George H. Gellrich
Vice President

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CENG
a joint venture of
Constellation Energy · eDF

CALVERT CLIFFS
NUCLEAR POWER PLANT

November 19, 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

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. 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 some of the questions were provided in Reference (a). The responses to the remainder of the questions are contained in Attachment (1). This supplement does not change the No Significant Hazards determination previously provided in Reference (b).

Attachment (1) contains information that is proprietary to AREVA and Westinghouse, therefore, it is accompanied by affidavits signed by AREVA and Westinghouse, owners of the information (Attachment 2). The affidavits set forth the basis on which information may be withheld from public disclosure by the Commission, and address, with specificity, the considerations listed in 10 CFR 2.390(b)(4). Accordingly, it is requested that the information that is proprietary to AREVA and Westinghouse 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 St. Mary's, this 19th day of November, 2010.

JANET L. SCULLY NOTARY ST. MARY'S COUNTY

WITNESS my Hand and Notarial Seal:

Notary Public

My Commission Expires:

March 1, 2011
Date

GHG/PSF/bjd

Attachment:
(1) Proprietary Supplement to License Amendment Request: Transition to AREVA Nuclear Fuel
Enclosure: 1 Draft Revision of EOP-0, Post Trip Immediate Actions
(2) AREVA Proprietary Affidavit/Westinghouse Proprietary Affidavit
(3) Non-Proprietary Supplement to License Amendment Request: Transition to AREVA Nuclear Fuel
Enclosure: 1 Draft Revision of EOP-0, Post Trip Immediate Actions

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

ATTACHMENT (2)

AREVA PROPRIETARY AFFIDAVIT /

WESTINGHOUSE PROPRIETARY AFFIDAVIT

Calvert Cliffs Nuclear Power Plant, LLC
November 19, 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 November 19, 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 19th

day of November, 2010.

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

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

Before me, the undersigned authority, personally appeared J. M. Brennan, who, being by me duly sworn according to law, deposes and says that he is authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of his knowledge, information, and belief:

J. M. Brennan, Vice President
Engineering Services

Sworn to and subscribed before me
this 19th day of November 2010

Notary Public

COMMONWEALTH OF PENNSYLVANIA
Notarial Seal
Katherine W. McGinnett, Notary Public
Cranberry Twp., Butler County
My Commission Expires Jan. 4, 2013
Member, Pennsylvania Association of Notaries

(1) I am Vice President, Engineering Services, in Nuclear Services, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitutes Westinghouse policy and provides the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's

Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iii) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390; it is to be received in confidence by the Commission.

(iv) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(v) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in "Letter from 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" (Proprietary) for submittal to the Commission, being transmitted by Constellation Energy Nuclear Group letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with the results of the calculation of overpower margin for Westinghouse fuel in the Unit 2 Cycle 19 mixed core of AREVA and Westinghouse fuel and may be used only for that purpose.

This information is part of that which will enable Westinghouse to:

(a) Validate the use of Westinghouse Turbo fuel in the Calvert Unit 2 Cycle 19 mixed core.

Further this information has substantial commercial value as follows:

(a) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar calculations and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

ATTACHMENT (3)

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

Calvert Cliffs Nuclear Power Plant, LLC
November 19, 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 some responses were provided in Reference 10. This response combined with Reference 10 provides the complete set of responses.

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 1^{st} 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. Response provided in Reference 10.

b. The description of the current methodology follows:

   The core inlet temperature distribution is dictated by temperature differences in the cold legs and the amount of mixing that occurs in the downcomer, inlet plenum, and inlet flow skirt. Symmetric events generally result in an evenly distributed core inlet temperature.

   The inlet temperature is biased by the uncertainty in a direction that will provide more adverse results.

   For the Control Element Assembly (CEA) Withdrawal, Excess Load, and pre-trip Steam Line Break events, the radial power peak is penalized to account for changes in core inlet temperature.

Asymmetric events result in tilted cold side temperatures. Although the Seized Rotor event occurs on one cold leg, the transient is terminated before significant core inlet temperature asymmetries occur. A pre-trip Steam Line Break analysis becomes asymmetric upon main steam isolation valve (MSIV) closure, but the reactor trip occurs prior to MSIV closure for the limiting cases. The post-trip Steam Line Break models a stuck CEA and analyzes a return to power. Cold-edge temperatures are used to calculate moderator reactivity feedback. A loss of alternating current power to the reactor coolant pumps upon reactor trip may be assumed. The decreased flow tends to decrease mixing, which maximizes the Reactor Coolant System (RCS) cooldown used in the moderator reactivity calculation. A CEA Drop causes an asymmetric power distribution, but the core inlet temperature is not affected due to fluid mixing in the core exit and entrance.

The Asymmetric Steam Generator event is an asymmetric event. An Asymmetric Steam Generator event causes a severe asymmetry in the reactor core inlet temperatures. A difference in the amount of steam generator tube plugging is modeled to exacerbate the core inlet temperature tilt. No core inlet mixing is conservatively assumed. A skewed power distribution in the core occurs due to the influence of the moderator temperature coefficient. This in turn may increase core power peaking.

The Asymmetric Steam Generator event core response in the transient code is calculated as core average. In order to conservatively capture the local power and peaking factor increase on the cold side of the core, event-specific radial peak temperature dependence is used. This is applied to the entire core temperature tilt at the core inlet to yield a radial peaking factor that will result in a conservative calculation of margin. The core inlet temperature for the hot channel minimum DNBR calculation is based on the core average inlet temperature.

Core inlet temperature is input as an initial condition to the transient analyses. The transient codes model changes in cold leg temperatures due to the performance of the secondary system. The amount of mixing that occurs in the inlet to the core is determined by a mixing factor. The mixing factor is based on fluid mixing in a scaled reactor vessel flow model. For the Asymmetric Steam Generator event, this value is conservatively set to zero to exacerbate the temperature asymmetry in the core.

The AREVA method for treating core temperature asymmetry is conservative and appropriate because it is either explicitly modeled or bounded by conservative assumptions. All events are generally uniform with respect to core inlet temperature distribution with the exception of the Asymmetric Steam Generator and Steam Line Break events.

The Asymmetric Steam Generator event limiting case is modeled as a uniform, non-segmented, S-RELAP5 model which exhibits rapid progression and the Reactor Protective System trips the reactor prior to the development of any significant asymmetry at the core inlet. Additionally, the moderator reactivity feedback is conservatively based on the cold leg temperature in the unaffected loop rather than the more representative core average fluid temperature. This approach conservatively produces a core power response which offsets any augmentation due to power redistribution effects.

For the Steam Line Break event, a sectorized S-RELAP5 core model, with affected and unaffected regions, is utilized, which explicitly accounts for core inlet temperature asymmetry. The core inlet temperature asymmetry is also accounted for in the core neutronics and thermal-hydraulics models.

A uniform, but changing core inlet temperature distribution results in changes in the power distribution. The current methodology penalizes the radial peaking factor for inlet temperature changes for uniform and asymmetric events. The hot full power Excess Load event results in a uniform change in core inlet tempeature of about 15°F. This large temperature change results in a 2% increase in F_r and a corresponding decrease in minimum DNBR when % F_r/F, as described in Updated Final Safety Analysis Report (UFSAR) Table 14.4-1, is applied.

The current and proposed methodologies are consistent in the identification of events that result in uniform versus asymmetric inlet temperature distributions. The temperature uncertainty is applied consistently between the two methodologies. Both methodologies account for peaking factor changes associated with CEA position changes. The current methodology penalizes the radial peaking factor for inlet temperature changes for uniform and asymmetric events.

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. The 9% F_r margin between the maximum F_r in AREVA fuel and the maximum F_r in the Westinghouse fuel is explicitly verified for the limiting cases for transient events that involve radial power redistribution. These events include:

   - Pre-trip Steam Line Break - Verified at maximum power for each break case (asymmetric cases subsequently turned out to be non-limiting)
   - Post-trip Steam Line Break - Verified for all cases exhibiting a return to power

- CEA Withdrawal (from low power or subcritical) - Verified for several snapshots from all-regulating-rods-in to all-rods-out
- CEA Drop - Verified every 30 minutes for 3 hours following initiation

The CEA Ejection event was not analyzed for F_r margin using AREVA methodologies because Westinghouse has retained responsibility for this event for the mixed cores.

