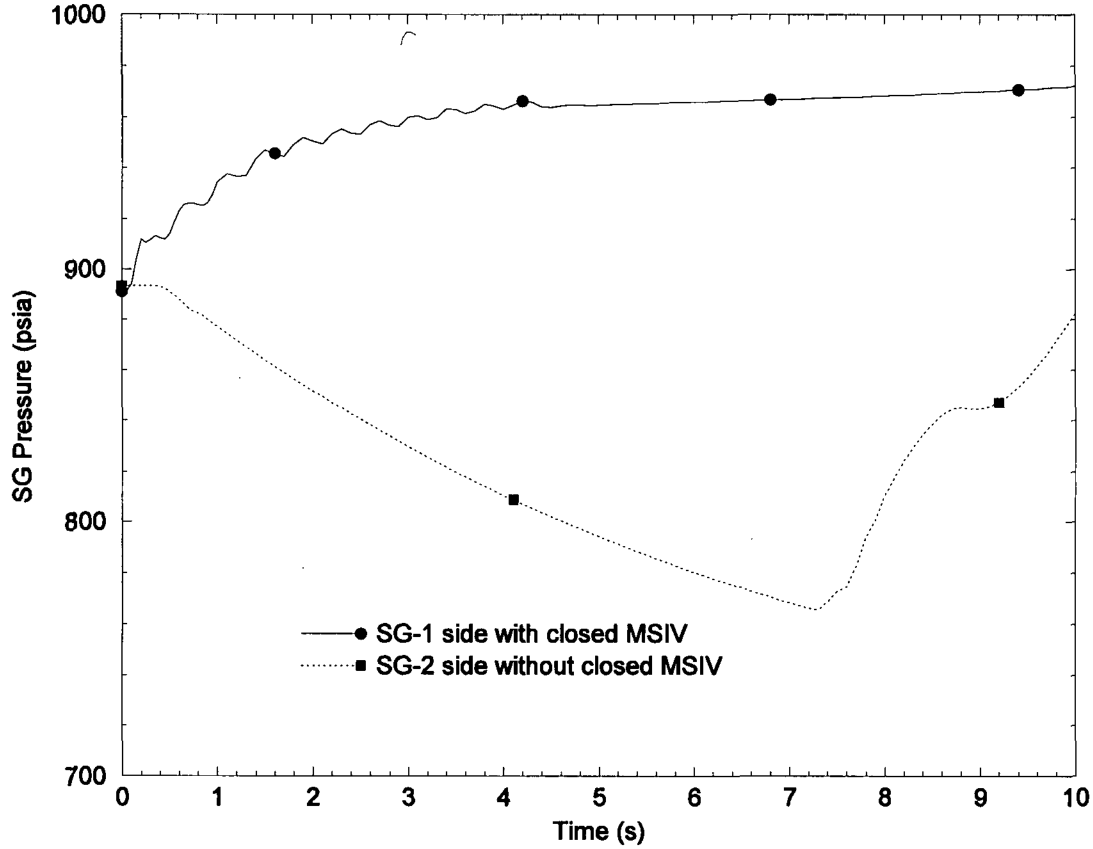

Figure 1-3, SG Pressures (Instantaneous MSIV Closure)

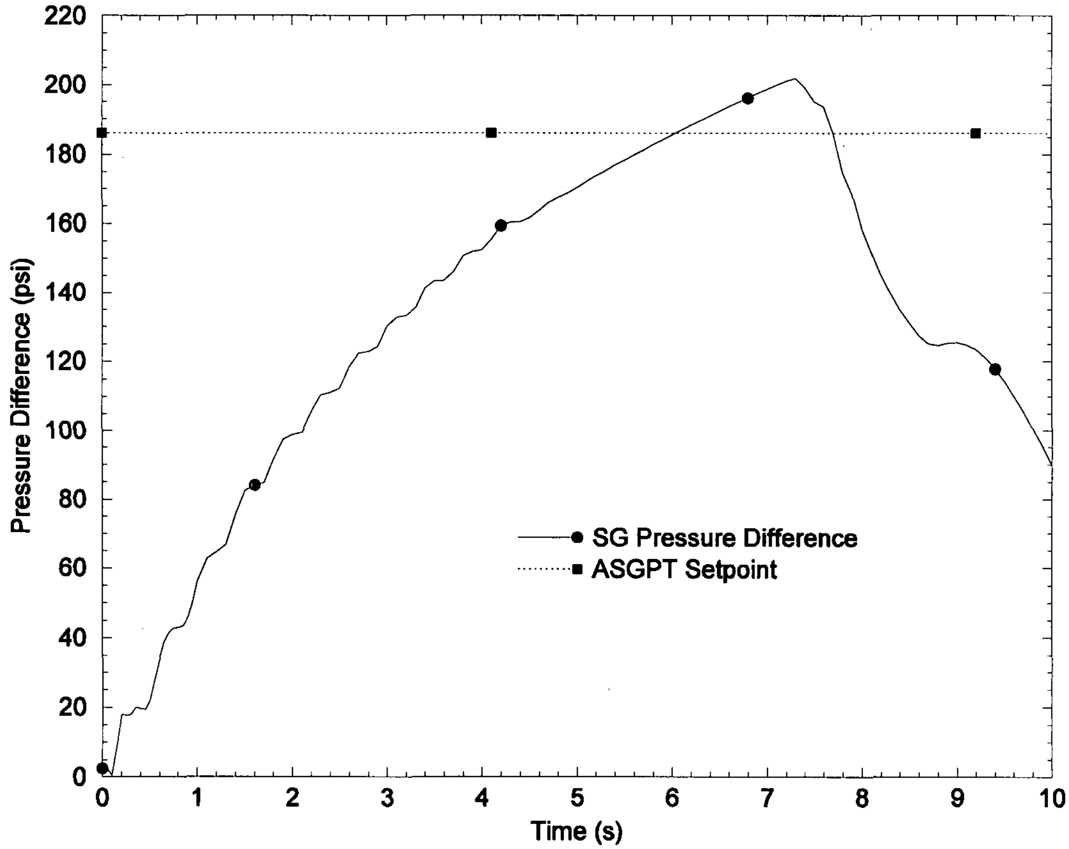

Figure 1-4, SG Pressure Difference vs. Asymmetric Steam Generator Transient Setpoint (Instantaneous MSIV Closure)

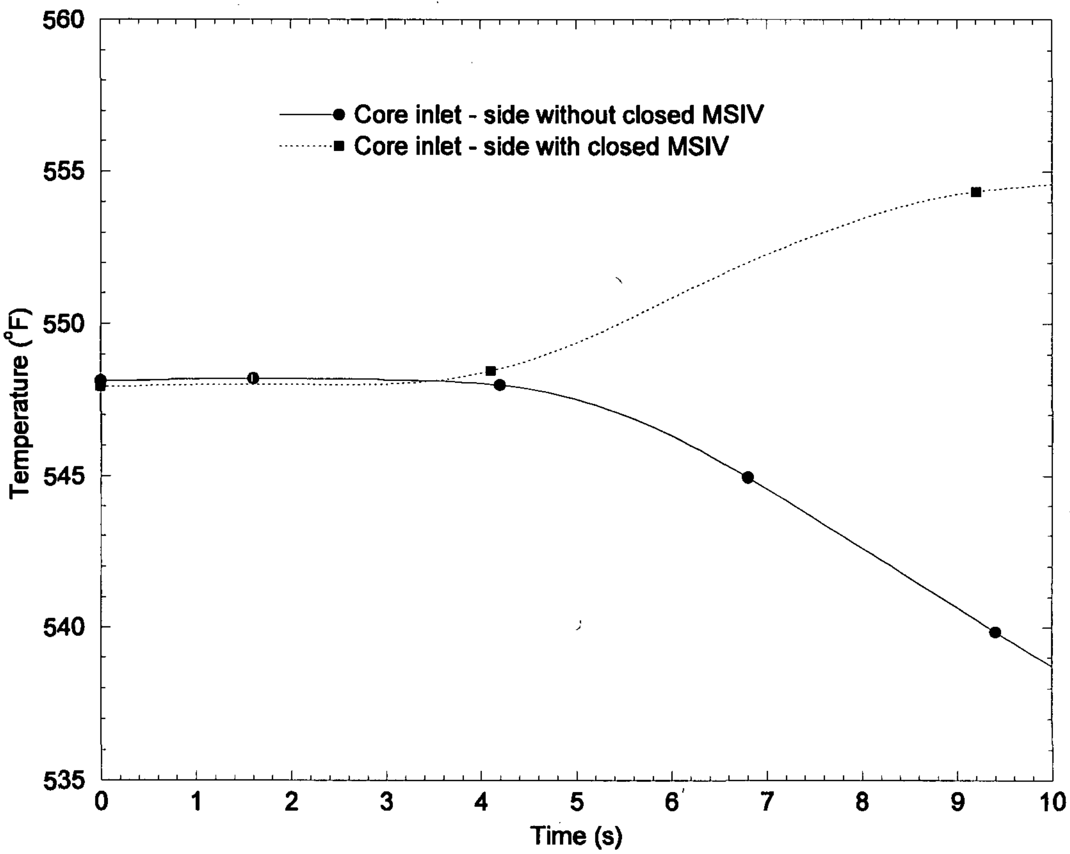

Figure 1-5, Core Inlet Temperatures (Instantaneous MSIV Closure)

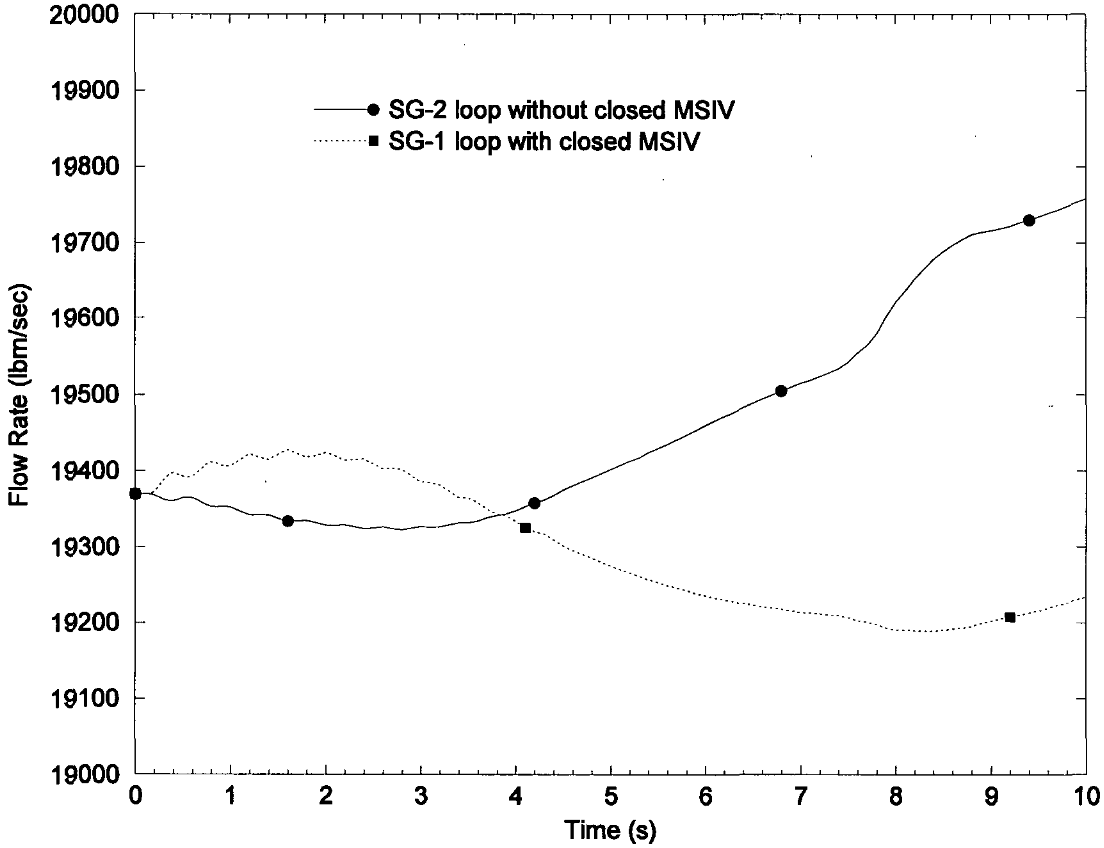

Figure 1-6, RCS Loop Flow Rates (Instantaneous MSIV Closure)

