September 2, 2005

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

SUBJECT: CORRECTION TO AMENDMENT NOS. 237 AND 216, FOR NORTH ANNA POWER STATION (TAC NOS. MC5711 AND MC5712)

Dear Mr. Christian:

On August 20, 2004, and April 1, 2004, the U.S. Nuclear Regulatory Commission (NRC) issued Amendment Nos. 237 and 216 to Renewed Facility Operating License Nos. NPF-4 and NPF-7 for the North Anna Power Station, Units 1 and 2. The NRC staff issued these amendments in response to your application dated March 28, 2002, as supplemented by letters dated May 13, June 19, July 9, July 25, August 2, August 16, and November 15, 2002, May 6, May 9, May 27, June 11 (2 letters), July 18, August 20, August 26, September 4, September 5, September 22, September 26 (2 letters), November 10, December 8, and December 17, 2003, and January 6, January 22 (2 letters), February 12, February 13, March 1, June 16, and June 18 (2 letters), 2004. The November 15, 2002, submittal replaced the submittals dated July 9, July 25, and August 16, 2002. These amendments revised the Technical Specifications to allow for the use of Framatome Advanced Mark-BW fuel at North Anna, Units 1 and 2.

By letter dated January 12, 2005, Virginia Electric and Power Company informed the NRC staff of inaccuracies and editorial changes that were found in the safety evaluations (SEs) supporting the amendments for North Anna, Units 1 and 2. The NRC staff has resolved this by revising the appropriate sections in the SEs. The corrected pages to the SEs for North Anna, Units 1 and 2 are included as Enclosures (1) and (2) to this letter. These revisions are identified by lines in the margin.

The NRC regrets any inconvenience this may have caused. If you have any questions, please contact me at (301) 415-1544.

Sincerely,

/RA/
Stephen R. Monarque, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-338 and 50-339

Enclosures: 1. Pages 12, 13, 28, 29, and 30 of SE for North Anna, Unit 1
2. Pages 13, 27, 28, and 29 of SE for North Anna, Unit 2

cc w/encls: See next page

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ENCLOSURE 1

CORRECTIONS TO SAFETY EVALUATION FOR NORTH ANNA, UNIT 1

Westinghouse NAIF is acceptable because it accounts for the Westinghouse NAIF in a conservative fashion by not including a mixed-core benefit. However, because the mixed-core penalty for the Framatome ANP Advanced Mark-BW fuel and the PCT and oxidation values for the NAIF fuel were calculated using different methodologies, neither methodology correctly reflects the difference in results between the fuels. Pursuant to 10 CFR 50.46(a)(3)(ii), the licensee must report changes to, or errors discovered in both LOCA methodologies, as applied to both fuel types, that affect the temperature calculation. The licensee makes these reports in its 10 CFR 50.46 Annual Reports for the ECCS Evaluation Model Changes for North Anna, Unit 1. In addition, Section 50.46(a)(3)(ii) requires a report within 30 days if the change or error is significant. Accordingly, should the mixed-core penalty change such that the proposed treatment is no longer conservative, the licensee would be required to report this to the NRC. Therefore, the difference in the results obtained from the methodologies does not change the NRC staff's conclusions.

3.3.3 Overall Applicability of LOCA Analysis Methodologies

In its letter dated September 5, 2003, the licensee provided a statement showing that it has ongoing processes with Framatome ANP for the purpose of assuring that the ranges and values of input parameters for the LOCA analysis, described in Topical Report EMF-2103 (P), bound the ranges and values of the as-operated plant values for those parameters. The licensee provided this information to show that it would properly model North Anna, Unit 1, the reported results would specifically represent the ECCS performance for North Anna, Unit 1, and these results would be within the applicability range of the model. Therefore, the NRC staff concludes that, in applying the LOCA methodology described in Topical Report EMF-2103 (P), the licensee will meet the requirements of 10 CFR 50.46(c).

By letter dated January 22, 2003, as supplemented by letter dated February 12, 2004, the licensee reported that certain rod bow effects had been inadvertently omitted in the licensee's fuel assessments, including the LOCA analyses. The licensee indicated that the effects associated with the Framatome ANP Advanced Mark-BW fuel assembly bow have been accommodated in the LBLOCA and SBLOCA analyses for both the resident NAIF fuel and the Framatome ANP Advanced Mark-BW fuel. The NRC staff concludes that the licensee's approach will assure that the rod bow effects issue will be bounded and is, therefore, acceptable.