For Asymmetric Steam Generator events, there is insufficient time for significant asymmetry to develop in the core inlet temperature distribution and produce any significant increase in radial peaking. Therefore this event will not challenge the F_r penalty for Westinghouse fuel and is covered by the steady-state F_r margin verification.

The 9% margin specifically includes a component that offsets the impact of cross flow on Westinghouse fuel due to the mixed core, thus re-establishing the conditions for the previous analysis of record. This ensures that Westinghouse fuel is bounded by the previous cycle.

b. Response provided in Reference 10.

c. Table 2-1 provides the F_r data.

Table 2-1, F_r for U2C19 Nominal Hot Full Power Conditions

Cycle BU (GWd/MTU)	Max. F_r (fresh)	Max. F_r (burned)	F_r Margin
0.15	1.541	1.336	13.3
0.50	1.535	1.336	13.0
1.00	1.533	1.337	12.8
1.50	1.532	1.341	12.5
2.00	1.531	1.343	12.3
2.50	1.528	1.346	11.9
3.00	1.526	1.348	11.7
3.50	1.527	1.350	11.6
4.00	1.534	1.351	11.9
4.50	1.541	1.351	12.3
5.00	1.545	1.350	12.6
5.50	1.545	1.348	12.8
6.00	1.544	1.346	12.8
6.50	1.540	1.342	12.9
7.00	1.536	1.339	12.8
7.50	1.532	1.336	12.8
8.00	1.527	1.334	12.6
8.50	1.523	1.332	12.5
9.00	1.518	1.329	12.5
9.50	1.518	1.327	12.6
10.00	1.522	1.324	13.0
10.50	1.525	1.321	13.4
11.00	1.526	1.317	13.7
11.50	1.526	1.312	14.0
12.00	1.524	1.306	14.3
12.50	1.523	1.299	14.7

Cycle BU (GWd/MTU)	Max. F_r (fresh)	Max. F_r (burned)	F_r Margin
13.00	1.532	1.292	15.7
13.50	1.537	1.285	16.4
14.00	1.540	1.279	16.9
14.50	1.540	1.274	17.3
15.00	1.536	1.270	17.3
15.50	1.529	1.267	17.1
16.00	1.519	1.264	16.8
16.50	1.509	1.261	16.4
17.00	1.498	1.258	16.0
17.50	1.488	1.255	15.7
18.00	1.477	1.252	15.2
18.50	1.466	1.249	14.8
19.00	1.456	1.245	14.5
19.50	1.447	1.242	14.2
20.00	1.437	1.239	13.8
20.50	1.430	1.236	13.6
21.00	1.424	1.233	13.4
21.50	1.419	1.230	13.3
22.00	1.413	1.227	13.2
22.10	1.412	1.227	13.1
22.50	1.407	1.225	12.9
23.09	1.400	1.221	12.8

The overpower margin calculated using AREVA codes and deterministic methods and using nominal thermal hydraulic conditions at hot full power with the limiting axial power shape is shown below. The Unit 2 Cycle 19 mixed core of AREVA and Westinghouse fuel was assumed.

Nominal F_r	1.65
Max Allowed F_r (at correlation limit)	2.075
Overpower DNB Margin	(2.075 – 1.65) / 1.65 = 25.8%

The values shown below use Westinghouse codes and the methods consistent with the current UFSAR analysis, and use the same nominal thermal hydraulic conditions at hot full power and axial shape used to determine the AREVA values. The Unit 2 Cycle 19 mixed core of AREVA and Westinghouse fuel was assumed.

[ ]

d. For Westinghouse fuel, the 1% limit is a self-imposed limit. The approach is to take the worst AOO and determine if the strain meets the 1% limit. The overpower required to reach the specified acceptable fuel design limits was not calculated for the AOOs because of the conservative input used and because the most limiting AOOs (hot zero power CEA Withdrawal and CEA Withdrawal at 20% power) result in a total strain of less than 1%. The strain capability of ZIRLO™ is projected to be in excess of 1% at the burnup limit.

The 1% limit is a vendor limit applied as a strain criterion in fuel mechanical design analysis. The limit only addresses that amount of mechanical strain, independent of the possibility of DNB, induced in the clad as a result of pellet swelling due to thermal expansion caused by power density changes.

For postulated accidents, the strain limit imposed by DNB propagation concerns is a clad ballooning issue due to the stresses caused by clad temperature changes. For the DNB propagation limit, the time in DNB is used. There is no overpower calculation to determine the conditions to meet the strain limit. The AOO and accident limits are independent of each other.

Per Chapter 4.2 of the Standard Review Plan and per AREVA's generic fuel design criteria topical (Reference 11), the fuel system is required to not fail as a result of normal operation and AOOs. Accordingly, AREVA's cladding strain methodology is designed to show that the cladding does not reach [ ] strain during Condition I (normal operation) and Condition II (AOOs) events over the lifetime of the fuel. For accident conditions, it is not required that the cladding strain be below [ ]. Instead, it is required that deflection or failure of components not interfere with reactor shutdown or emergency cooling of the fuel rods. Fuel rod failures are permitted during postulated accidents, but they must be accounted for in the dose analysis.

The Condition II events that are limiting with respect to peak linear heat rate are Excess Load, CEA Drop, and CEA Withdrawal. Review of the peak attainable linear heat rate from each of these events shows that the CEA Drop event is the most severe. For Cycle 19 specifically, it is noted from the fuel centerline melt calculation for the event that a fuel rod operating at the Technical Specification peaking limit will experience an increase in power by a factor of 1.17 as a result of this event. Multiplying this value by the maximum axial peaking factor of 1.485, also given in the same calculation, the peak node will experience a power level equal to 1.74 times the average power of the hot fuel rod operating at the Technical Specification radial peaking limit.

The CEA Drop transient was therefore simulated for the Cycle 19 specific core by imposing the transient power factor calculated above to every burnup step for each of the predicted design power histories for a three-cycle operation of the Cycle 19 fuel. The objective of this calculation is to calculate the maximum cladding strain attained at various times in life. Results show that the CEA Drop event produces a maximum cladding strain of [ ].

The above calculation shows that an occurrence of the most severe Condition II event will not result in violation of the cladding transient strain criterion for the fuel to be inserted into Calvert Cliffs Unit 2 Cycle 19.

e. The 9% F_r margin consists of a 4% factor to account for flow changes due to the mixed core, and 5% to ensure that AREVA fuel is limiting from a power peaking standpoint. Westinghouse has determined that enforcing the 4% reduction in F_r offsets the negative effect of expected cross flow conditions in the mixed core. Therefore, Westinghouse fuel will perform no worse than in the previous cycle. Cycle 19 analyses demonstrate that there will be no failures in AREVA fuel. Therefore, for each safety event, the total number of fuel failures in the core will not increase from the previous cycle.

f. Response provided in Reference 10.

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:

As described in Reference 8, events that can be potentially limiting at part-power initial conditions are bounded by the envelope formed by the analysis performed at hot full power and hot zero power. A design feature that differentiates a Combustion Engineering plant from other designs is the Rate of Change of Power-High function. At lower initial power levels, the Rate of Change of Power-High trip reset function precludes the RCS from reaching the Power Level-High trip and the temperature obtained in the hot full power case.

The exposure dependency of the non-loss-of-coolant accident (LOCA) event analysis is dictated by the time in life conditions at the initiation of the transient. As a general guideline, beginning of cycle or end of cycle exposure conditions are used depending upon the nature of the transient response - based on the fact that the core has the least amount of negative feedback at beginning of cycle and the most amount of negative feedback at end of cycle. For events that exhibit RCS heatup, beginning of cycle neutronics parameters would add to the severity of the event response because the least amount of negative feedback would occur for these events. On the other hand, events that predominantly result in cooldown of the RCS are bounded by the use of end of cycle parameters since the most negative reactivity conditions would be detrimental for this category of events.

