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10 CFR 50.59
10 CFR 72.48

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U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2
Renewed Facility Operating License Nos. DPR-53 and DPR-69
NRC Docket Nos. 50-317 and 50-318

Calvert Cliffs Nuclear Power Plant
Independent Spent Fuel Storage Installation, License No. SNM-2505
NRC Docket No. 72-8

Subject: Report of Changes, Tests, and Experiments – 10 CFR 50.59 and 10 CFR 72.48

In accordance with 10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2), a report of changes, tests and experiments is provided as Attachment (1). The attachment contains brief descriptions of changes, tests, and experiments approved under the provisions of 10 CFR 50.59 and 10 CFR 72.48 between January 1, 2015 and December 31, 2016.

There are no regulatory commitments contained in this correspondence.

Should you have questions regarding this matter, please contact Mr. Larry D. Smith at (410) 495-5219.

Respectfully,

George Gellrich

George H. Gellrich
Site Vice President

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Attachment: (1) Calvert Cliffs Nuclear Power Plant Report of Changes, Tests, and Experiments [10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2)]

cc: NRC Project Manager, Calvert Cliffs
NRC Regional Administrator, Region I
NRC Resident Inspector, Calvert Cliffs
S. Gray, MD-DNR
M. Dapas, NMSS

ATTACHMENT (1)

CALVERT CLIFFS NUCLEAR POWER PLANT

REPORT OF CHANGES, TESTS, AND EXPERIMENTS

[10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2)]

ATTACHMENT (1)

CALVERT CLIFFS NUCLEAR POWER PLANT REPORT OF CHANGES, TESTS, AND EXPERIMENTS [10 CFR 50.59(d)(2) and 10 CFR 72.48(d)(2)]

Document Id SE00513	Doc Type 72.48	Rev Status Approved	Revision 0000	Date Issued 6/27/2016
Subject	ECP-09-000128 License Amendment for 32PHB, HSM-HB and SPMT			
Summary	Proposed Activity: Expansion of the ISFSI total capacity from 120 HSMs to 132 HSMs (and increase in the licensed MTU which can be stored there) is being undertaken to ensure sufficient storage capacity between the spent fuel pool and ISFSI to allow for 60 years of operation for both CCNPP units. In addition, a number of changes in the CE 14xl4 fuel design used at CCNPP have occurred since the ISFSI was originally licensed, which must be accounted for in the design of the ISFSI. These include introduction of Value Added Pellets (larger diameter, higher assembly MTU) in 2000, Zirlo cladding and TURBO grids in 2002, peak pin enrichments above the current 4.5% limit for ISFSI (up to 4.95%), 6" top and bottom low enriched (2.6%) axial blankets, and zirconium diboride (ZrB2) burnable absorber in 2005, and most recently the change to AREVA fuel with M5 cladding, gadolinia burnable absorber, and HTP grids in 2011. In addition 24-month cycle operation and high plant capacity factors for more than a decade have resulted in a continuing trend towards higher assembly discharge burn ups which need to be considered in the ISFSI design. To support higher heat load fuel and potentially shorter spent fuel cooling times, the 32PHB will be employed as the high burn-up storage cask in the HSM-HB. The HSM-HB is equipped with thicker concrete walls to maintain the aggregate off-site dose under federal limits. Since the original ISFSI footprint will be maintained while increasing the density of modules from 120 to 132 modules, the spacing between the HSM-HB back-to-back rows will be reduced thereby necessitating the use of a modular trailer capable of tighter maneuverability. The SPMT is designed to meet all critical design characteristics of the originally licensed transporter while providing augmented maneuverability to accommodate the reduced area for travel between the HSM-HB rows. Conclusions: SE00513 answered questions 6, 7 and 8 as 'yes'. These items were sent to the NRC for approval as part of the License Amendment Request. Below is a summary of those items screened in as 'Yes' and subsequently reviewed/approved as part of SER Docket No. 72-8 ISFSI Renewed Materials License No. SNM-2505 Amendment No. 11.				

