Calvert Cliffs Nuclear Power Plant

1650 Calvert Cliffs Parkway
Lusby, Maryland 20657

November 23, 2009

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
License Amendment Request – Transition from Westinghouse Nuclear Fuel to AREVA Nuclear Fuel

The Calvert Cliffs Nuclear Power Plant, LLC hereby requests an Amendment to Renewed Operating License Nos. DPR-53 and DPR-69 for Calvert Cliffs Unit Nos. 1 and 2, respectively, with the submittal of the proposed changes to the Technical Specifications.

Calvert Cliffs currently uses Westinghouse Turbo 14x14 fuel assemblies in both Units 1 and 2. As a result of continued grid-to-rod fretting fuel failures, we evaluated the use of an alternate fuel supplier to help us achieve our goal of defect-free fuel performance. The result of this evaluation led to the selection of AREVA Advanced CE-14 High Thermal Performance (HTP) fuel for use in the Calvert Cliffs reactors. Therefore, this license amendment request seeks to amend the licensing basis and the Technical Specifications to allow the use of AREVA Advanced CE-14 HTP fuel in the Calvert Cliffs reactors. The AREVA Advanced CE-14 HTP fuel design consists of standard uranium dioxide (UO_2) fuel pellets with gadolinium oxide (Gd_2O_3) burnable poison and M5® cladding. The NRC has previously approved the use of similar fuel at other Combustion Engineering plants.

Attachment (1) describes the requested Technical Specification changes needed to support the requested fuel change. Attachment (2) provides the marked-up Technical Specification pages. An exemption request per 10 CFR 50.12 to use M5® cladding is contained in Attachment (3). Attachment (4) contains the technical basis to support the requested fuel change and the associated Technical Specification changes.

Two AREVA evaluations are provided as Enclosures (1) and (2) to Attachment (4). These evaluations contain information that is proprietary to AREVA, therefore, they are accompanied by separate affidavits signed by AREVA, the owner of the information (Enclosure 3). The affidavits set forth the basis on which the information may be withheld for 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 be withheld from public disclosure. The non-proprietary versions of the evaluations (Enclosures 4 and 5) are included for public disclosure.

Calvert Cliffs plans to refuel and operate with AREVA Advanced CE-14 HTP fuel beginning with the refueling outages in 2011 for Unit 2 and 2012 for Unit 1. The transition is planned to occur over three refueling cycles on each Unit. Calvert Cliffs requests review and approval for the use of AREVA Advanced CE-14 HTP fuel in Units 1 and 2, including use in a mixed core by January 1, 2011 with implementation to occur for Unit 2 no later than the startup from the Unit 2 spring 2011 refueling outage.

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Unit 1 is scheduled to begin loading the AREVA Advanced CE-14 HTP fuel during the spring 2012 refueling outage. Because the affected Technical Specifications are common to both Units, we request a staggered implementation be allowed for Unit 1, with Unit 1 implementation to occur no later than startup from the Unit 1 spring 2012 refueling outage.

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

Very truly yours,

Thomas E. Trepanier
Plant General Manager

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

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

Subscribed and sworn before me, a Notary Public in and for the State of Maryland and County of Calvert, this 23 day of November, 2009.

WITNESS my Hand and Notarial Seal:

Wendy L. Hunter
Notary Public

My Commission Expires:

01/01/10
Date

TET/PSF/bjd

Wendy L. Hunter
NOTARY PUBLIC
Calvert County, Maryland
My Commission Expires 01/01/10

Attachments:
(1) Evaluation of the Proposed Technical Specification Changes
(2) Marked Up Technical Specification Pages
(3) M5® Cladding Exemption Request
(4) Reload Transition Report
Enclosures
(1) RLB LOCA Evaluation (Proprietary version)
(2) Sample Application for Non-LOCA Analysis (Proprietary version)
(3) AREVA Proprietary Affidavit
(4) RLB LOCA Evaluation (Non-proprietary version)
(5) Sample Application for non-LOCA Analysis (Non-proprietary version)

cc: **[Without Enclosures (1) and (2)]**
D. V. Pickett, NRC
S. J. Collins, NRC
Resident Inspector, NRC
S. Gray, DNR

ATTACHMENT (1)

EVALUATION OF THE PROPOSED TECHNICAL SPECIFICATION CHANGES

1.0 SUMMARY DESCRIPTION

2.0 DETAILED DESCRIPTION

3.0 TECHNICAL EVALUATION

4.0 REGULATORY EVALUATION

- 4.1 Applicable Regulatory Requirements
- 4.2 Precedent
- 4.3 Significant Hazards Determination
- 4.4 Conclusion

5.0 ENVIRONMENTAL CONSIDERATION

6.0 REFERENCES

1.0 SUMMARY DESCRIPTION

This evaluation supports a request to amend Renewed Operating Licenses DPR-53 and DPR-69 for Calvert Cliffs Nuclear Power Plant, LLC (Calvert Cliffs). This license amendment request seeks to amend the licensing basis and the Technical Specifications to allow the use of AREVA Advanced CE-14 High Thermal Performance (HTP) fuel in the Calvert Cliffs reactors. Calvert Cliffs plans to refuel and operate with AREVA Advanced CE-14 HTP fuel beginning with the refueling outage in 2011 for Unit 2 and 2012 for Unit 1. The transition is planned to occur over three refueling cycles on each Unit.

The AREVA Advanced CE-14 HTP fuel design consists of standard uranium dioxide (UO_2) fuel pellets with Gd_2O_3 burnable poison and M5® cladding.

2.0 DETAILED DESCRIPTION

Calvert Cliffs currently uses Westinghouse Turbo fuel assemblies with each assembly consisting of 176 rods (pins) and 5 guide tubes. The pins may contain fuel or a fuel/neutron poison mixture. The assembly is held together by spacer grids and is closed at the top and bottom by end fittings. Lateral support and positioning of the fuel rods within an assembly is provided by spacer grids with either cantilever tab springs or I-springs. The spacer grids are welded to five full-length guide tubes. The guide tubes provide channels which guide the control element assemblies (CEAs) over their entire length of travel and form the longitudinal structure of the assembly. In selected fuel assemblies, the central guide tube houses incore instrumentation. The fuel is low enrichment UO_2 in the form of ceramic pellets clad in Zircaloy or ZIRLO tubes which are welded into a hermetic enclosure.

Calvert Cliffs has experienced fuel failures in numerous operating cycles, including recent operating cycles, related to grid-to-rod fretting wear of the fuel cladding. Design changes have been made to address failure mechanisms as different mechanisms manifested themselves. Despite changes in fuel design features, failures continue to be observed. The causes of these failures were determined to be consistent with prior failure mechanisms. Some of the most recent failures are also attributable to manufacturing defects in the fuel rods. Therefore, Calvert Cliffs determined that a fundamental change in fuel design and fuel vendors was desirable to eliminate failures due to both design and manufacturing issues. We have chosen the AREVA Advanced CE-14 HTP fuel assemblies, manufactured by AREVA, as replacements for the current Westinghouse Turbo fuel assemblies. While AREVA does not have zero-defect performance in all of the reactors they provide fuel for, they have had relatively few fuel failures in these reactors, and no fuel failures in the reactors with the fuel design proposed for Calvert Cliffs. The AREVA Advanced CE-14 HTP fuel consists of dimensionally similar fuel as the current Westinghouse Turbo fuel. Additional design details and evaluations of the AREVA Advanced CE-14 HTP fuel are in Attachment (4).