The combination of the above described conditions with other neutronics parameters, such as rod insertion limits, axial power shapes, radial peaking factors, etc., has a varying effect depending upon the event being analyzed. A description of these effects is provided for the different types of events.

Different Types of Events: The setpoint analysis verifies the limiting condition for operation and limiting safety system setting limits for the power level and burnup variations expected for each cycle. The overall goal of the setpoint analysis is to demonstrate for each cycle that there is margin to the specified acceptable fuel design limits at the trip setpoints for all allowed/possible power levels, peaking distributions, temperatures, pressures, and flows. The events are classified into two different categories, quasi-steady-state symmetric events and fast events.

The quasi-steady-state symmetric category represents conditions for transients that are in a quasi-steady-state thermal condition. This condition is applicable for most of the UFSAR Chapter 14 events. These include CEA Withdrawal at power, Boron Dilution at power, overcooling without trip, overheating, and depressurization events. All other events are either asymmetric or too fast to evaluate with the quasi-steady-state model. These events include loss of flow transients, CEA Ejection, CEA Withdrawal from subcritical conditions, Post Trip Steam Line Break, and CEA Drop. These fast events are handled directly with transient simulations.

[

] Based on these studies, full power and lower power operation has been verified for the quasi-steady-state transients.

a. The combination of initial power level and instrument uncertainty that provides the greatest challenge to Reactor Protective System effectiveness is the combination that begins with the least initial margin and then allows the greatest margin degradation. The analyses force the hot full power conditions to be limiting, as opposed to lower power conditions, by imposing peaking factors and a limiting axial shape that bounds those afforded by the limiting conditions for operation at hot full power. The hot zero power conditions provide the fastest power rise prior to trip. Inputs for hot zero power events are calculated using CEA insertions that result in conservative axial shapes and peaking. Therefore, the greatest challenge to reactor protection system effectiveness is analyzed.

   The greatest rise in power between event initiation and trip completion occurs at hot zero power conditions where the largest difference between initial power and the Variable High Power Trip setpoint occurs, and where the power rise is accelerated due to initiation of the event at an extremely low power level. These events are fast events and are dependent upon fuel temperature feedback to arrest the initial power rise. The peaking factors and axial shapes used in these events include distortion induced by CEA insertions.

b. The strategy for the fuel transition project has been to change plant parameters only if necessary. Therefore, the existing power dependent insertion limit definition as a function of power level (COLR Figure 3.1.6) has not been changed. Instead, analyses have been performed using the existing power dependent insertion limit curve as input. This includes verification of power peaking margins and shutdown margin, as well as neutronics input to safety analyses that are limiting at the deepest allowable rod insertion (power dependent insertion limit) for the relevant power level(s).

c. For the quasi-static events, there is margin to the specified acceptable fuel design limits as defined by the setpoint analysis. The worst margin loss during any possible event is not needed to define the limiting conditions for these classes of events. Discussion of the fast events is addressed on a case-by-case basis in CCNPP Response 3e.

d. Variation in peaking and cycle burnup is addressed above in the setpoint analysis. Overall, reactivity feedback parameters are selected to create bounding analysis inputs ased on beginning of cycle and end of cycle conditions. As such, mid-cycle cases are bounded by the beginning of cycle and end of cycle cases developed in support of the licensed methods.

e. Variation in peaking versus power level is addressed above in the setpoint analysis. The responses to the above requests for the events categorized here as fast events are summarized for the loss of flow transients, CEA Ejection, CEA Withdrawal from subcritical conditions, post trip Steam Line Break, and CEA Drop. These fast events are handled directly with transient simulations. Because these accidents are fast, they are more dependent upon the initial and final conditions of the event and the limiting safety system setting trips for linear power density and TM/LP alone may not preserve the limiting criteria. These evaluations are performed to verify that the initial limiting condition for operation limiting conditions with the changes associated with the accident meet the criteria for the event.

Loss of Coolant Flow

Beginning of cycle kinetics parameters for the transient simulation for Loss of Coolant Flow are most limiting. Based on this simulation it is shown that the initial conditions defined by the Technical Specifications have acceptable limits to the specified acceptable fuel design limits. The worst margin loss due to Loss of Coolant Flow occurs at full power since the flow trip is independent of power. [

]

CEA Ejection

CEA Withdrawal from Subcritical

Post-Trip Steam Line Break

This accident is most severe when initiated at hot zero power (largest feedwater inventory to maximize core cooldown) and with a most negative moderator temperature coefficient (end of cycle). [ ]

CEA Drop

Question 4:

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:

For the Calvert Cliffs fuel transition, only the events that would challenge the specified acceptable fuel design limits and fuel failure are analyzed by AREVA. Overpressure aspects of the events are not being analyzed by AREVA and the coolant pressure boundary integrity continues to be supported by the analyses presented in the UFSAR. The events analyzed by AREVA to replace the UFSAR analyses are selected on the basis of criteria that pertains to transition to AREVA Advanced CE-14 HTP fuel and changes in thermal-hydraulics performance and neutronics. The S-RELAP5 analysis performed to generate system response and statepoints for specified acceptable fuel design limits and fuel failure considerations are biased to exacerbate the aspects being challenged by the event, as described in the analyses.

Question 5:

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:

Response provided in Reference 10.

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:

Response provided in Reference 10.

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:

For the current analysis, as discussed in CCNPP Response 1a, the 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 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.

For the AREVA analysis, the RCP Seized Rotor S-RELAP5 calculation shows that the RCP seized rotor event results in approximately 25% reduction from the Technical Specification minimum flow. This event specific flow reduction is included along with temperature, power and pressure in the DNB calculation for this event. Inlet flow asymmetries are further modeled using [ ] flow penalty [ ]. As discussed in CCNPP Response 1a, modeling assumptions for flow mixing in the lower plenum do not have a first order effect on the minimum DNBR for this event. The analytical method applies an exit-skewed axial power shape, and the inlet flow asymmetries quickly become more uniform in a pressurized water reactor open lattice core. While there will be significant flow asymmetry at the reactor vessel inlet nozzles due to the RCP seized rotor, this asymmetry in flow will be largely dissipated by the time the flow reaches the core inlet due to mixing as the flow moves from the downcomer into the lower plenum, through the holes in the lower core plate, and through the lower tie plate. Any small remaining flow asymmetry at the core inlet will be dissipated as the flow moves through the open-lattice core to the DNBR location, which will be located near the top of the core, consistent with use of a top skewed axial power profile. Therefore this modeling parameter is applicable and conservative.

Delaying the turbine trip was found to reduce the RCS pressure by approximately [ ]. Considering the available DNB margin for this event, the impact of this change has a negligible effect on the overall DNB results.

Pre-Trip Steam Line Break 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:

Response provided in Reference 10.

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:

Prior to MSIV closure, the steam flow from both steam generators increased for both the symmetric and asymmetric cases. The cases evaluated in the pre-trip Steam Line Break analysis are as follows:

- Breaks located downstream of a MSIV which allow steam to flow to the break from all SGs prior to MSIV closure. These are called "symmetric" breaks. Symmetric breaks are simulated as breaks at the exit of the common steam header, with break size ranging from [ ]. Since these breaks are outside the Containment, the Reactor Protective System is not impacted by a harsh environment.
- Breaks located upstream of a MSIV and outside Containment which allow steam to flow to the break from all steam generators prior to MSIV closure and only from the upstream steam generator after MSIV closure. These are called "asymmetric" cases. These cases result in augmented radial peaking which reduces the margin to DNB and fuel centerline melt. The reactor trips for these cases are not affected by containment harsh conditions. This set of cases does not involve the Containment Pressure - High trip. [ ]
- Breaks located upstream of a MSIV and inside Containment which allows steam to flow to the break from all steam generators prior to MSIV closure and from only the upstream steam generator after MSIV closure. These are called "asymmetric" cases. These cases result in augmented radial peaking. The availability and uncertainties associated with reactor trips for these cases is affected by containment harsh conditions. This set of cases does involve the Containment Pressure - High trip. This set of cases is required in addition to the set of asymmetric cases with breaks outside Containment. The reason is that harsh condition effects on reactor trips may make this set of cases more limiting than the set of asymmetric cases with breaks outside Containment (even though the

outside containment break cases do not involve the Containment Pressure - High trip). [

]

For symmetric pre-trip Steam Line Break cases, there is no single-failure that would worsen the event consequences. The worst single-failure for an asymmetric pre-trip Steam Line Break case is the failure of one nuclear instrument channel, while another channel is out-of-service.

