June 17, 2008

Mr. David A. Christian
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 THE INCREASE OF MAXIMUM SERVICE WATER TEMPERATURE LIMIT FROM 95 °F TO 100 °F (TAC NOS. MD5877 AND MD5878)

Dear Mr. Christian:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 259 Renewed Facility Operating License No. DPR-32 and Amendment No. 259 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 June 25, 2007, as supplemented by letters dated November 14, 2007, January 10 and April 11, 2008.

These amendments increase the maximum service water temperature limit from 95 °F to 100 °F.

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. 259 to DPR-32
2. Amendment No. 259 to DPR-37
3. Safety Evaluation

cc w/encls: See next page

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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. 259
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 June 25, 2007, as supplemented by letters dated November 14, 2007, January 10 and April 11, 2008, 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. 259, 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 within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Melanie C. Wong, 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: June 17, 2008

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-281

SURRY POWER STATION, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 259
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 June 25, 2007, as supplemented by letters dated November 14, 2007, January 10 and April 11, 2008, 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. 259, 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 within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Melanie C. Wong, 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: June 17, 2008

ATTACHMENT

TO LICENSE AMENDMENT NO. 259

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

DOCKET NO. 50-280

AND

TO LICENSE AMENDMENT NO. 259

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
3.8-4	3.8-4
Figure 3.8-1	Figure 3.8-1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 259 TO

RENEWED FACILITY OPERATING LICENSE NO. DPR-32

AND

AMENDMENT NO. 259 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 June 25, 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML071840169), as supplemented by letters dated November 14, 2007 (ADAMS Accession No. ML073190271), January 10 (ADAMS Accession No. ML080160070) and April 11, 2008 (ADAMS Accession No. ML081020691), 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). The requested changes would increase the maximum service water (SW) temperature limit from 95 °F to 100 °F. The supplements dated November 14, 2007, January 10, and April 11, 2008, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on July 17, 2007 (72 FR 39084).

Unusually hot summer temperature peaks, combined with decreased river flows due to low rainfall in the upper James River basin, coupled with a seasonal increase in radiation heating of the lower Chesapeake Bay during the July and August time period, have caused SW temperatures to approach the current TS limit of 95 °F during each of the last 2 years. Based on these isolated peaks in SW temperature, the potential exists for SW temperature to exceed the TS limit in the future. The licensee proposes to increase the TS limit to 100 °F.

The SW temperature limit was changed from 92 °F to 95 °F in 1993 to address previously experienced extended hot weather, minimal rainfall, and low tide that caused the SW temperature to approach the 92 °F limit. The increase in the SW temperature limit was incorporated into the Surry 1 and 2 TSs by Amendment Nos. 183/183 dated September 7, 1993 for Surry 1 and 2, respectively. The existing temperature limit of 95 °F was set 2 °F warmer than the highest river water temperature on record at that time.

The SW system is a safety-related (SR) system that transfers heat from other SR systems and equipment and rejects it to the ultimate heat sink (UHS) (i.e., James River). The SR equipment served by the SW system includes:

- the recirculation spray (RS) system heat exchangers (RSHX)
- the chemical and volume control (CH) system charging pump intermediate seal cooler and lube oil coolers
- the main control room and emergency switchgear room air conditioning system chiller condensers
- the component cooling water (CC) System heat exchangers
- the cooling water to the water jacket of the diesel engines that power the emergency service water pumps

2.0 BACKGROUND

The James River, by way of the intake canal, is the source of water for the Surry 1 and 2 SW system and is the UHS for Surry 1 and 2. Water is pumped from the river to the intake canal by the circulating water (CW) system pumps. Three diesel-driven emergency service water (ESW) pumps are also provided to ensure that water can be supplied to the intake canal when power is not available to the CW pumps. Water in the intake canal gravity flows to the high-level intake structure for each unit and enters the CW system piping. Service water then branches from the CW system piping and flows to the various heat loads and services associated with Surry 1 and 2. Service water returns to the James River by way of a discharge tunnel from each unit, which empties into a common discharge canal. An SW system is provided for each unit. Portions of each unit's SW system's components are common to both units.