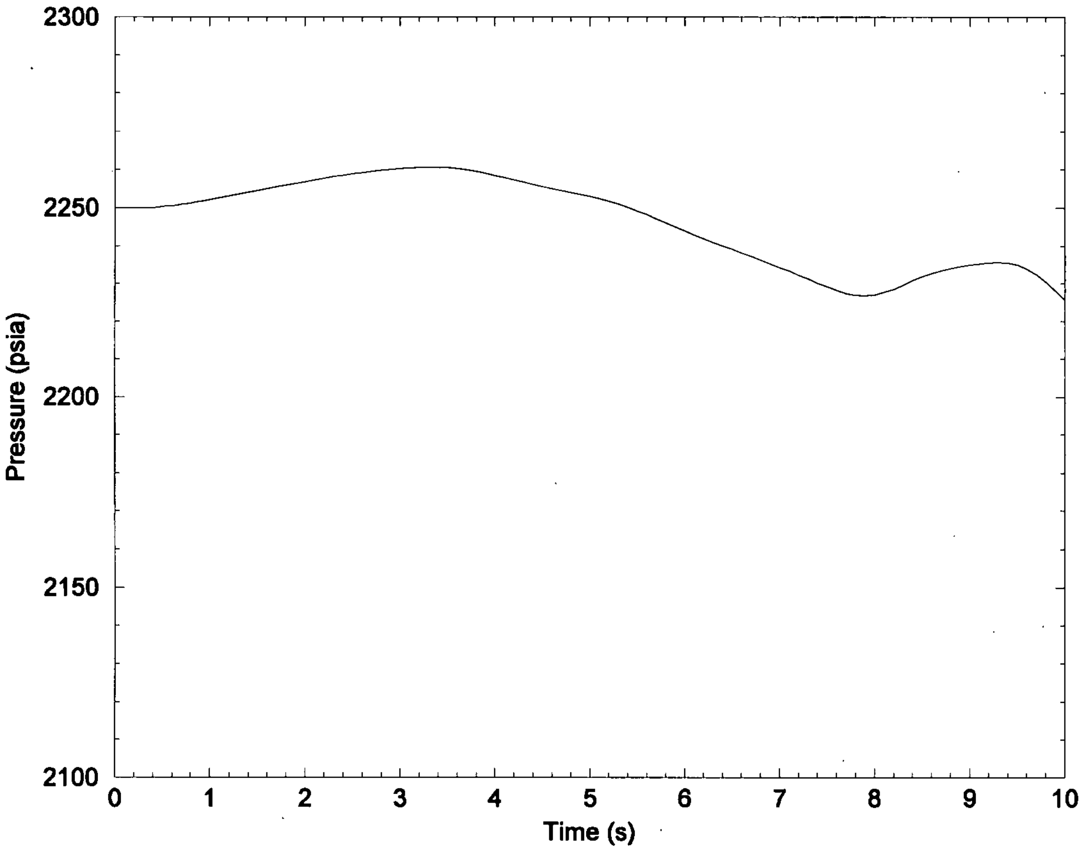

Figure 1-7, Pressurizer Pressure (Instantaneous MSIV Closure)

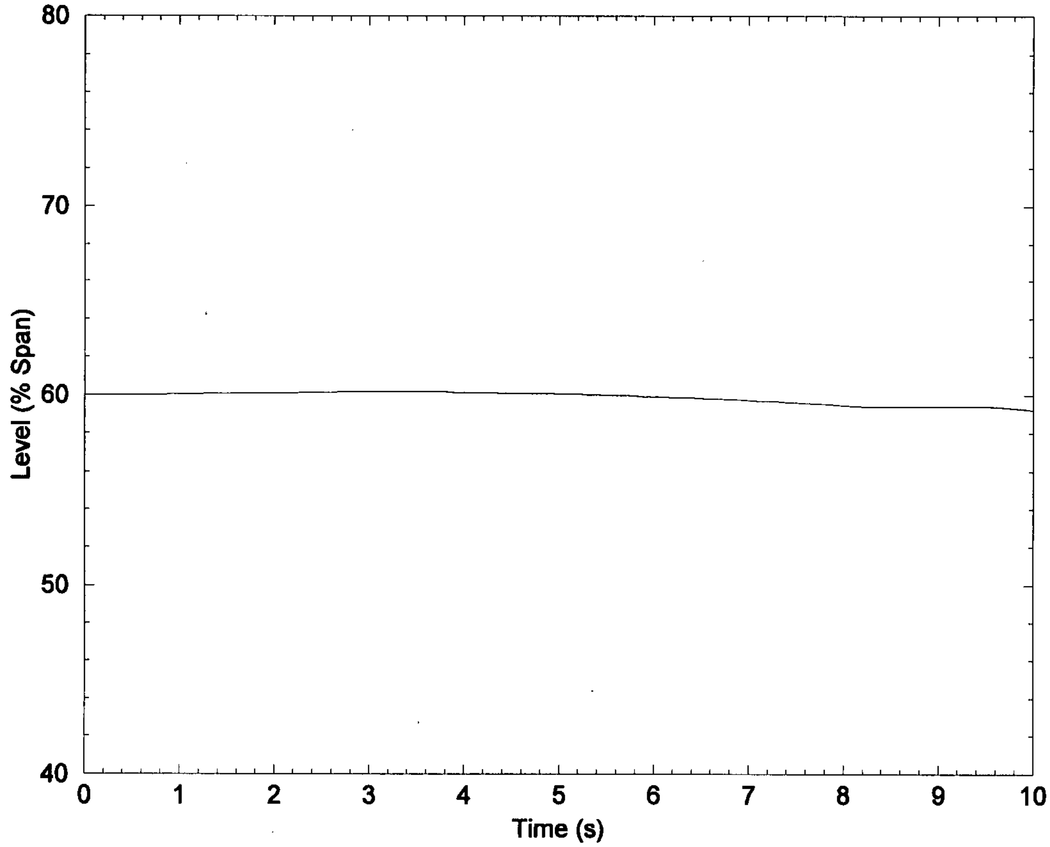

Figure 1-8, Pressurizer Level (Instantaneous MSIV Closure)

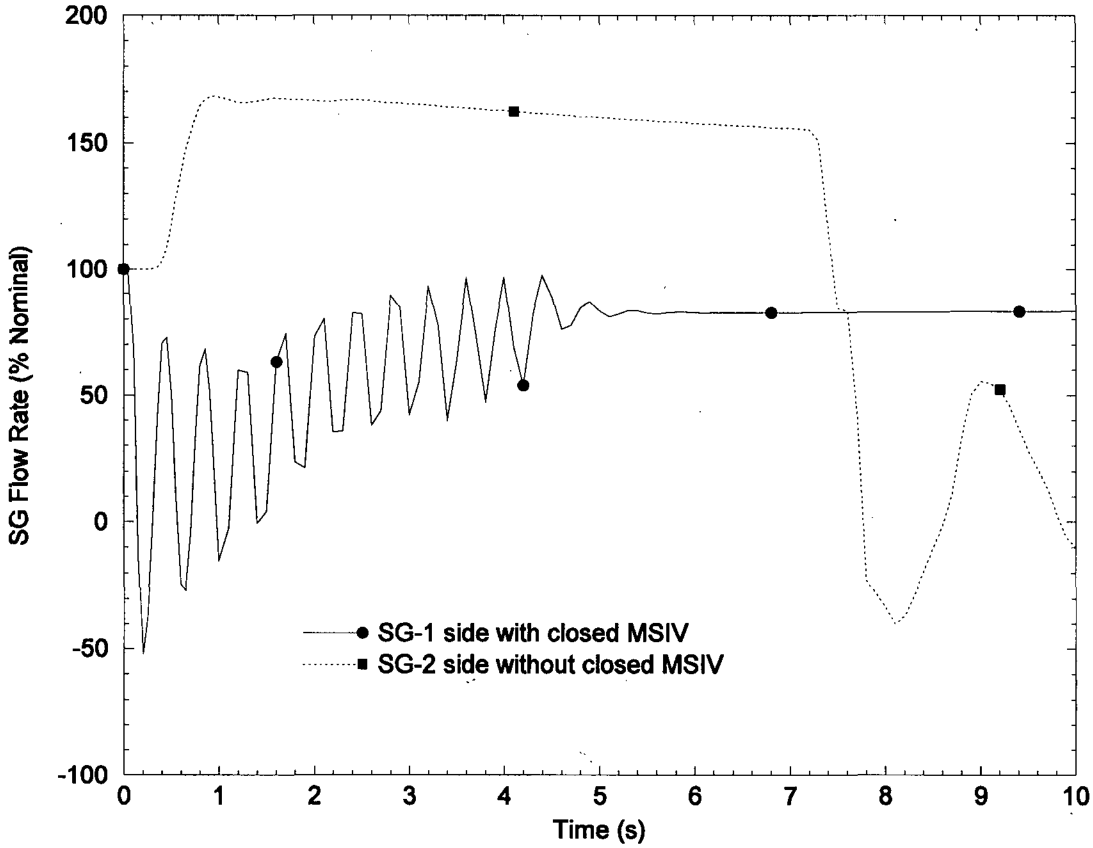

Figure 1-9, Steam Flow Rates (Instantaneous MSIV Closure)

Figure 1-10, Reactor Power (MSIV Closure = 7 sec.)

[ ]

Figure 1-11, Reactivity Feedback (MSIV Closure = 7 sec.)

[ ]

Figure 1-12, SG Pressures (MSIV Closure = 7 sec.)

[ ]

Figure 1-13, SG Pressure Difference vs. Asymmetric Steam Generator Transient Setpoint (MSIV Closure = 7 sec.)

[ ]

Figure 1-14, Core Inlet Temperatures (MSIV Closure = 7 sec.)

[ ]

Figure 1-15, RCS Loop Flow Rates (MSIV Closure = 7 sec.)

[ ]

Figure 1-16, Pressurizer Pressure (MSIV Closure = 7 sec.)

Figure 1-17, Pressurizer Level (MSIV Closure = 7 sec.)

[ ]

Figure 1-18, Steam Flow Rates (MSIV Closure = 7 sec.)

Question 2:

In response to Questions 1a and 7, the licensee states that "modeling assumptions for flow mixing in the lower plenum do not have a first order effect on the minimum DNBR." The response also states that inlet flow distributions are "washed out quickly" in an open lattice PWR core. Based on this position, the single Reactor Coolant Pump Locked Rotor event is modeled using a uniform core inlet flow and temperature distribution. This modeling assumption essentially removes the unique aspects of the asymmetric core inlet flow distribution resulting from a coast down from 4-pump to 3-pump conditions. The existing Calvert Cliffs licensing basis for the LR event specifically captures the asymmetric core inlet flow distribution. Without new information, the staff is unable to accept this change to the CCNPP licensing basis. The staff requests that the Locked Rotor minimum DNBR be re-calculated using the existing 3-pump limiting assembly inlet flow factor.

CCNPP Response 2:

Response provided in Reference 3.

Question 3:

Based on the licensee's letter dated November 19, 2010, the staff has identified the need for additional information. In response to Question 2d, AREVA appears to have run some RODEX-2 cases for the CEA drop transient. The maximum cladding strain is calculated to be 0.78% strain at 16.4 GWd/MTU. The CEA drop case was selected because it exhibited the "peak attainable linear heat rate." A peak

LHGR is limiting with respect to fuel centerline temperature. However, the maximum change in LHGR is limiting with respect to calculated cladding strain. For each AOO and accident, please provide the peak calculate cladding strain along with the pre- and post-LHGR. Specify radial (F_r) and axial power peaking (F_z) components.