Document Id SE00538	Doc Type 50.59	Rev Status Approved	Revision 0000	Date Issued 2/23/2015
Subject	Calvert Cliffs Unit 2 Cycle 21 Core Reload (2015 RFO)			
Summary	Proposed Activity: The proposed activity is required to support the biennial refueling of Unit 2 at Calvert Cliffs. The continued change from Westinghouse to AREVA manufactured fuel is to achieve improved fuel performance (i.e. eliminate chronic grid-to-rod fretting failures). U2C21 will be the first Calvert Cliffs core with only AREVA manufactured fuel.			

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SE00538	50.59	Approved	0000	2/23/2015

Conclusions:
The safety analyses justified the following significant core reload changes:

- 1st Full core of AREVA fuel assemblies.
- Increasing the Peak Linear Heat Rate (PLHR) limit from 14.3 to 15.0 kw/ft. This is a limit increase rather than actually operating the core at a higher LHR.
- Eliminated the COLR requirement to reduce power from 100% to 95% when incore monitoring is out of service and excores are used to monitor linear heat rate.
- Retired the Westinghouse methodology for CEA Eject on Unit 2 since no longer have Westinghouse fuel.

The currently reported UFSAR accident doses remain bounding for operation of Unit 2 Cycle 21.

EQ source terms and EQ qualification of equipment remain valid for a full core of AREVA fuel.

Pellet Clad Interaction (PCI) risk analyses have justified implementation of the AREVA power ascension ramp rates at Calvert Cliffs for the initial power ascension following a refueling outage.

The NRC License Conditions contained in Appendix C of the Technical Specifications associated with the implementation of AREVA fuel at Calvert Cliffs will continue to be satisfied.

The proposed activity has been evaluated against the eight criteria of 10 CFR 50.59. It is concluded that no License Amendment is required for Unit 2 Cycle 21.

Document Id	Doc Type	Rev Status	Revision	Date Issued
SE00539	50.59	Approved	0000	2/19/2015
Subject	Eliminate Unit 2 Turbine High Exhaust Temperature Trip			
Summary	Proposed Activity: The exhaust hood temperature trips have been identified as a Single Point Vulnerability that could result in an inadvertent reactor trip as a consequence of a failed instrument. Each LP turbine has a temperature switch that provides a trip at 245 F. The three temperature switches act independently; a high temperature on any single switch will initiate the trip. Conclusions: High temperature alarms provide control room indication of rising exhaust hood temperatures at 175 F. Upon receipt of the alarm, sufficient time is available for Operations to initiate corrective actions and, if corrective actions are not successful,			

Document Id	Doc Type	Rev Status	Revision	Date Issued
SE00539	50.59	Approved	0000	2/19/2015
	manually tripping the turbine from the Control Room per the approved Abnormal Operating Procedure before the onset of turbine damage. Therefore, there is not more than a minimal increase in the likelihood of occurrence of a malfunction resulting in potential damage to the main turbine as a result of the modifications to remove the automatic Unit 2 turbine high exhaust temperature trips			

Document Id	Doc Type	Rev Status	Revision	Date Issued
SE00540	50.59	Approved	0000	2/7/2015
Subject	Accept as-is lock wire FME in reactor after failed retrieval attempt.			
Summary	Proposed Activity: Core debris scans during the defuel window of the 2015 Unit 2 RFO, a piece of lock-wire was identified on top of the fuel alignment plate. An attempt to retrieve the FME was initially successful but upon removal of the tool/basket from the refuel pool the FME was not in these retrieval devices. A re-inspection of the plate could not identify the location of the lock-wire. Conclusions: Debris in the reactor could result in damage to the fuel cladding (first nuclear barrier). A 50.59 evaluation determined that there is not more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component (SSC) important to safety previously evaluated in the UFSAR. The response to Question No. 2 contains a comprehensive list of all malfunctions considered in this evaluation. The lock wire has been evaluated for its effect on fuel, the reactor coolant system boundary and interfacing systems. There are no malfunctions that cause any of the fuel cladding safety limits identified in Engineering Standard, ES-017, or discussed in the UFSAR to be exceeded or altered. The RCS pressure boundary has also been evaluated for the effects of fretting due to the lock wire and there are no plausible failures postulated for the RCS Pressure boundary. Finally, there are no malfunctions postulated that could result in altering or exceeding the Containment boundary design pressure. Accordingly, the lock wire does not result in a design basis limit for a fission product barrier as described in the UFSAR being exceeded or altered.			

