UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

December 19, 2012

Mr. George H. Gellrich, Vice President
Calvert Cliffs Nuclear Power Plant, LLC
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 – SAFETY EVALUATION OF THE REALISTIC LARGE-BREAK LOSS-OF-COOLANT ACCIDENT SUMMARY REPORT (TAC NOS. ME7672 AND ME7673)

Dear Mr. Gellrich:

On February 18, 2011, the Nuclear Regulatory Commission (NRC) issued Amendment No. 297 to Renewed Facility Operating License (FOL) No. DPR-53 and Amendment No. 273 to Renewed FOL No. DPR-69 for the Calvert Cliffs Nuclear Power Plant Unit, Nos. 1 and 2 (Calvert Cliffs) to revise the licensing basis and Technical Specifications to allow the use of AREVA Advanced CE-14 HTP fuel in the Calvert Cliffs reactors. The amendments contained a license condition prohibiting plant operation after the first operating cycle. By letter dated December 1, 2011, Calvert Cliffs Nuclear Power Plant, LLC, the licensee, submitted ANP-3043(P), Revision 0, "Calvert Cliffs RLBLOCA [realistic large-break loss-of-coolant accident] Summary Report," for Calvert Cliffs. By letter dated January 17, 2012, the licensee submitted ANP-3043(P), Revision 1, "Calvert Cliffs RLBLOCA Summary Report," for Calvert Cliffs. These AREVA reports, submitted for NRC approval, contain the RLBLOCA analysis for once and twice burnt AREVA fuel. The licensee submitted additional clarifying information and supporting data by letter dated July 25, 2012.

The NRC staff has concluded that the licensee's LOCA analyses are acceptable for the second and subsequent cycles of operation and demonstrate that the licensee complies with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46 (b)(1-4).

The NRC staff has determined that the related safety evaluation (SE) contains proprietary information pursuant to 10 CFR 2.390, "Public inspections, exemptions, requests for withholding." Accordingly, the NRC staff also has prepared a redacted, publicly-available, non-proprietary version of the SE. Copies of the proprietary and non-proprietary versions of the SE are enclosed.

Enclosures transmitted herewith contain sensitive unclassified information. When separated from enclosures, this document is decontrolled.

Please contact me at (301) 415-1016, if you have any questions regarding this issue.

Sincerely,

Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosures:
1. Non-Proprietary SE
2. Proprietary SE

cc w/ Enclosure 2: Distribution via Listserv

ENCLOSURE 2

(NON-PROPRIETARY)

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REGARDING REALISTIC LARGE-BREAK LOSS-OF-COOLANT ACCIDENT

CALVERT CLIFFS NUCLEAR POWER PLANT, LLC

CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-317 AND 50-318

Proprietary information pursuant to Title 10 of the *Code of Federal Regulations* Section 2.390 has been redacted from this document. Redacted information is identified by blank space enclosed within double brackets.

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REGARDING REALISTIC LARGE-BREAK LOSS-OF-COOLANT ACCIDENT

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CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2

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1.0 INTRODUCTION

On February 18, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML110390224), the Nuclear Regulatory Commission (NRC) issued Amendment No. 297 to Renewed Facility Operating License (FOL) No. DPR-53 and Amendment No. 273 to Renewed FOL No. DPR-69 for the Calvert Cliffs Nuclear Power Plant Unit, Nos. 1 and 2 (Calvert Cliffs) to revise the licensing basis and Technical Specifications to allow the use of AREVA Advanced CE-14 HTP fuel in the Calvert Cliffs reactors. The amendments contained a license condition prohibiting plant operation after the first operating cycle. By letter dated December 1, 2011 (ADAMS Accession No. ML113400246), Calvert Cliffs Nuclear Power Plant, LLC, the licensee, submitted ANP-3043(P), Revision 0, "Calvert Cliffs RLBLOCA [realistic large-break loss-of-coolant accident] Summary Report," for Calvert Cliffs. By letter dated January 17, 2012 (ADAMS Accession No. ML12020A218), the licensee submitted ANP-3043(P), Revision 1, "Calvert Cliffs RLBLOCA Summary Report," for Calvert Cliffs. These AREVA reports, submitted for NRC approval, contain the RLBLOCA analysis for once and twice burnt AREVA fuel. The licensee submitted additional clarifying information and supporting data by letter dated July 25, 2012 (ADAMS Accession No. ML12208A257).