CCNPP Response 3:

The anticipated operational occurrences applicable to Calvert Cliffs have been reviewed by AREVA. They include both the hot full power and the hot zero power events. The hot zero power events were examined by AREVA based on additional questions from the NRC subsequent to the initial set of responses that were sent to the NRC. The initial response was limited to the hot full power events. The events selected for cladding strain calculation are – [ ]. Note that the other anticipated operational occurrences such as [ ] are bounded by the events mentioned above in terms of peak linear heat generation rate and incremental power and are therefore not explicitly analyzed. Also note that the [ ] was analyzed as an anticipated operational occurrence even though it is a postulated accident. However, a review of the event shows that it is bounded by the [ ] event in terms of peak linear heat generation rate and incremental power and is therefore not explicitly analyzed. The [ ] event appears to be relatively benign compared to the [ ] event with respect to the energy deposited in the fuel during the event as is evident from the significantly lower peak centerline temperatures experienced by the fuel. This would indicate lower peak volumetric average fuel temperatures and therefore lower peak cladding strains resulting from the event. The [ ] event is therefore not explicitly analyzed and is considered to be bounded by the [ ].

Cladding strain for anticipated operational occurrences is calculated with the RODEX2 fuel performance code using the NRC-approved methodology in Reference 5. Each of the selected hot full power events is modeled in RODEX2 [ ]. The power factors for the three selected hot full power events are calculated as shown below.

Anticipated operational occurrence strain analysis is then run for each of these events. Maximum cladding strain results are reported in Table 3-1 for each event along with the pre and post-transient linear heat generation rate values. Results are reported for the 4.34% UO_2 rod and the 2.93% UO_2 enriched 8 wt% Gd rod. These results correspond to power histories that were transmitted to the NRC in Reference 11.

Table 3-1, Results for Hot Full Power Events

The [ ] is a rapid transient with the duration of the spike lasting only a few seconds. [ ].

[ ]. Maximum cladding strain results are reported in Table 3-2 for the 4.34% UO_2 rod and the 2.93% UO_2 enriched 8 wt% Gd rod. These results correspond to power histories that were transmitted to the NRC in Reference 11.

Table 3-2, Results for [ ]

Conservatisms in AOO Strain Analysis

Question 4.a:

The response to Question 3 does not provide sufficient information to address the staff's concerns. The staff has prepared this request for follow-up information using two examples: the quasi-steady state CEA withdrawal error at power (CWAP) and the transient CEA drop events.

a. CWAP

The CWAP transient is analytically terminated by the variable high power trip (VHPT). It is asserted that the limiting power ascension occurs with the zero-power transient because the trip ceiling at that point provides for the longest trip delay, and with reactor power greater than 30-percent, the trip setpoint is 10-percent greater than the reactor power level.

The response provided considers various phenomena as separate effects, including perturbations in peaking factor, core response characteristics, fractional power level, and setpoint methodology conservatisms. The response asserts that each of these effects provides sufficient conservatism in the analysis to assure that a hot zero power and a hot full power analytic case are bounding of power levels in between. The conservatism is unquantified and is not supported with analytic examples.

Please perform a sensitivity analysis of the CWAP transient to demonstrate the effectiveness of the VHPT as a function of reactor power.

It is stated, "The [ ] is proportional to the fraction of power, so that at powers below 60% this effect overshadows the increases in local peaking from the axial shape index and CEAs at lower powers." Demonstrate that this is true. At each power level, consider limiting achievable initial axial shape indices (ASIs) and address the attendant DNBR effects. Also include appropriate consideration of the transient radial power redistribution.

CCNPP Response 4.a:

Part-power CEA Withdrawal at power events can potentially challenge the specified acceptable fuel design limits for some plant designs due to specific Reactor Protective System trip functions in combination with allowed peaking factors and axial shapes. For Combustion Engineering designed plants like Calvert Cliffs, the Power Level-High trip minimizes the challenge to the specified acceptable fuel design limits for part-power initiated events. For decreasing powers, the Power Level-High trip is designed to automatically maintain a variable trip setpoint corresponding to the core power level plus a fixed power offset (e.g., typically about 10%). As indicated core power increases, the Power Level-High trip setpoint remains fixed at the lowest setting prior to power ascension. Operator action is needed to reset the Power Level-High trip setpoint during controlled power ascensions. Thus, for power ascension during a transient event, the Power Level-High trip remains fixed at the Power Level-High trip setpoint corresponding to the initial indicated core power level plus the power offset. The function of the Power Level-High trip is to limit the consequences of events like a CEA Withdrawal at power, especially when initiated from part-power conditions.

To demonstrate the effectiveness of the Power Level-High trip, part-power CEA Withdrawal at power sensitivity analyses were performed for a reference Combustion Engineering plant of the same fundamental design as Calvert Cliffs. Sensitivity calculations using the S-RELAP5 computer code to model system performance were performed at several part-power initial conditions. Core inlet temperatures and the Power Level-High trip setpoints were set according to each initial part-power level. The S-RELAP5 sensitivity calculations modeled reactivity feedback conditions spanning maximum to minimum feedback, and several reactivity insertion rates for each part-power initial condition to determine the limiting conditions. Boundary conditions from the S-RELAP5 system analysis were generated for each case for subsequent DNB and peak linear heat rate analyses.

Figures 4a-1 to 4a-4 show a comparison of the transient response for the core power (based on rod surface heat flux), core inlet temperature, RCS total loop flow rate and core exit pressure from the sensitivity analyses at various initial condition power levels (25%, 50%, 75%, and 100%). For this example, the S-RELAP5 reactivity insertion rate was 9.0 pcm/sec with end of cycle reactivity feedback.

Figure 4a-1 shows the responses of core power (based on rod surface heat flux) vs. time. Figure 4a-1 demonstrates that the Power Level-High trip effectively limits part-power transient power excursions resulting from a CEA Withdrawal at power event. Figure 4a-2 shows the core inlet temperature response vs. time. For part-power initial conditions, the plant's cold leg temperature vs. reactor power program decreases the initial core inlet temperatures relative to full power operation which tends to increase thermal margins for part-power operation. Figure 4a-3 shows that the RCS flow transient responses are similar for each power level (note that the flow rates in the S-RELAP5 analyses conservatively did not account for density differences with decreasing initial core inlet temperature as a function of power). Figure 4a-4 shows that the core exit pressure responses also are similar for each power level (note that reactor trip on Pressurizer Pressure-High coincident with reaching the PORV setpoint was conservatively ignored for the purpose of predicting conservative thermal margins). Based on a comparison of the transient response of the system transient boundary conditions used in the DNB and peak linear heat rate analyses, thermal margin tends to increase as initial power decreases.

To evaluate the effect of part-power radial peaking and axial shapes, the system transient boundary conditions from the S-RELAP5 sensitivity analyses were used in subsequent DNB and peak linear heat rate analyses. Key parameters affected by part-power operation for the DNB and peak linear heat rate analyses include radial peaking factors and axial shapes. The DNB and peak linear heat rate sensitivity calculations were run for the reference plant. The radial peaking and axial shape index limiting safety system setting functions for the reference plant are shown in Figure 4a-5 and Figure 4a-6, respectively. The sensitivity analyses demonstrated that the CEA Withdrawal at power thermal margins for DNB increase as the initial power level decreases (Table 4a-1). For the reference plant sensitivity analysis, the peak linear heat rate margin at [

].

The trend in DNB and peak linear heat rate from the reference plant sensitivity analysis is evaluated to be applicable to Calvert Cliffs based on comparisons of key parameters. Table 4a-2 compares general plant parameters between the reference plant and Calvert Cliffs. Relative to the system transient response to a CEA Withdrawal at power event for a given part-power initial condition, the key parameters that affect the timing of the Power Level-High trip are: the power offset, the Power Level-High trip delay time, the Power Level-High trip floor power, and the Power Level-High trip ceiling power. Table 4a-3 provides a comparison of these parameters for the reference plant and Calvert Cliffs. The Power Level-High power offset value for Calvert Cliffs is 10% whereas the value for the reference plant is [

] Figure 4a-7 depicts the Technical Specification Power Level-High trip functions for the reference plant and Calvert Cliffs. [

]

Figure 4a-8 shows that the cold leg temperature vs. reactor power program for Calvert Cliffs and the reference plant [ ].

Figure 4a-9 shows a comparison of the calorimetric power measurement uncertainties. [

]

The axial power shapes [

] Figure 4a-6 compares the axial shape index limiting safety system setting functions for the reference plant and Calvert Cliffs. [

]

A sensitivity analysis was performed [

]

Based on results of the CEA Withdrawal at power sensitivity analysis for a reference plant, accounting for differences for Calvert Cliffs, it is concluded that the thermal margins for a CEA Withdrawal at power event generally increase with decreasing power; thus demonstrating the effectiveness of the Power Level-High trip function.

To examine the F_r dependence during a CEA Withdrawal event, sensitivity calculations are presented in which the CEA bank is withdrawn in ~14 inch increments until fully withdrawn. Since the CEA Withdrawal could occur anywhere within the Technical Specification limits during the cycle, the CEA bank is withdrawn at a matrix of conditions representing these possible combinations. These CEA bank withdrawals are simulated at powers of 70%, 97%, and 110%, for beginning of cycle, end of cycle, and middle of cycle; assuming equilibrium Xenon, top-skewed Xenon, and bottom-skewed Xenon. The F_r for all the cases that are within the power dependent insertion limits at the respective power are shown in Figure 4a-10. Also, the Technical Specification limit on F_r versus power, augmented by the F_r factor for rods at the power dependent insertion limit (F_r*F_{PDIL}) is shown on this figure. All of the possible conditions are less than the assumption of peaking used in the transient thermal analysis (F_r*F_{PDIL}). If the CEAs stayed at the 70% power insertion dependent limit position and power increased to 97% or 110% power, the F_r for those conditions are shown in Figure 4a-11. All of the possible conditions are less than the assumption of peaking used in the transient analysis. Therefore, regardless of the power level, axial shape index, burnup or rod configuration, the F_r peaking used in the transient thermal analysis bounds the calculated values.

Table 4a-1, DNB and Peak Linear Heat Rate Results for CEA Withdrawal at Power Sensitivity Analysis (Reference Plant)

	Beginning of Cycle		End of Cycle	
Core Power	**Minimum DNBR**	**Peak LHR (kW/ft)**	**Minimum DNBR**	**Peak LHR (kW/ft)**
25% RTP	[ ]	[ ]	[ ]	[ ]
50% RTP	[ ]	[ ]	[ ]	[ ]
75% RTP	[ ]	[ ]	[ ]	[ ]
100% RTP	1.427	17.7	1.239	17.7

Table 4a-2, Comparison of General Plant Parameters

Parameter	Reference Plant	Calvert Cliffs
Plant Design	CE 2 x 4	CE 2 x 4
Fuel Design	CE 14x14	CE 14x14
Total Number of Fuel Assemblies in the Core	217	217
Rated Core Power	[ ]	2737 MWt
Maximum Core Inlet Temperature	[ ]	548°F
RCS Flow Rate (TS minimum flow, incl. uncertainty)	[ ]	370,000 gpm
Core bypass flow	[ ]	3.9%
Pressurizer Pressure	[ ]	2250 psia
Pressurizer Level (hot full power)	[ ]	60.0%
Moderator Temperature Coefficient	[ ]	+ 1.5 pcm/°F (100% RTP) + 7.0 pcm/°F (≤ 70% RTP) -33 pcm/°F (HFP)

Table 4a-3, Comparison of Power Level-High Trip Parameters

Parameter	Reference Plant	Calvert Cliffs
Power Level-High Setpoint (ceiling)	107% (TS value) [ ]	107% (TS value) 110.3% (analytical value)
Power Level-High Setpoint (floor)	[ ] [ ]	30% (TS value) 40% (analytical value)
Power Level-High Power Offset	[ ]	10%
Power Level-High Delay	0.4 sec.	0.4 sec.