Document Id	Doc Type	Rev Status	Revision	Date Issued
SE00541	50.59	Approved	0001	3/7/2016
Subject	Replacement CEAs for Calvert Cliffs Unit 1 and 2			
Summary	Proposed Activity: ECP-13-000911 provides acceptance to replacement of the resident CEAs with the AREVA CEAs.			

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SE00541	50.59	Approved	0001	3/7/2016

Calvert Cliff's current inventory of CEAs is approaching the end of their lifetime and must be replaced at least one more time so that both units can operate to their current licensed limit (2034 and 2036). The CEA lifetime analysis indicates we need some replacement for the RFO 2016. AREVA has been selected to manufacture replacement CEAS for Calvert Cliffs. The AREVA CEAs are equivalent to the resident CEAs in form, fit, and function. For Calvert Cliffs, there are two CEA design; a Full Strength CEA and a Part Strength CEA. The Part Strength CEA is only used in the center core location. The Full Strength CEA has four outer rods and one center rod each loaded with two annular Ag-In-Cd bars, one stack support assembly, and a 120.65 inch column of B4C pellets. The Part Strength CEA has four outer rods each loaded with an equivalent stack of Stainless Steel tubes and one center rod loaded the same as the Full Strength CEA.

Conclusions:

1. The comparison between the AREVA 14x14 CEA design and the resident 14x14 CEA used at Calvert Cliffs shows that the key parameters are the same for both designs. Therefore, the AREVA CEA is compatible with Calvert Cliffs' plant systems and fuel assemblies and will perform in the same manner as the current CEAs.
2. The AREVA CEAs are equivalent to the resident CEA design at Calvert Cliffs from a mechanical and nuclear performance standpoint. Since the CEA weights and reactivity worths of the AREVA CEAs are similar, the control rod drop time and the reactivity insertion rate assumed in the safety analysis will not be affected. Therefore, the change to AREVA CEAs can be reviewed under the provisions of 10 CFR 50.59. The basis for this approach is that the change to AREVA supplied CEAs does not require a significant change to the UFSAR, the technical specifications or result in an un-reviewed safety question. There is no increase in the likelihood of occurrence or consequences of an accident or malfunction of equipment important to safety that has been previously evaluated. The possibility for an accident or malfunction of a different type than previously evaluated is not created. There are no changes to a design basis limit for a fission product barrier or departure from a method of evaluation used to establish the design bases or in the safety analyses. By replacing the resident CEAs with the AREVA CEAs there will be no increase in the likelihood of occurrence or consequences of an accident or malfunction of equipment important to safety.
3. There is not more than a minimal increase in the likelihood of occurrence of a malfunction of a structure, system or component (SSC) important to safety previously evaluated in the UFSAR.

Document Id	Doc Type	Rev Status	Revision	Date Issued
SE00543	50.59	Approved	0000	5/30/2015
Subject	Evaluation of CCW System Performance for certain Single Failure Scenarios During a LOCA			
Summary	Proposed Activity: It has been identified that under certain single-failure scenarios, where salt water cooling has been lost to a CCW HX, that it may be difficult to maintain the CCW HX outlet temperature to the 120°F limit identified in Tech Spec Bases section 3.7.5, and UFSAR Section 9.5.2. Maintaining the CCW HX outlet temperature below 120°F would conservatively ensure that the containment response pressure and temperature profiles do not exceed their respective EQ envelopes. This activity is to			

Document Id	Doc Type	Rev Status	Revision	Date Issued
SE00543	50.59	Approved	0000	5/30/2015

identify under what conditions it will be necessary to isolate a non-functioning CCW HX to ensure that the EQ envelopes are not exceeded.

