August 31, 2007

Robert J. Duncan II, Vice President
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Code: Zone 1
New Hill, North Carolina 27562-0165

SUBJECT: SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1 - ISSUANCE OF AMENDMENT TO REVISE STEAM GENERATOR WATER LEVEL TRIP SETPOINT VALUES (TAC NO. MD2723)

Dear Mr. Duncan:

The Nuclear Regulatory Commission has issued Amendment No. 126 to Facility Operating License No. NPF-63 for the Shearon Harris Nuclear Power Plant, Unit 1. This amendment changes the technical specifications in response to your application dated August 2, 2006, as supplemented by letters dated March 9 and May 8, 2007.

The amendment increases the statistical summation error term “Z” and the allowable value for steam generator trip setpoints used in the reactor trip system and engineered safety feature actuation system.

A copy of the related safety evaluation is enclosed. Notice of Issuance will be included in the Commission's regular biweekly *Federal Register* notice.

Sincerely,

Marlayna Vaaler, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-400

Enclosures:
1. Amendment No. 126 to NPF-63
2. Safety Evaluation

cc w/enclosures: See next page

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Sincerely, ***/RA/***

Lisa M. Regner, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
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CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-400

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 126
License No. NPF-63

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

   A. The application for amendment by Carolina Power & Light Company, (the licensee), dated August 2, 2006, as supplemented by letters dated March 9 and May 8, 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 Title 10 of the *Code of Federal Regulations* (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 2.C.(2) of Facility Operating License No. NPF-63 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 126, are hereby incorporated into this license. Carolina Power & Light Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA Eva Brown for Thomas H. Boyce/

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to License No. NPF-63 and
the Technical Specifications

Date of Issuance: August 31, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 126

FACILITY OPERATING LICENSE NO. NPF-63

DOCKET NO. 50-400

Replace page 4 of Facility Operating License No. NPF-63 with the attached revised page 4.

Replace the following pages of the Appendix A Technical Specifications 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
2-5	2-5
3/4 3-32	3/4 3-32

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 126 TO FACILITY OPERATING LICENSE NO. NPF-63

CAROLINA POWER & LIGHT COMPANY

SHEARON HARRIS NUCLEAR POWER PLANT, UNIT 1

DOCKET NO. 50-400

1.0 INTRODUCTION

By application dated August 2, 2006, as supplemented by letters dated March 9, 2007, and May 8, 2007, Carolina Power and Light Company (CP&L or the licensee), which is now doing business as Prgress Energy, requested a change to the technical specifications (TSs) for the Shearon Harris Nuclear Power Plant, Unit 1 (HNP). The supplements dated March 9 and May 8, 2007, 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 February 27, 2007 (72 FR 8801).

The changes would revise the statistical summation error term "Z" and the allowable value for steam generator (SG) water level trip setpoints used in the reactor trip system (RTS) and engineered safety features actuation system (ESFAS) instrumentation. Specifically, the TS changes would modify Items 13 and 14 of TS Table 2.2-1, "Reactor Trip System Instrumentation Trip Setpoints," and Items 5.b and 6.c of TS Table 3.3-4, "Engineered Safety Features Actuation System Instrumentation Trip Setpoints."

The changes increase the "Z" term for:

1. the SG water level low-low RTS trip setpoint;
2. the SG water level low RTS trip setpoint;
3. the SGWL high-high turbine trip and feedwater (FW) isolation ESFAS trip setpoint; and
4. the SGWL low-low auxiliary FW (AFW) actuation ESFAS trip setpoint.

The changes also increase the allowable value (AV) for the SG water level low RTS trip setpoint.

The proposed TS changes are necessary to address generic issues involving SG water level measurement uncertainty considerations associated with Westinghouse-designed SGs.

2.0 REGULATORY EVALUATION

The NRC regulatory requirements related to the content of the TSs are set forth in Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.36, “Technical Specifications,” which requires that the TSs include limiting safety system settings. This regulation requires, in part, that “where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting must be chosen so that automatic protective action will correct the abnormal situation before a safety limit is exceeded.” Accordingly, the limits for instrument channels that initiate protective functions must be included in the TSs.