All three scenarios credit the Asymmetric Steam Generator Transient trip in that it is modeled for all cases. The trip is actuated however only for the smallest break sizes analyzed (0.5 ft^2 and 1.00 ft^2) for an asymmetric break inside or outside Containment. Larger break sizes result in either a Containment Pressure-High or Power Level-High trip prior to actuation of the Asymmetric Steam Generator Transient trip.

Figures 9-1 through 9-8 are representative plots of steam flow vs. time for the range of break sizes for each scenario. The steam flows are at the steam generator exit nozzles. For asymmetric breaks, SG-1 is unaffected and SG-2 is the affected generator. For symmetric breaks, there is no distinction.

[ ]

Figure 9-1, Steam Flow vs Time – 1.00 ft^2

[ ]

Figure 9-2, Steam Flow vs Time – 3.00 ft^2

[ ]

Figure 9-3, Steam Flow vs Time – 5.62 ft^2

[ ]

Figure 9-4, Steam Flow vs Time – 1.00 ft^2

[ ]

Figure 9-5, Steam Flow vs Time – 3.99 ft^2

Figure 9-6, Steam Flow vs Time – 1.00 ft^2

Figure 9-7, Steam Flow vs Time – 3.00 ft^2

[ ]

Figure 9-8, Steam Flow vs Time – 5.62 ft^2

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:

Response provided in Reference 10.

Post-Trip Steam Line Break 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:

Response provided in Reference 10.

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:

Technical Specification 3.3.1 requires all Reactor Protective System trip functions to be operable in Modes 1 and 2 with the following exception pertinent to Steam Line Break. The Steam Generator Pressure-Low reactor trip may be bypassed when steam generator pressure is < 785 psig. Technical Specification 3.3.2 requires only the Rate of Change of Power–High reactor protective system trip to be operational in Modes 3, 4, and 5 with any reactor trip circuit breakers closed and any CEA capable of being withdrawn. Since the post-trip Steam Line Break analyses assume a reactor trip at the initiation of the event, the above Technical Specification requirements for the Reactor Protective System trips do not affect the Steam Line Break calculations.

Similarly, Technical Specification 3.3.4 requires all engineered safety features actuation system instrumentation to be operable in Modes 1, 2, and 3, with the following exceptions:

The Steam Generator Pressure–Low function may be bypassed when the steam generator pressure is < 785 psia. The Steam Generator Isolation Signal function and the Steam Generator Pressure–Low function are not required to be operable when all associated valves isolated by the steam generator isolation function are closed and de-activated. This indicates that the MSIVs will be closed if the steam generator isolation function is not operable. Thus, simulation of MSIV closure in the Mode 2 hot zero power Steam Line Break calculations bounds operation in lower modes.

Technical Specification 3.3.4 indicates that the Pressurizer Pressure–Low function may be manually bypassed when pressurizer pressure is < 1800 psia. Also, Technical Specification 3.3.5 indicates that the high pressure safety injection (HPSI) pumps are only required to start automatically on a Safety Injection Actuation Signal when the RCS temperature is ≥ 385°F for Unit 1 and ≥ 325°F for Unit 2. Thus, operation of HPSI may require manual action in the case of a Steam Line Break event in lower modes. In addition, Technical Specification 3.5.3 only requires one HPSI train to be operable in Modes 3 and 4 when the pressurizer pressure is < 1750 psia. Thus, the requirements for HPSI flow for Mode 3 and below are less than for Modes 1 and 2. In order to bound the reduced HPSI requirements in the case of a Steam Line Break in Mode 3 and below, the following approach has been taken.

To bound a Steam Line Break event in Mode 3 and below with reduced Emergency Core Cooling System operational requirements (i.e., only one HPSI train available and where automatic initiation of HPSI is not required per the Technical Specifications), Mode 2 hot zero power calculations were performed assuming no HPSI flow to mitigate the event. With no HPSI flow to mitigate the event, RCS cooldown gradually ceases, which terminates positive moderator reactivity feedback, and the core power asymptotically reaches a new quasi-steady power level. The peak return to power for the hot zero power case with offsite power available is about 6% of rated thermal power. The peak return to power for the hot zero power case with loss of offsite power is about 3% of rated thermal power. The return to power is not significant for these cases and there is ample DNBR and fuel centerline melt margin. These results are bounding for Mode 3 and below because; a Steam Line Break event initiated in Mode 3 and below would be initiated from a reduced RCS temperature relative to the Mode 2 calculations, the cooldown would be less severe than in Mode 2, the positive moderator reactivity feedback would be less than in Mode 2, and the return to power would be less than in Mode 2. These calculations also demonstrate that, relative to the core response, HPSI flow is not required to mitigate a Steam Line Break event in Modes 2, 3, and 4, thus supporting the reduced Emergency Core Cooling System requirements in the Technical Specifications for Mode 3 and below. The operators are assumed to take appropriate action to mitigate the event in the long-term.

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:

Response provided in Reference 10.

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:

See the discussion on CEA Ejection presented under CCNPP Response 3.

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:

Response provided in Reference 10.

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:

Reference 12 describes the CEA Ejection event, and is generically applied to all Combustion Engineering plants analyzed by AREVA. Trip reactivity does not factor into the calculation of deposited enthalpy. This event is turned around by Doppler reactivity—not trip worth.

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:

Response provided in Reference 10.

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:

Response provided in Reference 10.

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:

Response provided in Reference 10.

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:

Response provided in Reference 10.

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:

See the CCNPP Response 22, below.

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:

AREVA's generic fuel system design criteria topical (Reference 11) was reviewed to assess the applications affected by the lack of a fuel thermal conductivity degradation model in the RODEX2/2A and RODEX3A fuel performance codes. Fuel centerline melt, cladding strain, cladding fatigue, and LOCA analyses were identified as being affected for pressurized water reactor fuel. The impact on each of these applications as well as the development of penalties that bound the impact is discussed below.

Fuel Centerline Melt

The impact on fuel centerline melt temperatures and limits resulting from the lack of modeling of the degradation of fuel thermal conductivity with burnup was calculated with a code-to-code comparison between the RODEX2 and the COPERNIC fuel performance codes. COPERNIC (Reference 11) is a contemporary NRC-approved fuel performance code that models exposure dependent degradation of fuel thermal conductivity. By comparing RODEX2 and COPERNIC results using the appropriate approved methodologies for each of the two codes and a consistent set of conditions, penalty factors were developed as a function of burnup for application to RODEX2 fuel centerline melt temperature.

Temperature penalties to the melt limit were calculated as a function of fuel rod average burnup and fuel rod type (urania-gadolinia concentration) in a manner such that the burnup dependent fuel centerline melt limits predicted by RODEX2 with the reduced melt limits are bounded by the fuel centerline melt limits calculated with COPERNIC.

Cladding Strain and Fatigue

Under-prediction of fuel temperatures due to fuel thermal conductivity degradation at higher exposures by the RODEX2 code, results in calculation of non-conservative steady-state and AOO strain margins as well as margins to the cladding fatigue criterion. This is due to the fact that higher fuel temperatures will result in an increase in the pellet outer diameter (due to thermal expansion) as well as an increase in fuel swelling. A larger fuel pellet will in turn result in larger cladding strains and cladding fatigue.

AREVA has two different criteria for pressurized water reactor cladding strain (References 13 and 14); a maximum steady-state total strain (elastic + plastic) of [ ], and a maximum AOO strain of [ ]. AREVA's criterion for pressurized water reactor cladding fatigue (Reference 14) limits the cumulative fatigue usage in the cladding to [ ].

