

VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

August 13, 2008

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

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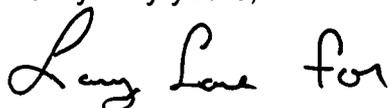
Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)
NORTH ANNA POWER STATION UNITS 1 AND 2
SUMMARY OF FACILITY CHANGES, TESTS AND EXPERIMENTS

Dominion letter dated March 28, 2008 (Serial No. 08-0125) provided a summary description of Facility Changes, Tests and Experiments identified in Regulatory Evaluations implemented at the North Anna Power Station during 2007 in accordance with 10 CFR 50.59(d)(2). It was recently identified that summary descriptions for two additional evaluations were not included in the previous submittal. Summary descriptions for these evaluations are provided in the attachment to this letter. Process enhancements have been initiated to ensure future reports include the required information.

If you have any questions, please contact Page Kemp at (540) 894-2295.

Very truly yours,



Daniel G. Stoddard, P.E.
Site Vice President

Attachment

cc: U. S. Nuclear Regulatory Commission
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NRC Senior Resident Inspector
North Anna Power Station

IE47
NRR

ATTACHMENT

**10 CFR 50.59 SUMMARY DESCRIPTION OF
FACILITY CHANGES, TESTS AND EXPERIMENTS**

North Anna Power Station Units 1 and 2

**Virginia Electric and Power Company
(Dominion)**

NORTH ANNA UNITS 1 & 2

10 CFR 50.59 SUMMARY DESCRIPTION OF FACILITY CHANGES, TESTS AND EXPERIMENTS

REGULATORY EVALUATION: 07-SE-ST-02

Documents Evaluated: ET-NAF-07-0020, UFSAR Change Request FN-2005-043, Reconstitution Methodology for AREVA Advanced Mark-BW-Fuel Assemblies, North Anna Power Station Units 1 and 2

Brief Description: The North Anna Power Station Technical Specifications contains references to the use of limited substitutions of stainless steel filler rods for fuel rods using NRC approved codes and methods. The North Anna UFSAR currently allows the use of reconstituted Westinghouse fuel assemblies but does not discuss the use of reconstituted AREVA Advanced Mark-BW (AMBW) fuel assemblies. The proposed activity is the implementation of a new methodology for the evaluation of reconstitution of AREVA Advanced Mark-BW fuel assemblies. The proposed UFSAR changes incorporate references to the NRC approved AREVA methodology for the analysis of reconstituted fuel assemblies and to allow for the use of reconstituted AREVA Advanced Mark-BW fuel assemblies.

Technical Report NE-1463, Revision 0 documents the technical and licensing information that supports the implementation of AREVA Topical Report BAW-2149-A, "Evaluation of Replacement Rods in BWFC Fuel Assemblies," for the reconstitution of the AREVA Advanced Mark-BW fuel product at North Anna Units 1 and 2. BAW-2149-A permits the use of up to ten solid stainless steel replacement rods per assembly.

Reason for Change: The purpose of the Engineering Transmittal (ET) was to provide the appropriate documentation to support the implementation of AREVA Topical Report BAW-2149-A, "Evaluation of the Replacement Rods in BWFC Fuel Assemblies," for the reconstitution of the AREVA Advanced Mark-BW fuel product at North Anna Units 1 and 2. The ET also provided a UFSAR Change Request (FN-2005-043) for North Anna Power Station.

Summary: Technical Report NE-1463, Revision 0 documents the technical and licensing information that supports the implementation of AREVA Topical Report BAW-2149-A, "Evaluation of Replacement Rods in BWFC Fuel Assemblies," for the reconstitution of the AREVA Advanced Mark-BW fuel product at North Anna Units 1 and 2. AREVA Topical Report BAW-2149-A describes the NRC approved methodology for evaluating replacement rods in BWFC fuel assemblies. Pacific Northwest Laboratory (PNL) acted as a consultant to the NRC in the review of the report. The NRC Safety Evaluation Report (SER) and PNL Technical Evaluation Report (TER) are included in the approved version of BAW-2149-A. The report also includes the responses to two rounds of Requests for Additional Information (RAIs).