2.0 REGULATORY EVALUATION

LOCAs are postulated accidents that would result in the loss of reactor coolant from piping breaks in the reactor coolant pressure boundary at a rate in excess of the capability of the normal reactor coolant makeup system to replenish it. Loss of significant quantities of reactor coolant would prevent heat removal from the reactor core, unless the water is replenished. The reactor protection system and the emergency core cooling system (ECCS) are provided to mitigate these accidents.

The NRC's acceptance criteria are based on:

(1) 10 CFR § 50.46, insofar as it establishes standards for the calculation of ECCS performance and acceptance criteria for that calculated performance;

(2) 10 CFR Part 50, Appendix K, insofar as it establishes required and acceptable features of evaluation models (EMs) for heat removal by the ECCS after the blowdown phase of a LOCA;

(3) General Design Criteria (GDC) 4, insofar as it requires that systems, structures, and components important to safety be protected against dynamic effects associated with flow instabilities and loads such as those resulting from water hammer;

(4) GDC 27, insofar as it requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the ECCS, of reliably controlling reactivity changes under postulated accident conditions, with appropriate margin for stuck rods, to assure the capability to cool the core is maintained; and

(5) GDC 35, insofar as it requires that a system to provide abundant emergency core cooling be provided to transfer heat from the reactor core following any LOCA at a rate so that fuel clad damage that could interfere with continued effective core cooling will be prevented.

Specific review criteria are contained in Sections 6.3 and 15.6.5 of the Standard Review Plan.

3.0 TECHNICAL EVALUATION

Large-Break Loss-of-Coolant Accident

The NRC staff reviewed the implementation of the methodology to ensure it was correctly implemented; the input assumptions to ensure they reflect the plant licensing basis; and the results to ensure that they meet the 10 CFR 50.46 requirements.

Methodology implementation and the analytical model

The licensee's best estimate, LBLOCA analyses supporting EPU [extended power uprate] operation was performed by Areva NP and documented in ANP-3043(P) Revision 1, "Calvert Cliffs Realistic Large Break LOCA Summary Report." The analysis was conducted in accordance with EMF-2103(P)(A), "Realistic Large Break LOCA Methodology for Pressurized Water Reactors," with exceptions as discussed throughout the safety evaluation (SE). EMF-2103(P)(A) received generic approval by NRC SE on April 9, 2003 (ADAMS Accession No. ML030760312), which contained 13 conditions and limitations restricting its use. The licensee provided Table 3-4 in ANP-3043(P) Rev. 1, which listed the conditions and limitations and included information to demonstrate that adherence to each had been attained. The NRC staff reviewed this information and concluded that the licensee's implementation of EMF-2103 adheres to each of the conditions and limitations.

Subsequent to the approval of EMF-2103(P)(A), the NRC staff has found that certain modeling assumptions and constitutive relationships contained in the EMF-2103 methodology are not suitable for demonstrating compliance with the 10 CFR 50.46(b) acceptance criteria, as described in a draft SE dated April 3, 2007 (ADAMS Accession No. ML070940125).[1]

Therefore, the NRC staff also considered the conditions and limitations in this draft SE, and the corresponding departures from NRC-approved EMF-2103(P)(A), Revision 0, required to adhere to these conditions and limitations. The licensee provided information to address these issues, as well as, issues identified in the February 2011 SE approving the Calvert Cliffs transition to AREVA fuel and methods in Sections 4, "Generic Support for Transition Package" and 6, "Supplemental Information Regarding NRC Issues with the Use of AREVA RLBLOCA Methodology."