Conclusions:
By allowing / directing Operations to isolate a non-functioning (i.e., non-heat removing) CCW HX during a LOCA, this activity provides Operations an additional means to improve the decay heat removal from containment during a LOCA, and ensure that the containment response pressure and temperature EQ profiles are not exceeded.

This activity uses the methodology and results of existing calculation M-93-041 which validated the adequacy of the CCW HX design and sizing. This includes evaluating the heat load on the CCW system from the SDC HXs for various CCW inlet fluid temperatures, and the limiting containment sump fluid temperature. It also includes predicting the outlet temperature from the functioning CCW HX which would exist if the CCW HX is greater than or equal to the limiting fouling factor of 0.00105 developed in calculation M-93-041.

The two limiting single failure scenarios include:

1) Loss of salt water cooling to a CCW HX (still have both SDC HXs adding heat to the CCW system).
2) Failure of an EDG, or 125V DC Bus (lose an entire train of safety equipment).

For both Units 1 and 2 the loss of salt water cooling under the minimum CCW flow to the SDC HX of 1800 gpm does not result in a CCW fluid temperature exceeding 120°F at the outlet of the functioning CCW HX, and therefore the containment response equipment profiles will not be exceeded. The minimum CCW flow of 1800 gpm is the value considered in the UFSAR Chapter 14.20 accident analysis. At the maximum flow condition of the heat removal from containment is more than double the minimum assumed in the accident analysis. The CCW HX outlet temperature of the functioning CCW can rise to approximately 130°F; however, this has no deleterious effects. In this failure scenario isolating the non-functioning CCW HX is not necessary, but will improve decay heat removal, and therefore should only be considered after measures to restore cooling to the affected CCW HX have not been successful.

For Unit 2 loss of an entire train of safety equipment due to a EDG failure, or failure of a 125V DC bus puts the system in single train operation which is already an approved line-up per Tech Spec Bases 3.7.5, and calculation M-93-041. Again, isolating the non-functioning CCW HX is not necessary, but will improved decay heat removal, and therefore should be considered only after measures to restore the lost train back to service have not been successful.

For Unit 1 the 11 CCW HX is aligned with 12 SDC HX and 12 CCW HX is aligned with 11 SDC HX. With this configuration loss of an entire train of safety equipment due to a EDG failure, or failure of a 125V DC bus puts the system in a configuration where one CCW loop has no cooling in the CCW HX, but will have heat input in the SDC HX, while the other CCW loop will

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	have cooling in the CCW HX, but will not have heat input in the SDC HX. Heat removal will still occur because the fluids will mix at the pump suction therefore, heat will be transferred to the salt water system; however, heat transfer will not be as efficient since the CCW Flow going to the functioning SDC HX will be the higher temperature fluid coming from the non-functioning CCW HX. In this case it will be necessary to isolate the non-functioning CCW HX in order to ensure that the containment response pressure and temperature profiles do not exceed the EQ profile. This action must be completed within 30 minutes, or prior to RAS whichever is longer.			

Document Id SE00544	Doc Type 50.59	Rev Status Approved	Revision 0000	Date Issued 9/21/2015
Subject	Unit 1 Digital Feedwater Control System and Steam Generator Feed Pump Speed Control Upgrade Modification			
Summary	Proposed Activity: The current Technical Specification Basis requires a 6-week rotation/overlap between SR 3.3.3.1 and SR 3.3.3.2, RTCB channel functional test and RPS logic test, respectively, to ensure the Reactor Trip Circuit Breakers (RTCB) are cycled as recommended by the Combustion Engineering topical report. Since the implementation of these recommendations, the RCTBs have been replaced with a different make/model (Square D Masterpact NT installed under ES200700035-001) and the concerns that drove the 6-week scheduling requirement are no longer the same. At the time of replacement, the surveillance requirements were not reevaluated. This evaluation is to support the removal of an obsolete recommendation in CCNPP Technical Specification Basis. Removing the 6 week intervals between the two tests will still ensure a RTCB channel test on a 92 day frequency required and cycles the breakers the same total number of times but allows flexibility in the scheduling of these tests. This also meets current vendor recommended annual cycle interval. Conclusions: Supporting documents have been reviewed and it is apparent that the driving factor in the Combustion Engineering topical report behind the 6-week overlap recommendation was RTCB failure rates and the performance issues with the previous model breakers at the time. Since then, these factors have changed with the replacement of the original breakers and significant improvement industry wide with breakers in this application. A similar approach to the one used in the license amendment request to determine failure likelihood of RTCBs was used in the reliability analysis and yielded results lower than the failure rate used to establish the 6-week overlap. Our current breakers have proven to be more reliable, site specific and industry-wide, than the previous with no failures to open on demand or slow timing responses at up to and exceeding 92 day intervals. The proposed change will still meet the current Technical Specifications requirement of a 92 day frequency of each test. The conclusion is to remove the scheduling requirement to allow these tests to be scheduled independent of each other.			