The proposed activity is not a departure from a method of evaluation described in the FSAR used in establishing the design basis or in the safety analyses. The implementation of the proposed activity is made under two provisions: USNRC Generic Letter (GL) 83-11, Supplement 1 and 10 CFR 50.59. The evaluation is a two step process. In the first step, an evaluation is performed to demonstrate that the requirements are met for implementation of the BAW-2149-A methodology under Generic Letter 83-11, Supplement 1. In the second step, an evaluation is performed to demonstrate that the changes to elements of the BAW-2149-A may be implemented without prior NRC review and approval methodology pursuant to 10 CFR 50.59, part (A)(2)(ii). A summary of this evaluation is presented below.

Generic Letter (GL) 83-11, Supplement 1, Attachment 1, Guidelines for Qualifying Licensees to Use Generally Approved Analysis Methods, presents a simplified approach for qualifying licensees to use NRC approved analysis methods. Typically, these methods are developed by fuel vendors, utilities, national laboratories, or organizations such as the Electric Power Research Institute, Incorporated (EPRI). This procedure establishes a formal program which satisfies the guidelines presented in GL 83-11, Supplement 1. The procedure applies to the use of an analytical method which has been approved by the NRC either generically or on another plant licensee's docket to replace or modify an existing Dominion method that has been (a) reviewed and approved by the NRC or (b) previously implemented by Dominion under the provisions of GL 83-11, Supplement 1.

Virginia Power Administrative Procedure (VPAP) 0313, Implementation of Analytical Methods Under the Provisions of Generic Letter 83-11, Supplement 1, establishes the formal program pursuant to GL 83-11, Supplement 1. The program delineates the process Dominion will use for implementing the analytical methods which have been approved by the USNRC either generically or on another plant licensee's docket. Implementation of BAW-2149-A, a generically approved vendor methodology, falls under the requirements of this program. Changes to some elements of the methodology described in BAW-2149-A are required however refer to the current set of NRC approved computer codes and models necessary to apply the methodology to North Anna. The changes to BAW-2149-A for application to North Anna AMBW reconstitution are evaluated under 10 CFR 50.59.

Technical Report NE-1463 concludes that BAW-2149-A can be implemented under the provisions of GL 83-11, Supplement 1, because the GL provisions are applicable. This is done by demonstrating that the method under consideration has been approved by the NRC for the intended application and type of analysis being conducted, and that applicable terms, conditions and limitations for its use are satisfied. Specifically, the NRC SER approved the generic application of AREVA methodology in BAW-2149-A for the reconstitution of the AREVA fuel product. Therefore, BAW-2149-A has been approved for use by the NRC on a non-plant specific basis (i.e., generic basis). By its title and introduction, BAW-2149-A is limited to the use of replacement rods for B&W Fuel Company (BWFC) supplied fuel assemblies. AREVA is the subsequent owner of the BWFC technologies, methodologies, and fuel manufacturing. Hence, BAW-2149-A

is applicable to the AREVA supplied AMBW fuel assemblies. Thus, implementation of BAW-2149-A at North Anna falls under the requirements of the VPAP-0313 program. Attachment A of NE-1463 provides the Code/Method Implementation Checklist for implementation under the VPAP-0313 (i.e., GL 83-11, Supplement 1) program.

The proposed activity substitutes the AREVA specific elements of the methodology for nuclear core design, thermal-hydraulic design and reload design analyses with NRC approved Dominion specific elements. Lists of the NRC approved methodologies (elements of the BAW-2149-A method) for BAW-2149-A, the current AREVA analyses, and the current Dominion analyses are provided in Section 5.0 of NE-1463. The current Dominion methodologies (elements of BAW-2149-A, e.g., codes) used to evaluate reconstituted fuel assemblies with stainless steel replacement rods are essentially the same as those discussed in BAW-2149-A, in that the approved codes are benchmarked to measured data and the analysis methodology requires inclusion of appropriate uncertainties to account for differences between measures and predicted values, such a radial and total peaking nuclear uncertainty factors. With these uncertainties appropriately accounted for, these NRC approved codes yield either conservative or accurate power distribution solutions; from this standpoint, the "reconstitution methodologies" yield equivalent results.