In accordance with General Design Criterion (GDC) 20, “Protection System Functions,” of Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, the protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. Adherence to acceptable fuel design limits is called for in GDC 10, “Reactor Design;” these limits are specified in each plant’s Core Operating Limits Report (COLR) and maintained as a part of the Administrative TSs.

Regulatory Guide (RG) 1.105, Revision 3, “Setpoints for Safety-Related Instrumentation,” describes a method acceptable to the NRC staff for complying with the NRC regulations for assuring that setpoints for safety-related instrumentation are initially within and remain within the TS limits.

3.0 TECHNICAL EVALUATION

3.1 Introduction

3.1.1 Background

The NRC issued Information Notice (IN) 2002-10 on March 7, 2002, to alert holders of operating licenses to the potential for non-conservative setpoints of the SG water level. The IN was issued as a result of a February 9, 2002, occurrence at Diablo Canyon Power Plant, Unit 2, where the SG water level narrow range (NR) instrumentation did not respond as expected to initiate an automatic reactor trip and AFW system actuation on the SG water level low-low signal during a plant trip. This event prompted Westinghouse, the SG manufacturer, to issue various Nuclear Safety Advisory Letters (NSALs).

As discussed in NSAL-02-3 and its revision, Westinghouse attributed the water level uncertainties mainly to a differential pressure (ΔP), previously unaccounted for, created by steam flow past the mid-deck plate in the moisture separator section of the SG. Westinghouse-designed SGs incorporate a mid-deck plate at the top of the primary separator assembly between the upper and lower taps used for the SG water level NR instruments. The installation of the mid-deck plate is to reduce moisture carryover. When some of the steam flows through the separator downcomer, instead of the primary separator orifice, this steam with some entrained moisture will flow upwards through the flow area in the mid-deck plate, creating a ΔP. The mid-deck plate ΔP, which is a function of steam flow, causes the SG water level NR instrumentation to read higher than the actual water level, and adversely affects the SG water level low-low trip with an uncertainty bias in a non-conservative direction. Therefore, the SG water level instrumentation without accounting for this ΔP phenomenon could be non-conservative during certain transients.

NSAL-02-4 dealt with uncertainties in the measurement created because the void content of the two-phase mixture above the mid-deck plate is not reflected in the calculation. The uncertainties may adversely affect the SG water level high-high trip signal for actuation of turbine trip and FW system isolation in a non-conservative direction.

NSAL-03-09 indicated that Westinghouse had developed a program for the Westinghouse Owners Group that evaluates the effects on the SG water level control system uncertainties from various items. These items include the mid-deck plate, FW ring and FW ring supports, lower-deck plate supports, non-recoverable losses due to carryunder, decrease in subcooling due to carryunder, as well as transient conditions due to events such as the loss of normal FW, or a steamline break outside containment. Under the program, Westinghouse evaluated the design features of Westinghouse-designed SGs and other phenomena associated with Westinghouse SGs as they affect uncertainties in terms of the SG water level control system, and the SG water level low-low and high-high trip functions.

To address generic issues involving SG water level measurement uncertainties associated with Westinghouse-designed SGs, the licensee proposed TS changes to revise the setpoint values for the RTS and ESFAS trips summarized in the table below. The proposed changes would revise TS Section 2.2.1 (Table 2.2-1, items 13 and 14) and TS Section 3/4.3.2 (Table 3.3-4, items 5.b and 6.c). Specifically, the changes increase the minimum AV for the SG water level low RTS trip setpoint from 23.5 percent NR span to 24.05 percent NR span, and also change the statistical summation error term “Z” as summarized in the table below.

Trip Setpoint	**current value**	**new value**
SG water level low-low RTS trip	16.85 percent	17.45 percent
SG water level low RTS trip	5.35 percent	5.95 percent
SG water level high-high turbine trip and FW isolation ESFAS trip	8.05 percent	8.15 percent
SG water level low-low AFW actuation ESFAS trip	16.85 percent	17.45 percent

The NRC staff evaluated the licensee’s proposed TS changes in comparison to the regulatory requirements and the guidance discussed in Section 2.0 of this Safety Evaluation, and verified that the licensee had appropriately addressed the issues discussed in the IN and NSALs.