The intake canal is elevated and has an intrinsic storage capacity that provides a reservoir of water for the SW and CW systems. Surry 1 and 2 are located at the end of the intake canal approximately 1.5 miles downstream of the intake structure. Water from the intake canal is directed to Surry 1 and 2 by CW system lines that originate at the high-level intake structure. SW system piping originates from branches in the 96-inch CW lines upstream of the CW motor-operated-valves (MOVs) that supply water to the main condensers.

The SW system supplies cooling water through Surry 1 and 2 by way of several supply headers. Return headers collect the SW from the cooled components and subsystems and return the water to the James River via the discharge tunnel and discharge canal. The elevation difference between the intake canal and the discharge tunnel provides the motive force for flow of the service water to various loads. Various components are supplied directly by gravity flow, and other components are supplied via booster pumps.

3.0 REGULATORY EVALUATION

The regulatory requirements and the guidance upon which the NRC staff based its review of the effect on containment analysis due to a proposed increase in the SW temperature maximum limit from 95°F to 100°F are based on the following:

Appendix A to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, in particular,

(1) General Design Criterion (GDC 1) as it relates to structures and components being designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the importance of the safety function to be performed,

(2) GDC 2 as it relates to structures and components important to safety being designed to withstand the effects of earthquakes combined with the effects of normal or accident conditions,

(3) GDC 5 as it relates to the sharing of structures, systems, and components should not significantly impair the ability to perform their safety functions,

(4) GDC 16 as it relates to the containment and associated systems establishing a leak-tight barrier against the uncontrolled release of radioactivity to the environment and assuring that the containment design conditions important to safety are not exceeded for as long as the postulated accident requires,

(5) GDC 19 as it relates to the capability of the control room to remain functional to the degree that actions can be taken to operate the nuclear plant safely under normal conditions and to maintain it in a safe condition under accident conditions,

(6) GDC 38 as it relates to the containment heat removal system safety function which shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident (LOCA) and to maintain them at acceptably low levels,

(7) GDC 44 as it relates to cooling water systems transferring heat from structures, systems, and components important to safety to an ultimate heat sink under normal operating and accident conditions,

(8) GDC 50 as it relates to the containment heat removal system which shall be designed so that the containment structure and its internal compartments can accommodate without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any LOCA.

Some of the review areas are contained in the NRC Standard Review Plan (SRP), NUREG-0800, Chapter 6, Section 6.2.1, “Containment Functional Design”, Section 6.2.1.1.A, “PWR [Pressurized Water Reactors] Dry Containments, Including Subatmospheric Containments,” Section 6.2.2, ”Containment Heat Removal Systems,” Chapter 9, Section 9.4.1, “Control Room Area Ventilation System,” and Section 9.2.1, “Station Service Water System.”

The NRC staff’s review covers the structural integrity of the SW system piping and its associated supports. Technical areas covered by this review include the code allowable stresses within the SW piping system. The affected piping systems were designed in accordance with the rules of the American Society of Mechanical Engineers, *Boiler and Pressure Vessel Code* (ASME Code), Section III, and the American National Standards Institute (ANSI) Power Piping Code B31.1. The NRC staff’s review focused on verifying that the licensee has provided reasonable assurance that the affected piping and pipe supports due to the proposed change will remain structurally adequate to perform their intended function.

The NRC staff's review of the proposed change to the SW temperature focused on the impact the proposed increase in temperature will have on the ability of the SR structures, systems, and components (SSCs) to perform their safety functions. Acceptability of the proposed change is judged based upon continued conformance with the plant licensing basis as reflected primarily in Updated Final Safety Analysis Report (UFSAR) Section 9.9, "Service Water System," and other sections that are referred to in the Technical Evaluation Section (as applicable).