Along with the physical fuel change, a change from Westinghouse Turbo fuel design and evaluation methods to AREVA Advanced CE-14 HTP fuel design and evaluation methods is also required. These design and evaluation methods and their acceptance criteria are described in Attachment (4).

To support the change in fuel from Westinghouse Turbo fuel to AREVA Advanced CE-14 HTP fuel (and transition cores with both fuel types) certain Technical Specifications require changes. These changes are described below and are shown on the marked up pages in Attachment (2).

A. Safety Limit (SL) 2.1.1.2, Reactor Core SLs – The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. The current SL uses a peak linear heat rate limit to prevent overheating of the fuel. The limit was chosen because it is the highest steady-state linear heat rate

at which the fuel can operate without causing the fuel centerline temperature to reach the melting point. This limit adequately addresses steady-state normal operation. However, as noted in Technical Specification Task Force traveler (TSTF)-445-A (Reference 1), some anticipated operational occurrences may result in exceeding the peak linear heat rate limit without reaching the fuel centerline melt temperature. Therefore, a more representative SL would be one that is based on peak fuel centerline melt temperature. This would address both normal operations and anticipated operational occurrences.

Therefore, we are taking this opportunity to change this specification to match TSTF-445-A, as well as adding the limit for the AREVA Advanced CE-14 HTP fuel. Specifically, the peak linear heat rate is replaced with the peak fuel centerline melt temperature in SL 2.1.1.2. Two peak fuel centerline melt temperatures are proposed to be added, one for the Westinghouse Turbo fuel currently approved for operation in the Calvert Cliffs reactors and one for the AREVA Advanced CE-14 HTP fuel specifically addressed in this request. The peak centerline melt temperature limit for Westinghouse Turbo fuel will remain in the Technical Specifications until Westinghouse Turbo fuel is no longer available for use in either Calvert Cliffs reactor core. The proposed fuel centerline melt temperature safety limits are shown below.

- For AREVA fuel, the peak centerline temperature must be maintained < 5081°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per XN-NF-79-56(P)(A), Revision 1, Supplement 1.
- For Westinghouse fuel, the peak centerline temperature must be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per CENPD-382-P-A.

The melting point of the fuel is dependent on fuel burnup and the amount and type of burnable poison used in the fuel. It is based on fuel material properties that are independent of fuel geometry and therefore, including both limits in the Technical Specification is acceptable for any co-resident fuel. The design melting point of unirradiated fuel containing no burnable poison is 5080°F for Westinghouse Turbo fuel and 5081°F for AREVA Advanced CE-14 HTP fuel. The melting point is adjusted downward from this temperature depending on the amount of burnup and amount and type of burnable poison in the fuel. The adjustment for burnup of 58°F per 10,000 MWD/MTU was accepted by the Nuclear Regulatory Commission (NRC) in topical report CENPD-382-P-A (Reference 2) for Westinghouse Turbo fuel containing erbium absorbers and XN-NF-79-56(P)(A), Revision 1, Supplement 1 (Reference 3) for AREVA Advanced CE-14 HTP fuels containing gadolinium. The specific formula for adjustment to these burnable poisons is considered proprietary and therefore cannot be included in the Technical Specifications.

B. Technical Specification (TS) 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^T) – The purpose of this Technical Specification is to limit the core power distribution to the initial values assumed in the accident analyses. The limits on linear heat rate (TS 3.2.1), total planar radial peaking factor (F_{xy}^T, TS 3.2.2), total integrated radial peaking factor (F_r^T, TS 3.2.3), azimuthal power tilt (T_q, TS 3.2.4), and axial shape index (TS 3.2.5) represent limits within which the linear heat rate algorithms are valid. These limits are obtained directly from the core reload analysis.

This proposed change would remove TS 3.2.2 in its entirety.

10 CFR 50.36(c)(2)(ii) provides the criteria for determining when a limiting condition for operation must be established in the Technical Specifications. Each criteria is addressed below.

(A) Criterion 1. *Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary.* The total planar peaking factor describes the limits for core power distribution. It does not address instrumentation of any kind, and in particular, does not address instrumentation related to detection of abnormal degradation of the reactor coolant pressure boundary.

(B) Criterion 2. *A process variable, design feature, or operating restriction that is an initial condition of a design basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.* The total planar peaking factor was previously used as an initial value in the design basis accident analyses. During and following the transition to AREVA Advanced CE-14 HTP fuel, the core reload analyses are performed using AREVA methodologies as described in Attachment (4). These methodologies do not use the total planar radial peaking factor (F_{xy}^T) as an initial value in the accident analyses. The linear heat rate algorithm limits are provided by the total integrated radial peaking factor (F_r^T, TS 3.2.3), azimuthal power tilt (T_q, TS 3.2.4), and axial shape index (TS 3.2.5). Since F_{xy}^T is not used in any relevant AREVA methodology related to accident analysis for either AREVA-fueled cores or transition cores, it does not meet Criterion 2.

(C) Criterion 3. *A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.* The total planar peaking factor only describes core power distribution limits. It does not provide any accident mitigation function. As noted above, in the current accident analyses, it is used as an initial value, but will no longer be used during or after the transition to AREVA Advanced CE-14 HTP fuel.

(D) Criterion 4. *A structure, system, or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety.* The total planar peaking factor is not significant to public health and safety. Linear heat rate algorithms are provided by other factors, all of which are currently described in the Technical Specifications.

We are requesting that the total planar peaking factor (F_{xy}^T) be removed from the Technical Specifications because it does not meet any of the requirements in 10 CFR 50.36(c)(2)(ii) for inclusion in the Technical Specifications. Note that the term F_{xy}^T and TS 3.2.2 are referenced in other Technical Specifications. With the proposed removal on this Technical Specification, these other Technical Specifications also need to be revised to remove reference to F_{xy}^T or TS 3.2.2. These requests are described below.

- Limiting Condition for Operation (LCO) 3.1.8, Special Test Exception (STE) – Modes 1 and 2 – This Technical Specification allows the suspension of the requirements of several Technical Specifications during physics testing. One of the Technical Specifications referenced in this section is LCO 3.2.2, Total Planar Radial Peaking Factor (F_{xy}^T). With the proposed deletion of TS 3.2.2, the reference to this LCO also needs to be removed from this specification.
- Surveillance Requirement (SR) 3.2.1.1, Linear Heat Rate - This periodic Surveillance Requirement verifies the value of F_{xy}^T to ensure that it remains within the range assumed in the analyses. As noted above, with the transition to AREVA Advanced CE-14 HTP fuel and

analysis methods, the value of F_{xy}^T is no longer used in the accident analyses. This amendment requests the removal of this Surveillance Requirement since it is no longer needed.