3.3.4 Use of the Forslund-Rohsenow Correlation

The NRC staff's SE dated April 9, 2003, for Topical Report EMF-2103 (P) questioned the validity of the use of the Forslund-Rohsenow correlation. Subsequently, during its review of this amendment request, the NRC staff requested additional information regarding use of the Forslund-Rohsenow correlation in the LBLOCA methodology. By letter dated September 26, 2003, the licensee provided the results of a sensitivity study quantifying the effect of the Forslund-Rohsenow correlation on the result of North Anna, Unit 1, LBLOCA analyses. The results indicate that using the RLBLOCA methodology causes the Forslund-Rohsenow correlation to have a significant (per 10 CFR 50.46) nonconservative effect on the North Anna, Unit 1, LBLOCA analyses. In a letter dated November 10, 2003, the licensee proposed to compensate for this nonconservatism by adding a 64°F penalty to the PCT that was calculated with the methodology described in Topical Report EMF-2103 (P). In its November10, 2003, letter, the licensee committed to include this Forslund-Rohsenow penalty assessment in the

North Anna-specific LBLOCA models, and document this PCT penalty in the North Anna UFSAR. The licensee generated this penalty by disabling the Forslund-Rohsenow correlation to eliminate its nonconservative effect. The NRC staff finds that this provision is justified and acceptable. The analyses also exhibited a significant increase in time to quench. As such, the NRC staff requested that the licensee address the effect of the extended time to quench on oxidation results. The licensee addressed this concern in the November 10, 2003, letter by identifying the LBLOCA oxidation associated with the “non-Forslund-Rohsenow” case in the sensitivity study as the North Anna, Unit 1, oxidation value of record. However, the licensee did not perform this penalty calculation for this application. Instead, the licensee used a more conservative approach by using the calculated penalty of 64° F. Because the presently calculated PCTs are much lower than those for which the 64° F penalty was determined, the influence of the Forslund-Rohsenow correlation less, and the penalty significantly smaller, the NRC staff has determined that the use of the 64° F penalty added conservatism. As such, the NRC staff finds the licensee’s actions acceptable.

3.3.5 Inclusion of Radiative Effects in Convective Heat Transfer Coefficient

The RLBLOCA methodology described in Topical Report EMF-2103 (P) does not provide radiative heat transfer correlations. Instead, the convective heat transfer correlations in the methodology are derived from empirical data that include a significant amount of radiative heat transfer. By including a significant amount of radiative heat transfer in the calculation of total heat transfer for a plant that does not have a significant amount of post-LOCA radiative heat transfer, a licensee could significantly overestimate the total post-LOCA radiative heat transfer for its plant. Accordingly, the licensee would nonconservatively underestimate the PCT.

To address the NRC staff’s concern about this issue, the licensee, in its submittal dated January 22, 2004, provided calculations that compared the estimated radiative contribution to heat transfer for the North Anna, Unit 1, core to the radiative contribution to heat transfer for representative tests upon which the RLBLOCA methodology heat transfer models are based. The comparison indicates that the radiative heat transfer expected for this core is greater than the amount calculated using the RLBLOCA methodology for North Anna, Unit 1. The NRC staff concludes that the RLBLOCA methodology conservatively calculates less post-LOCA radiative heat transfer than would be expected for the core at North Anna, Unit 1, and, as a result, the NRC staff finds this to be acceptable.

3.3.6 Pellet Fragmentation and Relocation

The proposed LBLOCA methodology described in Topical Report EMF-2103 (P) did not provide for calculation of fuel pellet relocation. As such, the NRC staff postulated that ignoring these effects could lead the RLBLOCA methodology to underestimate the limiting PCT and oxidation values.