3.1.2 SG Water Level Trip Functions

SG water level low-low channels are part of the RTS and ESFAS. They are designed to trip the reactor (as an RTS function) and start the AFW pumps (as an ESFAS function) for protection of the reactor core from a loss of heat sink in the event of a sustained steam and FW flow mismatch.

SG water level low channels are part of the RTS which is designed to actuate a reactor trip to protect against a loss of heat sink. The SG water level low RTS trip setpoint is not classified as a limiting safety system setting because this setpoint is not assumed to occur in any of the transient or accident analysis; therefore, this setpoint is not safety limit related. To ensure that the associated instrument channel is capable of performing its specified function in accordance with applicable design requirements and associated analysis, the licensee performs periodic surveillance requirements that are defined in the TSs.

SG water level high-high channels are part of the ESFAS to mitigate an excessive FW flow event. The trip is used to prevent the SGs from overfilling with water and passing water into steam piping which could overload the steam piping support design. Initiation of a SG water level high-high signal trips the main turbine and isolates the AFW system.

3.1.3 Setpoint Methodology

To determine baseline instrumentation setpoints, HNP used the methodology described in ISA-S67, “Methodologies for the Determination of Setpoints for Nuclear Safety-Related Instrumentation,” which was endorsed by the NRC in RG 1.105. Certain SG level setpoint uncertainties were addressed by Attachment 5 to the licensee’s August 2, 2006, submittal. The attachment documents a revision to the licensee’s calculation for determining the total loop uncertainty (referred to as ‘channel statistical allowance’ by Westinghouse), trip setpoints, “Z” terms, AVs, and other values. The incorporation of these new values is consistent with the approach outlined by the Westinghouse Owners Group in the documents described in Section 3.1.1 of this Safety Evaluation to address measurement uncertainties, and is therefore acceptable to the staff.

The SG level setpoint uncertainties may be caused by several process effect terms that were not previously accounted for in many plants with Westinghouse-designed SGs. In its new setpoint calculations, the licensee considered the following process measurement accuracy items: (1) process pressure variations; (2) reference leg temperature variations; (3) fluid velocity effects; (4) downcomer subcooling effects; (5) mid-deck plate ΔP effects; (6) intermediate deck plate ΔP effects; (7) feedline ΔP effects; (8) lower deck plate and deck support ΔP effects, and (9) non-recoverable losses due to carryunder into the lower downcomer.

The calculation of the “Z” term is composed of these process effects, adverse containment environmental effects, and instrumentation loop uncertainty. The process measurement accuracies and environment effects were combined algebraically. The instrumentation loop uncertainties address the accuracy of instruments, such as transmitter pressure and temperature effects, and instrument temperature effects. The licensee statistically combined these independent and random accuracies using the square-root-of-the-sum-of squares technique.

The AV is the limiting value that the trip setpoint can have when tested. The AV is determined by subtracting the instrument uncertainty allowances from the nominal trip setpoint (NTS). Instrument uncertainty allowances are composed of: (1) instrument calibration; (2) instrument drift; and (3) instrument uncertainties during normal operation.

The licensee’s maintenance surveillance test procedures require that a trip setpoint be within the acceptable as-left tolerance after completion of the periodic surveillance. When a setpoint is found outside the acceptable as-left tolerance, but inside the acceptable as-found tolerance, the licensee's procedures require that the setpoint be adjusted to within the acceptable as-left tolerance per the surveillance requirements.

When a setpoint is found to be outside the acceptable as-found tolerance, also called the AV for this particular trip setpoint, then the channel is declared inoperable and evaluated to verify that it is functioning as required before returning the channel to an operable status. Procedures require documentation of this condition in the Corrective Action Program. The licensee's Corrective Action Program provides the requirements for identifying the cause, correcting, and trending adverse conditions.

Per the HNP procedure, an engineering change is performed anytime that an instrument uncertainty calculation, individual scaling calculation, or maintenance surveillance test procedure is revised; this process is controlled under 10 CFR 50.59, "Changes, Tests, and Experiments." The setpoint calculations cannot be changed without an engineering evaluation.

Based on the licensee's:

1) adequate incorporation into the setpoint calculations of factors that may promulgate SG level setpoint uncertainties;
2) appropriate use of maintenance surveillance testing and Corrective Action Program procedures to address as-left and as-found tolerances; and
3) proper use of the engineering change and 10 CFR 50.59 processes to ensure setpoint calculations are not changed without proper analysis,

the NRC staff concluded that the licensee's proposed changes to address measurement uncertainty in the calculations for SG water level trip setpoints are acceptable.