- Required Action 3.2.4.A.2, Azimuthal Power Tilt (T_q) – This Technical Specification establishes power distribution limits that are based on correlations between power peaking and the measured variables used as inputs the linear heat rate. As noted above, the Total Planar Radial Peaking Factor (F_{xy}^T) will no longer be used as an input into this determination. Required Action 3.2.4.A.2 requires verification of F_{xy}^T and F_r^T (total integrated radial peaking factor) if the azimuthal power tilt is not within limits. With the use of AREVA analytical methodologies, only F_r^T needs to be verified to support the accident analyses. Therefore, the reference to F_{xy}^T is deleted from Required Action 3.2.4.A.2 and the grammar is corrected to reflect a single variable.
- TS 5.6.5.a, Core Operating Limits Report (COLR) – This section lists the sections of the Technical Specifications that rely on the COLR to establish limits for each of those Technical Specifications. Included on that list is TS 3.2.2, Total Planar Radial Peaking Factor. As noted above, we are requesting that this Technical Specification be deleted. With the deletion of the Technical Specification, the reference to it must be removed. Therefore, we are requesting removal of the reference to TS 3.2.2 from the list in TS 5.6.5.a.

C. TS 4.2.1, Fuel Assemblies – This Technical Specification describes the fuel assemblies that are allowed to be used in the Calvert Cliffs cores. The description includes the materials allowed for use on fuel rods. This proposed change will add the allowance to use AREVAs M5® advanced alloy for fuel rod cladding and end plugs to TS 4.2.1. Attachment (4) describes these fuel assembly components. A permanent exemption to 10 CFR 50.46 and 10 CFR Appendix K is also proposed to support the use of the M5® advanced alloy. The exemption request is included as Attachment (3). A number of temporary exemptions are described in this section as well. These temporary exemptions are related to the recent lead test assembly program that will be completed in 2010. With the completion of the program, these temporary exemptions are no longer effective and the references to them can be removed from this Technical Specification.

D. TS 5.6.5.b, Core Operating Limits Report – This section provides a list of the analytical methods used to determine the core operating limits. These methods have been previously reviewed and approved by the NRC. With the change in fuel design, an accompanying change in methods used to derive the core operating limits is necessary. These approved methods must now be listed in TS 5.6.5.b. We are proposing to add the appropriate AREVA methods to the list. Additionally, a number of analytical methods can be removed from TS 5.6.5.b since they are no longer used with the change to AREVA methods. The form of each method listed follows the Improved Technical Specification format. Note that the COLR will contain the complete identification for each of the methods referenced in TS 5.6.5.b, including the report number, title, revision, date, and any supplements.

With the extensive revision to the listing, the list will be renumbered as shown on the attached marked up pages (Attachment 2).

With the transition to AREVA Advanced CE-14 HTP fuel, an accompanying change in analytical methods will occur. AREVA methodologies will be used to calculate the core operating limits for core reloads (both transition cores and full AREVA cores). The following approved methods are proposed to be added to TS 5.6.5. The use of these methods is described in Attachment (4).

1. ANF-88-133 (P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnup of 62 GWd/MTU"
2. BAW-10240(P)(A), "Incorporation of M5 Properties in Framatome ANP Approved Methods"
3. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs"
4. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel"
5. EMF-96-029(P)(A), "Reactor Analysis System for PWRs"
6. EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors"
7. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors"
8. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors"
9. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based"
10. XN-NF-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing"
11. XN-NF-78-44A, Generic Analysis of the Control Rod Ejection Transient for PWRs
12. XN-NF-79-56(P)(A), "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation"
13. XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup"
14. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations"
15. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results"

A review was also performed of the existing TS 5.6.5.b COLR list to determine if updating was needed. The review focused on methodologies that are no longer used with the transition to AREVA methods. The following methods will no longer be used to derive the core operating limits following the insertion of AREVA Advanced CE-14 HTP fuel in Calvert Cliffs' cores. These methods were used to evaluate cores loaded with Westinghouse Turbo fuel and do not apply to mixed Westinghouse/AREVA cores or full AREVA-fueled cores. We are therefore requesting that these methods be deleted from TS 5.6.5.b.

1. CENPD-199-P, "C-E Setpoint Methodology: C-E Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems"
2. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 1: C-E Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for Calvert Cliffs Units I and II"
4. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 3: C-E Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for Calvert Cliffs Units 1 and 2"
7. CEN-348(B)-P, "Extended Statistical Combination of Uncertainties"
8. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated October 21, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-348(B)-P, Extended Statistical Combination of Uncertainties"

10. CENPD-162-P-A, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution"
11. CENPD-207-P-A, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non- Uniform Axial Power Distribution"
14. CENPD-266-P-A, "The ROCS and DIT Computer Code for Nuclear Design"
15. CENPD-275-P-A, "C-E Methodology for Core Designs Containing Gadolinia - Urania Burnable Absorbers"
22. Letter from Mr. A. E. Scherer (CE) to Mr. J. R. Miller (NRC), dated December 15, 1981, LD-81-095, Enclosure 1-P, "C-E ECCS Evaluation Model Flow Blockage Analysis"
23. CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS"
24. CENPD-133, Supplement 5, "CEFLASH-4A, a FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis"
25. CENPD-134, Supplement 2, "COMPERC-II, a Program for Emergency Refill-Reflood of the Core"
26. Letter from Mr. D. M. Crutchfield (NRC) to Mr. A. E. Scherer (CE), dated July 31, 1986, "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports"
28. Letter from Mr. R. L. Baer (NRC) to Mr. A. E. Scherer (CE), dated September 6, 1978, "Evaluation of Topical Report CENPD-135, Supplement 5"
29. CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"
30. CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident"
31. Letter from Mr. K. Kniel (NRC) to Mr. A. E. Scherer (CE), dated September 27, 1977, "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P"
32. CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup"
33. Letter from Mr. C. Aniel (NRC) to Mr. A. E. Scherer, dated April 10, 1978, "Evaluation of Topical Report CENPD-138, Supplement 2-P"
34. Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. J. R. Miller (NRC) dated February 22, 1985, "Calvert Cliffs Nuclear Power Plant Unit 1; Docket No. 50-317, Amendment to Operating License DPR-53, Eighth Cycle License Application"
35. Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated May 20, 1985, "Safety Evaluation Report Approving Unit 1 Cycle 8 License Application"
36. Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. R. A. Clark (NRC), dated September 22, 1980, "Amendment to Operating License No. 50-317, Fifth Cycle License Application"
37. Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated December 12, 1980, "Safety Evaluation Report Approving Unit 1, Cycle 5 License Application"
38. Letter from Mr. J. A. Tiernan (BG&E) to Mr. A. C. Thadani (NRC), dated October 1, 1986, "Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2, Docket Nos. 50-317 & 50-318, Request for Amendment"
39. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated July 7, 1987, Docket Nos. 50-317 and 50-318, Approval of Amendments 127 (Unit 1) and 109 (Unit 2)