EMF-2103 is a best estimate code and uses a statistical method based on order statistics to produce an estimate of the upper tolerance limit for predicted peak cladding temperature, consistent with the "high level of probability" statement contained in 10 CFR 50.46(a)(1)(i). Upper tolerance limit estimates are also produced for the maximum local oxidation and hydrogen generation, as well.

The power assumed in the analyses, 2754 MWt, represents 100 percent primary power (2737 MWt) plus 0.62 percent of primary power (17 MWt) to account for the measurement uncertainty. This departure from the previously approved methodology is acceptable because it is conservative in that the previously approved methodology permitted ranging the assumed power level, meaning that some cases could have initiated at a power level less than 2754 MWt had the analysis been performed using the previously approved methodology. It is also acceptable because it is consistent with the NRC staff's position that parametrically ranging the assumed initial power level is inconsistent with 10 CFR 50.46 requirements, whereas deterministically including uncertainty in the assumed initial power level is acceptable.

The RLBLOCA analysis was performed with a version of S-RELAP5 that requires both the void fraction to be less than 0.95 and the clad temperature to be less than 900 °F before the rod is allowed to quench. During its review of EMF-2103(P), Revision 1, the NRC staff determined that the S-RELAP5 EM could allow rod quench to occur once the temperature drops below the minimum film boiling temperature regardless of the void fraction in the channel. Contrarily, NUREG-0915 demonstrated that the void fraction must also be less than 0.95 for rod rewet to occur. To address this concern for Calvert Cliffs, the licensee stated that the clad temperature must be less than the minimum temperature for film boiling heat transfer (T_{min}) and the void fraction must be less than 0.95 before the rod is allowed to quench. T_{min} is a sampled parameter and according to Table I-3 of the response to RAI 4, T_{min} was never sampled above 696.31 K (793.69 °F). The NRC staff finds this departure from the methodology acceptable because the departure provides for analytic predictions that are not only more consistent with observed data, but also more conservative than predictions obtained using the previously NRC-approved methodology.

[1] This SE was never formally issued, and the vendor withdrew the topical report revision that it supported. Therefore, there is no publicly available copy of this report. Nonetheless, the NRC staff has continued to use adherence to the conditions and limitations listed in this SE as a basis for its approval of requests to implement EMF-2103, Revision 0, as an interim review approach until a second revision to EMF-2103 is submitted to the NRC for review and approval. While EMF-2103(P)(A) is an acceptable EM as described in 10 CFR 50.46, the NRC staff requires these deviations for plant-specific application of the methodology.

The RLBLOCA analysis was performed with a version of S-RELAP5 that limits the contribution of the Forslund-Rohsenow model to no more than 15 percent of the total heat transfer at and above a void fraction of 0.9. This departure from the approved methodology accounts for experimentally observed phenomena that appear to inhibit droplet contact with heated fuel rods at high void fractions. Thus, this departure is conservative relative to the approved methodology because it corrects for any potential to over-predict heat transfer through conduction to entrained droplets, which experimental observations have shown not to come in contact with the fuel rods at such high void fractions. The net effect of this conservatism would serve to increase the overall predicted PCT compared to evaluations performed using the previously NRC-approved methodology.

The analyses addressed the availability of offsite power correctly by ranging each case separately. This is acceptable because it satisfies GDC 35, in that each distribution type has been accounted for separately with its own set of cases, thereby addressing possible concerns associated with the mixing of two separate statistical spectra. The NRC staff finds this treatment acceptable because it is consistent with the NRC staff's position regarding compliance with GDC 35, as noted in ANP-3043(P).

Downcomer boiling is caused by metal heat release from vessel and core barrel walls to fluid in the downcomer gap. Metal heat from the vessel lower head and structures in the lower plenum also contribute to downcomer boiling. As heat is released to the downcomer fluid, its temperature is gradually increased, eventually subcooled, and saturated boiling takes place. Voids generated by these processes displace water in the downcomer and reduce the driving head that forces water into the core during the reflood phase of a large break LOCA. This loss in head can significantly reduce the core flooding rate, and increase the peak cladding temperature.