The methodology for calculating the impact of the under-prediction of fuel temperatures on the strain and fatigue margins is as follows:

The under-prediction of fuel temperatures as a function of burnup was first ascertained by benchmarking the RODEX2 code to an extended base that included thermal data beyond the approved burnup range of [ ]. This extended data base was used for the benchmarking of the modern RODEX4 fuel performance computer code that models the degradation of fuel thermal conductivity with burnup. The results of the benchmark of the RODEX2 code to the extended data base are shown in Figure 22-1. The under-prediction of fuel centerline temperature with burnup is shown beginning at a burnup of approximately [ ]

Figure 22-1 shows that there is a single test fuel rod that falls below the predictions for the rest of the data base. The RODEX2 predictions for this single test fuel rod were used to calculate the adjustment factor for the degradation of fuel thermal conductivity with exposure. This bounding (lower) data set was then analyzed to provide a temperature uncertainty at the 95/95 confidence level which also bounds the remainder of the benchmark data.

[ ]

Figure 22-1, RODEX2 Temperature Prediction Deviation in % versus Exposure

Figure 22-2 shows the RODEX2 benchmark results with the burnup dependent adjustment factor applied to the predicted temperatures. The RODEX2 best estimate temperature has been effectively increased to account for the degradation of fuel thermal conductivity over the burnup range. Note that the adjustment factor was derived based on the benchmark results of the lowest under predicted test fuel rod and is greater than 95% of the (single fuel rod's) distribution of temperatures with a 95% confidence.

[ ]

Figure 22-2, RODEX2 Measured / Adjusted Temperature Prediction versus Exposure

As expected, the factor increases with burnup, with a maximum value of [ ] at the end-of-life burnup. This temperature uncertainty factor was used to compute an increase in fuel pellet diameter due to thermal expansion and swelling effects. The resulting larger fuel pellet was very conservatively used as input to the RODEX2 code at beginning-of-life and the strain and fatigue calculations were repeated to calculate new strain values at various times in life encompassing the entire burnup range.

As expected, the results showed an increase in the cladding strains and fatigue. The maximum increases in AOO strains and steady-state strains were [ ] respectively. The maximum increase in the fatigue usage factor was [ ]. These values are applied as penalties to the original strain and fatigue values calculated using the approved RODEX2 code and methodology. Sufficient margins to criteria limits exist to offset the impact due to degradation of fuel thermal conductivity.

Realistic Large Break LOCA

The RODEX3A fuel model has been incorporated into the S-RELAP5 code to calculate fuel response for transient analyses. The S-RELAP5/RODEX3A model does not calculate the burnup response of the fuel. Instead, fuel conditions at the burnup of interest are transferred via a binary data file from RODEX3A to S-RELAP5, establishing the initial state of the fuel prior to the transient. The data transferred from RODEX3A describes the fuel at zero power. A steady-state S-RELAP5 calculation is required to establish the fuel state at power. The transient fuel pellet radial temperature profile is computed by solving the conduction equation of S-RELAP5. Material properties are taken from RODEX3A and incorporated into S-RELAP5. The RODEX3 topical report, (Reference 15, Appendix B) details the calculation of the radial temperature distribution.

A polynomial transformation adjustment is applied to the entire fuel pellet. The polynomial transformation provides a bias adjustment to the fuel centerline temperature. A sampled parameter provides a random assessment and adjustment of the centerline temperature uncertainty. These are combined and the total adjustment is achieved by iterating a multiplicative adjustment to the fuel thermal conductivity until the desired fuel centerline temperature is reached. Thus, the adjustment is applied to the entire pellet but with variance according to the nodal pellet temperature and the distance from the node to the pellet surface.

Original:

The first transformation applies a linear adjustment if the analysis is being performed for fuel which has an exposure of 10 GWd/MTU or higher.

[ ]

where:

T_{new} = New fuel centerline temperature (°F)
B = Burnup (GWd/MTU)
$T_{original}$ = Base fuel centerline temperature (°F)

2nd Transformation — Adds a value which is determined from a random sampling range of [ ] from a Gaussian distribution.

Revised:

1st Transformation — Instead of adding the linear transformation after 10 GWd/MTU, a different form of correction factor should be applied.

[ ]

where:

T_{new} = New fuel centerline temperature (K)
B = Burnup (GWd/MTU)
$T_{original}$ = Base fuel centerline temperature (K).

2nd Transformation — Remains the same as the original method.

The justification for this process comes from analyzing the fuel rod database used for the development of RODEX4. A calculation was created that used RODEX3A to compute fuel centerline temperatures using all the points in the RODEX4 database (Reference 16). Three cases (cases 432R2, 432R6, and 597R8) were not used from the RODEX4 database. Case 597R8 was not needed for the present application. Cases 432R2 and 432R6 were rod studies that were not configured in a manner which are to be used in these types of comparisons.

The fractional difference between the RODEX3A calculated results and the data in the RODEX4 database was calculated. The temperature fraction for each point in the database was computed as follows:

$$T_{fraction} = \frac{T_{rodex3A} - T_{data}}{T_{rodex3A}}$$

where:

$T_{fraction}$ = Delta fractional temperature of computed to data (K)
$T_{rodex3A}$ = Temperature computed by RODEX3A (K)
T_{data} = Temperature from the RODEX4 database (K).

A polynomial curve fit was generated from this data set. Figure 22-3 is the plot of this data and the curve fit.

[ ]

Figure 22-3, Fractional Fuel Centerline Temperature Delta between RODEX3A and Data

The curve fit was then inverted about the zero axis. This new polynomial correction is applied regardless of fuel exposure. Figure 22-4 shows how the new correction factor changes the results. The data for this plot were created by subtracting $T_{rodex3A}$ from T_{data} as a function of burnup.

Figure 22-4, Fuel Centerline Temperature Delta of RODEX3A Calculations versus Data (Original and Using the New Correlation)

The new fuel centerline temperatures no longer have a bias off of the zero error line. The approach to use a fractional based correction algorithm was requested by the NRC. Based on the plot of T_{rodex} - T_{data}, the uncertainty used in the original basis does not need to be altered. No specific temperature bias is identified in the uncertainty of the data. Therefore retaining the current Gaussian distribution sampled from [ ] is acceptable.

Small Break LOCA

The RODEX2 code is used to determine the initial core and hot pin stored energy for small break LOCA evaluations. Small break LOCAs evolve through a pump coastdown and natural circulation phase to a loop draining phase followed by a boil-down and refill phase. The pump coastdown phase lasts approximately 100 seconds. For most of this phase a single or two phase forced circulation exists within the RCS which prevents a cladding temperature excursion and acts to remove the initial energy of the fuel and deposit it in the steam generators or the Containment. In either case, the energy content of the fuel has been reduced to that required to transport decay heat out of the fuel by the end of the coastdown phase. Thus, the peak cladding temperatures, which occur later in the transient depend on decay heat versus heat transfer and have no relationship to the initial stored energy within the fuel. This was demonstrated in a recent sensitivity study performed for the U.S. EPR. In this study, the centerline fuel temperature of a reference case was raised by 600°F with a negligible impact on the peak cladding temperature which occurred at an extended time. Thus, the adjustments (RODEX2 corrections) made to the initial fuel temperature there will have no significant effect on the small break LOCA cladding temperatures or the local oxidation.

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 degraded 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. Response provided in Reference 10.

b. Response provided in Reference 10.

c. Response provided in Reference 10.

d. Response provided in Reference 10.

e. Response provided in Reference 10.

f. Response provided in Reference 10.

g. Response provided in Reference 10.

h. A comparative evaluation of the 0.08 ft^2 break size (the limiting break size in the Combustion Engineering analysis) was performed based on the available information for the Combustion Engineering analysis in the UFSAR and the AREVA analysis.

   The primary reason for the large difference in peak cladding temperatures is a difference in loop seal clearing behavior. The Combustion Engineering analysis clears one loop seal for this break size, while the AREVA calculation clears two loop seals. This difference in loop seal clearing behavior results in the following differences between the analyses.