Table 4a-4, DNB and Peak Linear Heat Rate Results for 25% RTP CEA Withdrawal at Power with 40% RTP Power Level-High Floor

	Beginning of Cycle		End of Cycle	
Core Power	Minimum DNBR	Peak LHR (kW/ft)	Minimum DNBR	Peak LHR (kW/ft)
25% RTP (40% Power Level-High floor)	[ ]	[ ]	[ ]	[ ]

[ ]

Figure 4a-1, Core Average Power Based on Rod Surface Heat Flux (Reference Plant)

Figure 4a-2, Core Inlet Temperatures (Reference Plant)

[

]

Figure 4a-3, RCS Mass Flow Rates (Reference Plant)

[ ]

Figure 4a-4, Core Exit Pressures (Reference Plant)

[ ]

Figure 4a-5, Radial Peaking Limits vs. Power

Reference Plant		Calvert Cliffs	
RTP (%)	F_r	RTP (%)	F_r
100	[ ]	100	1.65
82.3	[ ]	80	1.7325
20	[ ]	20	1.819
5	[ ]	5	1.819

[ ]

Figure 4a-6, Allowed Rated Thermal Power vs. Axial Shape Index

Reference Plant		Calvert Cliffs	
ASI	RTP (%)	ASI	RTP (%)
-0.6	[ ]	-0.6	15
-0.6	[ ]	-0.6	40
-0.2	[ ]	-0.2	100
0	[ ]	0	117
0.2	[ ]	0.2	100
0.6	[ ]	0.6	40
0.6	[ ]	0.6	15

[ ]

Figure 4a-7, Power Level-High Function (Technical Specification)

	Power Level-High Setpoint (from Table 4a-3)	
RTP (%)	Reference Plant	Calvert Cliffs
120	[ ]	107
100	[ ]	107
50	[ ]	60
40	[ ]	50
30	[ ]	40
20	[ ]	30
15	[ ]	30
12	[ ]	30
10	[ ]	30
5	[ ]	30
0	[ ]	30

[ ]

Figure 4a-8, Cold Leg Temperature vs. Reactor Power Program

	Cold Leg Temperature (°F)	
RTP (%)	Reference Plant	Calvert Cliffs
100	[ ]	548
0	[ ]	532

[ ]

Figure 4a-9, Calorimetric Power Measurement Uncertainty

Reference Plant		Calvert Cliffs	
RTP (%)	Power Measurement Uncertainty (%)	RTP (%)	Power Measurement Uncertainty (%)
100	[ ]	100	0.6211
75	[ ]	95	0.82
25	[ ]	85	0.84
0	[ ]	30	1.98
		20	3.14
		15	3.74

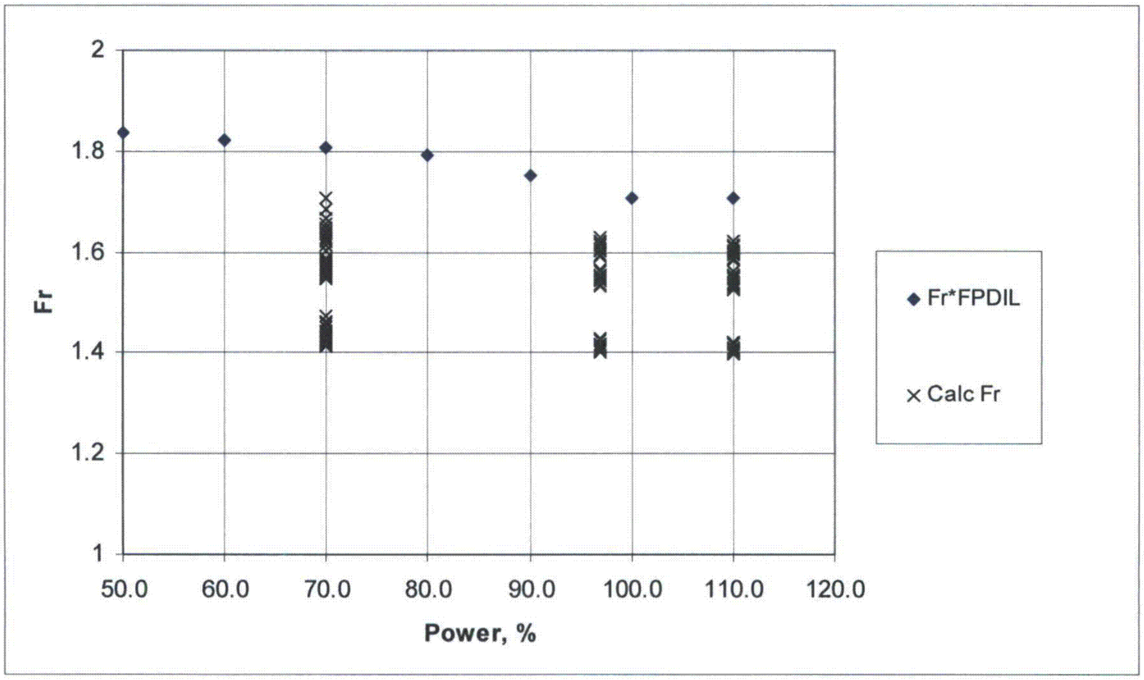

Figure 4a-10, F_r for CEA Withdrawal Events

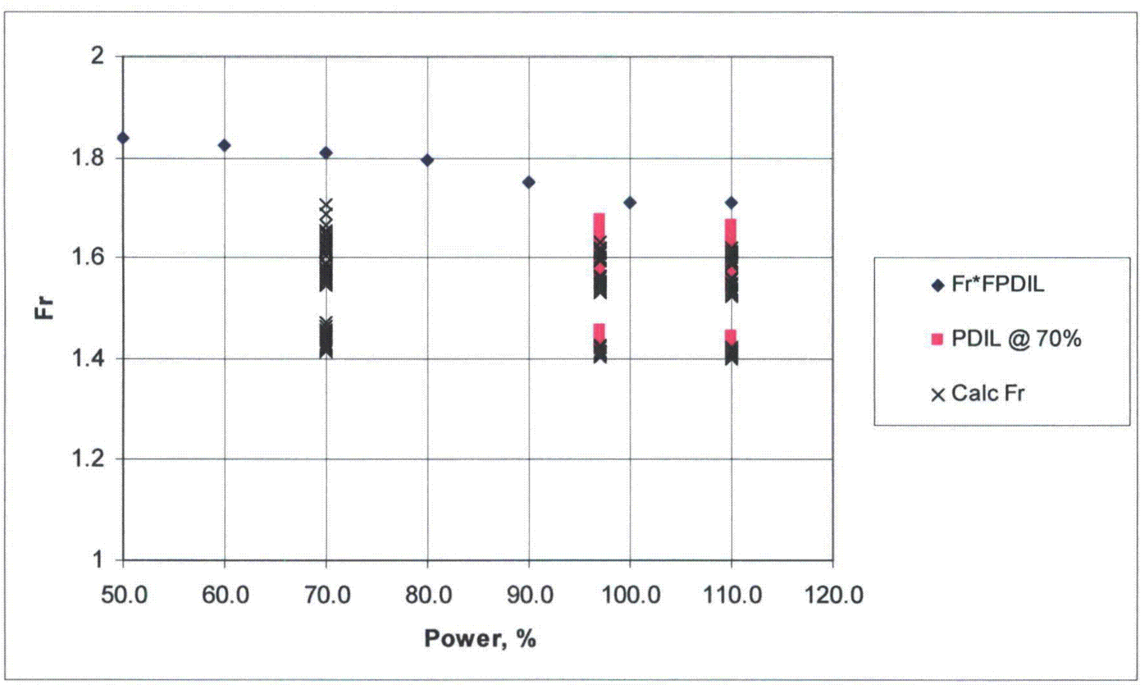

Figure 4a-11, F_r for CEA Withdrawal Events at a 70% Power Dependent Insertion Limit

Question 4.b:

b. CEA Drop

The CEA drop event is evaluated for both minimum DNBR and for fuel centerline melt. The peak linear heat generation rate is predicted based on a steady-state evaluation of the end state power level, and a linear heat rate calculation factoring in the maximum allowable Technical Specification peaking factor values. The CEA drop event, however, causes a transient change to the peaking factors.

While the response to RAI 3 states that setpoint verification calculations account for transient variations in local power distribution, this transient is unmitigated by a reactor trip. The response to RAI 3 also asserts that proportionality in linear heat rate provides adequate margin to SAFDLs to assure that lower-power transients remain bounded by the HFP analyses.

Demonstrate that the above discussed phenomenology holds true for the CEA Drop transient. Analyze the hot full-power sensitivities to dropped rod worth, initial and final power level, and limiting power distributions and redistributions at lower power levels. Confirm that the limiting results from these studies are bounded by the return to power and radial peaking augmentation factors obtained from the full-power neutronic analyses [ ].

CCNPP Response 4.b:

The CEA Drop event is analyzed at hot full power conditions to estimate the closest approach to the specified acceptable fuel design limits. The transient simulation demonstrates that the initial power level and initial inlet temperature conservatively bounds the transient response versus time so that a steady state evaluation can be performed to estimate the peaking effects. An augmentation factor representing the increase in F_r due to the CEA Drop event is calculated at full power to define limiting conditions for the event. This maximum factor is obtained from any single CEA dropped from either the power dependent insertion limit position, or CEAs fully withdrawn at beginning of cycle and end of cycle conditions. To demonstrate the adequacy of this method, the maximum augmentation factors for the CEA Drop event are similarly generated at 80 and 20 percent power. These factors are shown in Figure 4.b-1. The behavior is nearly linear with power and the augmentation factor increases slightly as power decreases (~1.5% increase from 100 to 80% power). The peaking increase due to a dropped CEA is primarily a 2-D effect since the largest effect is for a fully withdrawn CEA that is dropped. Figure 4.b-2 illustrates the major F_r peaking effect versus power. The first line represents the Technical Specification F_r times the F_r multiplier for the power dependent insertion limit. The second line represents the previous line times the peaking augmentation factor for a dropped CEA versus power from Figure 4.b-1. The third line represents how large that F_r would have to be at the power level, adjusted for power measurement uncertainty, to have the same average kW/ft as the pin analyzed at full power. The distance between the second and third line represents the difference in the average pin power between the full power case and a case that would be run at a lower power. This verifies that the hot full power dropped CEA peaking will be closer to the specified acceptable fuel design limit than the lower power conditions.

The allowed F_q also is power dependent. As power decreases, the allowed axial shape index band becomes larger. The increase in F_q that accompanies larger absolute axial shape index values is quantified by generating additional power distributions at 40% and 50% power. These are added to the normal set of cases generated in the reload process that represent possible initial conditions. The coverage of the peaking shapes of the axial shape index bands and power levels relative to the limiting safety system setting axial shape index limits are shown in Figure 4.b-3. Even though these shapes are outside the trip envelope for axial shape index and power, the maximum F_z from all these shapes are obtained. This maximum F_z is multiplied by the allowance of F_r with power to obtain the maximum F_q as a function of power. The power dependency of this maximum F_q and the fuel centerline melt limit in terms of F_q is illustrated in Figure 4b-4. The setpoint analysis shows that zero margin occurs above 95% power, which is illustrated in Figure 4.b-4, even though this F_q simulation does not include other uncertainties considered in the setpoints such as axial shape index errors. As the power varies below 60%, the distance increases significantly between the calculated peak and the peaking limit which verifies the adequacy of margins for the Technical Specification and limiting safety system setting axial shape index limits at lower powers. If the worst dropped CEA peaking effect at 20% occurred at 40% power, the new F_q would be ~5.3 in Figure 4b-4, which still remains substantially below the 7.67 limit line for fuel centerline melt (31%). So, for the CEA Drop event, the limiting safety system setting limits (not the Technical Specification limits) could have been assumed and the dropped CEA peaking effect at 20% power and still would be 31% away from the fuel centerline melt limits. The full power case at the Technical Specification limits is [ ] from the fuel centerline melt limit which makes it the most limiting condition. More than 40% margin exists at lower power because the limit increases inversely with fraction of power and the other effects are nearly linear. Hence, the hot full power dropped rod results provides the closest approach to the specified acceptable fuel design limits.