40. CENPD-188-A, "HERMITE: A Multi-Dimensional Space-Time Kinetics Code for PWR Transients"
41. The power distribution monitoring system referenced in various specifications and the BASES, is described in the following documents:
    i. CENPD-153-P, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System"
    ii. CEN-119(B)-P, "BASSS, Use of the Incore Detector System to Monitor the DNB-LCO on Calvert Cliffs Unit 1 and Unit 2"
    iii. Letter from Mr. G. C. Creel (BG&E) to NRC Document Control Desk, dated February 7, 1989, "Calvert Cliffs Nuclear Power Plant Unit No. 2; Docket 50-318, Request for Amendment, Unit 2 Ninth Cycle License Application"
    iv. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. G. C. Creel (BG&E), dated January 10, 1990, "Safety Evaluation Report Approving Unit 2 Cycle 9 License Application"
42. Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated May 11, 1995, "Approval to Use Convolution Technique in Main Steam Line Break Analysis - Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M90897 and M90898)
44. CENPD-199-P, Supplement 2-P-A, Appendix A, "CE Setpoint Methodology"
46. CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model"
47. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model"
53. WCAP-15604-NP, "Limited Scope High Burnup Lead Test Assemblies"

The remaining methods will be retained in TS 5.6.5.b because they are required methods as long as Westinghouse Turbo fuel is in the core or could be placed in the core. They may be removed in a future amendment request if it is determined that Westinghouse Turbo fuel is no longer available for placement in either reactor core.

These remaining methods will be renumbered as shown on the marked-up pages.

3.0 TECHNICAL EVALUATION

Both of the Calvert Cliffs reactors are the same design. The reactor is of the pressurized water type, using two reactor coolant loops. The reactor core is composed of 217 fuel assemblies and 77 CEAs. The fuel assemblies are arranged to approximate a right circular cylinder with an equivalent diameter of 136" and an active height of 136.7".

Westinghouse Turbo fuel is currently installed in both Calvert Cliffs Units and will be phased out with AREVA Advanced CE-14 HTP fuel. Analyses and evaluations of the change to AREVA Advanced CE-14 HTP fuel are described in Attachment (4). These evaluations address reactor core designs with both AREVA-only fuel, and mixed cores of Westinghouse Turbo fuel and AREVA Advanced CE-14 HTP fuel. The discussion in Attachment (4) includes an overview of mechanical design features, neutronics, thermal hydraulics, and accident analyses.

As required by their respective NRC-approved topical reports, a summary of the realistic large-break loss-of-coolant accident (LOCA) evaluation and a representative non-LOCA evaluation summary are provided as Enclosures to Attachment (4).

4.0 REGULATORY EVALUATION

4.1 Applicable Regulatory Requirements

As described in the Calvert Cliffs Updated Final Safety Analysis Report, Calvert Cliffs was designed and constructed to comply with the proposed Atomic Energy Commission general design criteria (GDC). The construction of Calvert Cliffs was significantly complete prior to the issuance of the current 10 CFR Part 50, Appendix A, General Design Criteria. Similarly, Calvert Cliffs was not designed or constructed with the benefit of the Standard Review Plan (SRP). Although Calvert Cliffs is not a GDC or SRP plant, the following SRP sections provide guidance in assessing the fuel transition. These sections specify the GDC that are applicable.

SRP 4.2	Fuel System Design
SRP 4.3	Nuclear Design
SRP 4.4	Thermal and Hydraulic Design
SRP 6.3	Emergency Core Cooling Systems
SRP 15	Accident Analyses (various sections, as applicable)

Attachment 4 identifies specific GDC and SRP sections which address specific fuel evaluations.

4.2 Precedent

Although similar AREVA fuel designs are licensed for other Combustion Engineering plants, there is not precedent that covers all aspects of the fuel design proposed for this request.

4.3 Significant Hazards Determination

The proposed amendment for Calvert Cliffs Units 1 and 2 requests changes to the Technical Specifications which support a change in fuel type from Westinghouse Turbo fuel to AREVA Advanced CE-14 High Thermal Performance (HTP) fuel. The design criteria for the new fuel (AREVA Advanced CE-14 HTP) are consistent with those for the existing fuel and ensure that the new fuel is compatible with the Calvert Cliffs reactors and the existing fuel on the basis of coolant flow and neutronic characteristics as well as departure from nucleate boiling (DNB) and peak cladding temperature requirements. The new fuel design also ensures mechanical compatibility with the existing fuel, reactor core, control rods, steam supply system, and fuel handling system. The following Technical Specification changes are requested to support the fuel transition. Technical Specification 2.1.1.2 (Safety Limits) is revised to include a limit for AREVA Advanced CE-14 HTP fuel and to change the existing Westinghouse Turbo fuel limit format to match TSTF-445-A. Technical Specification 3.2.2 (Total Planar Radial Peaking Factor) is eliminated because, using AREVA analysis methodology, it no longer meets any of the 10 CFR 50.36 (c)(2)(ii) criteria for inclusion in the Technical Specifications. With the elimination of this Technical Specification, a number of other Technical Specifications must be updated to remove references to this Technical Specification. AREVA Advanced CE-14 HTP fuel uses M5® cladding material. This cladding material must be added to Technical Specification 4.2.1 (Fuel Assemblies). The list of methods used in the development of the core operating limits must be updated to add methods to support the AREVA fuel analysis and to remove the methods no longer needed to support Westinghouse fuel analysis.

Calvert Cliffs has evaluated whether or not a significant hazards consideration is involved with the proposed amendment by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of Amendment," as discussed below:

i. *Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?*

No.

The reactor fuel and the analyses associated with it are not accident initiators. The response of the fuel to an accident is analyzed using conservative techniques and the results are compared to approved acceptance criteria. These evaluation results will show that the fuel response to an accident is within approved acceptance criteria for both cores loaded with the new AREVA Advanced CE-14 HTP fuel and cores loaded with both AREVA and Westinghouse Turbo fuel. Therefore, the change in fuel design does not affect accident or transient initiation or consequences.

The proposed change to the Safety Limit Technical Specification (2.1.1.2) does not require any physical change to any plant system, structure, or component. The change to establish the peak fuel centerline temperature as the safety limit is consistent with the Standard Review Plan (SRP) for ensuring that the fuel design limits are met. Operations and analysis will continue to be in compliance with Nuclear Regulatory Commission (NRC) regulations. The peak fuel centerline temperature is the basis for protecting the fuel and is consistent with the analogous wording for other pressurized water reactor (PWR) plants. Providing the peak fuel centerline melt temperature as the safety limit does not impact the initiation or the mitigation of an accident.

