September 6, 2007

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

SUBJECT: SURRY POWER STATION, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS REGARDING USE OF WESTINGHOUSE BEST-ESTIMATE LARGE BREAK LOSS-OF-COOLANT ACCIDENT ANALYSIS METHODOLOGY (TAC NOS. MD3609 AND MD3610)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 254 to Renewed Facility Operating License No. DPR-32 and Amendment No. 253 to Renewed Facility Operating License No. DPR-37 for the Surry Power Station, Unit Nos. 1 and 2, respectively. The amendments change the Technical Specifications (TSs) in response to your application dated November 16, 2006, as supplemented by letters dated March 29 and July 31, 2007.

These amendments add a reference in Technical Specification (TS) Section 6.2.C, "Core Operating Limits Report (COLR)," to permit the use of the Westinghouse Best-Estimate Large Break Loss-of-Coolant Accident (BE-LBLOCA) analysis methodology using the Automated Statistical Treatment of Uncertainty Method (ASTRUM) for the analysis of LBLOCA.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Siva P. Lingam, Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosures:
1. Amendment No. 254 to DPR-32
2. Amendment No. 253 to DPR-37
3. Safety Evaluation

cc w/encls: See next page

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Package No.: ML071430067 Amendment No.: ML071430074
Tech Spec No.: ML 072490614 *transmitted by memo dated.

OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	DSS/SPWB/BC	DIRS/ITSB/BC	OGC	NRR/LPL2-1/BC
NAME	SLingam:nc	MO'Brien	GCranston	TKobetz	BMizuno	EMarinos
DATE	8/8/07	8/9/07	8/1/07*	8/25/07	8/23/07	/ /07

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VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

SURRY POWER STATION, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 254
Renewed License No. DPR-32

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated November 16, 2006, as supplemented by letters dated March 29 and July 31, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-32 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 254, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented at the completion of Unit 1 fall 2007 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. DPR-32
and the Technical Specifications

Date of Issuance: September 6, 2007

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 253
Renewed License No. DPR-37

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Virginia Electric and Power Company (the licensee) dated November 16, 2006, as supplemented by letters dated March 29 and July 31, 2007, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Renewed Facility Operating License No. DPR-37 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 253, are hereby incorporated in the renewed license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance and shall be implemented at the completion of Unit 1 fall 2007 refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Evangelos C. Marinos, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes License No. DPR-37
and the Technical Specifications

Date of Issuance: September 6, 2007

ATTACHMENT

TO LICENSE AMENDMENT NO. 254

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

AND

TO LICENSE AMENDMENT NO. 253

RENEWED FACILITY OPERATING LICENSE NO. DPR-37

DOCKET NO. 50-281

Replace the following pages of the Licenses and the Appendix A Technical Specifications (TSs) with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove Pages	Insert Pages
License	License
License No. DPR-32, page 3	License No. DPR-32, page 3
License No. DPR-37, page 3	License No. DPR-37, page 3
TSs	TSs
6.2-2	6.2-2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 254 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AND

AMENDMENT NO. 253 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY

SURRY POWER STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated November 16, 2006 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML063210473), as supplemented by letters dated March 29 (ADAMS Accession No. ML070930084) and July 31, 2007 (ADAMS Accession No. ML072190482), Virginia Electric and Power Company (the licensee) submitted a request for changes to the Surry Power Station, Unit Nos. 1 and 2 (Surry 1 and 2), Technical Specifications (TSs). On May 24, 2007, the licensee informed the Nuclear Regulatory Commission (NRC) that a computer code error affecting the determination of the limiting peak clad temperature (PCT) was discovered by Westinghouse, and this error does not affect the conclusions of the analyses submitted by letter dated November 16, 2006. The supplements dated March 29, 2007, and July 31, 2007, provided clarifying information that did not change the scope of the original application and the initial proposed no significant hazards consideration determination.

The amendment requests to apply the NRC-approved Westinghouse best-estimate large break loss-of-coolant accident (BE-LBLOCA) methodology described in WCAP-16009-P-A, “Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM),” January 2005 (ADAMS Nos. ML050910159 (Non-Proprietary), ML050910161(Non-Proprietary), ML050910162 (Proprietary) and ML050910163 (Proprietary)), to Surry 1 and 2. The licensee also requested a license amendment to include the ASTRUM LBLOCA methodology in TS 6.2.C, ”Core Operating Limits Report (COLR)” for each unit.

The NRC staff reviewed the licensee’s demonstration evaluations of the emergency core cooling system (ECCS) performance analyses, done in accordance with the ASTRUM methodology, for Surry 1and 2 operating at their currently licensed core powers of 2546 MWt (the analyses were conducted at the rated power of 2546 MWt plus 2 % measurement uncertainty or 2597 MWt). These specific analyses were performed to demonstrate the suitability of the ASTRUM methodology for application to the Surry 1 and 2. Also the specific

analyses, when approved herein will be acceptable and specifically applicable to Surry 1 and 2 operated with the fuel(s) identified in Table 1 that follows. For Surry 1and 2 the BE-LBLOCA analyses were conducted assuming that the plants use cores containing Vantage$^+$ (ZIRLO - clad fuel) assemblies.