3.2 <u>SG Water Level Low-Low Reactor Trip and AFW Actuation Setpoint</u>
(Item 13 of TS Table 2.2-1 and Item 6.c of TS Table 3.3-4, respectively)

In regards to the SG Water Level Low-Low Reactor Trip and AFW Actuation Setpoint, the NRC staff found that the licensee first appropriately identified events that credited a particular SG water level trip function for consequence mitigation. The licensee determined the physical effects of the transient conditions with respect to process measurement accuracy terms and the overall impact of SG water level setpoint uncertainties for those affected events.

The licensee used this information to calculate the SG water level low-low uncertainties based on the transient conditions involved in a single loop loss of normal FW and small/intermediate feedline break outside containment. The licensee calculated a value for "Z" of 17.45 percent and an AV of 23.5 based on the limiting case, the FW line break event. The NRC staff concluded that the proposed values of "Z'" and AV were calculated based on the approved methods discussed in section 3.1.3 of this SE, and are therefore acceptable.

Furthermore, appropriately to address these changes, the calculated value for "Z" of 17.45 percent was correctly reflected in item 13 of revised TS Table 2.2-1 and item 6.c of revised TS Table 3.3-4 for the SG water level low-low RTS, and AFW actuation setpoint, respectively; and the calculated AV value of 23.5 percent was correctly reflected in item 13 of revised

TS Table 2.2-1 and item 6.c of revised TS Table 3.3-4, and remained unchanged for the SG water level low-low reactor trip setpoint and AFW actuation setpoints.

3.3 SG Water Level Low Trip Setpoint
(Item 14 of TS Table 2.2-1)

For the SG Water Level Low Trip Setpoint, the licensee presented the calculated "Z" value of 5.95 percent and AV of 24.05 percent in Attachment 5 to the August 2, 2006, submittal. In addition, the calculated result for total rack uncertainty allowance was found to be 1.5 percent based on the acceptable methodology discussed in section 3.1.3 of this SE.

The value of 5.95 percent for "Z" was also calculated using the acceptable methodology discussed in section 3.1.3 of this SE. Per these methods, an uncertainty of 0.95 percent was calculated by subtracting the "Z" term and the total sensor uncertainty from the total allowance. Due to the "Z" term change from 5.35 to 5.95 percent, the 0.95 percent uncertainty was less than the total rack uncertainty of 1.5 percent. Therefore, a new AV was calculated by subtracting 0.95 percent from the NTS (i.e., 25.0 - 0.95 = 24.05 percent). The NTS and acceptable as-left tolerance were not changed.

The NRC staff found that: (1) the values of "Z'" and AV were calculated based on the acceptable methods discussed in section 3.1.3 of this SE; and (2) the calculated values were correctly reflected in item 14 of revised TS Table 2.2-1 for the SG water level low trip setpoint. Therefore, the NRC staff concluded that the proposed values for "Z" of 5.95 percent and AV of 24.04 percent are acceptable.

3.4 SG Level High-High Trip Setpoint
(Item 5.b of TS Table 3.3-4)

NSAL-02-4 indicated that the void content of the two-phase mixture above the mid-deck plate was not reflected in the SG level setpoint calculations. This would result in a non-conservative actuation of the SG water level high-high trip signal. In a May 8, 2007, letter, the licensee provided its calculation of the NTS with inclusion of the effects of void content above the mid-deck plate.

For the determination of the SG water level high-high NTS, the safety analysis limit (SAL) used was 91.2 percent and the total allowance was 13.2 percent of the SG NR span for HNP. The SAL of 91.2 percent of the span was bounded by the maximum reliable indicated level of 96.5 percent, which considered the effects of the void content. In addition, the calculated SG water level high-high NTS of 78 percent NR span was calculated by subtracting the total allowance of 13.2 percent from the void-content adjusted SAL of 91.2 percent. This lower NTS (as compared to the SAL) would result in an earlier turbine trip and FW isolation during an excessive FW flow event, and thus, would provide a greater margin to protect the SG from overfilling with water.