The proposed change to remove the Total Planar Radial Peaking Factor (F_{xy}^{T}, Technical Specification 3.2.2) is based on a methodology change. During and after the transition to AREVA Advanced CE-14 HTP fuel, the core analyses are performed using AREVA methodologies. These methodologies do not use the total planar radial peaking factor (F_{xy}^{T}) as an initial value in the accident analyses. The linear heat rate algorithm limits are provided by the total integrated radial peaking factor, azimuthal power tilt, and axial shape index. The linear heat rate is evaluated in accordance with NRC-approved methodology and meets acceptance criteria. The total planar radial peaking factor is not an accident initiator and does not play a role in accident mitigation. A number of other changes are also made to remove references to Technical Specification 3.2.2 throughout the Technical Specifications.

Topical reports have been reviewed and approved by the NRC for use in determining core operating limits. The core operating limits to be developed using the new methodologies will be established in accordance with the applicable limitations as documented in the appropriate NRC Safety Evaluation reports. The proposed change to add and remove various topical reports to Technical Specification 5.6.5 enables the use of appropriate methodologies to re-analyze certain events. The proposed methodologies will ensure that the plant continues to meet applicable design criteria and safety analysis acceptance criteria.

The proposed change to the list of NRC-approved methodologies listed in Technical Specification 5.6.5 is administrative in nature and has no impact on any plant configuration or system performance relied upon to mitigate the consequences of an accident. The proposed change will update the listing of NRC-approved methodologies to remove methods no longer used and add new methods consistent with the transition to AREVA Advanced CE-14 HTP fuel. Changes to the calculated core operating limits may only be made using NRC-approved methods, must be consistent

with all applicable safety analysis limits and are controlled by the 10 CFR 50.59 process. The list of methodologies in the Technical Specifications does not impact either the initiation of an accident or the mitigation of its consequences.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

ii. *Does the proposed amendment create the possibility of a new or different type of accident from any accident previously evaluated?*

No.

Use of AREVA Advanced CE-14 HTP fuel in the Calvert Cliffs reactor cores is consistent with the current plant design bases and does not adversely affect any fission product barrier, nor does it alter the safety function of safety systems, structures, or components, or their roles in accident prevention or mitigation. The operational characteristics of AREVA Advanced CE-14 HTP fuel are bounded by the safety analyses. The AREVA Advanced CE-14 HTP fuel design performs within fuel design limits and does not create the possibility of a new or different type of accident.

The proposed change to the Safety Limit Technical Specification (2.1.1.2) does not require any physical change to any plant system, structure, or component, nor does it require any change in safety analysis methods or results. The existing analyses remain unchanged and do not affect any accident initiators that would create a new accident.

The proposed change to remove the total planar radial peaking factor (F_{xy}^{T}, Technical Specification 3.2.2) is based on a change in analytical methods needed to support the physical fuel change. These methodologies do not use the total planar radial peaking factor (F_{xy}^{T}) as an initial value in the accident analysis. The total planar radial peaking factor does not play a role in accident mitigation and cannot create the possibility of a new or different kind of accident. A number of other changes are made to remove references to Technical Specification 3.2.2 throughout the Technical Specifications.

The proposed change to the list of topical reports used to determine the core operating limits is administrative in nature and has no impact on any plant configuration or on system performance. It updates the list of NRC-approved topical reports used to develop the core operating limits. There is no change to the parameters within which the plant is normally operated. The possibility of a new or different accident is not created.

Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

iii. *Does the proposed amendment involve a significant reduction in a margin of safety?*

No.

Use of AREVA Advanced CE-14 HTP fuel is consistent with the current plant design bases and does not adversely affect any fission product barrier, nor does it alter the safety function of safety systems, structures, or components, or their roles in accident prevention or mitigation. The operational characteristics of AREVA Advanced CE-14 HTP fuel are bounded by the safety analyses. The AREVA Advanced CE-14 HTP fuel design performs within fuel design limits. The proposed

changes do not result in exceeding design basis limits. Therefore, all licensed safety margins are maintained.

The proposed change to the Safety Limit Technical Specification (2.1.1.2) does not require any physical change to any plant system, structure, or component, nor does it require any change in safety analysis methods or results. Therefore, by changing the safety limit from peak linear heat rate to peak fuel centerline temperature, the margin as established in the current licensing basis remains unchanged.

The proposed change to remove the total planar radial peaking factor (F_{xy}^{T}, Technical Specification 3.2.2) is based on a methodology change. The linear heat rate algorithm limits are provided by the total integrated radial peaking factor, azimuthal power tilt, and axial shape index. The linear heat rate is evaluated in accordance with NRC-approved methodology and meets acceptance criteria. Therefore, the margin as established for the linear heat rate remains unchanged. A number of other changes are made to remove references to Technical Specification 3.2.2 throughout the Technical Specifications.

The proposed change to the list of topical reports does not amend the cycle specific parameters presently required by the Technical Specifications. The individual Technical Specifications continue to require operation of the plant within the bounds of the limits specified in the COLR. The proposed change to the list of analytical methods referenced in the COLR is administrative in nature and does not impact the margin of safety.

Therefore, the proposed change does not involve a significant reduction in the margin of safety.

Based on the above, we conclude that the proposed amendments do not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

4.4 Conclusion

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or the health and safety of the public.

5.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR Part 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or a significant increase in the amounts of any effluents that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental assessment need be prepared in connection with the proposed amendment.

6.0 REFERENCES

1. TSTF-445-A, Revision to Peak Linear Heat Rate Safety Limit, Revision 1, March 18, 2003
2. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993
3. XN-NF-79-56(P)(A), "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation," Revision 1, Supplement 1

ATTACHMENT (2)

MARKED UP TECHNICAL SPECIFICATION PAGES

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the combination of THERMAL POWER, pressurizer pressure, and the highest operating loop cold leg coolant temperature shall not exceed the limits shown in Figure 2.1.1-1.

2.1.1.2 In MODES 1 and 2, the peak ~~linear heat rate (LHR)~~ fuel centerline temperature[a] shall be ~~≤ 22.0 kW/ft.~~ maintained at:

2.1.2 Reactor Coolant System (RCS) Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psia.

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, or 5, restore compliance within 5 minutes.

a. For AREVA fuel, < 5081°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per XN-NF-79-56(P)(A), Revision 1, Supplement 1.
b. For Westinghouse fuel, < 5080°F, decreasing by 58°F per 10,000 MWD/MTU and adjusted for burnable poison per CENPD-382-P-A.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Special Test Exception (STE)-MODES 1 and 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC);"
LCO 3.1.4, "Control Element Assembly (CEA) Alignment;"
LCO 3.1.5, "Shutdown Control Element Assembly (CEA) Insertion Limits;"
LCO 3.1.6, "Regulating Control Element Assembly (CEA) Insertion Limits;"
~~LCO 3.2.2, "Total Planar Radial Peaking Factor (F^T_{xy});"~~
LCO 3.2.3, "Total Integrated Radial Peaking Factor (F^T_r);" and
LCO 3.2.4, "AZIMUTHAL POWER TILT (T_q)"

may be suspended, provided THERMAL POWER is restricted to the test power plateau, which shall not exceed 85% RTP.