4.0 TECHNICAL EVALUATION

Surry 1 and 2 are three-loop Westinghouse pressurized water reactors (PWRs) with a subatmospheric containment design. The James River serves as the ultimate heat sink for removing the operating and decay heat produced by various Surry 1 and 2 plant components during normal operation, anticipated operational occurrences, and accidents. As described in the Surry 1 and 2 UFSAR Section 6.1, following a design-basis accident, the engineered safeguards system provided for performing the function of limiting the driving potential, including differential pressure and time duration, for leakage out of the containment structure, is the spray system for condensation of steam released inside the containment, the depressurization of the containment below atmospheric pressure, and means for maintaining the containment at subatmospheric conditions for an extended period of time. Two separate subsystems of the spray system (containment spray and recirculation spray) operate together to reduce the containment temperature, return the containment pressure to subatmospheric, and remove heat from the containment. The recirculation spray (RS) subsystem maintains the containment subatmospheric and transfers heat from the containment to the SW system. The proposed change to TS 3.8-4, to increase the maximum SW temperature limit from 95 °F to 100 °F affects the containment pressure and temperature analysis documented in UFSAR Section 5.4.2.

4.1 Containment Pressure and Temperature Response

In a previous license amendment request (LAR) dated January 31, 2006[1], approved by the NRC, the licensee revised the containment analysis by using the Generation of Thermal Hydraulic Information for Containments (GOTHIC) computer code version 7.2 methodology. The SW temperature used in this analysis was 95 °F. In the present LAR, the licensee states that the proposed change of increasing the TS maximum SW temperature limit from 95 °F to 100 °F does not affect the (a) LOCA peak containment pressure because these peaks occur before the RS subsystem actuation, (b) main steam line break analysis because it does not depend on the use of the RS subsystem, (c) net positive suction head available (NPSHA) for the low head safety injection (LHSI) pumps because their NPSHA is limiting at SW temperature lower than 70 °F due to a higher containment backpressure from warmer RS spray, and increases for SW temperatures higher than 70 °F, and (d) NPSHA for the RS pumps because their NPSHA is limiting at the minimum SW temperature due to a higher containment backpressure from warmer RS spray. Therefore a revised analysis for SW temperature of 100 °F is not required for these items. The NRC staff accepts the licensee's conclusions for these items. However the LOCA containment depressurization analysis is adversely affected because the higher SW temperature reduces the RS heat exchanger effectiveness, increases containment pressure and temperature and increases the containment depressurization time.

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[1] Letter from Leslie N. Hartz, Vice President- Nuclear Engineering, Virginia Electric and Power Company, to the NRC, "Surry Power Station Units 1 and 2, Proposed Technical Specification Change and Supporting Safety Analyses Revisions to Address Generic Safety Issue 191," dated January 31, 2006 (ADAMS Accession No. ML060370098).

4.2 LOCA Containment Depressurization Analysis

As per LAR dated January 31, 2006, the revised containment depressurization design criterion in conjunction with implementation of the alternative source term analysis requires that following a LOCA, the containment pressure should drop to less than 1.0 psig within 1 hour and should not exceed 1.0 psig for the interval from 1 to 4 hours following a LOCA, and beyond 4 hours the containment pressure should remain less than 0.0 psig, which terminates the leakage from the containment.

In order to demonstrate that the above requirements are met, the licensee performed a containment depressurization analysis to determine the depressurization peak pressure (DPP), depressurization pressure profile, and the containment depressurization time (CDT) using the GOTHIC methodology for a SW temperature of 100 °F. In these analyses, the licensee used the same RS pump start logic as in the LAR dated January 31, 2006. For the RS heat exchangers, the licensee assumed a 10-percent tube plugging and a tube fouling factor of 0.0005 ft^2-hr-°F, and states that the SW side (tube side) is kept in a clean dry condition and therefore these values are conservative. The licensee performed the analysis using two RS trains out of the four available to meet the containment depressurization requirement. The licensee also confirmed analytically that only one RS train is required after 24 hours from LOCA initiation to maintain the subatmospheric containment conditions for the proposed 100 °F SW limit. The NRC staff considers the licensee's assumptions acceptable because the licensee states that the SW side of the RS heat exchanger is kept clean and dry and also there is sufficient available heat exchanger reserve capacity.