By letter dated September 5, 2003, the licensee referred to previous sensitivity studies that justified the omission of fuel pellet relocation from the LBLOCA methodology, and provided confirmation that these studies assumed a peak linear heat rate that remains bounding for the conditions at North Anna, Unit 1. The NRC staff concludes that this comparison demonstrates the applicability of the sensitivity studies to North Anna, Unit 1.

density and pre-pressurized the fuel rods. As shown in Topical Report BAW-10163-P-A, the analyses used to determine the effect of the inter-pellet gaps on power peaking demonstrated that the peaking factor increase due to spike densification is negligible for the Mark-BW fuel design. Since the Advanced Mark-BW design is an evolution of the Mark-BW design and the characteristics that were used to support the determination that the spike densification peaking factor is negligible remain applicable, not applying the spike densification peaking factor to Framatome Advanced Mark-BW fuel assemblies is acceptable.

3.6.4.2.8 Core Power Distributions

The reference core axial power distribution for the Advanced Mark-BW fuel design is the 1.55 cosine. This is also the power distribution that is currently in use for the resident Westinghouse NAIF fuel. Therefore, the core axial power distribution is consistently modeled for both fuel types.

The nuclear enthalpy rise hot-channel factor is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power. By letter dated March 28, 2002, the licensee incorrectly stated that the nuclear enthalpy rise hot-channel factor of 1.49 was a TS limit. Subsequently, the licensee informed the NRC staff on January 12, 2005, that the nuclear enthalpy rise hot-channel factor was not in fact a TS limit, instead it is a value that is bounded by the methodologies listed in TS 5.6.5, “Core Operating Limits Report.” Eventually, the licensee would like to increase the nuclear enthalpy rise hot-channel factor. However, a change to these methodologies that support the nuclear enthalpy rise hot-channel factor was not requested with this application. Therefore, the higher nuclear enthalpy rise hot-channel factor limits shall continue to be bounded by the methodologies in TS 5.6.5.

3.6.4.2.9 Core Bypass Flow

The Advanced Mark-BW fuel assemblies create a slight increase in the pressure drop across the core. This will increase the flow through alternate routes around the core. This increase in core bypass flow was analyzed, and the bounding core bypass flow was determined to be less than 5 percent when core inserts were not present. Similarly, the minimum core bypass flow was determined to be greater than 3 percent when 1500 core inserts were present. A bypass flow of 5.5 percent is specified for use in the statistical DNB analysis for the Advanced Mark-BW fuel design, and 6.5 percent is specified for non-statistical DNB applications and other deterministic nuclear steam supply system (NSSS) evaluations. The minimum bypass flow of 3 percent is used for the lift force calculations. Since the values for the core bypass flow used in the previous analysis methods are more conservative than the plant-specific values, the results of the previous analyses remain valid for the use of Framatome Advanced Mark-BW fuel.

3.6.4.3 Hydraulic Compatibility

To evaluate the pressure drop, hydraulic loads, and cross flow velocities in mixed core configurations, Framatome’s mixed-core methodology as outlined in section 3.6.9 of this SE is used. The calculations were performed using the LYNXT code. Four core configurations were considered: a full core of Advanced Mark-BW fuel assemblies, a full core of Westinghouse NAIF assemblies, the most limiting configuration for the Framatome Advanced Mark-BW fuel assemblies, and the most limiting configuration for the NAIF fuel assemblies. The most limiting configuration for the Advanced Mark-BW fuel is a full core of Advanced Mark-BW fuel assemblies, while the most limiting configuration for the NAIF fuel is a single fuel assembly in the center of the core with the rest of the core composed of Advanced Mark-BW fuel assemblies. The core operational conditions used for the analyses included cold zero power, hot zero power, hot full power, hot overpower, normal flow, mechanical design flow, and high flow.

3.6.4.3.1 Mixed-core Nominal Pressure Drop Results

Pressure drop evaluations demonstrated that the most limiting core configurations were obtained when a full core of Advanced Mark-BW fuel assemblies is used for the Advanced Mark-BW fuel design and when one NAIF assembly is in the center of the core with the rest of the core composed of Advanced Mark-BW fuel assemblies for the NAIF fuel design. These limiting pressure drop results are used in the evaluation of the non-LOCA transients.