APPLICABILITY: MODES 1 and 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Test power plateau exceeded.	A.1 Reduce THERMAL POWER to less than or equal to test power plateau.	15 minutes

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------
Either the Excore Detector Monitoring System or the Incore Detector Monitoring System shall be used to determine LHR.

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	~~----------NOTE---------- Only applicable when the Excore Detector Monitoring System is being used to determine LHR. ----------~~ Verify the value of F^T_{xy}.	~~Each 72 hours of accumulated operation in MODE 1~~
SR 3.2.1.2	----------NOTE---------- Only applicable when the Excore Detector Monitoring System is being used to determine LHR. ---------- Verify ASI alarm setpoints are within the limits specified in the COLR.	31 days

Deleted

3.2 POWER DISTRIBUTION LIMITS

3.2.2 Total Planar Radial Peaking Factor (F^T_{xy})

LCO 3.2.2 The calculated value of F^T_{xy} shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. F^T_{xy} not within limits.	A.1 Restore F^T_{xy} to within limits.	6 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	----------NOTE---------- F^T_{xy} shall be determined by using the incore detectors to obtain a power distribution map with all full length control element assemblies at or above the long-term steady state insertion limit, as specified in the COLR. This determination shall be limited to core planes between 15% and 85% of full core height inclusive and shall exclude regions influenced by grid effects. ---------- Verify the value of F^T_{xy}.	Once prior to operation above 70% RTP after each fuel loading AND Each 31 days of accumulated operation in MODE 1

3.2 POWER DISTRIBUTION LIMITS

3.2.4 AZIMUTHAL POWER TILT (T_q)

LCO 3.2.4 T_q shall be ≤ 0.03.

APPLICABILITY: MODE 1 with THERMAL POWER > 50% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Indicated T_q > 0.03 and ≤ 0.10.	A.1 Restore T_q to ≤ 0.03.	4 hours
	OR	
	A.2 Verify ~~F^T_{xy} and~~ F^T_r ~~are~~ is within the limits of ~~LCO 3.2.2, "Total Planar Radial Peaking Factor (F^T_{xy})," and~~ LCO 3.2.3, "Total Integrated Radial Peaking Factor (F^T_r)" ~~respectively~~.	4 hours AND Once per 8 hours thereafter
B. Indicated T_q > 0.10.	B.1 Restore T_q to ≤ 0.10.	2 hours
C. Required Action and associated Completion Time not met.	C.1 Reduce THERMAL POWER to ≤ 50% RTP.	4 hours

4.0 DESIGN FEATURES

4.1 Site Location

The site for the Calvert Cliffs Nuclear Power Plant is located on the western shore of the Chesapeake Bay in Calvert County, Maryland, about 10-1/2 miles Southeast of Prince Frederick, Maryland. The site is approximately 45 miles southeast of Washington, DC, and 60 miles south of Baltimore, Maryland. The exclusion area boundary has a minimum radius of 1,150 meters from the center of the plant.

4.2 Reactor Core

4.2.1 Fuel Assemblies

MS,

The reactor shall contain 217 fuel assemblies. Each assembly shall consist of a matrix of Zircalloy or ZIRLO fuel rods with an initial composition of natural or slightly enriched uranium dioxide (UO_2) as fuel material. Limited substitutions of zirconium alloy or stainless steel filler rods for fuel rods, in accordance with approved applications of fuel rod configurations, may be used. Fuel assemblies shall be limited to those fuel designs that have been analyzed with applicable NRC staff approved codes and methods and shown by tests or analyses to comply with all fuel safety design bases. A limited number of lead test assemblies that have not completed representative testing may be placed in nonlimiting core regions. ~~For Unit 2 Cycle 14 only, advanced cladding material may be used in one lead test assembly as described in an approved temporary exemption dated March 6, 2001. For Unit 1 Cycle 19 or Unit 2 Cycle 17 only, advanced cladding material from Framatome ANP may be used in up to two lead test assemblies as described in approved temporary exemption dated November 9, 2006. For Unit 1 Cycle 19 or Unit 2 Cycle 17 only, advanced cladding material from Westinghouse may be used in up to two lead test assemblies as described in approved temporary exemption dated November 9, 2006. For Unit 1 Cycle 19 only, advanced cladding material from AREVA may be used in up to two lead test assemblies as described in approved temporary exemption dated December 17, 2007. For Unit 1 Cycle 19 only,~~

4.0 DESIGN FEATURES

~~advanced cladding material from Westinghouse may be used in up to two lead test assemblies as described in approved temporary exemption dated December 17, 2007.~~

4.2.2 Control Element Assemblies

The reactor core shall contain 77 control element assemblies.

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.00 weight percent;

b. $k_{eff} < 1.00$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.7.2 of the Updated Final Safety Analysis Report (UFSAR) and $k_{eff} \leq 0.95$ if fully flooded with water borated to 350 ppm, which includes an allowance for uncertainties as described in Section 9.7.2 of the UFSAR;

c. A nominal 10-3/32-inch center-to-center distance between fuel assemblies placed in the high density fuel storage racks;

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. Fuel assemblies having a maximum U-235 enrichment of 5.0 weight percent;

5.6 Reporting Requirements

5.6.3 Radioactive Effluent Release Report

----------NOTE----------
A single submittal may be made for both units. The submittal should combine sections common to both units at the station.

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a, as modified by approved exemptions. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the units. The material provided shall be consistent with the objectives outlined in the ODCM, Process Control Program, and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Deleted

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

3.1.1	SHUTDOWN MARGIN
3.1.3	Moderator Temperature Coefficient
3.1.4	CEA Alignment
3.1.6	Regulating Control Element Assembly Insertion Limit
3.2.1	Linear Heat Rate
~~3.2.2~~	~~Total Planar Radial Peaking Factor~~
3.2.3	Total Integrated Radial Peaking Factor
3.2.5	AXIAL SHAPE INDEX
3.3.1	RPS Instrumentation - Operating
3.9.1	Boron Concentration

INSERT

1. ANF-88-133 (P)(A), "Qualification of Advanced Nuclear Fuels' PWR Design Methodology for Rod Burnup of 62 GWd/MTU"
2. BAW-10240(P)(A), "Incorporation of M5 Properties in Framatome ANP Approved Methods"
3. EMF-92-116(P)(A), "Generic Mechanical Design Criteria for PWR Fuel Designs"
4. EMF-92-153(P)(A), "HTP: Departure from Nucleate Boiling Correlation for High Thermal Performance Fuel"
5. EMF-96-029(P)(A), "Reactor Analysis System for PWRs"
6. EMF-1961(P)(A), "Statistical Setpoint/Transient Methodology for Combustion Engineering Type Reactors"
7. EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors"
8. EMF-2310(P)(A), "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors"
9. EMF-2328(P)(A), "PWR Small Break LOCA Evaluation Model, S-RELAP5 Based"
10. XN-NF-75-32(P)(A), "Computational Procedure for Evaluating Fuel Rod Bowing"
11. XN-NF-78-44A, Generic Analysis of the Control Rod Ejection Transient for PWRs
12. XN-NF-79-56(P)(A), "Gadolinia Fuel Properties for LWR Fuel Safety Evaluation"
13. XN-NF-82-06(P)(A), "Qualification of Exxon Nuclear Fuel for Extended Burnup"
14. XN-NF-82-21(P)(A), "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations"
15. XN-NF-85-92(P)(A), "Exxon Nuclear Uranium Dioxide/Gadolinia Irradiation Examination and Thermal Conductivity Results"