The NRC staff has historically identified differences in results between NRC staff confirmatory calculations and those produced using the AREVA RLBLOCA EM, attributable to downcomer boiling modeling, that cause significant differences in peak cladding temperature results. As discussed in Section 1.0 of ANP-3043(P) Rev. 1, AREVA attributes this to an under-prediction of cold leg condensation efficiency. To correct for this, AREVA has identified appropriate multipliers to force fluid entering the downcomer to saturated conditions following the deployment of the safety injection tanks. The NRC staff finds this departure from the previously approved methodology acceptable because (1) the artificially saturated fluid conditions will conservatively reduce both the downcomer driving head and the core flooding rate, which becomes conducive to portions of the fuel remaining in a vapor-cooled environment, thus presenting a greater challenge to clad surface cooling, and (2) conditions in the downcomer following safety injection tank discharge are expected to be slightly subcooled, meaning that assuming fully saturated conditions is conservative.

The NRC staff requested sensitivity studies to demonstrate the adequacy of the axial and azimuthal nodding in the downcomer to predict the effects of downcomer boiling on peak cladding temperature (PCT). As a result, the licensee increased the number of downcomer axial nodes in the boiling region adjacent the core from two to four cells. The increased noding did not alter the void distribution relative to the base case, so that downcomer hydrostatic head remained close to the base noding scheme result.

The licensee also cited a three loop downcomer sensitivity study where the azimuthal noding was increased from three azimuthal cells to nine, resulting in two additional cells created between each azimuthal sector connected to each cold leg. As a consequence, fluid entering the downcomer would potentially not readily mix with the adjacent cells, due to lateral flow resistance, increasing the potential for boiling. Results of the calculation showed no noticeable difference in the PCT for the more detailed nodalization. The NRC staff notes, however, that the increased bypass and loss of additional downcomer fluid out the break with the fewer, larger cells is expected to have the potential to further reduce the inventory in the downcomer during the long term. This reduced inventory can cause the downcomer static head to decrease, causing the PCT to be higher for the base nodalization downcomer. Thus, the base downcomer noding scheme, with the potential to increase the effect of bypass, is expected to maximize PCT for those cases experiencing downcomer boiling.

A final sensitivity study was performed increasing the number of downcomer wall mesh spacings from 9 to 18. There was no noticeable change in the PCT plot for these two wall mesh size cases. Furthermore, comparison of the finite difference solution for wall heat conduction in the downcomer wall reproduce the exact solution to the conduction equation in this region.

The NRC staff finds that these sensitivity studies, including the conservative model employed to fully saturate the fluid entering the downcomer, are sufficient to justify the downcomer boiling model used to simulate downcomer boiling following a LBLOCA.

The NRC staff continues to have concerns that AREVA's omission of a model representing fuel rod swell, rupture, and pellet relocation is unjustified and possibly non-conservative. However, the Calvert Cliffs' argument provided by the licensee stated that [[

]]. The fact that the predicted PCTs at Calvert Cliffs do not exceed 1700 °F, the NRC staff finds that there is reasonable assurance that blowdown cladding ruptures would not occur during a postulated LOCA at Calvert Cliffs, and thus, the model is acceptable because it provides an acceptable representation of the LBLOCA progression without modeling a blowdown cladding rupture.

NRC Information Notice 2009-23, "Nuclear Fuel Thermal Conductivity Degradation," describes a recently identified issue concerning the ability of legacy thermal-mechanical fuel modeling codes to predict the exposure-dependent degradation of fuel thermal conductivity accurately. Some legacy codes, including RODEX-3A, non-conservatively over-predict fuel thermal conductivity at higher burnups. A safety concern with fuel thermal conductivity degradation (TCD) in a LOCA would be that fuel temperatures modeled incorrectly would affect the heat transfer to the cladding surface, causing the LOCA EM to predict potentially erroneous PCTs. To correct for this issue, AREVA has applied a polynomial transformation that is used to bias the fuel pellet centerline temperature based on empirical data collected supporting the more recent RODEX4 fuel performance code.