In summary, BAW-2149-A is a generally approved methodology for the evaluation of reconstituted fuel assemblies which can be implemented under GL 83-11, Supplement 1 without prior NRC review or approval. The changes to BAW-2149-A, which consist of the substitution of AREVA specific elements of the methodology for nuclear core design, thermo-hydraulic design and reload design analyses with NRC approved Dominion specific elements, can also be implemented under 10 CFR 50.59 without prior NRC review and approval since they are essentially the same as those discussed in the BAW-2149-A.

Therefore, the proposed activity (i.e., NE-1463, Revision 0, UFSAR Change Request FN 2005-043 and ET-NAF-07-0020) may be implemented without NRC review/approval.

REGULATORY EVALUATION: 07-SE-ST-03

Documents Evaluated: Licensing Basis Document Change Request 18 and UFSAR Change Request FN-2007-010

Brief Description:

North Anna Technical Specifications (TS) currently require periodic measurement of response times for Reactor Trip System (RTS) and Engineered Safety Feature (ESF) instrumentation channels. The TS definitions for ESF Response Time and RTS Response Time state that in lieu of measurement, response times may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

The proposed change would revise TS Bases for Response Time Testing (RTT) to allow for the use of allocated times for sensors, process protection system (7300), the nuclear instrumentation system (NIS), and the solid state protection system (SSPS).

The allocated times are determined using the guidance in WCAP-13632-P-A, revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" and WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests" which have been approved by the NRC. The allocated times will be added to the actual measured times for the final output device and compared to the accident analysis limits.

Reason for Change:

Eliminate response time testing for selected instrumentation loops.

Summary: North Anna Technical Specification (TS) Table 3.3.1-1 and Table 3.3.2-1 require response time testing (RTT) for selected Reactor Trip System (RTS) and Engineered Safety Feature (ESF) functions. During the implementation of the Improved Technical Specifications (ITS), North Anna TS 1.1 definitions for Engineered Safety Feature (ESF) Response Time and Reactor Trip System (RTS) Response Time were revised to allow for the verification of response times versus actual measurement provided the components and methodology for the verification were previously reviewed and approved by the NRC. The change to the definitions was based on Technical Specification Task Force Improved Standard Technical Specification Change Traveler, TSTF-111, Revision 6, which was approved by the Nuclear Regulatory Commission (NRC) on June 29, 1999.

A decision was made during the implementation of ITS to not incorporate the required TS Bases changes, as specified in TSTF-111, because the technical evaluation to determine that the terms, conditions, and limitations for use of the analysis based on the NRC Safety Evaluation Reports (SERs) had not been completed.

The proposed change being evaluated would revise TS Bases B 3.3.1 and B 3.3.2 to allow for the use of allocated times for sensors, process protection system (7300), the nuclear instrumentation system (NIS), and the solid state protection system (SSPS). The allocated times will be added to the actual measured times for the actuated devices and compared to the accident analysis limits.

The change to response time testing is in accordance with WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements", which was approved by the NRC on September 5, 1995 and WCAP-14036-P-A, Revision 1, "Elimination of Periodic Protection Channel Response Time Tests", which was approved by the NRC on October 6, 1998. The technical evaluation is documented in Engineering Transmittal (ET) N-07-0049.

The change to response time testing does not change the design or operation of the RTS or ESF systems. The RTS will continue to be the primary system for detecting conditions that may cause a nuclear safety event and automatically initiates a reactor trip to mitigate the consequences of such events. The ESF systems will still actuate as required to mitigate the consequences of an accident and to protect the public by preventing or minimizing the release of fission products. Periodic surveillance of the systems will continue and may be used to detect degradation that could cause the response time to exceed the total allowance as specified in the periodic tests. The response time that is allocated for each instrument string in this change bounds all degradation that cannot be detected by periodic surveillance tests.

The new method of evaluation of response times for selected RTS and ESF components does not represent a departure from a method of evaluation as described in 10 CFR 50.59 and guidance in NEI 96-07, Revision 1. The new methodology does not represent a departure since the method is appropriate for the intended application and type of analysis conducted. The applicable terms, conditions, and limitations for its use as specified in the related NRC SERs have been satisfied, and the method has been approved by the NRC.