Using the same method discussed in Section 3.1.3 above, the licensee presented the calculated "Z" value of 8.15 percent in Attachment 5 to the August 2, 2006, submittal. The AV of 79.5 percent was calculated as described in Section 3.3 by adding the total rack uncertainty allowance of 1.5 percent to the NTS of 78 percent, which did not change.

The NRC staff found that: (1) the values of "Z'" and AV were calculated based on the acceptable methods discussed in section 3.1.3 of this SE; and (2) the calculated values were correctly reflected in item 5.b of revised TS Table 3.3-4 for the SG water level high-high trip setpoint. Therefore, the NRC staff concluded that the proposed values for "Z" of 8.15 percent and AV of 79.5 percent are acceptable.

3.5 Summary

The NRC staff reviewed the licensee's proposed TS changes to Tables 2.2-1 and 3.3-4 to increase:

(1) the SG water level low-low RTS trip setpoint value of the statistical summation error term "Z" from 16.85 percent to 17.45 percent in Item 13 of Table 2.2-1;
(2) the SG water level low-low AFW actuation setpoint value of "Z" from 16.85 percent to 17.45 percent in Item 6.c of Table 3.3-4;
(3) the SG water level low trip setpoint values of "Z" and AV from 5.35 percent and 23.5 percent to 5.95 percent and 24.05 percent, respectively, in Item 14 of Table 2.2-1; and
(4) the SG water level high-high trip setpoint values of "Z" from 8.05 percent to 8.15 percent in Item 5.b of Table 3.3-4.

Based on the evaluation discussed in Section 3.0, the NRC staff found that the proposed TSs appropriately address the issues specified in the NSALs, and the proposed values of the "Z" terms and AVs were based on acceptable setpoint methodologies. Therefore, the staff concluded that the licensee's proposed TS changes comply with 10 CFR 50.36 requirements and are acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement 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 amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (72 FR 8801). Accordingly, the amendment meets 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 amendment.

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.

Principal Contributors: Sang Rhow
Summer Sun

Date: August 31, 2007

cc:
David T. Conley
Associate General Counsel II -
Legal Department
Progress Energy Service Company, LLC
Post Office Box 1551
Raleigh, North Carolina 27602-1551

Resident Inspector/ Harris NPS
c/o U. S. Nuclear Regulatory Commission
5421 Shearon Harris Road
New Hill, North Carolina 27562-9998

Ms. Margaret A. Force
Assistant Attorney General
State of North Carolina
Post Office Box 629
Raleigh, North Carolina 27602

Public Service Commission
State of South Carolina
Post Office Drawer 11649
Columbia, South Carolina 29211

Ms. Beverly Hall, Section Chief
Division of Radiation Protection
N.C. Department of Environment
and Natural Resources
3825 Barrett Drive
Raleigh, North Carolina 27609-7721

Mr. J. Paul Fulford
Manager, Performance Evaluation and
Regulatory Affairs PEB 5
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602-1551

Mr. Chris L. Burton
Director of Site Operations
and Acting Plant General Manager
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
Post Office Box 165, Mail Zone 1
New Hill, North Carolina 27562-0165

Mr. Robert P. Gruber
Executive Director
Public Staff NCUC
4326 Mail Service Center
Raleigh, North Carolina 27699-4326

Chairman of the North Carolina
Utilities Commission
Post Office Box 29510
Raleigh, North Carolina 27626-0510

Mr. Tony Gurley, Chair
Board of County Commissioners
of Wake County
P. O. Box 550
Raleigh, North Carolina 27602

Mr. Carl Thompson, Chair
Board of County Commissioners
of Chatham County
P. O. Box 87
Pittsboro, North Carolina 27312

Mr. Thomas J. Natale, Manager
Support Services
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
P. O. Box 165, Mail Zone 1
New Hill, North Carolina 27562-0165

Mr. David H. Corlett, Supervisor
Licensing/Regulatory Programs
Shearon Harris Nuclear Power Plant
Carolina Power & Light Company
P. O. Box 165, Mail Zone 1
New Hill, NC 27562-0165

Mr. John H. O'Neill, Jr.
Pillsbury Winthrop Shaw Pittman, LLP
2300 N Street NW.
Washington, DC 20037-1128