Information provided by the licensee explained the polynomial transformation applied to the fuel centerline temperature and that the clad surface temperature is unaffected by the fuel centerline temperature bias because pellet power is not adjusted. The clad surface temperature is affected by changes in the heat transfer properties from the pellet surface through the gap, through the cladding, and to the coolant. The polynomial transformation does not change any of these heat transfer properties and therefore has a minor effect on the clad surface temperature. The bias does provide an adjustment to the entire fuel pellet. The licensee stated that the

polynomial transformation provides a bias adjustment to the fuel centerline temperature and 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. This treatment causes an increase to the cladding heat load during the initial phases of the transient that corrects for TCD and treats the uncertainty associated with the correction. The NRC staff finds that this polynomial bias adequately accounts for fuel thermal conductivity degradation.

Consistent with NRC regulatory guidance for realistic LOCA EMs described in Regulatory Guide (RG) 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," the licensee uses the ANS-1979 standard for decay heat power in light-water reactors. In the analysis provided, the uncertainty for the decay heat parameter is set to zero and no sampling is done on this parameter, resulting in the decay heat being used with a 1.0 multiplier. This is a change from the EMF- 2103 EM. The decay heat in the analysis is always the 1979 ANS standard for decay heat from U^{235} with fully saturated decay chains, corresponding to infinite operation, assuming 200 MeV per fission. The infinite operation decay chain was compared to several finite decay chains that incorporated a 2-sigma uncertainty. The licensee provided a detailed comparison of the infinite case without uncertainty to various finite cases that included uncertainty in order to demonstrate that uncertainty is properly accounted for in the infinite chain by bounding the finite chains. It was shown that, two seconds after shutdown, the infinite chain bounds all of the finite chains. During the first two seconds when the infinite chain is not bounding, it was determined that the difference in energy deposited between the infinite chain and the bounding finite chain would have an insignificant effect on clad temperature because stored energy will have a much greater influence on PCT that early in the transient. The NRC staff finds this treatment of decay heat acceptable because the licensee appropriately incorporated an acceptable model identified in RG 1.157 and demonstrated that uncertainty was accounted for using a bounding method.

Determination of break locations and break sizes

The analyses ranged in area between the minimum break area and an area of twice the size of the broken pipe. The licensee stated that minimum break area was calculated to be 28.67 percent of the double-ended guillotine break area. This information demonstrates that the total number of sampled cases is appropriate because the phenomenology dominating the limiting aspects of the event for all sampled break areas is consistent. That is, a certain number of sampled cases are appropriate, because the limiting results of the accident for pipe ruptures ranging from about 20 to 100 percent of the double-ended pipe rupture size are all limited by dispersed flow film boiling ahead of the quench front. If the sampled break area included a greater range (i.e., break sizes less than 20 percent of the double-ended guillotine rupture), additional phenomenology would govern the limiting events and additional cases would be required to provide the necessary high level of statistical confidence that a bounding upper tolerance limit had been identified.

The licensee stated that the worst break location was generically addressed by deterministic studies, and determined to be in the cold leg between the reactor coolant pump and the reactor vessel for the reactor coolant system loop containing the pressurizer. The NRC staff reviewed the methodology and confirmed this to be true. The method did not consider slot breaks because Calvert Cliffs does not have any loop seals with bottom elevation below the top

elevation of the core. The NRC reviewed this information and confirmed that condition 8 of the SE approving EMF-2103 was satisfied. On this basis, the NRC staff finds the licensee's conclusion that the Calvert Cliffs PCT-limiting transient is a double-ended cold leg guillotine break acceptable, because uncertainties related to break type and size were included in the modeling approach.