For the DPP analysis, the licensee used conservative containment initial conditions in order to determine the depressurization pressure profile during the period from 1 hour to 4 hours from a LOCA. The analysis confirmed that for a minimum SW flow to each RS heat exchanger at a temperature of 100 °F, the containment was depressurized to less than 1.0 psig in 3080 seconds, and less than zero psig in 3432 seconds. The depressurization peak pressure (DPP) of 0.58 psig was reached at 5425 seconds. The licensee determined that the containment pressure stays permanently below 0 psig after 11,490 seconds (3.2 hours).

For the CDT analysis, the licensee used conservative containment initial conditions in order to determine the time when the containment pressure decreases below 1.0 psig. The analysis confirmed that for a minimum SW flow to each RS heat exchanger at a temperature of 100 °F, the containment was depressurized to less than 1.0 psig in 3149 seconds, and less than zero psig in 3420 seconds. The DPP of -0.78 psig was reached at 5145 seconds. The licensee determined that the containment pressure stays permanently below 0 psig after 3376 seconds (0.94 hours).

The licensee states that the temperature effect due to increase in the service temperature to 100 °F has no significant impact on the rooms where SW and CW lines are located. These rooms were evaluated for room temperatures at 100 °F or above. The licensee further states that short term increase of 2 °F in normal ambient temperatures in various areas of the plant will not have significant effect on the qualified life.

The licensee also states that the containment pressures and temperature for the LOCA containment depressurization analysis were confirmed to be bounded by the analyzed limits for the environmentally qualified (EQ) equipment inside the containment.

The NRC staff reviewed and evaluated the licensee's submittal, and considers the analyses acceptable because they meet the containment depressurization design criterion, and the limits for the EQ equipment have been confirmed to bound 1) the containment pressure and temperatures from the LOCA containment depressurization analyses, and 2) the room temperatures where SW and CW lines are located.

4.3 Main Control Room and Emergency Switchgear Air Conditioning System

The licensee states that the increase in SW temperature to 100 °F results in a small decrease in the capacity of chillers supplying cooling water to the main control room (MCR) and emergency switchgear room (ESGR) air conditioning system (ACS). The licensee evaluated the impact of the reduced chiller capacity on the MCR and ESGR space temperatures and confirmed that the ACS is able to maintain the space temperature within the design limits under maximum heat load conditions. The licensee also performed special tests for the MCR and ESGR ACS and confirmed this result.

4.4 SW Piping and Supports

The metallic (carbon steel) portions of the Surry 1 and 2 SW system piping were designed in accordance with the Power Piping Code ANSI B31.1, 1967 Edition. The fiberglass portions of the SW piping system were qualified in accordance with the criteria of the ASME Code Case N155-2. The licensee reviewed the revised design conditions of a higher SW temperature for impact on the existing design basis analyses for the affected portions of the SW piping systems. It was indicated that the fiberglass portions of the SW piping system were analyzed with additional scrutiny due to the fact that the fiberglass portions of the SW system had the most limiting stress margins of any of the piping systems at the plant when analyzed against the revised SW intake temperature.

In response to NRC staff Requests for Additional Information (RAIs), the licensee provided a quantitative stress summary of the fiberglass piping analysis which was performed to demonstrate compliance with the ASME Code Case N155-2 requirements, where the piping qualification is performed with combined primary and secondary stresses. The licensee's quantitative summary provided the code allowable stresses along with the calculated stresses for the fiberglass piping before and after the SW temperature increases produced from the analysis of the SW piping system. The results of this summary indicate that the increased SW intake temperature conditions proposed by the licensee are within ASME Code allowable values for the fiberglass piping. The NRC staff concurs with the licensee's assessment that the fiberglass portion of the SW piping is acceptable for operation at the proposed increased TS allowable limit of 100 °F for SW.

The licensee also indicated that the thermal expansion stresses for the metallic piping increase slightly. However, given the relatively low temperature range (95 °F – 100 °F) within which these stresses increase, the licensee deemed an analysis unnecessary and indicated that the existing analyses show sufficient stress margins to accommodate the increased thermal stresses. The NRC staff concurs with the licensee's assessment that the carbon steel portions of the SW piping system and the SW piping supports are acceptable for operation at the increased TS allowable limit of 100 °F for SW.