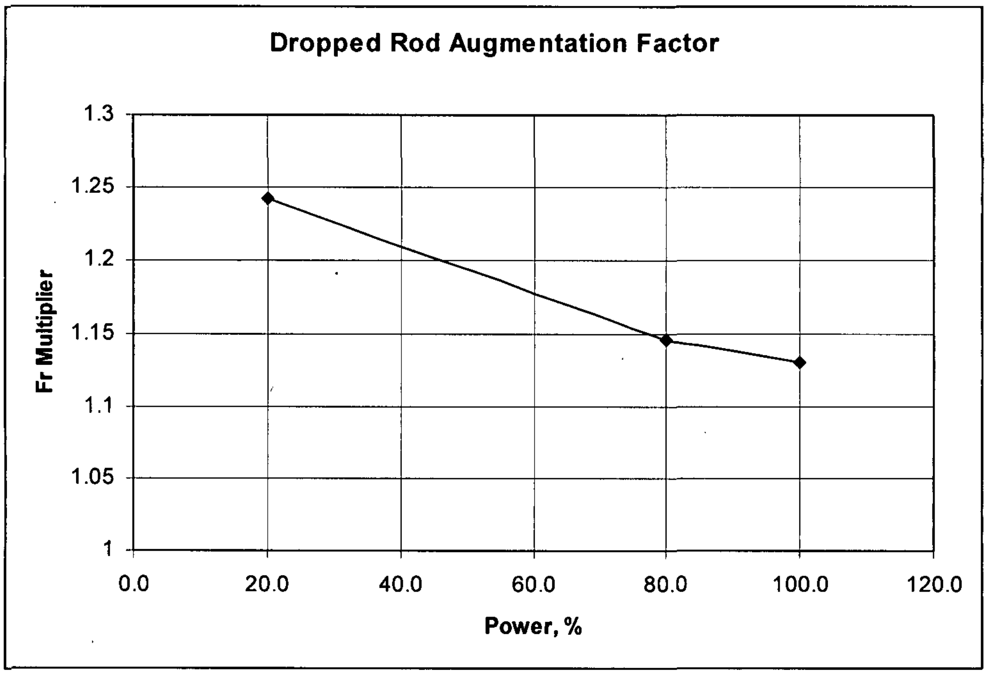

Figure 4b-1, Power Dependence of the Dropped CEA Augmentation Factor

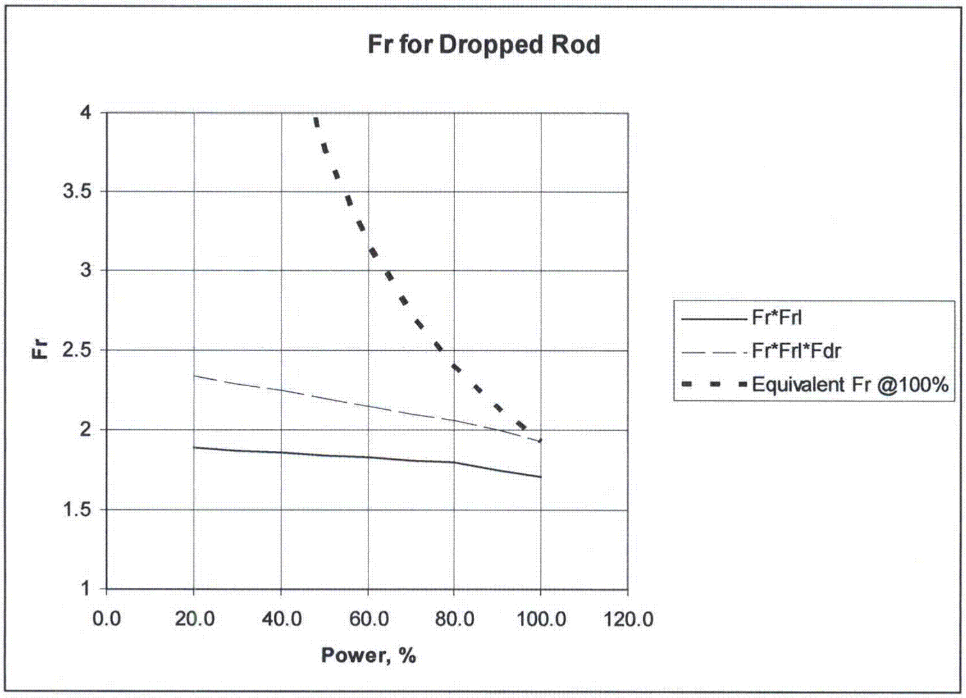

Figure 4b-2, Power Dependence of F_r for CEA Drop Event

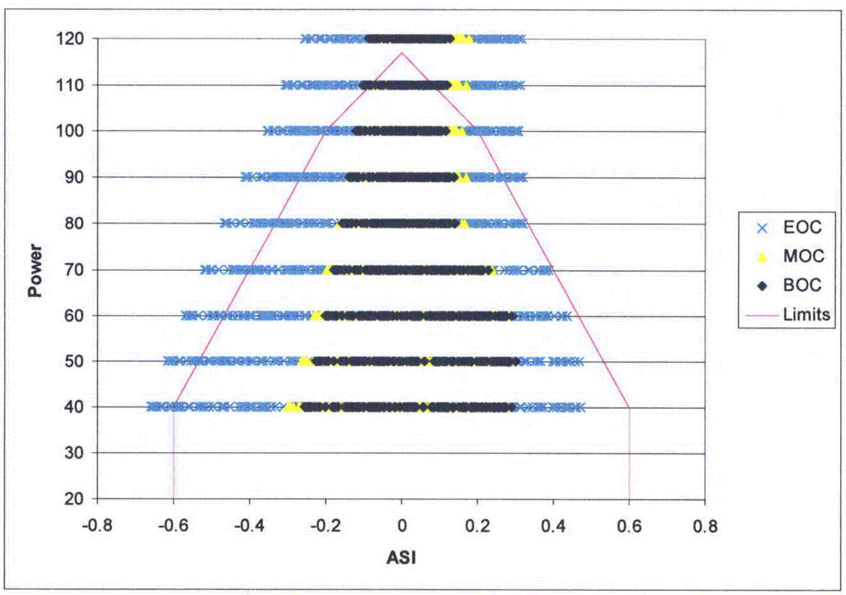

Figure 4b-3, Range of Axial Shape Index and Power

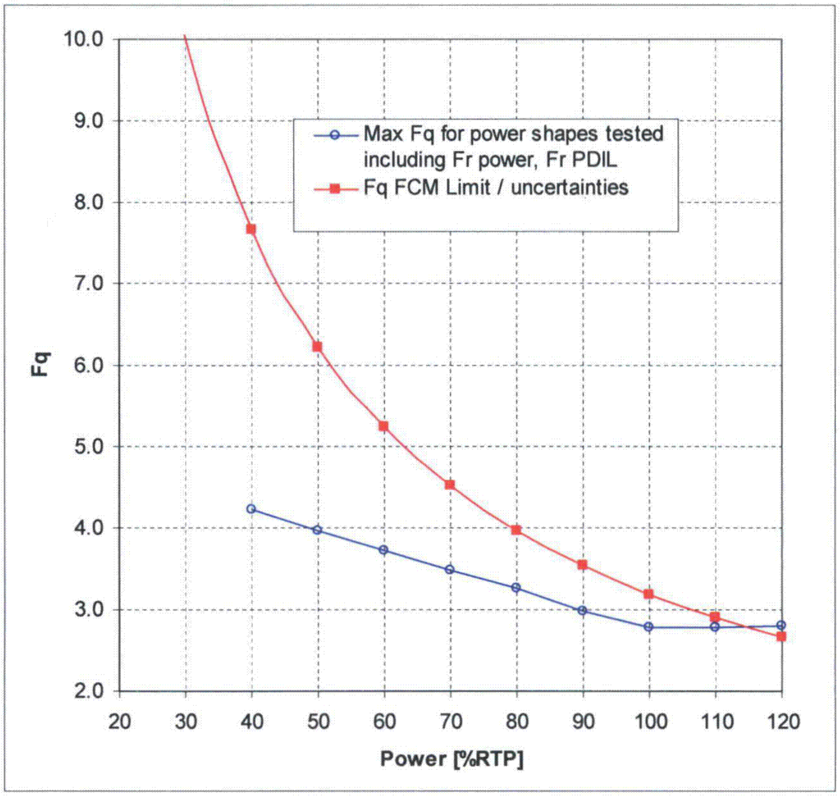

Figure 4b-4, Power Dependence of F_q

Question 5:

Withdrawn

CCNPP Response 5:

N/A

Question 6:

The staff will request a license condition from CCNPP to limit itself from changing Technical Specification COLR Figures 3.1.6, 3.2.1-2, 3.2.3, or 3.2.5 without prior NRC review and approval until an NRC-accepted, generic or CCNPP-specific basis is developed for analyzing power level-sensitive transients (Control Rod Bank Withdrawal, CEA Drop, and CEA Ejection) at full power conditions only.

CCNPP Response 6:

Response provided in Reference 3.

Question 7:

Withdrawn

CCNPP Response 7:

N/A

Question 8:

The response to Question 3 provides no evidence that CEA ejection events initiated at mid-power conditions along the COL PDIL are bounded HZP and hot full power conditions. Please provide CEA ejection cases at several at-power conditions which capture power-dependent parameters such as PDIL LCOs, ASI LCOs, power peaking LCOs, LSSSs, and power measurement uncertainty. For example, one scenario might be allowable initial conditions at 19.9% power (e.g., Ejected rod worth spanning up to Bank 3 60% inserted, most severe AXPD, etc.) while initiated at a higher power corresponding to 19.9% plus power measurement uncertainty (e.g., secondary calorimetric uncertainty) along with a similarly decalibrated high power trip.

CCNPP Response 8:

Response provided in Reference 3.

Question 9:

The response to Question 4 appears to be non-responsive to the staff's technical concern.

In Question 4, the staff requested that the licensee address pressurization effects of allowable initial conditions other than nominal. In response, the licensee stated that the S-RELAP5 code is being used to analyze for conformance to specified acceptable fuel design limits, and not to analyze reactor coolant pressure boundary integrity. Because this response does not address the staff's technical concern, we would need to indentify a way to restrict our approval of S-RELAP5 to only those safety analyses that confirm acceptable transient performance relative to the specified acceptable fuel design limits, and require prior, transient-specific NRC review and approval of any use of S-RELAP5 to demonstrate reactor coolant pressure boundary integrity.

If the licensee is amenable to proceeding in this fashion, we would close Question 4 as resolved, pending the development of the appropriate license condition. If not, we will need to identify an alternative path forward.

CCNPP Response 9:

Response provided in Reference 3.

Question 10:

For the SBLOCA please provide the plots of the key system parameters for the breaks re-analyzed that are provided in the UFSAR. No plots were provided nor were the tables summarizing the timing for the key events provided.

CCNPP Response 10:

Response provided in Reference 3.

Question 11:

An HPSI flow delivery curve with 5% more flow was used in the AREVA analysis compared to the previous SBLOCA submittal by CE. Please justify this new HPSI delivery curve and demonstrate that it meets the latest surveillance measurement for HPSI pressure and flow. The HPSI curve is adjusted to account for measurement error (approximately 5%) for pressure and flow when the surveillance pressure/flow measurements are taken. Please demonstrate that these errors are accounted for in the HPSI pressure and flows used in the re-analysis. Please provide the HPSI pressure vs flow curve used in the analysis.

CCNPP Response 11:

Response provided in Reference 3.

Question 12:

It was stated that the AREVA SBLOCA analysis results in multiple loop seals clearing relative to the previously approved CE analysis where only one loop seal clears. Please provide the plots of the liquid levels in the loop seals for the limiting break and the steam mass flow and velocity entering the loop seal from the horizontal portions of the suction legs. Please show that the conditions in the loop seals in the unbroken loop support clearing of this additional loop. Loop seal clearing phenomena following SBLOCA is very difficult to predict correctly and has historically been poorly predicted by all T/H codes, including the RELAP5 series of codes. As such, loop seal clearing only in the broken loop has been the accepted approach by the NRC staff during the review of evaluation models. Please provide benchmarking of the S-RELAP5 model against loop seal clearing separate effects tests as well as integral experimental data. While it is recognized that AREVA is using an approved RELAP5 model, it is still necessary to demonstrate that the model is performing correctly in all plant specific calculations, with a physically based thermal hydraulic behavior that supports the loop seal clearing behavior. It is not clear that additional loop seals will clear once the broken loop seal has cleared for such small break sizes. Please provide justification for the RELAP5 multiple loop seal clearing following a small break in the discharge leg. As a comparison, please provide the results of the limiting SBLOCA with only the single broken cold leg loop seal cleared.

CCNPP Response 12:

The small break loss-of-coolant accident (SB LOCA) methodology delineated in Reference 9 was submitted to the NRC in January of 2000 and approved in March 2001. There was an extensive review by both the NRC staff and the ACRS.