Postulated initial conditions and sequence of events

The assumed reactor core power used in the RLBLOCA analysis is 2754 MWt, which represents 100 percent primary power plus 17 MWt (0.62 percent of primary power) to account for the measurement uncertainty. The RLBLOCA analysis also assumed a steam generator (SG) tube plugging level of 10 percent in all SGs, a total linear heat generation rate of 15.0 kW/ft (no axial dependency), a total peaking factor up to a value of 2.37, and a radial peaking factor up to a value of 1.810 (including 6 percent measurement uncertainty and 3.5 percent control rod insertion uncertainty).

The single failure assumption was based on of the approved RLBLOCA EM, EMF-2103(P)(A) Rev. 0. The single failure scenario is a diesel loss with fully functional containment sprays at Technical Specification (TS) minimum temperature and one train of pumped ECCS injection at TS maximum temperature. Containment pressure is indirectly sampled through containment volume. A sensitivity study on the limiting case for a maximum ECCS injection scenario is provided in Section 6.3 of ANP-3043(P) Rev. 1. While the maximum ECCS scenario resulted in the same PCT as the RLBLOCA EM single failure, the quench time calculated was greater for the RLBLOCA EM single failure case, therefore the RLBLOCA EM single failure remains the most limiting.

3.1 Results

The RLBOCA analyses were performed to demonstrate that the system design would provide sufficient ECCS flow to transfer the heat from the reactor core following a LBLOCA at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) the clad metal-water reaction would be limited to less than would compromise cladding ductility and result in excessive hydrogen generation. The NRC staff reviewed the analyses to assure that they reflected suitable redundancy in components and features; and suitable interconnections, leak detection, isolation, and containment capabilities were available such that the safety functions could be accomplished, assuming a single failure, for LBLOCAs considering the availability of onsite and offsite electric power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available, with offsite electric power available). The acceptance criterion for ECCS performance, provided in 10 CFR 50.46, was used by the NRC staff in assessing the acceptability of the AREVA EMF-2103 methodology for Calvert Cliffs.

In its submittal, the licensee provided the analysis results for the best estimate LOCA analyses at the proposed extended power uprate conditions, which were produced in accordance with the EMF-2103 methodology and NRC requested deviations. The licensee's results for the calculated PCTs, the maximum cladding oxidation (local), and the maximum core-wide cladding oxidation are provided in the following table along with the acceptance criteria of 10 CFR 50.46(b).

Parameters	Fresh Fuel 8% Gd_2O_3 Rod	Once Burned UO_2 Fuel	10 CFR 50.46 Limits
Cladding Material	M5	M5	
Peak Clad Temperature (°F)	1620	1545	2200 (10 CFR 50.46(b)(1))
Pre-transient Local Oxidation (%)	1.214	1.997	
Maximum Transient Local Oxidation (%)	0.543	0.463	
Maximum Local Oxidation (%)	1.757	2.460	17.0 (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel) (%)	0.0111		1.0 (10 CFR 50.46(b)(3))

The licensee's analytic limiting local maximum oxidation is 2.46 percent, and the transient oxidation contribution was shown to decrease from the beginning of life (BOL). This result is expected because fuel is generally more susceptible to transient oxidation at the BOL. The table demonstrates that the sum of pre-transient plus transient oxidation remains below 17 percent at all times in life for the AREVA NP HTP fuel.

The limiting core-wide oxidation is based on the maximum values of core-wide transient oxidation computed for the case set. Because the oxidation is 0.011 percent, there is significant margin to the regulatory limit, and the NRC staff finds that the licensee has adequately demonstrated that the core-wide oxidation will remain less than 1 percent.

4.0 CONCLUSION

The NRC staff has concluded that the licensee's LOCA analyses are acceptable for the second and subsequent cycles of operation and demonstrate that the licensee complies with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) 50.46 (b)(1-4).

Principle Contributors: Jennifer Gall
Leonard Ward

Date: December 19, 2012

Please contact me at (301) 415-1016, if you have any questions regarding this issue.

Sincerely,

/RA/

Nadiyah S. Morgan, Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

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