4.5 Generic Letter (GL) 96-06 Considerations

The licensee indicated in its June 25, 2007, submittal that the impact of the higher SW temperature limit on the analyses performed with respect to the issues outlined in GL 96-06 were considered. GL 96-06 required licensees to address water hammer and two-phase flow with respect to containment air cooler cooling water systems during accident conditions and also required licensees to address the susceptibility of thermal overpressurization of piping which penetrates containment. The licensee indicated that, with respect to the first concern of GL 96-06, the higher SW temperature limit would have no effect on the potential for voiding in the containment cooling systems for which SW is used given the current head tank elevation and the piping arrangement present for the system. In response to NRC staff RAIs, the licensee addressed the second concern of GL 96-06 in the context of the higher SW temperature limit. It was stated by the licensee that the GL 96-06 piping penetration analyses were performed for peak piping temperatures following a design basis accident which occur at approximately 10 minutes following the accident. Based on the fact that the only portion of the SW piping system applicable to these analyses is used for the recirculation spray heat exchangers, the licensee states that the GL 96-06 penetration analyses remain valid for the proposed SW temperature increase since the recirculation spray system is not used until after the 10-minute time period for the peak piping temperature has occurred. The NRC staff concurs with the licensee's assessment that the GL 96-06 evaluations remain valid for the increased TS SW temperature limit.

4.6 Component Cooling Water (CC) System

The CC system is an intermediate cooling system which serves both units. It transfers heat from CC heat exchangers (CCHXs) containing reactor coolant, other radioactive liquids, and other fluids to the SW water system. The CC system is designed to (1) provide cooling water for the removal of residual and sensible heat from the reactor coolant system during shutdown, cooldown, and startup, (2) cool the containment recirculation air coolers and reactor coolant pump motor coolers, 3) cool the letdown flow in the chemical and volume control system during power operation, and during residual heat removal for continued purification, (4) cool the reactor coolant pump seal water return flow, (5) provide cooling water for the neutron shield tank, and (6) provide cooling to dissipate heat from other reactor unit components.

The CC subsystem has four component cooling water pumps and four component cooling water heat exchangers. Each of the component cooling water heat exchangers is designed to remove during normal operation the entire heat load from one unit plus one half the heat load common to both units. Two pumps and two heat exchangers are normally operated during the removal of residual and sensible heat from one unit during cooldown. Failure of a single component may extend the time required for cooldown but does not affect the safe operation of the station.

The licensee stated that it is expected that the CCHX outlet temperature will remain at or below the normal design temperature of 105 ºF noted in UFSAR Table 9.4-1, "Component Cooling Water System Component Design Data." The licensee performed an evaluation that demonstrated that with: 1) two units at power, 2) normal CC heat loads, 3) minimal tube blockage, and 4) a normal SW flow of 9000 gpm per CCHX, the CC water outlet temperature is predicted to remain below the normal limit of 105 ºF. The UFSAR states in Section 9.4.3.1, "Component Cooling Water System Description," that the temperature immediately downstream of the heat exchangers is monitored in the control room. The licensee stated that the CCHX outlet temperature is expected to remain below the alarm value of 110 ºF for SW flows as low as 6500 GPM assuming tubesheet and tube blockage in minimal. The licensee also stated that with fully open SW inlet valves to the CCHXs, flows greater than 9000 gpm have been measured during previous tests with the CW intake canal (which supplies water to the SW system) at normal operating levels.

In evaluating the proposed increase in SW temperature to 100 ºF, the licensee constrained the CCHX outlet temperature to 120 ºF, which the licensee stated was the same value used in the previous evaluations in which the SW temperature limit was 95 ºF.