As identified on page 14 of the Safety Evaluation Report for Reference 9, AREVA (formally FRA-ANP) benchmarked to the [ ]. The conclusion in the Safety Evaluation Report (SER) states;

> "Comparisons of the S-RELAP5 calculations to these tests demonstrated, respectively, the capability of S-RELAP5 to conservatively predict...loop seal clearing; and overall SBLOCA behavior and temperature results."

Additionally, the Safety Evaluation Report states on page 14;

[ ]

On page 15 of the Safety Evaluation Report, the NRC continues to discuss S-RELAP5 behavior and predictions as it relates to loop seal clearing;

These benchmarks are documented in Section 5.4 for the [ ], and Section 5.5 for the [ ] of Reference 9.

Correct prediction of the loop seal clearing phenomena is a very difficult computational task for all thermal hydraulic codes, including S-RELAP5. As such, an approach was specified in Reference 9 to address this issue;

With this modeling approach, as discussed in Reference 9, AREVA produced SB LOCA calculations with [ ]. These results were transmitted to the NRC in Reference 1.

A review of the applicable tests performed to investigate loop seal clearing phenomena provided some guidance in the context of categorizing how many loop seals will clear for a given break size. The test data supports an argument that only a single loop seal shall be allowed to clear [ ]. For breaks greater [ ], multiple loop seals would be expected to clear. The conclusions of the review prompted AREVA to reanalyze the previously submitted SB LOCA break spectrum [ ].

Additionally, AREVA has addressed the following:

As previously discussed, AREVA has chosen to conservatively apply the augmented loop seal biasing in a manner to remain consistent with the intent and philosophy of Reference 9, whereby the [ ]. To remain within the intent of the previously approved methodology, results for the limiting SB LOCA with only the single broken cold leg loop seal cleared are not provided. The application of the [ ].

The summary of results for the SB LOCA break spectrum is presented in Table 12-1. From this table, it can be seen that the limiting break size is 0.09 ft^2, and with the [ ]

The sequence of key events for the limiting break of 0.09 ft^2 is presented in Table 12-2. The following plots were obtained from the limiting case which was the 0.09 ft^2 break. For this case, [ ]. Figures 12-1 to 12-4 show the thermal hydraulic conditions in each individual loop seal. Consistent with the [ ]. Thus, the [ ]. As seen in Table 12-1, all break sizes met the acceptance criteria of 10 CFR 50.46.

Figures 12-5 to 12-12 were included to reproduce the type of parameters presented in the Calvert Cliffs Updated Final Safety Analysis Report (UFSAR).

Table 12-1, Summary of SB LOCA Break Spectrum Results (0.02 ft^2 to 0.14 ft^2)

Table 12-2, Timing of Key Events for 0.09 ft^2 Break

[ ]

Figure 12-1, Loop Seal Liquid Levels for 0.09 ft^2 Break

Figure 12-2, Loop Seal Steam Flow Rates for 0.09 ft^2 Break

Figure 12-3, Loop Seal Steam Velocities for 0.09 ft^2 Break

[ ]

Figure 12-4, Loop Seal Void Fractions for 0.09 ft^2 Break

[ ]

Figure 12-5, Core Power for 0.09 ft^2 Break

Figure 12-6, Reactor Vessel Upper Head Pressure for 0.09 ft^2 Break

[ ]

Figure 12-7, Break Flow Rate for 0.09 ft^2 Break

[ ]

Figure 12-8, Cladding Temperature at Hot Spot for 0.09 ft^2 Break

[ ]

Figure 12-9, Coolant Temperature at Hot Spot for 0.09 ft^2 Break

[ ]

Figure 12-10, Heat Transfer Coefficient at Hot Spot for 0.09 ft^2 Break

Figure 12-11, Inner Vessel Inlet Flow Rate for 0.09 ft^2 Break

Figure 12-12, Core Two-Phase Mixture Level for 0.09 ft^2 Break

Question 13:

An analysis of the severed injection leg is needed. Please provide the results of an analysis of the severed injection line with the degraded injection into the RCS since one of the line spills to containment while the others inject at the much higher RCS pressures. Breaks up to and including approximately 1.0 ft^2 are considered in the small break spectrum even though they are in the transition region.

<u>CCNPP Response 13</u>:

The SB LOCA methodology delineated in Reference 9 contains [ ]. In Section 6.0, Conclusions, of the NRC Safety Evaluation Report, [ ]. Given the piping layout of the emergency core cooling systems (HPSI, LPSI, and SIT) at Calvert Cliffs, the postulated severance of an injection line will be a break in the SIT piping which has an inner diameter of approximately 10.5 inches (0.6 ft^2). [ ]. To address this specific question (Question 13), AREVA [ ].

From an emergency core cooling system performance perspective, the likely scenario would be a diversion of more than 25% (100%/4 injection locations) of the total injected HPSI flow due to the flow diversion to the break. However, the consequent asymmetric flow distribution is highly dependent upon the specifics of each plant's piping network and the components in that piping system (e.g., cavitating venturis, throttle valves, etc.).

The analysis of record emergency core cooling system flow rates specific to an injection line break were not available for inclusion into the response to this question. Therefore, without a specific licensing basis for asymmetric HPSI flows due to the ruptured line injecting to the containment backpressure as opposed to the RCS system pressure for this postulated SB LOCA scenario, AREVA has performed [ ]. Additionally, since the break was assumed to be the severance of the SIT piping, only the three intact SITs were modeled to inject into the RCS. The SIT on the severed pipe was assumed to dump directly to the containment.

The analysis demonstrated that the severance of an injection line was [ ]. The peak cladding temperature was [ ]. Table 13-1, provides provides the sequence of events for the SIT line break. Figures 13-1 through 13-4 show the break mass rate, the core two-phase mixture, the SIT injection time and flow rate, and the peak centerline temperature.

Table 13-1, Sequence of Events for SIT Line Break (0.6 ft^2)

	Time (sec)
Break in SIT line to CL-2B	0.0
Pressurizer pressure reached TM/LP floor (1790 psia)	7.8
Reactor trip, offsite power lost, RCPs, main feedwater pumps, and turbine tripped	9.2
Pressurizer pressure reached SIAS setpoint (1640 psia)	10.9
HPSI flow began	--- (none injected)
SG level reached AFAS setpoint (29.26% WR)	---
Steam-driven AFW delivery began	---
Loop seal 1A cleared	72
Loop seal 1B cleared	59
Loop seal 2A cleared	72
Loop seal 2B cleared	60
Break uncovered	72
SIT flow began to intact loops	186
Minimum RV mass occurred	188
Hot rod rupture occurred	---
Peak Centerline Temperature occurred	188
Non-condensable gas reached break	---

Figure 13-1, RCS-side Break Flow Rate for SIT Line Break

[ ]

Figure 13-2, Core Two-Phase Mixture Level for SIT Line Break

[ ]

Figure 13-3, SIT Injection Rates for SIT Line Break

[ ]

Figure 13-4, PCT Node Cladding Temperature for SIT Line Break

Question 14:

Please perform an analysis of hot leg breaks to demonstrate that the limiting break location for the RCP trip timing criteria has been identified.

CCNPP Response 14:

A delayed operator action time RCP trip sensitivity study has been completed for the revised SB LOCA analysis, addressing a [ ]. This study utilized the Appendix K evaluation model to assess the maximum delay time for operator action to trip the RCPs that would not violate 10 CFR 50.46 acceptance criteria. Table 14-1 provides a summary of the peak centerline temperature results from this study. The operator action time in Table 14-1 is the time span or delay from the indication of less than 20°F subcooling (including allowances for uncertainty) in the RCS and the time of RCP trip. The conclusion of the study is that [ ]

For both hot leg and cold leg pump discharge breaks, the RCS evolves as a reasonably homogeneous mixture of steam and water while the RCPs are running. The break flow is representative of the mixture flowing within the RCS and thus continuously extracting water from the RCS while the pumps are running. This results in a gradual buildup of the RCS void fraction such that if the RCPs are tripped, or fail, late in the transient there will be insufficient water within the RCS to fill a substantial portion of the reactor vessel and provide effective core cooling. There are two main differences between the hot leg and cold leg pump discharge breaks:

- For hot leg breaks is there is no spillage of ECCS directly to the break in the broken loop. For the cold leg pump discharge breaks, the ECCS injection within the broken cold leg can flow directly through the break and not be available to provide core cooling. Thus, the reactor vessel liquid inventory can start to recover sooner post-RCP trip for hot leg breaks.
- On the other hand, the void fraction within the hot leg is somewhat lower than in the cold leg such that the liquid loss rate can be more rapid for a hot leg break. Thus, at the same break size and time of RCP trip, the post-trip inventory may be substantially lower for a hot leg break.

The importance of these effects differs with break size and the timing of RCP trip. In some cases the reactor vessel liquid inventory post-RCP trip is overshadowed by the added viable injection and the hot leg break is less severe than the cold leg pump discharge break. In other cases, the extra viable injection is not sufficient to compensate for the lower reactor vessel inventory and the hot leg break is more severe.

The limiting cases, as defined by meeting 10 CFR 50.46 acceptance criteria, were [

] to trip the RCPs. The peak cladding temperature transients for these two cases are shown in Figures 14-1 and 14-2. For Calvert Cliffs, the limiting condition occurs for the [

]. An evaluation with more realistic modeling, such as was done by Combustion Engineering in response to TMI Action Item II.K.3.5, would lead to longer allowed RCP trip delay times.

Table 14-1, Comparison of Cold Leg Pump Discharge and Hot Leg Peak Centerline Temperatures

[

]

[

]

[

]

Figure 14-1, Peak Centerline Temperature for [ ]

[

]

Figure 14-2, Peak Centerline Temperature for [

]

Question 15:

Withdrawn

CCNPP Response 15:

N/A

Question 16:

In response to RAI #24, Constellation described the regulatory basis for removing Surveillance Requirement 3.2.1.1, but did not address the technical basis. During the audit, please be prepared to describe (1) how "peripheral" ASI will be used to confirm "interior" linear heat rate limits (TS 3.2.1), (2) will peripheral ASI (F_z) be combined with another local power surveillance (e.g., F_r) to validate peak LHGR (F_q)?, and (3) how does the removal of F_{xy} surveillance guarantee the same level of protection?

CCNPP Response 16:

Response provided in Reference 3.

Question 17:

Realistic LBLOCA

During the audit, we'll need to speak to individuals familiar with the RLBLOCA analysis. The staff doesn't fully understand CCNPP's disposition and RAI response concerning the single failure selection. ANP-2834(P), Section 1, states that "A conservative loss of a diesel assumption is applied in which LPSI inject into the broken loop and one intact loop and HPSI inject into all four loops." CCNPP response to RAI 2 (letter dated August 9, 2010), as well as Section 3.1 of ANP-2834(P) state that the limiting single failure has been determined to be the loss of one ECCS pumped injection train. The staff needs to confirm that these failures are one in the same.

The staff also needs to understand how the limiting single failure for the CE NSSS was determined, since the basis for the RAI response defers to NRC-approved methodology. Poring through EMF-2103, the staff only located sensitivity results on 3-loop W systems. In some cases, the limiting failure would be a single LPSI and in others it was a diesel. The staff could not locate a clear, generic disposition for the single failure at any place in EMF-2103.

What was done under the auspices of EMF-2103 development to ensure that the containment analysis produced a sufficiently conservative prediction that a no failure, max SI spillage case, for a CE NSSS, is bounded by the chosen single failure? The staff will need to see that work.

CCNPP Response 17:

Response provided in Reference 3.

Question 18:

The response to RAI 23.o discussed the 7-minute operator action that is credited to secure the reactor coolant pumps. Please address the following:

a. Will operators need to know that there is a time-constraint of 7 minutes associated with this action?

b. How have these actions been validated to be feasible and reliable?

c. Who was, or will be, involved in the validation?

d. Describe the changes, if any, to the plant-reference simulator and training that are planned to support these actions.

e. The response to the RAI states, "The new step is in the Pressure and Inventory Safety Function, normally the second safety function performed by the Reactor Operator, and normally within two minutes of a reactor trip." Where did the two minute time frame originate and what is its basis? Is this from another procedure that has previously been validated?