2.0 REGULATORY EVALUATION

The BE-LBLOCA analyses were performed to demonstrate that the ECCS design would provide sufficient ECCS flow to remove the heat from the reactor core following an 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 the amounts that 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 that suitable interconnections, leak detection, isolation, and containment capabilities were available such that the safety functions could be accomplished, assuming a single failure, for LBLOCAs and considering the availability of only onsite or offsite electric power (i.e., 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 NRC staff used the acceptance criteria for ECCS performance provided in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.46, in assessing the acceptability of the Westinghouse ASTRUM methodology for Surry 1 and 2.

In its assessment of the acceptability of the methodology for Surry 1 and 2, the NRC staff also reviewed the limitations and conditions stated in its safety evaluation supporting general approval of the Westinghouse ASTRUM methodology and the range of parameters described in the ASTRUM topical report.

3.0 TECHNICAL EVALUATION

In its March 29, 2007, submittal, the licensee stated, “Dominion and Westinghouse Electric Company (analysis vendor) have ongoing processes that assure that the ranges and values of the input parameters for the Surry Units 1 and 2 Best-Estimate Large Break LOCA (BE-LBLOCA) analysis using ASTRUM bound the ranges and values of the as-operated plant values for those parameters.” The NRC staff finds that this statement, along with the generic acceptance of ASTRUM, provides assurance that ASTRUM and its LBLOCA analyses apply to Surry 1 and 2, respectively, operated at their current licensed power levels.

In its submittal, the licensee provided the results for the Surry 1 and 2 BE-LBLOCA analyses, each operating at the rated power of 2546 MWt (plus 2 % measurement uncertainty or 2597 MWt) performed in accordance with the ASTRUM methodology. The licensee’s results for the calculated peak cladding temperatures (PCTs), the maximum cladding oxidations (local), and the maximum core-wide cladding oxidations for each unit are provided in the following table along with the acceptance criteria of 10 CFR 50.46(b).

TABLE 1

LARGE BREAK LOCA ANALYSES RESULTS - Surry Units 1 and 2

Parameter	Unit 1 ASTRUM Vantage+ Results	Unit 2 ASTRUM Vantage+ Results	10 CFR 50.46 Limits
Limiting Break Size/Location	DEG/PD	DEG/PD	N/A
Cladding Material	Zirlo	Zirlo	(Cylindrical) Zircaloy or Zirlo
Peak Clad Temperature	2044 °F	2044 °F	2200 °F (10 CFR 50.46(b)(1))
Maximum Local Oxidation	5.26 %	5.26 %	17.0 % (10 CFR 50.46(b)(2))
Maximum Total Core-Wide Oxidation (All Fuel)	> 0.32 %	> 0.32%	1.0 % (10 CFR 50.46(b)(3))

DEG/PD is a double-ended guillotine break at the pump discharge.

In its analyses, the licensee also addressed the concern that the Vantage+ fuel cladding may have pre-existing oxidation that must be considered in its LOCA analyses. In its March 29, 2007, response to an NRC staff's request for additional information, the licensee indicated that it considered whether the fuel cladding has both pre-existing oxidation and oxidation resulting from the LOCA (pre- and post-LOCA oxidation).

In the March 29, 2007, letter, the licensee indicated that the calculated pre-LOCA oxidation was factored into the licensee's BE-LBLOCA analyses for the Zirlo clad fuel, consistent with the Westinghouse ASTRUM methodology. Both plants will be operated as governed by the licensee's program consistent with the Westinghouse recommendations to limit operational duty within fuel duty limits, even during a fuel pin's final cycle in the core, such that the sum of the calculated pre- and post-LOCA oxidation will be sufficiently small that the total local oxidation will remain less than the 17 % acceptance criterion of 10 CFR 50.46(b)(2) as noted above.

The concern with core-wide oxidation relates to the amount of hydrogen generated during a LOCA. Because hydrogen that may have been generated pre-LOCA (during normal operation) will be removed from the reactor coolant system throughout the operating cycle, the NRC staff noted that pre-existing oxidation does not contribute to the amount of hydrogen generated post-LOCA; and, therefore, it does not need to be addressed when determining whether the calculated total core-wide oxidation meets the 1.0 % criterion of 10 CFR 50.46(b)(3).

As discussed previously, the licensee had Westinghouse conduct BE-LBLOCA analyses for Surry 1 and 2 operating at about 102 % of the current licensed power level of 2546 MWt using an NRC-approved Westinghouse methodology (ASTRUM). The NRC staff concluded that the results of these analyses demonstrated compliance with 10 CFR 50.46(b)(1) through (b)(3) for licensed power levels of up to 2546 MWt. Meeting these criteria provides reasonable assurance that at the current licensed power level, the Surry 1 and 2 cores will be amenable to cooling as required by 10 CFR 50.46(b)(4). The capability of Surry 1 and 2 to satisfy the long-

term cooling requirements of 10 CFR 50.46(b)(5) is unaffected by this amendment, and will be addressed if needed in a future NRC safety evaluation.