   With only one loop seal clearing, the Combustion Engineering analysis results in more mass in the loop seals, less mass in the reactor vessel, and a larger pressure drop from the core exit to the break to relieve steam from the core, which produces an earlier and deeper core uncovery. At the point of minimum two-phase mixture level in the respective analyses, the minimum two-phase mixture level for the Combustion Engineering analysis is about 3.5 ft deeper in the core than for the AREVA calculation. The peak cladding temperature node in the Combustion Engineering analysis is at the most top node in the core. The peak cladding temperature node in the AREVA analysis is approximately [ ] lower in the core and has approximately [ ] higher power than the peak cladding temperature node in the Combustion Engineering analysis. The initial heatup rate for the peak cladding temperature node in the AREVA calculation is over twice the initial heatup rate for the peak cladding temperature node in the Combustion Engineering calculation. However, this does not last long, as the rate of cladding heatup for the AREVA peak cladding temperature node tapers off significantly with time. The most dominant effect is a significantly lower coolant temperature at the peak cladding temperature node for the AREVA calculation compared to the Combustion Engineering calculation. The peak coolant temperature at the peak cladding temperature node in the AREVA calculation is about [ ], while the peak coolant temperature at the peak cladding temperature node in the Combustion Engineering calculation is about [ ]. The major reason for this large difference is a significantly shorter distance from the two-phase mixture level to the peak cladding temperature node in the AREVA calculation.

   Other contributing factors are as follows: The time of beginning of cladding heatup at the peak cladding temperature node is about [ ] later for the AREVA calculation. At this time, the decay heat is about [ ] lower than at the beginning of heatup for the Combustion Engineering calculation. Also, with the delayed time of beginning of heatup in the AREVA calculation, the saturation temperature is about [ ] lower for the AREVA calculation (due to RCS depressurization). That means the cladding temperature at the time of beginning of heatup is about [ ] lower for the AREVA calculation. Also, the Combustion Engineering calculation applied a [ ] penalty to the HPSI flow rate that was not incorporated in the AREVA calculation. The AREVA calculation used the HPSI flow rate reported in the UFSAR. The reduced HPSI flow rate in the Combustion Engineering calculation contributed to the deeper core uncovery compared to the AREVA calculation. The reduced HPSI flow in the Combustion Engineering calculation is estimated to account for about [ ] difference in the peak cladding temperature.

With the difference in loop seal clearing behavior, the [ ] prior to initiation of the safety injection tank flow, which then turns over the cladding temperature. Considering the combined effect of loop seal clearing, RCS depressurization, HPSI flow, and safety injection tank flow, the overall peak cladding temperature in the AREVA analysis occurs at a larger break size than for the Combustion Engineering analysis.

i. The base AREVA small break LOCA analysis was re-performed following the receipt of the NRC staff questions to [ ], consistent with the AREVA approved small break LOCA methodology. As a result, a new break spectrum and RCP trip analysis supporting Calvert Cliffs was created.

The above analysis model is used as the baseline for the additional small break LOCA sensitivity study that was conducted in response to the NRC staff's questions 23i, 23k, and 23n as presented below.

The S-RELAP5 model that was utilized to address the NRC staff questions and perform the separate small break LOCA break spectrum sensitivity analysis contains the following modifications:

- removal of the hot leg nozzle gaps and upper core barrel leakage paths
- introduction of reactivity feedback effects
- higher safety injection tank temperature

[ ] Moderator density and Doppler fuel temperature reactivity effects were added to the model, basing the moderator feedback curve on the most positive moderator temperature coefficient ([ ], see CCNPP Response 23d) and the Doppler temperature coefficient on a conservatively low value for minimum feedback at beginning of cycle [ ]. Safety injection tank temperature was raised to 120°F for this analysis; the maximum expected ambient containment temperature, which is also the value assumed in the Combustion Engineering small break LOCA analysis (see CCNPP Response 23n).

With moderator and Doppler reactivity feedback, the core power response for the [ ] break (Figure 23-5) is similar to the response shown for the Combustion Engineering analysis.

In general, the peak cladding temperature difference between the base small break LOCA analysis and the sensitivity study performed in response to the NRC staff questions was observed to be insignificant.

Table 23-1, Summary of Small Break LOCA Break Spectrum Results

[ ]

[ ]

[ ]

Table 23-2, Event Times for Break Spectrum Cases

[ ]

[

[ ]

Figure 23-5, Core Power Behavior

j. Response provided in Reference 10.

k. Moderator and Doppler reactivity feedback has been included in the break spectrum sensitivity study performed in response to NRC staff question 23i.

l. Response provided in Reference 10.

m. Response provided in Reference 10.

n. The injection lines at the cold leg connections are 10.5-inch inner diameter pipes. The cold leg flow area for Calvert Cliffs is 4.9087 ft^2. The small break LOCA methodology (Reference 9) is applicable up to 10-percent of the cold leg area. Therefore, the severed injection line falls outside of the range of applicability of Reference 9, since a 10.5-inch pipe (0.60 ft^2) is greater than 0.491 ft^2. The severed injection line falls into the break category known as an intermediate or transition break range.

   Smaller safety injection piping exists upstream of the final reverse flow check valves in the 10.5" injection line piping which are typically situated within about 12 ft of the cold leg connection. Severance of this smaller piping upstream of the check valve would be of negligible consequence due to the action of the check valve closing due to high RCS pressure on the cold leg side of the valve.

The RLB LOCA Evaluation (Enclosure 1 of Reference 17) discusses the intermediate break sizes and provides a disposition in Section 4.6.3. Table 4-5 in Enclosure 1 of Reference 17 shows the minimum peak cladding temperature difference between true large breaks and intermediate size breaks. Since the submittal of Reference 17, the supporting calculation for Enclosure 1, Table 4-5 was updated with new plant data from recent analyses. The updated data is shown below.

[ ]

As can be seen from the table, the intermediate break range will be several hundred degrees below those in the true large break LOCA range. Therefore, these breaks will not provide a limit or a critical measure of emergency core cooling system performance. Given that the large break spectrum bounds the intermediate spectrum, the use of only the large break spectrum meets the requirements of 10 CFR 50.46 for breaks within the intermediate break LOCA spectrum, and the method demonstrates that the Emergency Core Cooling System for the plant meets the criteria of 10 CFR 50.46 with high probability.

Based on the discussion above, the analysis of a 10.5-inch severed injection line is not needed and the intermediate break LOCA spectrum meets the 10 CFR 50.46 criteria.

o.a A step is being added to the existing emergency operating procedure, EOP-0, Post Trip Immediate Actions, to secure the reactor coolant pumps in the event of a small break LOCA. 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. The location of the step within the procedure assures that the RCPs will be secured in accordance with the accident analyses. The new step's action is identical to the existing Pressure and Inventory steps for conditions of containment isolation and failure to meet RCP pump curves, which also require securing the RCP's. Therefore, the actions required by the Reactor Operator are familiar and previously trained upon, both in the classroom and the simulator.

The specific step to be added to the Pressure and Inventory Safety Function is: "IF RCS subcooling drops below 15°F, THEN trip ALL RCP's." Note that this new step is based upon RCS subcooling which is a redundant, clearly visible to the operator, and alarmed indication. In addition the step contains succinct and clear language which requires unambiguous action. The importance of performing the step is easily understood and clear to the operator.

Additionally, the training scope is clear and the task uncomplicated. This procedure revision is scheduled to be completed and incorporated into operator training during the training cycle that occurs just before the refueling outage when the new AREVA fuel is loaded into the reactor vessel.

o.b Portions of the draft revision of EOP-0 which incorporates this step is enclosed (Enclosure 1).

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:

In Technical Specification 3.2.1, there are two surveillance requirements that are applicable when using the excore detector monitoring system to determine linear heat rate. One surveillance requirement (SR 3.2.1.1) verifies the value of F_{xy}, while the other surveillance requirement (SR 3.2.2.2) verifies the axial shape index alarm setpoints. We have proposed removing the surveillance requirement that verifies the value of F_{xy} (SR 3.2.1.1) for two reasons.

First, the total planar radial peaking factor (F_{xy}) is not determined as part of the AREVA analytical methodology. The total planar radial peaking factor limit and associated curves will be removed from the COLR when the transition to AREVA fuel begins. The analyses for the mixed Westinghouse and AREVA nuclear fuel cores (and the full AREVA nuclear fuel cores) is performed by AREVA using their approved methodologies, which does not include F_{xy}.