The licensee determined that the CCHXs were sensitive to two types of fouling: macro-fouling and micro-fouling. Macro-fouling is any major blockage occurring when biological growth begins to foul the tubesheet and tubes. When testing the CCHXs, a relationship for SW flow versus tubesheet pressure drop is used to evaluate the amount of macro-fouling in the CCHX. Since the SW supply to the CCHXs is gravity fed, as opposed to pump driven, any plugged tubes reduce the SW flow in direct proportion to the number of tubes (or area) lost to plugging. Micro-fouling was assumed to be at the design specification data sheet fouling values. A performance test was conducted to define the fouling that the chemically treated CCHXs could experience under expected operating conditions. The testing of the CCHXs demonstrated that, for micro-fouling, the CCHX design specification data sheet values of $R_i = 0.008$ and $R_o = 0.002$ can be used in the development of the CCHX test acceptance criteria.

A review of the heat loads on the CC system was performed to ensure that the CCHX acceptance criteria are based on the most recent plant system heat loads. The maximum expected heat load (worst case heat load) was determined to be for normal shutdown of two units following a loss of offsite power. This heat load was used as a constant heat load in the development of the new CCHX acceptance criteria for operation.

The licensee performed a calculation to develop curves showing the operable limits for the CCHXs for the SW flow expected during normal plant operations. The curves identify an operable range for each CCHX indicating the CCHXs are capable of removing the necessary heat load for a given SW flow. The curves also identify an alert condition that indicates the CCHX is capable of removing the required heat, but consideration must be given to CCHX cleaning. An inoperable region of the curves is identified where the CCHX has degraded below the minimum capability to remove the required heat load. The curves show the annubar differential pressure that measures the SW flow through the CCHX versus the tubesheet differential pressure that indicates the pressure drop across the tubesheet. The line designating the area of inoperability was developed by determining the extent of tube fouling (as indicated by tubesheet differential pressure) at various SW total flow values.

The licensee developed operability limit curves for the CCHXs for SW temperatures ranging from 60 ºF to 100 ºF in five degree increments. The operability limit curves for 95 °F and 100 °F were submitted to the NRC staff. The licensee committed to incorporating the operability limit curves into the station CCHX surveillance test to verify the ability of the CCHXs to adequately perform their required safety function at various SW temperatures.

The NRC staff identified two areas requiring additional information. First, the licensee stated that the CCHX outlet temperature was constrained to 120 ºF, which they stated was the same value used in the previous evaluations in which the SW temperature limit was 95 ºF. The NRC staff requested the technical basis for this statement. The licensee responded by stating that the CCHX outlet temperature was limited to 120 °F, consistent with the original Westinghouse functional requirements for balance of plant interface systems.

The NRC staff also requested the licensee to confirm that with an SW temperature of 100 ºF each CCHX is "…capable of removing half of the heat load occurring four hours after a shutdown of one unit under conditions representing the maximum allowable cooldown rate…" as stated in Section 9.4.1.1.1 of the UFSAR. The licensee responded that a special test was performed on the CCHX in February 2007 to determine heat exchanger fouling and tube-side differential pressure at various SW flows. From this data operability curves, discussed above, for different levels of tube plugging and SW temperatures were developed. Data from this testing was also used to analyze the cooldown scenarios. The calculations demonstrated that the CCHX would be capable of meeting cooldown requirements with an inlet SW temperature of 101 ºF for the required cooldown scenarios. These calculations also demonstrated that the cooling capability of a single CCHX exceeded one half of the projected heat load occurring 4 hours after a shutdown of one unit under conditions representing the maximum allowable cooldown rate.

The NRC staff assessed the licensee's statement that the expected CCHX outlet temperature would remain below 105 ºF with the SW temperature of 100 ºF. The NRC staff finds that there is reasonable assurance that the CCHX outlet temperature will remain at or below the design basis temperature of 105 ºF and if it does exceed 105 ºF the alarm at 110 ºF will alert the operators prior to reaching the 120 ºF CCHX outlet temperature consistent with the original functional requirements for balance of plant interface systems. Therefore, the NRC staff finds there is reasonable assurance that the CCW system will be capable of performing its safety-related functions with an increase in the SW temperature to 100 ºF.

4.7 Charging Pump Service Water System

A charging pump service water system for each reactor unit provides water to cool the charging pump intermediate seal coolers and the charging pump lubricating oil coolers.