Document Id	Doc Type	Rev Status	Revision	Date Issued
SE00545	50.59	Approved	0000	7/26/2015
Subject	Temporary powering of CEA #18 Lower Gripper Coil to de-energize the Upper Gripper Coil			
Summary	Proposed Activity: During performance of the weekly Unit 1 upper gripper coil trace readings (WO C93025778), it was evident of an increase in the deviation (high to low) for CEA #18's upper gripper coil current. Previous operating experience (CR-2012-010304) at CCNPP have indicated that with an increase in the coil trace deviation, is positive indication of future coil failure and a dropped rod (CR-2012-007619). Conclusions: After performing a 50.59 Screen, it has been confirmed that this temporary modification utilizes the lower gripper coil of CEA #18 to provide the primary holding function due to the degraded condition of CEA #18 upper gripper coil. This modification will also disable the function to lift the CEA adversely affecting the design function of the CEA #18, requiring a 50.59 Evaluation. It has been evaluated and concluded that the effect of this change has minimal impact on the design bases and safety analyses described in the UFSAR not requiring a License Amendment Request.			

Document Id	Doc Type	Rev Status	Revision	Date Issued
SE00553	50.59	Approved	0000	2/17/2016
Subject	Unit 1 Cycle 23 Core Reload (2016 RFO)			
Summary	Proposed Activity: The proposed activity is the Calvert Cliffs Unit 1 Cycle 23 (U1C23) core reload during the 2016 RFO. Conclusions: • 1st Unit 1 core containing only AREVA manufactured fuel assemblies. • Increasing the PLHR limit from 14.3 kw/ft to 15.0 kw/ft. This is a limit increase rather than actually operating the core at a higher LHR. • A higher PLHR limit provides the margin to eliminate the requirement to reduce power from 100% to 95% when incore monitoring is out of service and excore detectors are used to monitor linear heat rate. • Allows a slightly more negative MTC limit of -3.1 x 10-4 Δρ/oF. • Retired the Westinghouse methodology for CEA Eject on Unit 1 since no longer have Westinghouse fuel. • First mixed vendor core of CEAs (AREVA and Westinghouse). • Implementing the AREVA thermal power ascension rates. • First Calvert core to use the AREVA UFSAR Chapter 14 peak pressure safety analyses. • Implement newer version of EMF-92-116 methodology. • Crediting the negative reactivity of CEAs to reduce the soluble boron requirements for refueling. • Resolution of IR on Clad Strain.			

Document Id	Doc Type	Rev Status	Revision	Date Issued
SE00553	50.59	Approved	0000	2/17/2016

The currently reported UFSAR accident doses remain bounding for operation of Unit 1 Cycle 23.

EQ source terms and EQ qualification of equipment remain valid for a full core of AREVA fuel.

The NRC License Conditions contained in Appendix C of the Technical Specifications associated with the implementation of AREVA fuel at Calvert Cliffs will continue to be satisfied.

The proposed activity has been evaluated against the eight criteria of 10 CFR 50.59. It is concluded that no License Amendment is required for Unit 1 Cycle 23.