CCNPP Response 18:

The response for this question provided in Reference 3 is amended as follows. The response in Reference 3 stated that the operator response time was 7 minutes from achieving 1600 psia to shut off all RCPs. Based on the analysis described in Response 14 (above), that time is now 4 minutes from the indication of less than 20°F subcooling to turn off all RCPs. The verification actions described in Reference 3 continue to support this reduced operator action time.

Also note, Calvert Cliffs is developing a formal Time Critical Action Program based on the elements of Reference 12. This program will provide a means to ensure that the time critical actions (such as the RCP trip discussed here) can be accomplished by plant personnel. In addition, the program will provide a means to document periodic validation of credited action times. And this program will provide a means to ensure that changes to the plant or to procedures or protocols do not invalidate the credited action times.

While developing the formal program, a job/simulator performance measure is being established for this time critical action (turning off all RCPs based on subcooling) and this simulator/ job performance measure will be implemented independently of the formal Time Critical Action program. This simulator/job performance measure includes pass/fail criteria for training on the simulator. This criterion was developed using the Systematic Approach to Training process and the guidance of Reference 12. The job/simulator performance measure will be incorporated into the job/simulator scenarios prior to operation of AREVA fuel in the reactor vessel.

Question 19:

In response to RAI #9, the licensee provided Figures 9-4 and 9-6 illustrating steam flow versus time for a 1.0 ft^2 break outside containment and inside containment respectively. Both breaks are located upstream of the MSIV.

a. Explain the asymmetric steam flow prior to MSIV closure which is not exhibited in larger breaks.

CCNPP Response 19:

Response provided in Reference 3.

Question 20:

In response to RAI #10, the licensee described various credited trip functions. Please discuss the harsh environment uncertainty for the ASGT and Containment Pressure High trip functions.

CCNPP Response 20:

Due to the development of a potentially harsh environment in the containment, which can impact the performance of Reactor Protective System-related instrumentation, the pre-trip SLB cases from the previous analysis that tripped on Asymmetric Steam Generator Transient were rerun since they did not include an allowance for harsh conditions. To account for a harsh environment, the new analysis was performed with the conservative assumption that the Asymmetric Steam Generator Transient trip is not credited. The previous AREVA cases that tripped on Asymmetric Steam Generator Transient were the 0.5 ft^2 asymmetric inside-containment breaks. Even though instrumentation performance is not affected by harsh conditions for outside containment breaks, the 0.5 ft^2 and 1.0 ft^2 asymmetric outside-containment break cases were also rerun without credit for the Asymmetric Steam Generator Transient trip for completeness.

Table 20-1 provides a comparison of the results from the original and revised inside-containment asymmetric break cases. Table 20-2 provides a comparison of the results from the original and revised outside-containment asymmetric break cases.

Based on the results of the revised analyses which do not credit the Asymmetric Steam Generator Transient trip, the overall limiting pre-trip SLB case (2.0 ft^2 outside containment symmetric break with an moderator temperature coefficient of -16 pcm/°F) is not impacted and, therefore, the minimum DNBR and peak linear heat rate for this case remain limiting.

Table 20-1, Summary of Results for Inside-Containment Asymmetric Breaks

Table 20-2, Summary of Results Outside-Containment Asymmetric Breaks

Question 21:

Withdrawn.

CCNPP Response 21:

N/A

Question 22:

For the PCT-limiting RLBLOCA case, please provide:

a. Corrected and uncorrected radial temperature profile of the hot rod at the time and location of peak cladding temperature.

b. Temperature vs. time for the limiting PCT case at the limiting location, including the fuel centerline, fuel average, and clad surface temperatures. Indicate the end of blowdown, start of refill, and start of reflood on this graph.

c. Burnup for the limiting rod.

CCNPP Response 22:

Response provided in Reference 3.

Question 23:

The issue described in IN 2009-23 invalidates AREVA's generic disposition for analyzing fresh fuel only, which is based on sensitivity studies indicating that mid-second-cycle fuel had a PCT of 80°F lower than the limiting PCT. This work needs to be repeated accounting for fuel thermal conductivity degradation. Please provide several cases run at various times-in-life for once-burnt fuel, with information similar to the above list provided; Item 1.c is only necessary for the most limiting second-cycle case analyzed.

CCNPP Response 23:

Response provided in Reference 3.

Question 24:

In RAI #18, the staff requested an explanation of apparent differences between the UFSAR AOR and the AREVA analysis. The response was insufficient to understand differences in analytical techniques, assumptions, and initial conditions. Please expand this response.

a. Provide a comparison of Cycle 18 versus Cycle 19 calculated values for initial ASI, initial F_q, ejected rod worth, post ASI, and post F_q for the limiting rod configuration at HZP and HFP, and
b. Using prior reload cycle calculated parameters (ejected worth, post- F_q, fuel enthalpy), populate the plots of Deposited Enthalpy vs. Rod Worth in XN-NF-78-44.

CCNPP Response 24:

a. The requested information that was readily available is listed in Table 24-1. No uncertainties are applied to the data. A significant difference between the methods is the assumption for the initial CEA configuration. The Cycle 15 and 16 data are generated assuming full insertion of the CEAs that would be partially inserted at the power dependant insertion limit. The AREVA method places the CEAs at the power dependent insertion limit. For comparison purposes, the Cycle 19 values are generated both at power dependent insertion limit and Bank 5 fully inserted at hot full power. The significant difference between the hot full power results is greatly reduced when evaluated at consistent initial conditions. In general, the values with AREVA methods and/or core designs are higher for beginning of cycle and are lower for end of cycle.

Table 24-1, Ejected CEA Parameters

Parameter	Time in Life	Power	Unit 1 Cycle 15 W*	Unit 1 Cycle 16 W*	Unit 2 Cycle 19 AREVA
Ejected Rod Worth, %$\Delta\rho$	BOC	HZP	0.1957	0.1802	0.2363
	BOC	HFP	0.1326	0.1265	0.0458 0.1994*
	EOC	HZP	0.3416	0.2793	0.2329
	EOC	HFP	0.1730	0.1708	0.0428 0.1641*

F_q Post Ejection	BOC	HZP	-	4.9	6.12
	BOC	HFP	-	2.54	2.42 3.33*
	EOC	HZP	-	11.07	8.40
	EOC	HFP	-	2.82	2.18 2.66*

* Initial position- Partial banks at power dependent insertion limit are fully inserted.

b. Only one set of ejected CEA data was readily available from the previous analysis bases to compare to the XN-NF-78-44 plots. The parameters from this set are used to estimate the total enthalpy (cal/g) using the curves provided in XN-NF-78-44 and are provided in Table 24-2. For the same inputs the AREVA methods appear to be conservative to the previous methods.

Note: A clarification may be helpful to understand the AREVA methodology. The hot full power total cal/g is based on an initial cal/g of 40.8 which is under the ejected CEA. This delta cal/g calculated from the ejected CEA location is conservatively added to the initial peak cal/g in the core. The ejected CEA location exhibits the maximum change in peaking for any location in the core from a CEA-in to a CEA-out configuration. It is nearly impossible for the initial peak to be in the ejected CEA assembly location prior to ejection.

Table 24-2, Calculated Enthalpy (cal/g) for STRIKIN versus AREVA

Power	Time in Life	Ejected CEA Worth, pcm	B_{eff}, %	Initial F_q	Final F_q	STRIKIN cal/g Initial / Final	AREVA cal/g Initial / Final
HZP	All	870	0.44	-	24.7	16.5 / 180.0	18.0 / 263.9*
HFP	All	310	0.44	2.475	5.04	90.4 / 184.7	106 / 227.2

* The reported value corresponds to an F_q of 24.7 which is well beyond the extrapolation range of the method. If XTRAN was run with an F_q of 24.7, significantly more Doppler feedback would occur which would lower this cal/g. Extrapolation is probably adequate up to the delta F_q from the last two values. The points at 9.0 and 13.0 are used to extrapolate to an F_q of 17.0 which yields 181.5 cal/g.

<u>Question 25</u>:

Withdrawn

<u>CCNPP Response 25</u>:

N/A

<u>Question 26</u>:

In RAI #14, the staff requested an explanation of the use of BOC and EOC physics parameters. The response (provided in RAI #3) was insufficient to justify analyzing only these two extreme cases. At EOC conditions, FTC is most negative (turns event around) and Beff is smallest (promotes power excursion). These parameters act to offset each other, so the limiting scenario may be at another BU point where FTC is not so negative. This is probably the reason why the UFSAR AOR combines limiting parameters, regardless of exposure dependence.

CCNPP Response 26:

The cycle dependence of the Doppler coefficient, effective delayed neutron fraction (β), and ejected CEA worth are shown in Figures 26-1 through 26-3. After ~5 GWd/MTU, the Doppler coefficient and the β are nearly linear with cycle burnup. The Doppler coefficient remains nearly constant for fuel burnups less than 5 GWd/MTU. The ejected CEA worth is the driving parameter for this event and the beginning of cycle and end of cycle endpoints reasonably bound its behavior over the whole cycle. Prior to obtaining the ejected CEA worth (cal/g), the uncertainty is applied to the endpoint values for the beginning of cycle and the end of cycle, and the CEA worth is further increased to a minimum CEA worth analyzed for the cal/g that conservatively bounds the data as shown in Figure 26-4.

A sensitivity calculation is performed to determine whether this conservatism is adequate to cover the variation in Doppler coefficient and β. The moderator temperature coefficient is assumed to be zero for all calculations so that the end of cycle case contains more conservatism than the beginning of cycle case. In order to bound the beginning of cycle case, a β of 0.6% is selected to bound the behavior from 0-5 GWd/MTU. The ejected CEA worth (cal/g) is re-calculated with a β of 0.6% at hot full power and hot zero power, and the changes in ejected CEA worth (cal/g) at hot zero power and hot full power are +2.2 and +2.7 cal/g, respectively. The change in ejected CEA worth that would yield the same cal/g change at hot zero power and hot full power are 12.0 and 53.5 pcm, respectively. As shown in Figure 26-4, these values are small compared to the conservatism already included in the analyzed CEA worth. Hence, the evaluation of the beginning of cycle and the end of cycle endpoints remains bounding.