TECHNICAL SPECIFICATIONS CHANGE

In support of its November 16, 2006, application, the licensee proposed to make amendments to the Surry 1 and 2 TS 6.2.C, "Core Operating Limits Report (COLR)," for each unit to reflect use of a new LBLOCA and analysis methodology to perform LBLOCA analyses in support of Surry 1 and 2 operation. The licensee also provided a proposed page TS 6.2.C, "CORE OPERATING LIMITS REPORT (COLR)," amended to reflect the implementation of ASTRUM as the licensing basis LBLOCA methodology for the Surry 1 and 2. The NRC staff reviewed the TS provision, assessed it for consistency against NUREG-1431, Revision 3, as stated below, and found its content acceptable and compatible with a proposed COLR (when the cycle-specific COLR is submitted, including the reference as discussed below).

Proposed Technical Specifications change:

2a. WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)," (Westinghouse Proprietary)

This methodology was found to apply to all Westinghouse and Combustion Engineering pressurized water reactor (PWR) designs in the NRC generic safety evaluation of the ASTRUM methodology. Therefore, ASTRUM is acceptable for application to Surry 1 and 2, which are PWRs of Westinghouse design, and for inclusion in Surry 1 and 2 TSs for each unit. The above listed TS Reference 2a, was presented in the licensee's submittal as a TS addition. This reference does not include the WCAP-16009-P-A revision number (i.e., "0"); nor does it include the date of approval for the methodology. The licensee will list the topical report, including the latest revision number (used at Surry), and date of approval in the COLR for each of the Surry units consistent with guidance provided in NUREG-1431, "Standard Technical Specifications, Westinghouse Plants," Revision 3, TS 5.6.5.

The NRC staff finds that ASTRUM is applicable to Surry 1 and 2 and that the limitations and conditions of the NRC's safety evaluation approving ASTRUM were satisfied. The NRC staff concludes that the proposed addition of WCAP-16009-P-A to TS 6.2.C is acceptable.

Based on its review as discussed above, the NRC staff concluded that the Westinghouse ASTRUM methodology, as described in WCAP-16009-P-A, is acceptable for use for Surry 1 and 2 in demonstrating compliance with the requirements of 10 CFR 50.46(b). The NRC staff's conclusion is based on the staff's verification that the Surry 1 and 2 design is among the designs for which ASTRUM application was approved.

The NRC staff's review of the acceptability of the ASTRUM methodology for Surry 1 and 2 focused on assuring that the licensee and its vendor have processes to assure that plant-specific input parameters or bounding values and ranges (where appropriate) are used to conduct the Surry 1 and 2 LBLOCA analyses, that the analyses will be conducted within the conditions and limitations of the NRC-approved Westinghouse ASTRUM methodology, and that the results will satisfy the requirements of 10 CFR 50.46(b) for Surry 1 and 2.

This safety evaluation documents the NRC staff's review and acceptance of the Westinghouse ASTRUM BE-LBLOCA analysis methodology for application to Surry 1 and 2, for inclusion in the Surry 1 and 2 TS 6.2.C and COLRs, and of the specific LBLOCA analyses discussed above that were performed with the ASTRUM methodology for Surry 1 and 2 operated at powers up to their licensed power level of 2546 Mwt.

REPORT OF ERROR IN ANALYSES

On May 24, 2007, in a conference call, the licensee informed the NRC that a computer code error was discovered in executing the ASTRUM methodology. In subsequent telephone calls with the NRC on June 28, July 2, July 10, and July 11, 2007, the licensee confirmed that the error was such that the regulation would require a reanalysis of BE-LBLOCA for Surry 1 and 2. Using an unreviewed version of ASTRUM, the licensee estimated that the PCT would be about 2095 °F.

On July 11, 2007, in a conference call, the licensee proposed to provide full corrected ASTRUM reanalyses of BE-LBLOCA prior to startup following its next refueling outage in 2008. The licensee proposed to use the 2095 °F estimated temperature as its interim estimated PCT for continued operation, per 10 CFR 50.46(a)(3)(ii).

In its July 31, 2007, letter, the licensee reported the details of the errors discussed during July 11, 2007, phone call, and documented its previously estimated results: 2095 °F PCT, 10.8% maximum local oxidation, and 0.32% core-wide oxidation. Further, the licensee committed to perform a full BE-LBLOCA reanalyses for Surry 1 and 2 and to submit the reanalyses results to the NRC by December 31, 2008.

The NRC staff finds that the licensee's proposed actions are acceptable, because they comply with the provisions of 10 CFR 50.46(a)(3)(ii). Because of this compliance with 10 CFR 50.46, the above proposed TS changes continue to be acceptable for implementation in the Surry 1 and 2 TSs and COLRs.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Virginia State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (71 FR 70564). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

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

7.0 REFERENCE

1. Surry Power Station, Updated Final Safety Analysis Report (UFSAR) Revision 38, dated 9/29/2006, Pages 14.1-1 and 14.1-2

Principal Contributor: Frank Orr

Date: September 6, 2007

cc:

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