5.6 Reporting Requirements

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

Insert

~~1. CENPD-199-P, "C-E Setpoint Methodology: C-E Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems"~~

~~2. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 1: C-E Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for Calvert Cliffs Units I and II"~~

16. ~~3.~~ CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for Calvert Cliffs Units 1 and 2"

~~4. CEN-124(B)-P, "Statistical Combination of Uncertainties Methodology Part 3: C-E Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for Calvert Cliffs Units 1 and 2"~~

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

18. ~~6.~~ Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated June 24, 1982, Unit 1 Cycle 6 License Approval (Amendment No. 71 to DPR-53 and SER)

~~7. CEN-348(B)-P, "Extended Statistical Combination of Uncertainties"~~

~~8. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated October 21, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-348(B)-P, Extended Statistical Combination of Uncertainties"~~

5.6 Reporting Requirements

19. ~~9.~~ CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core"

~~10. CENPD-162-P-A, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution"~~

~~11. CENPD-207-P-A, "Critical Heat Flux Correlation of C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non-Uniform Axial Power Distribution"~~

20. ~~12.~~ CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods"

21. ~~13.~~ CENPD-225-P-A, "Fuel and Poison Rod Bowing"

~~14. CENPD-266-P-A, "The ROCS and DIT Computer Code for Nuclear Design"~~

~~15. CENPD-275-P-A, "C-E Methodology for Core Designs Containing Gadolinia - Urania Burnable Absorbers"~~

22. ~~16.~~ CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers"

23. ~~17.~~ CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report"

24. ~~18.~~ CEN-161-(B)-P-A, "Improvements to Fuel Evaluation Model"

25. ~~19.~~ CEN-161-(B)-P, Supplement 1-P, "Improvements to Fuel Evaluation Model"

26. ~~20.~~ Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated February 4, 1987, Docket Nos. 50-317 and 50-318, "Safety Evaluation of Topical Report CEN-161-(B)-P, Supplement 1-P, Improvements to Fuel Evaluation Model"

27. ~~21.~~ CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure"

5.6 Reporting Requirements

22. ~~Letter from Mr. A. E. Scherer (CE) to Mr. J. R. Miller (NRC), dated December 15, 1981, LD-81-095, Enclosure 1-P, "C-E ECCS Evaluation Model Flow Blockage Analysis"~~

23. ~~CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS"~~

24. ~~CENPD-133, Supplement 5, "CEFLASH-4A, a FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis"~~

25. ~~CENPD-134, Supplement 2, "COMPERC-II, a Program for Emergency Refill-Reflood of the Core"~~

26. ~~Letter from Mr. D. M. Crutchfield (NRC) to Mr. A. E. Scherer (CE), dated July 31, 1986, "Safety Evaluation of Combustion Engineering ECCS Large Break Evaluation Model and Acceptance for Referencing of Related Licensing Topical Reports"~~

28. ~~27.~~ CENPD-135, Supplement 5-P, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program"

28. ~~Letter from Mr. R. L. Baer (NRC) to Mr. A. E. Scherer (CE), dated September 6, 1978, "Evaluation of Topical Report CENPD-135, Supplement 5"~~

29. ~~CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model"~~

30. ~~CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss-of-Coolant Accident"~~

31. ~~Letter from Mr. K. Kniel (NRC) to Mr. A. E. Scherer (CE), dated September 27, 1977, "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P"~~

5.6 Reporting Requirements

32. ~~CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup"~~

33. ~~Letter from Mr. C. Aniel (NRC) to Mr. A. E. Scherer, dated April 10, 1978, "Evaluation of Topical Report CENPD-138, Supplement 2-P"~~

34. ~~Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. J. R. Miller (NRC) dated February 22, 1985, "Calvert Cliffs Nuclear Power Plant Unit 1; Docket No. 50-317, Amendment to Operating License DPR-53, Eighth Cycle License Application"~~

35. ~~Letter from Mr. D. H. Jaffe (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated May 20, 1985, "Safety Evaluation Report Approving Unit 1 Cycle 8 License Application"~~

36. ~~Letter from Mr. A. E. Lundvall, Jr. (BG&E) to Mr. R. A. Clark (NRC), dated September 22, 1980, "Amendment to Operating License No. 50-317, Fifth Cycle License Application"~~

37. ~~Letter from Mr. R. A. Clark (NRC) to Mr. A. E. Lundvall, Jr. (BG&E), dated December 12, 1980, "Safety Evaluation Report Approving Unit 1, Cycle 5 License Application"~~

38. ~~Letter from Mr. J. A. Tiernan (BG&E) to Mr. A. C. Thadani (NRC), dated October 1, 1986, "Calvert Cliffs Nuclear Power Plant Unit Nos. 1 & 2, Docket Nos. 50-317 & 50-318, Request for Amendment"~~

39. ~~Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. J. A. Tiernan (BG&E), dated July 7, 1987, Docket Nos. 50-317 and 50-318, Approval of Amendments 127 (Unit 1) and 109 (Unit 2)~~

40. ~~CENPD-188-A, "HERMITE: A Multi-Dimensional Space-Time Kinetics Code for PWR Transients"~~

5.6 Reporting Requirements

41. ~~The power distribution monitoring system referenced in various specifications and the BASES, is described in the following documents:~~

~~i. CENPD-153-P, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System"~~

~~ii. CEN-119(B)-P, "BASSS, Use of the Incore Detector System to Monitor the DNB LCO on Calvert Cliffs Unit 1 and Unit 2"~~

~~iii. Letter from Mr. G. C. Creel (BG&E) to NRC Document Control Desk, dated February 7, 1989, "Calvert Cliffs Nuclear Power Plant Unit No. 2; Docket 50-318, Request for Amendment, Unit 2 Ninth Cycle License Application"~~

~~iv. Letter from Mr. S. A. McNeil, Jr. (NRC) to Mr. G. C. Creel (BG&E), dated January 10, 1990, "Safety Evaluation Report Approving Unit 2 Cycle 9 License Application"~~

42. ~~Letter from Mr. D. G. McDonald, Jr. (NRC) to Mr. R. E. Denton (BGE), dated May 11, 1995, "Approval to Use Convolution Technique in Main Steam Line Break Analysis - Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 (TAC Nos. M90897 and M90898)~~

29. ~~43.~~ CENPD-387-P-A, "ABB Critical Heat Flux Correlations for PWR Fuel"

44. ~~CENPD-199-P, Supplement 2-P-A, Appendix A, "CE Setpoint Methodology"~~

30. ~~45.~~ CENPD-404-P-A, "Implementation of ZIRLO™ Cladding Material in CE Nuclear Power Fuel Assembly Designs"