Second, the Standard Improved Technical Specifications (NUREG-1432) for Combustion Engineering plants does not include a surveillance requirement to monitor linear heat rate using the total planar radial peaking factor. Instead, there is a surveillance requirement to monitor linear heat rate using the axial shape index alarm setpoints. This surveillance requirement was determined to be acceptable as a method of monitoring linear heat rate when using excore detectors. No additional surveillance was determined to be necessary in the Standard Improved Technical Specifications. When Calvert Cliffs converted to the Improved Technical Specifications, we retained an original Technical Specification surveillance requirement to use F_{xy} to monitor linear heat rate, as well as, including the NUREG-1432 surveillance requirement to monitor linear heat rate using axial shape index setpoints.

Since we will no longer be calculating F_{xy}, there is no limit in the COLR to compare it to for the surveillance requirement. Other Combustion Engineering plants that do not calculate F_{xy}, do not have a surveillance requirement to monitor linear heat rate using F_{xy} and do not have an additional surveillance requirement. Therefore, it is acceptable and within the industry norm to remove the surveillance requirement to monitor linear heat rate using F_{xy}, retain the surveillance requirement to monitor linear heat rate using the axial shape index alarm setpoints, and not add an additional surveillance requirement.

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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

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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

10. 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

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16. EMF-2994(P), Revision 4, "RODEX4: Thermal-Mechanical Fuel Rod performance Code – Theory Manual," December 2009

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ENCLOSURE 1

Draft Revision of EOP-0, Post Trip Immediate Actions

Calvert Cliffs Nuclear Power Plant, LLC
November 19, 2010

IV. ACTIONS

IMMEDIATE ACTIONS	**ALTERNATE ACTIONS**
A. VERIFY THE REACTIVITY CONTROL SAFETY FUNCTION IS SATISFIED.	
1. Depress ONE set of Manual RX TRIP buttons.	
2. Check the Reactor has tripped by the following: • Prompt drop in NI power • Negative SUR	2.1 **IF** the Reactor has **NOT** tripped, **THEN** perform the following actions: a. De-energize the CEDM Motor Generator Sets: • Open 22A 480V BUS FDR • Open 22A-22B 480V BUS TIE • Open 23A 480V BUS FDR • Open 23A-23B 480V BUS TIE b. Check the Reactor has tripped by the following: • Prompt drop in NI power • Negative SUR **NOTE** When re-energizing 22A and 23A 480V Buses, the breaker lineup should be returned to that existing prior to the trip. c. Energize 22A and 23A 480V Buses as follows: (1) Energize 22A 480V Bus by closing its normal feeder breaker **OR** its tie breaker. (2) Energize 23A 480V Bus by closing its normal feeder breaker **OR** its tie breaker.
(continue)	

IV. ACTIONS

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
A. (continued)	
	NOTE When Boration has been commenced, the Immediate Action for this step is considered complete.
3. Check that **NO** more than ONE CEA is **NOT** fully inserted.	3.1 **IF** more than ONE CEA fails to fully insert, **THEN** borate the RCS to at least 2300 ppm as follows: a. Shut the VCT M/U valve, 2-CVC-512-CV. b. Open the BA DIRECT M/U valve, 2-CVC-514-MOV. c. Open the BAST GRAVITY FD valves: • 2-CVC-508-MOV • 2-CVC-509-MOV d. Verify the M/U MODE SEL SW, 2-HS-210, is in MANUAL. e. Start a BA PP. f. Shut the VCT OUT valve, 2-CVC-501-MOV. g. Start **ALL** available CHG PPs.
4. Verify demineralized water makeup to the RCS is secured as follows: • 21 and 22 RCMU PPs are secured • VCT M/U valve, 2-CVC-512-CV, is shut • **IF** RCS Makeup is in Direct Lineup, **THEN** the RWT CHG PP SUCT, 2-CVC-504-MOV, is shut	

IV. ACTIONS

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
B. ENSURE TURBINE TRIP.	
1. Check the Reactor has tripped.	
2. Ensure the Turbine has tripped by performing the following actions:	
a. Depress the U-2 MAIN TURB TRIP button.	
b. Check the TURBINE THROTTLE valves shut.	b.1 **IF ANY** TURBINE THROTTLE valve failed to shut, **THEN** shut **BOTH** MSIVs.
c. Check Turbine speed drops.	c.1 **IF** Turbine speed does **NOT** drop, **THEN** shut **BOTH** MSIVs.
	NOTE If Coastdown is initiated, the Turbine Generator Output breakers will remain closed for up to 20 seconds.
d. **IF** the Turbine was paralleled to the grid, **THEN** check the Turbine Generator Output breakers open: • 21 GEN BUS BKR, 0-CS-552-61 • 21 GEN TIE BKR, 0-CS-552-63	d.1 **IF** the Unit 2 "COAST DOWN INTIATED" alarm is clear, **OR** the Turbine has been tripped for 20 seconds, **THEN** open the Turbine Generator Output breakers: • 21 GEN BUS BKR, 0-CS-552-61 • 21 GEN TIE BKR, 0-CS-552-63
e. Verify 21 GEN EXCITER FIELD BKR, 2-CS-41, is open.	
(continue)	

IV. ACTIONS

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
B. (continued) 3. Isolate the MSR as follows: a. Depress the RESET Button on the MSR Control Panel. b. Verify the Main Steam to MSR Second Stage Control Valves shut: • VALVE #1 (2-MS-4018-CV) • VALVE #2 (2-MS-4019-CV) • VALVE #3 (2-MS-4017-CV) • VALVE #4 (2-MS-4020-CV)	b.1 **IF ANY** Main Steam to MSR Second Stage Control Valve fails to shut, **THEN** dispatch an operator to shut the MAIN STEAM SUPPLY TO MSR 2ND STAGE ISOLATION VALVE: • (2-MS-4018-CV) 2-MS-341 • (2-MS-4019-CV) 2-MS-343 • (2-MS-4017-CV) 2-MS-346 • (2-MS-4020-CV) 2-MS-348

C. VERIFY THE VITAL AUXILIARIES SAFETY FUNCTION IS SATISFIED.

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
1. Check 21 **OR** 24 4KV Vital Bus is energized. (continue)	**CAUTION** **Attempts should NOT be made to re-energize a bus if a fault is suspected.** 1.1 **IF BOTH** 4KV Vital Buses are de-energized, **THEN** energize 21 **OR** 24 4KV Vital Bus from a DG by performing the following: a. Start the 0C DG using the 0C DG EMERGENCY START PB, 0-HS-0707. b. Verify 2A or 2B DG is running. (1) Verify the associated DG OUT BKR closed: • 21 Bus - 152-2103 • 24 Bus - 152-2403 (continue)

IV. ACTIONS

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
C.1 (continued) (continue)	C.1 (continued) 1.2 **IF** 2A and 2B DGs can **NOT** be loaded **AND** 13KV is available, **THEN** energize 21 **OR** 24 4KV Vital Bus as follows: a. Verify the DG OUT BKR is open. b. Place the 4KV BUS LOCI/SD SEQUENCER MANUAL INITIATE keyswitch in ON. c. Insert the sync stick **AND** close the alternate 4KV feeder breaker. **NOTE** Exit from EOP-0 shall **NOT** be delayed in anticipation of 0C DG availability. 1.3 **IF** 21 and 24 4KV Buses are de-energized, **THEN** use the 0C DG to energize 21 **OR** 24 4KV Bus as follows: a. Verify the 0C DG is running. b. Verify 07 4KV BUS FDR, 152-0704 is open. c. Place 0C DG 21 (24) 4KV BUS FDR, 152-2106 (152-2406) in PULL TO LOCK. d. Place 2A (2B) DG OUT BKR, 152-2103 (152-2403) in PULL TO LOCK. (continue)

IV. ACTIONS

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
C.1 (continued) 2. **IF EITHER** 21 **OR** 24 4KV Vital Bus is **NOT** energized **AND** the 0C DG is **NOT** running, **THEN** depress the 0C DG EMERGENCY START PB, 0-HS-0707. (continue)	C.1.3 (continued) **NOTE** Performance of EOP-0 should continue concurrently while an operator is operating disconnects. e. Dispatch an operator to operate disconnects as follows: (1) Obtain the 189-2106 (189-2406) disconnect keys from the CR key locker. (2) Close 0C DG 21 (24) 4KV BUS DISC, 189-2106 (189-2406). f. **WHEN** the 0C DG is up to rated speed and voltage, **THEN** verify the 0C DG OUT BKR, 152-0703 closed. g. **WHEN** disconnect 189-2106 (189-2406) is closed **AND** breaker 152-0703 is closed, **THEN** perform the following: (1) Close 07 4KV BUS TIE, 152-0701. (2) Insert the sync stick **AND** close the 0C DG 21 (24) 4KV BUS FDR, 152-2106 (152-2406).