Either of two 100%-capacity charging pump service water pumps delivers water from the service water system to the charging pump intermediate seal coolers and the charging pump lubricating oil coolers, thereby maintaining the charging pump lubricating oil and the component cooling water used to cool the charging pump mechanical seals at the proper temperature. To ensure that service water is continually available, one pump is in operation and the other on standby. The standby pump is automatically actuated on low pump discharge pressure to supply service water in the event of failure of the operating pump.

4.7.1 Charging Pump Lube Oil Coolers

The charging pump lube oil coolers are designed to remove heat from the charging pump bearings and gear drive under operating and accident conditions. There is one cooler per charging pump with oil on the shell side and service water on the tube side. The charging pump and gear box manufacturers have provided an operating limitation on bearing temperature at 185 ºF. To maintain the bearings below this temperature, the upper temperature limit for lube oil supplied to the charging pump is 160 ºF. Bearing temperatures are continuously checked via a trend recorder that is monitored by the plant computer system (PCS). The PCS alarms prior to the bearing temperature reaching 180 ºF. Also, plant operators log the oil temperature at the outlet of the cooler. If the oil temperature is above 110 ºF, the assigned system engineer is notified.

In addition, engineering evaluations have shown that these coolers have considerable design heat load margin such that micro-fouling would not be a concern. The coolers are flushed bi-weekly to remove any silt which may have accumulated in the head or tubes of a cooler serving a non-running charging pump.

The NRC staff finds that an increase in the maximum SW temperature from 95 ºF to 100 ºF will not adversely affect the ability to maintain the charging pump bearings below their limiting temperature of 185 ºF because of the continuous monitoring of the bearing temperature and the alarm prior to the bearings reaching a temperature of 180 ºF. Notification of the system engineer when the outlet temperature of the cooler is above 110 ºF in order to take appropriate action will also aid in maintaining the bearing temperature below the limit of 185ºF.

4.7.2 Charging Pump Intermediate Seal Cooler

The charging pump seal water is cooled by a closed system that is cooled by the SW via the charging pump intermediate seal coolers. There are two intermediate seal coolers per unit. The coolers have charging pump CC water on the shell side and SW on the tube side. The function of the charging pump intermediate seal coolers is to remove heat from the seal cooling loop, thereby maintaining the charging pump seals within their required temperature range. The charging pump seals have a maximum operating temperature of 250 ºF. The charging pump seal temperature is approximately 130 ºF during normal operation. During accident conditions, the seal temperature peaks at approximately 191 ºF.

The design heat transfer of the charging pump intermediate seal coolers is 760 Btu/min. A calculation was performed that demonstrated the design heat load could be removed even if the cooler is 40% fouled. The design margin of the coolers, as demonstrated by the calculation and operating experience, is very large. A previous engineering evaluation of these coolers concluded that performance testing is not even required to be performed because: 1) the thermal load under normal operating conditions is so low that meaningful data cannot be obtained, and 2) the heat exchanger specification data sheet shows a 2 ºF tube side temperature rise under design heat loads. Evaluation of heat exchanger performance for such a small change would not provide any meaningful information.

The NRC staff finds the licensee's determination that the seal temperature will peak at 191 ºF during accident conditions reasonable. The 191 ºF is well below the maximum operating temperature of 250 ºF. Therefore, an increase in the maximum SW temperature to 100 ºF is not expected to adversely affect the performance of the charging pump seals.

4.8 Emergency Service Water Pump (ESWP)

In the event of a loss of station power at the river intake, three diesel-driven, vertical ESWPs are provided for both units at the river intake structure to supply makeup water to the high-level canal. The SW system provides cooling water to the water jackets of the diesel engines that power the emergency cooling water pumps.

The licensee evaluated the effect of a SW temperature increase to 100 ºF on density, vapor pressure, and brake horsepower for the ESWPs and determined that the delivered flow is not affected significantly. Also, for a maximum SW temperature of 100 ºF, the diesel jacket outlet water temperature will not exceed 200 ºF, and a maximum jacket water temperature of 200 ºF remains below the current setting for the high jacket water temperature switches, which trip the diesels at approximately 210 ºF.