5.6 Reporting Requirements

46. ~~CENPD-132, Supplement 4-P-A, "Calculative Methods for the CE Nuclear Power Large Break LOCA Evaluation Model"~~

47. ~~CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB-CE Small Break LOCA Evaluation Model"~~

31. 48. WCAP-11596-P-A, "Qualification of the PHOENIX-P, ANC Nuclear Design System for Pressurized Water Reactor Cores"

32. 49. WCAP-10965-P-A, "ANC: A Westinghouse Advanced Nodal Computer Code"

33. 50. WCAP-10965-P-A Addendum 1, "ANC: A Westinghouse Advanced Nodal Computer Code; Enhancements to ANC Rod Power Recovery"

34. 51. WCAP-16072-P-A, "Implementation of Zirconium Diboride Burnable Absorber Coatings in CE Nuclear Power Fuel Assembly Designs"

35. 52. WCAP-16045-P-A, "Qualification of the Two-Dimensional Transport Code PARAGON"

53. ~~WCAP-15604-NP, "Limited Scope High Burnup Lead Test Assemblies"~~

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, ECCS limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Not Used

ATTACHMENT (3)

M5® CLADDING EXEMPTION REQUEST

Calvert Cliffs Nuclear Power Plant, LLC
November 23, 2009

BACKGROUND

The Calvert Cliffs Units 1 and 2 cores consist of 217 fuel assemblies each. Each standard fresh fuel assembly consists of 176 fuel rods, guide tubes, fuel rod spacer grids, and upper- and lower-end fittings. The rods are arranged in a square 14x14 array. The guide tubes, spacer grids, and end-fittings form the structural frame of the assembly.

In a standard fresh fuel assembly, the fuel rods consist of slightly enriched uranium dioxide cylindrical ceramic pellets and a round wire stainless steel compression spring located at the top of the fuel column, all encapsulated within a seamless ZIRLO™ tube with a Zircaloy-4 cap welded at each end.

Title 10 CFR 50.46(a)(1)(i) states, "Each boiling or pressurized light-water nuclear power reactor fueled with uranium oxide pellets within cylindrical Zircaloy or ZIRLO cladding must be provided with an emergency core cooling system (ECCS) that must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of this section. ECCS cooling performance must be calculated in accordance with an acceptable evaluation model and must be calculated for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the most severe postulated loss-of-coolant accidents are calculated." Section 10 CFR 50.46 goes on to delineate specifications for peak cladding temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long-term cooling.

Title 10 CFR Part 50, Appendix K, paragraph I.A.5, states, "The rate of energy release, hydrogen generation, and cladding oxidation from the metal/water reaction shall be calculated using the Baker-Just equation." Since the Baker-Just equation presumes the use of Zircaloy or ZIRLO™ cladding, the use of fuel with zirconium-based alloys that do not conform to either of these two designations requires an exemption from this section of the Code.

We plan to insert AREVA Advanced CE-14 High Thermal Performance (HTP) fuel assemblies into both Units containing cladding materials that do not meet the definition of Zircaloy or ZIRLO™. The AREVA Advanced CE-14 HTP fuel assemblies are scheduled to be inserted into the cores beginning with the Unit 2 refueling outage scheduled for spring 2011. We are requesting a permanent exemption to 10 CFR 50.46 and 10 CFR Part 50, Appendix K, to support the transition of our fuel to a fuel design that uses M5® cladding.

We believe that the standards of 10 CFR 50.12 are satisfied in this case. Special circumstances are present, as described in 10 CFR 50.12(a)(2), to warrant granting the permanent exemption. They are described below.

10 CFR 50.12 REQUIREMENTS

The standards set forth in 10 CFR 50.12 provide that specific exemptions may be granted that:

- are authorized by law;
- are consistent with the common defense and security;
- will not present an undue risk to the public health and safety; and
- are accompanied by special circumstances.

We believe that the activities to be conducted under the exemption are clearly authorized by law and are consistent with the common defense and security. The remaining standards for the exemption are also satisfied, as described below.

No Undue Risk

The exemption will not present an undue risk to the public health and safety. The M5® fuel rod cladding and fuel assembly structural material has been evaluated to confirm that operation of the plant with this fuel product does not significantly increase the probability or consequences of an accident. The evaluation also concluded that no new or different type of accident will be initiated that could pose a risk to the health and safety of the public. In addition, appropriate full-core and mixed-core analyses are performed to demonstrate that this fuel type does not present an undue risk to the public health and safety. Calvert Cliffs, in conjunction with AREVA, is utilizing Nuclear Regulatory Commission (NRC)-approved methods for the reload design process for reload cores containing M5® fuel rod cladding.

In the unlikely event that cladding failures occur, the environmental impact would be minimal and is bounded by previous environmental assessments. In addition, the insertion of fuel clad with M5® will not foreclose the option of reverting to the use of standard ZIRLO™ cladding. That is, the change is not irreversible. The long-term benefits expected from the fuel conversion to M5® cladding include reduced incidence of fuel failure, higher fuel burnup, and improved thermal margin.

Special Circumstances

This request involves special circumstances as set forth in 10 CFR 50.12(a)(2).

The underlying purpose of 10 CFR 50.46 is to ensure that nuclear power facilities have adequate acceptance criteria for ECCS. The effectiveness of the ECCS in Calvert Cliffs Units 1 and 2 will not be affected by the insertion of M5® clad fuel. Due to the similarities in the material properties of the M5® cladding to Zircaloy-4 or ZIRLO™, the approved AREVA methodologies conclude that the ECCS performance would not be adversely affected.

The intent of paragraph I.A.5 of Appendix K to 10 CFR Part 50 is to apply an equation for rates of energy release, hydrogen generation, and cladding oxidation from a metal-water reaction that conservatively bounds all post-loss-of-coolant coolant (LOCA) scenarios. The approved AREVA methodologies show that due to the similarities in the composition of the M5® cladding and Zircaloy-4 or ZIRLO™, the application of the Baker-Just equation will continue to conservatively bound all post-LOCA scenarios.

The wording of the regulations renders the criteria of 10 CFR 50.46 and 10 CFR Part 50, Appendix K inapplicable to the M5® cladding, even though the approved AREVA methodologies show that the intent of the regulations are met. Application of these regulations in this particular circumstance would not meet the underlying purpose of the rule nor is it necessary to achieve the underlying purpose of the rule, and therefore special circumstances exist.

Conclusion

Therefore, as described above, the requirements of 10 CFR 50.12 are met for the requested exemption to 10 CFR 50.46 and 10 CFR Part 50, Appendix K. We request this exemption be granted by January 1, 2011 to support core loading for the spring 2011 refueling outage.

PRECEDENT

The Nuclear Regulatory Commission has granted similar exemptions for M5® cladding to the following:

Ft. Calhoun - Letter from A. B. Wang (NRC) to R. T. Ridenoure (OPPD), dated August 17, 2006, Issuance of Exemption for Use of M5