3.6.4.3.2 Mixed-core Hydraulic Load Results

The results of the hydraulic load calculation is used for the evaluation of the fuel assembly holddown springs that was previously discussed in section 3.6.3.3 of this SE. The maximum hydraulic loads do not include the fuel assembly weight, buoyancy forces, or spring holddown forces. The maximum hydraulic load for the Advanced Mark-BW fuel assembly was obtained from the full-core Advanced Mark-BW core configuration. The maximum hydraulic load for the NAIF fuel assembly was obtained from the one NAIF fuel assembly in the center of the core with the rest of the core composed of Advanced Mark-BW fuel assemblies configuration.

3.6.4.3.3 Mixed-core Cross Flow Velocities

The design criterion for fretting for the Advanced Mark-BW fuel design is that the span average cross flow velocities must be less than 2 ft/sec. Analyses performed with a mixed-core configuration demonstrated that the maximum cross flow velocities were seen when one NAIF fuel assembly was in the center core location with the remainder of the core consisting of Advanced Mark-BW fuel assemblies. The results of the analysis of this limiting configuration demonstrated that the cross flow velocities remained below the criteria.

3.6.4.4 DNB Performance Evaluation

To demonstrate that the DNB performance of the Advanced Mark-BW fuel design is acceptable, full- and mixed-core configurations were evaluated. The full-core analysis uses the Framatome Statistical Core Design (SCD) methodology described in Babcock and Wilcox Topical Report BAW-10170-P-A, "Statistical Core Design For Mixing Vane Cores," dated December 23, 1988, while a mixed-core analysis uses the Mixed-Core Methodology outlined in section 3.6.9 of this SE. Both types of analyses use the LYNXT code. The design criterion for use with the SCD methodology is that the minimum DNBR must be equal to or greater than the thermal design limit (TDL), which is defined in Section 3.6.4.4.2. The design criterion for use with non-SCD methods is that the minimum DNBR must be equal to or greater than the CHF correlation design limits.

3.6.4.4.1 State Points for DNB Calculations

A set of state points were developed by the licensee for use in the DNB analyses. These state points represent points on the safety limit lines, limiting axial flux shapes at several axial offsets, and state points for several transient events, including misaligned rod, loss of flow, rod withdrawal at power, locked rotor, rod urgent failure, and rod withdrawal from subcritical and steamline break. In general, the state point conditions that were defined to evaluate the Advanced Mark-BW were used to evaluate the NAIF in the transition core analysis.

3.6.4.4.2 SCD

The Framatome SCD methodology, as described in Topical Report BAW-10170-P-A, together with the LYNXT code, is used to assess the thermal margin for the Advanced Mark-BW fuel design. The SCD method is independent of fuel type and does not specify an analysis code. The methodology in Topical Report BAW-10170-P-A has been approved for use in analyzing Framatome fuel in Westinghouse-designed reactors.

The SCD approach uses a technique that statistically combines uncertainties. In this method, the uncertainties on a group of input variables are subjected to a statistical analysis, and an overall DNBR uncertainty is established. This uncertainty is used to establish a DNBR design limit known as the SDL. Margin is then added to this limit for additional flexibility and the combination results in an analysis limit called the TDL. The calculated DNBR is compared to this TDL to demonstrate that the DNB protection is acceptable.

The TDL for the North Anna mixed core was determined to be 1.70. Section 4.4.2 of the licensee's submittal dated March 28, 2002, identified the SDL values as 1.61 for the BWU-N correlation and 1.31 for the BWU-Z correlation. Both of these values are less than the TDL and this demonstrates that thermal margin is retained with the use of the Advanced Mark-BW fuel design.

3.6.4.4.3 Full-core SCD DNB Analysis for Advanced Mark-BW

When a full-core SCD DNB analysis is performed for the Advanced Mark-BW fuel, calculations demonstrate that the minimum DNBR values are greater than or equal to the TDL of 1.70.

3.6.4.4.4 Mixed-core DNB Analysis

The mixed-core DNB analysis uses the Framatome mixed-core methodology discussed in Section 3.6.9 of this SE. The DNB calculations are performed using the LYNXT code. To determine the mixed-core penalty, the DNBR results from mixed-core configurations are compared to calculations performed at identical conditions with either a full-core Advanced Mark-BW model or a full-core NAIF model. The mixed-core penalty is equal to the largest differential value.