IV. ACTIONS

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
C. (continued)	
3. Check **ALL** 125V DC BUS VOLTS greater than 105 volts: • 11 • 12 • 21 • 22	
4. Check at least THREE 120V AC Vital Buses are energized: • 21 • 22 • 23 • 24	
5. Check **EITHER** 2Y09 **OR** 2Y10 energized.	
6. Verify Component Cooling flow to the RCPs.	6.1 **IF** Component Cooling flow can **NOT** be verified to the RCPs, **THEN** verify **ALL** RCPs are secured.
7. **IF ANY** electrical bus perturbations have occurred, **THEN** dispatch an operator to verify Switchgear Room Ventilation operating **PER** OI-22H, SWITCHGEAR VENTILATION AND AIR CONDITIONING.	

IV. ACTIONS

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
D. VERIFY THE RCS PRESSURE AND INVENTORY CONTROL SAFETY FUNCTION IS SATISFIED.	
1. Check pressurizer pressure stabilizes between 1850 and 2300 PSIA **AND** is trending to 2250 PSIA.	1.1 Operate heaters and sprays to restore and maintain pressurizer pressure between 1850 and 2300 PSIA. 1.2 **IF** pressurizer pressure is less than 2300 PSIA, **AND** the PORV(s) can **NOT** be verified closed as indicated by the PZR RV FLOW MON or red PORV energized light, **THEN** perform the following: a. Shut the associated PORV BLOCK valve(s). b. Place the associated PORV OVERRIDE handswitch(s) in the OVERRIDE TO CLOSE position. 1.3 **IF** pressurizer pressure drops to 1725 PSIA, **THEN** verify SIAS actuation.
(continue)	(continue)

IV. ACTIONS

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
D.1 (continued)	D.1 (continued) 1.4 Perform the RCP Trip Strategy as follows: a. **IF** RCS pressure drops to 1725 PSIA, **THEN** trip RCPs so **EITHER** of the following pairs remains running: • 21A and 22B RCPs • 21B and 22A RCPs b. **IF** CIS has actuated, **THEN** trip **ALL** RCPs. c. **IF** RCS subcooling drops below 15° F, **THEN** trip **ALL** RCPs. 01001 d. **IF** RCS pressure drops below the minimum pump operating limits **PER** ATTACHMENT (1), RCS PRESSURE TEMPERATURE LIMITS, **THEN** trip **ALL** RCPs.
2. Check pressurizer level stabilizes between 80 and 180 inches **AND** is trending to 160 inches.	2.1 Operate charging and letdown to restore and maintain pressurizer level between 80 and 180 inches.
3. Ensure RCS subcooling is greater than 30° F.	

IV. ACTIONS

IMMEDIATE ACTIONS	**ALTERNATE ACTIONS**
E. VERIFY THE CORE AND RCS HEAT REMOVAL SAFETY FUNCTION IS SATISFIED.	
	NOTE Shutting the MSIVs causes a loss of Main Feedwater and the Turbine Bypass Valves.
1. Verify the TURB BYP valves **OR** the ADVs operate to maintain the following: • S/G pressures between 850 and 920 PSIA • T_{COLD} between 525 and 535° F	1.1 **IF** S/G pressure drops to 800 PSIA, **THEN** shut both MSIVs. 1.2 **IF** S/G pressure drops to 685 PSIA, **THEN** verify SGIS actuated. 1.3 **IF** the pressure differential between 21 and 22 S/G is 115 PSID or greater, **THEN** verify AFAS Block is actuated to the S/G with the lower pressure. 1.4 **IF** SGIS has actuated, **AND** the cooldown terminates, **THEN** stabilize T_{COLD}.
2. Verify at least ONE S/G is available for controlled heat removal: • S/G level between (-)170 and (+)30 inches • Main or Auxiliary Feedwater operating in Auto or Manual control to maintain level • T_{COLD} greater than 525° F	2.1 **IF** S/G WR level drops to (-)170 inches, **THEN** verify AFAS actuation. 2.2 **IF** Feedwater flow is lost **OR** excessive, **THEN** perform the following actions: **CAUTION** **23 AFW PP flow limit is 300 GPM when supplied by a DG; otherwise the limit is 575 GPM.** • Start an AFW PP. • Trip the SGFPs. • Shut the SG FW ISOL valves. • Operate the AFW System to restore S/G levels to between (-)170 and (+)30 inches.
(continue)	

IV. ACTIONS

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
E. (continued)	
3. Check at least ONE RCP is operating in a loop with a S/G available for heat removal.	3.1 **IF NO** RCPs are operating in a loop with a S/G available for heat removal, **THEN** trip **ALL** RCPs.
4. **IF ANY** RCPs are operating, **THEN** check T_{HOT} minus T_{COLD} is less than 10° F in the loop(s) with a S/G available for heat removal.	4.1 **IF** T_{HOT} minus T_{COLD} is greater than 10° F in the loop(s) with a S/G available for heat removal, **THEN** trip **ALL** RCPs.
F. VERIFY THE CONTAINMENT ENVIRONMENT SAFETY FUNCTION IS SATISFIED.	
1. Check containment pressure is less than 0.7 PSIG. (continue)	1.1 **IF** containment pressure exceeds 0.7 PSIG, **THEN** perform the following: a. Verify **ALL** available CACs are operating. b. Open the CAC EMERGENCY OUT valves for the operating CACs. 1.2 **IF** containment pressure exceeds 2.8 PSIG, **THEN** verify ESFAS actuation of the following: • SIAS • CIS 1.3 **IF** CIS has actuated, **THEN** trip **ALL** RCPs. 1.4 **IF** containment pressure exceeds 4.25 PSIG, **THEN** verify CSAS actuation.

IV. ACTIONS

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
F. (continued)	
2. Check containment temperature is less than 120° F.	2.1 **IF** containment temperature exceeds 120° F, **THEN** perform the following: a. Verify **ALL** available CACs are operating. b. Open the CAC EMERGENCY OUT valves for the operating CACs. **NOTE** 23 IODINE FILT DISCs should be shifted, as required, to start at least TWO IODINE FILT FANs.
3. Check containment radiation monitor alarms are clear with **NO** unexplained rise.	3.1 **IF ANY** valid containment radiation monitor alarm is received, **THEN** start **ALL** available IODINE FILT FANs.
G. VERIFY THE RADIATION LEVELS EXTERNAL TO CONTAINMENT SAFETY FUNCTION IS SATISFIED.	
1. Check the following RMS alarms are clear with **NO** unexplained rise: • "U-2 WIDE RANGE NOBLE GAS MON" (2-RIC-5415) • "UNIT 2 CNDSR OFF-GAS" (2-RI-1752) • "UNIT 2 S/G B/D" (2-RI-4014) • "UNIT 2 MAIN VENT GASEOUS" (2-RI-5415)	1.1 **IF** a valid "UNIT 2 CNDSR OFF-GAS" or "UNIT 2 S/G B/D" alarm is received, **THEN** secure S/G Blowdown.

IV. ACTIONS

IMMEDIATE ACTIONS	ALTERNATE ACTIONS
H. PERFORM DIAGNOSTIC ACTIONS.	
1. Determine the appropriate Recovery Procedure **PER** the Diagnostic Flowchart. 2. **IMPLEMENT** the appropriate Emergency Operating Procedure. END of Section IV	