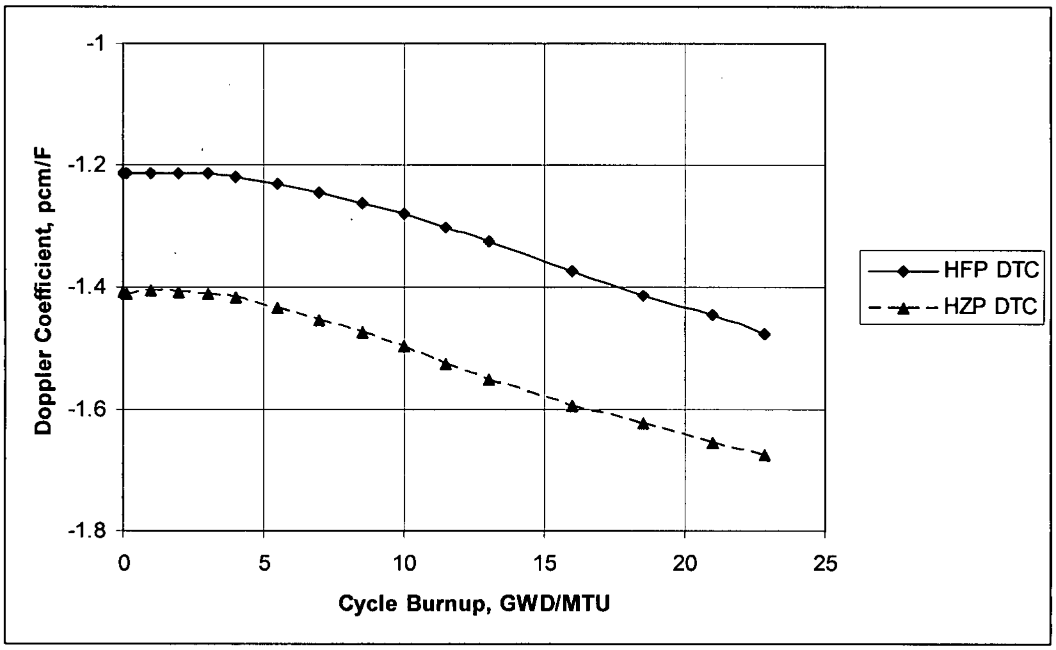

Figure 26-1, Doppler Coefficient versus Burnup

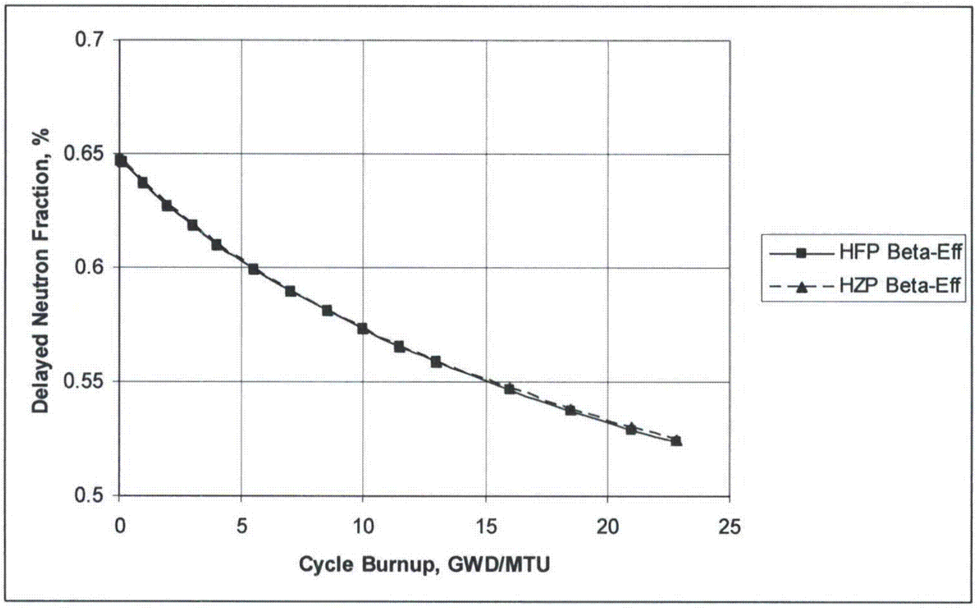

Figure 26-2, Delayed Neutron Fraction versus Burnup

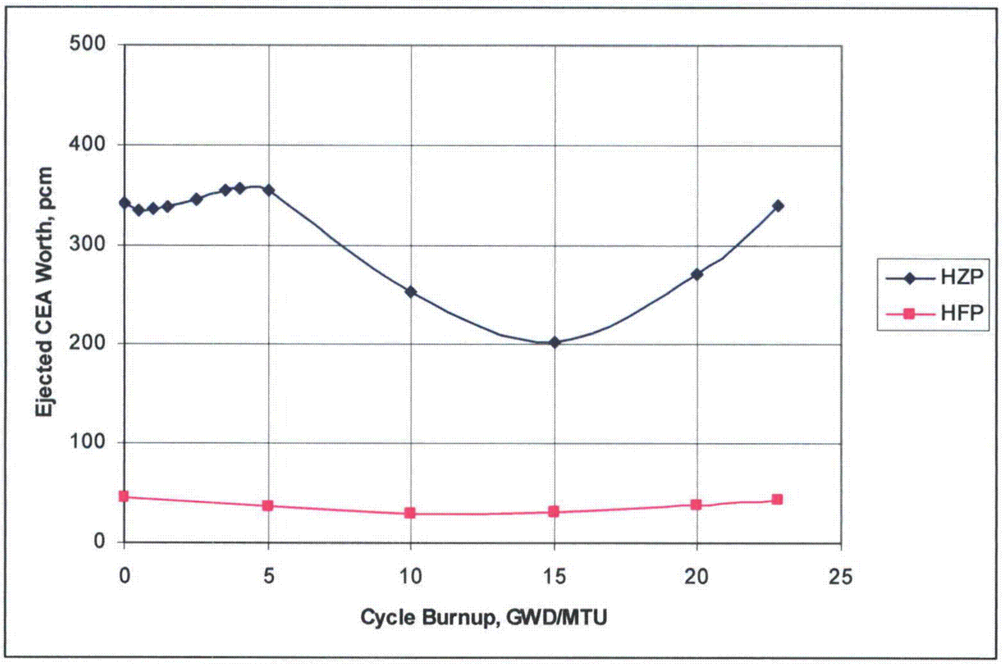

Figure 26-3, CEA Ejected Worth versus Burnup

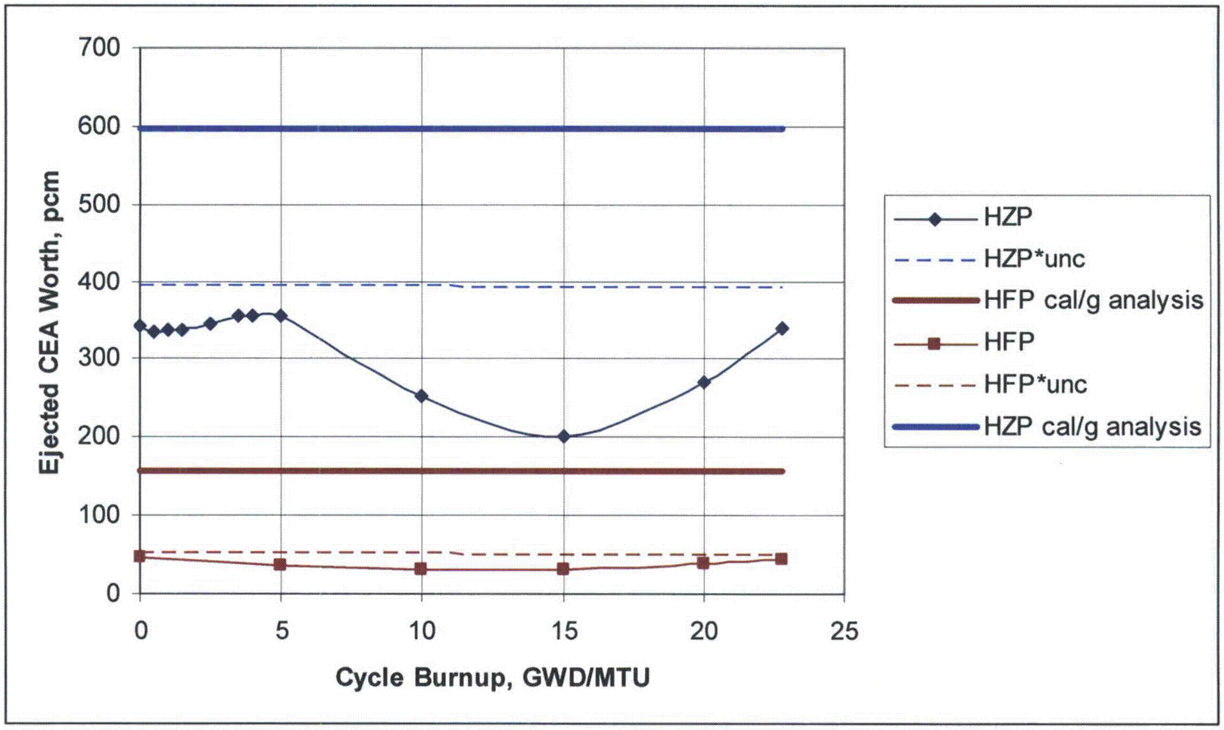

Figure 26-4, CEA Ejected Worth (Cal/g)

Question 27:

With respect to the performance of co-resident Westinghouse fuel assemblies:

a. Please explain how the rod power histories considered in the Westinghouse fuel rod thermal-mechanical performance calculations will be verified for future operating cycles.

b. Did the Post-Trip MSLB analysis consider the performance of Westinghouse fuel should the stuck CEA be located above a Westinghouse assembly?

CCNPP Response 27:

Response provided in Reference 3.

Question 28:

The staff has completed FRAPCON-3 benchmark calculations based upon the limiting CC2CY19 rod internal pressure power histories specified in 32-9135500-001. Code inputs and power histories have been coordinated, such that differences are most likely due to code-to-code variations and methodology (e.g., application of uncertainties). FRAPCON-3 benchmark calculations predict a nominal UO2 fuel rod internal rod pressure of 1714 psia (no AOO power ramps). A statistical evaluation simulating 500 cases each randomly sampling the manufacturing tolerances yields a 95/95 UTL internal pressure of 1820 psia. The addition of the fission gas release model uncertainty to this statistical evaluation yields a 95/95 UTL internal pressure of 2260 psia. Discuss the conservatism of the RODEX2 calculated rod internal pressure, relative to the staff's calculations for Calvert Cliffs.

CCNPP Response 28:

[

]. This level of agreement is within expectations for analytical code benchmarks.

Differences observed when FRAPCON-3 calculations are performed with a full set of statistical treatments are due primarily to the sensitivity of the FRAPCON-3 code to the models used and the accompanying power history. As each fuel performance code has been tuned to its own database and has its own models, sensitivities, conservative design methodology and uncertainties, difference in analytical design results are to be expected.

[

]. As can be seen from the results presented in Figures 28-1 through 28-3, RODEX2 becomes [ ]. Notably, RODEX2 predicts approximately [ ] at these conditions. This is due to the conservative nature of the RODEX2 methodology (Reference 4).

The deterministic design analyses methodology for determining limiting gas pressures, described and approved in Reference 4, are shown to be appropriate and conservative.

[

]

Figure 28-1, Comparison of Input Linear Heating Rates for 3 Rod Types

[ ]

Figure 28-2, Comparison of Internal Gas Pressures for 3 Rod Types

Figure 28-3, Comparison of Average Fission Gas Release Rates for 3 Rod Types

Question 29:

As part of a recent Fitzpatrick fuel transition, the staff developed penalties on calculated rod internal pressure and limitations on fuel rod burnup to address outdated fuel thermal-mechanical methods. The use of cycle-specific rod power histories in the AREVA RODEX-2 methodology makes this difficult.

CCNPP Response 29:

Response provided in Reference 3.

Question 30:

Withdrawn

CCNPP Response 30:

N/A

Question 31:

a. AREVA postulates that clad swelling and rupture produces a benefit to PCT, and because of this, the realistic large break loss of coolant accident (RLBLOCA) model does not include a clad swelling and rupture model. Does this conjecture include consideration of test data, which has shown that following fuel rupture, the ballooned region fills with fuel fragments? What analytic studies support this conclusion? How are they applicable to CCNPP? Please also address the potential for co-planar blockage with the fuel relocation evaluation.

b. Since blowdown ruptures can occur at end of life conditions, show that blowdown ruptures do not occur at the end of life for the postulated CCNPP large break LOCA.

CCNPP Response 31:

Response provided in Reference 3.

Question 32:

Provide information to illustrate the conservative nature of the single-side only oxidation model and its application to the CCNPP RLBLOCA analysis.

CCNPP Response 32:

Response provided in Reference 3.

Question 33:

Provide additional information to justify the use of the selected analytic treatment for decay heat uncertainty in the RLBLOCA model.

CCNPP Response 33:

Response provided in Reference 3.

REFERENCES

1. Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated November 19, 2010, Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel
2. Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated October 29, 2010, Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel
3. Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated December 30, 2010, Supplement to the License Amendment Request: Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel
4. XN-NF-82-06(P)(A) Revision 1, Supplement 5, "Qualification of Exxon Nuclear Fuel For Extended Burnup (PWR)," October 1986
5. ANF-88-133(P)(A) and Supplement 1, "Qualification of Advanced Nuclear Fuels PWR Design Methodology for Rod Burnups of 62 GWd/MTU," December 1991
6. XN-NF-81-58(P)(A) Revision 2 and Supplements 1 and 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," March 1984
7. XN-NF-81-58(P)(A) Revision 2 and Supplements 3 and 4, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," April 1990
8. BAW-10240(P)(A) Revision 0, "Incorporation of M5™ Properties in Framatome ANP Approved Methods," May 2004

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

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

11. Letter from Mr. G. H. Gellrich (CCNPP) to Document Control Desk (NRC), dated April 22, 2010, Response to Request for Additional Information – Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

12. WCAP-16755-NP, Revision 0, Operator Time Critical Action Program Standard, March 2007