Various core-loading patterns are evaluated to determine the most limiting core configurations for the Advanced Mark-BW fuel assembly. Similarly, the most limiting core configuration was determined for the resident NAIF fuel. Using these results, mixed-core penalties were determined for the first and second batch loads for the Advanced Mark-BW fuel, and a mixed-core penalty was determined for DNB for the NAIF fuel. The calculated mixed-core penalty factors are acceptable because they consider the important parameters in the most limiting core configurations with acceptable resolution to determine the impact of transitioning from one fuel type to another.

ENCLOSURE 2

CORRECTIONS TO SAFETY EVALUATION FOR NORTH ANNA, UNIT 2 representative tests upon which the RLBLOCA methodology heat transfer models are based. The comparison indicates that the radiative heat transfer expected for this core is greater than the amount calculated using the RLBLOCA methodology for North Anna, Unit 2. The NRC staff concludes from this that the RLBLOCA methodology conservatively calculates less post-LOCA radiative heat transfer than would be expected for the core at North Anna, Unit 2, and, as a result, the NRC staff finds this to be acceptable.

3.3.6 Pellet Fragmentation and Relocation

The proposed LBLOCA methodology described in Topical Report EMF-2103 (P) did not provide for calculation of fuel pellet relocation. As such, the NRC staff postulated that ignoring these effects could lead the RLBLOCA methodology to underestimate the limiting PCT and oxidation values.

By letter dated September 5, 2003, the licensee referred to previous sensitivity studies that justified the omission of fuel pellet relocation from the LBLOCA methodology, and provided confirmation that these studies assumed a peak linear heat rate that remains bounding for the conditions at North Anna, Unit 2. The NRC staff concludes that this comparison demonstrates the applicability of the sensitivity studies to North Anna, Unit 2.

3.3.7 Lower Limit in Ranging of Break Size in RLBLOCA Calculations

In response to the NRC staff's concern that the minimum split break size in the ranging of large breaks was well below the size that delineates large breaks from small breaks, the licensee, in its letter dated September 22, 2003, stated that the smallest break size used in the distribution was at a size boundary between large and small breaks "with the total break area defined as the sum from both break junctions." The NRC staff notes that since the friction loss coefficient at the break varies inversely with break size, the sum of the flows from the two small-break junctions might likely be less than the break flow at the smallest large-break junction.

The licensee's analyses indicated that during blowdown the core region became completely voided, thus exhibiting a large-break characteristic. Afterwards, the liquid level in the core region began to increase after the initiation of accumulator injection. Based on these results, the licensee concluded that SBLOCA phenomena did not occur in any of the cases that were run using the RLBLOCA methodology described in Topical Report EMF-2103 (P). Based on the licensee's analyses, the NRC staff concludes that for the RLBLOCA methodology the ranging of break size to a smaller size than is usually considered large is acceptable because the smallest break size analyses exhibited LBLOCA phenomena. The NRC staff also noted that this demonstration for the smaller break sizes is unique to the present LBLOCA analyses at North Anna, Unit 2, and possibly attributable to low containment back pressure and the plant-specific vessel internals design.

3.3.8 Counter Current Flow Warning

In a letter dated December 20, 2002, Framatome ANP committed to implement a counter current flow limitation (CCFL) warning in the Topical Report EMF-2103 (P) S-RELAP5 computer code in order to alert an analyst to a CCFL violation in the downcomer region should one occur. By letter dated August 20, 2003, the licensee discussed that a CCFL warning had been implemented in the RLBLOCA methodology in order to address this matter. Based on its cladding grows at a lower rate than Zircaloy cladding under irradiation conditions, the database for Zircaloy is conservative relative to the M5 performance. Therefore, use of this database for predicting the rod bow of M5 clad fuel and continuing to use the penalty generated by the Zircaloy database for M5 fuel is conservative and acceptable for use.

3.6.4.2.6 Active Fuel Stack Height

The Advanced Mark-BW fuel design uses an increased density fuel pellet. Therefore, under irradiation conditions the pellets experience less shrinkage and subsequently the overall fuel stack shrinks less. Thermal expansion of the fuel upon initial heatup is greater than the stack shrinkage with irradiation. Therefore, neglecting the stack shrinkage and assuming a nominal fuel stack height is conservative.