The NRC staff requested that the licensee provide the technical basis for the diesel jacket outlet water temperature not exceeding 200 °F. The licensee stated that this was discussed with a diesel field technical representative and documented in a plant calculation. The calculation documents that the jacket cooling system (heat exchanger and pump) is sized to limit the jacket cooling water temperature rise to 100 °F above the external cooling medium (i.e., SW) temperature. Thus, for a 101 °F SW inlet temperature, the jacket cooling water outlet temperature would be approximately 201 °F. Testing of the jacket cooling system was performed with the results indicated that the available SW cooling flow was greater than that required to limit the jacket cooling water temperature rise to 100 °F.

During operation, the jacket cooling water temperature is procedurally controlled within a range of

165 °F-185 °F by a thermostat in the diesel cooling loop and by the operator using a manual valve on the SW cooling loop. The jacket cooling water high temperature switches are set at 205 °F to maintain the maximum operating temperature of the jacket water below the 210 °F limit listed in the vendor technical manual.

Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," requested licensees to "Ensure by establishing a routine inspection and maintenance program for open-cycle service water system piping and components that corrosion, erosion, protective coating failure, silting, and biofouling cannot degrade the performance of the safety-related systems supplied by service water." In a letter dated October 2, 1991 (ADAMS Accession No. 9110080264, call NRC Public Document Room at 1-800-397-4209 or 301-415-4737), the licensee described the program as follows: "The ESW Pump Diesel Coolers' system inspection and maintenance include removal, inspection, and cleaning of the cooler core. Following maintenance, cooler SW temperature is verified to be within acceptance limits by adjusting the SW throttle valve."

Based on the information provided by the diesel field technical representative and documented in a plant calculation, and the testing of the jacket cooling system that indicated available SW cooling flow is greater than that required to limit the jacket cooling water temperature rise to 100 °F, and the routine inspection and maintenance program in response to GL 89-13, the NRC staff finds that the increase in the maximum SW temperature from 95 °F to 100 °F will not prevent the ESWPs from performing its safety-related functions.

4.9 SUMMARY

The licensee proposed to revise TS Section 3.8-4, by revising the maximum SW temperature from 95 °F to 100 °F. The NRC staff determined that the proposed changes meet the requirements of 10 CFR Part 50, Appendix A, (1) GDC 16, because the licensee showed that the containment design conditions important to safety are not exceeded during a postulated design basis LOCA, (2) GDC 19, because the licensee showed that the control room area space temperatures are maintained within design limits and actions can be taken to operate the nuclear plant safely under normal conditions and to maintain it in a safe condition under accident conditions, (3) GDC 38, because the licensee showed that the containment sprays would remove containment heat to reduce containment pressure and temperature rapidly, consistent with the functioning of the service water (SW) system and other associated systems, following a design-basis LOCA and would maintain them at acceptably low levels and (4) GDC 50, because the licensee showed that the containment heat removal system is designed so that the containment structure and its internal compartments can accommodate without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from the design basis LOCA. Therefore, the proposed license amendment is acceptable.

The NRC staff has reviewed the licensee's assessment of the impact of the proposed increase of the maximum TS allowable SW intake temperature with regard to the structural integrity of the affected piping and pipe supports at the proposed conditions. The NRC staff has also reviewed licensee's considerations of the increased SW temperature limit with respect to the evaluations performed in response to GL 96-06. On the basis of this review described above, the NRC staff finds that the proposed increase of the maximum TS allowable SW intake temperature will not have an adverse impact on the structural integrity of the SW piping system and supports and that the GL 96-06 evaluations remain unaffected by the proposed SW temperature limit increase from 95 ^{o}F to 100 ^{o}F.

Based on the results of the technical evaluation performed above in Sections 4.6, 4.7 and 4.8, the NRC staff finds that the proposed increase in the maximum SW temperature from 95 °F to 100 °F will not adversely impact the operation of the CC system, the charging pump service water system, and the ESWPs.

5.0 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.

6.0 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 (72 FR 39084). 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.

7.0 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.

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Date: June 17, 2008

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