3.6.4.2.7 Spike Densification Peaking Factor

The spike densification peaking factor is used to account for the peaking that is caused by inter-pellet gap formation as the result of fuel densification. To prevent the formation of large gaps between pellets and to prevent cladding creep collapse, Framatome increased the fuel density and pre-pressurized the fuel rods. As shown in Topical Report BAW-10163-P-A, the analyses used to determine the effect of the inter-pellet gaps on power peaking demonstrated that the peaking factor increase due to spike densification is negligible for the Mark-BW fuel design. Since the Advanced Mark-BW design is an evolution of the Mark-BW design and the characteristics that were used to support the determination that the spike densification peaking factor is negligible remain applicable, not applying the spike densification peaking factor to Framatome Advanced Mark-BW fuel assemblies is acceptable.

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To evaluate the pressure drop, hydraulic loads, and cross flow velocities in mixed core configurations, Framatome's mixed core methodology as outlined in section 3.6.9 of this SE is used. The calculations are performed using the LYNXT code. Four core configurations were considered: a full core of Advanced Mark-BW fuel assemblies, a full core of Westinghouse NAIF assemblies, the most limiting configuration for the Framatome Advanced Mark-BW fuel assemblies, and the most limiting configuration for the NAIF fuel assemblies. The most limiting configuration for the Advanced Mark-BW fuel is a full core of Advanced Mark-BW fuel assemblies, while the most limiting configuration for the NAIF fuel is a single fuel assembly in the center of the core with the rest of the core composed of Advanced Mark-BW fuel assemblies. The core operational conditions used for the analyses included cold zero power, hot zero power, hot full power, hot overpower, normal flow, mechanical design flow, and high flow.

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Pressure drop evaluations demonstrated that the most limiting core configurations were obtained when a full core of Advanced Mark-BW fuel assemblies is used for the Advanced Mark-BW fuel design and when one NAIF assembly is in the center of the core with the rest of the core composed of Advanced Mark-BW fuel assemblies for the NAIF fuel design. These limiting pressure drop results are used in the evaluation of the non-LOCA transients.

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Advanced Mark-BW fuel assemblies. The results of the analysis of this limiting configuration demonstrated that the cross flow velocities remained below the criteria.

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A set of state points were developed by the licensee for use in the DNB analyses. These state points represent points on the safety limit lines, limiting axial flux shapes at several axial offsets, and state points for several transient events, including misaligned rod, loss of flow, rod withdrawal at power, locked rotor, rod urgent failure, and rod withdrawal from subcritical and steamline break. In general, the state point conditions that were defined to evaluate the Advanced Mark-BW were used to evaluate the NAIF in the transition core analysis.

3.6.4.4.2 SCD

The Framatome SCD methodology, as described in Topical Report BAW-10170-P-A, together with the LYNXT code, is used to assess the thermal margin for the Advanced Mark-BW fuel design. The SCD method is independent of fuel type and does not specify an analysis code. The methodology in Topical Report BAW-10170-P-A has been approved for use in analyzing Framatome fuel in Westinghouse-designed reactors.

The SCD approach uses a technique that statistically combines uncertainties. In this method, the uncertainties on a group of input variables are subjected to a statistical analysis, and an overall DNBR uncertainty is established. This uncertainty is used to establish a DNBR design limit known as the SDL. Margin is then added to this limit for additional flexibility and the combination results in an analysis limit called the TDL. The calculated DNBR is compared to this TDL to demonstrate that the DNB protection is acceptable.

The TDL for the North Anna mixed core was determined to be 1.70. Section 4.4.2 of the licensee’s submittal dated March 28, 2002, identified the SDL values as 1.61 for the BWU-N correlation and 1.31 for the BWU-Z correlation. Both of these values are less than the TDL and demonstrate that thermal margin is retained with the use of the Advanced Mark-BW fuel design.

3.6.4.4.3 Full-core SCD DNB Analysis for Advanced MARK-BW

When a full core SCD DNB analysis is performed for the Advanced Mark-BW fuel, calculations demonstrate that the minimum DNBR values are greater than or equal to the TDL of 1.70.

North Anna Power Station, Units 1 & 2

cc:

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