VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 5, 1995

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

Serial No.	95-036E
NL&P/CGL	R0
Docket Nos.	50-280
	50-281
License Nos.	DPR-32
	DPR-37

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNITS 1 AND 2
CORE UPRATE - REACTOR SYSTEMS BRANCH INQUIRY

The Surry Core Uprate Technical Specification change request, including the Surry Core Uprating Licensing Report, was submitted for NRC review by an August 30, 1994 letter (Serial No. 94-509). On March 13, 1995, a conference call was held between Virginia Power personnel, Bart Buckley (NRC Project Manager), and Frank Orr (NRC Reactor Systems Branch (SRXB)). The items discussed during the call included:

- the basis for sizing of the pressurizer safety valves/reactor coolant system overpressure protection,
- the locked rotor analysis/2 second delay in RCP trip timing,
- RHR cooldown design basis/core uprate impact, and
- LOCA analyses/small break ECCS model issues.

As a follow-up to the conference call, discussion of the following two items is being provided:

- the existing RCS overpressure analysis basis and its continued applicability (Attachment 1) and
- the RCP trip assumption in the locked rotor incident analysis (Attachment 2).

If further information is required, please contact us.

Very truly yours,

James P. O'Hanlon

James P. O'Hanlon
Senior Vice President - Nuclear

Attachments

110017

A001
1/1

cc: U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, N. W.
Suite 2900
Atlanta, Georgia 30323

Mr. M. W. Branch
NRC Senior Resident Inspector
Surry Power Station

Commissioner
Department of Radiological Health
Room 104A
1500 East Main Street
Richmond, Virginia 23219

ATTACHMENT 1

Reactor Systems Branch (SRXB) Inquiry
Regarding Existing RCS Overpressure Analysis Basis -
Surry Power Station Units 1 and 2
Proposed Technical Specification Changes
to Accommodate Core Uprating

As a follow-up to the March 13, 1995 Virginia Power/NRC conference call, the following information is provided to (1) document the existing RCS overpressure analysis for Surry and (2) verify the continued applicability of the existing analysis for operation at uprated conditions:

WCAP-7769, Revision 1 Overpressure Analysis

The Westinghouse overpressure topical report (WCAP-7769, Revision 1, "Overpressure Protection for Westinghouse Pressurized Water Reactors," June 1972) includes a presentation of the capability of the Westinghouse-designed pressurized water reactor (PWR) to adequately relieve steam on the primary and secondary sides during accident conditions (anticipated operational occurrences - Condition II events). Section 2 of WCAP-7769, Revision 1, addresses the procedure by which the pressurizer safety valves are sized such that reactor coolant system (RCS) overpressurization does not occur. The text of Section 2 notes that representative 2-, 3-, and 4-loop plants were included in the pressurizer safety valve sizing analysis and, for each of the plants noted in Table 2-1 of WCAP-7769, Revision 1, conservative initial conditions were assumed including a power level at the Engineered Safeguards Design Rating (uprated power level). Surry was included in this analysis as a representative 3-loop plant. Based on the criterion describing the ratio of the capacity to relieve steam to the peak surge rate in the pressurizer, the Westinghouse 4-loop PWR designs have the least pressurizer safety valve overcapacity for steam relief. The RCS pressure relief capability of 2- and 3-loop plants is greater than that for a 4-loop plant which is presented in WCAP-7769, Revision 1. Thus, the conclusions delineated in Section 2 of the topical report are applicable and valid for designs described in WCAP-7769, Revision 1, including the Surry units at uprated conditions.

Section 2 of WCAP-7769, Revision 1, concludes by illustrating the capability of the Westinghouse 4-loop PWR to adequately relieve steam through the pressurizer and steam generator safety valves under loss of load transient conditions. The representative conclusions are also applicable to Westinghouse 2- and 3-loop plants. The conclusion is that overpressure protection is provided by the pressurizer and steam generator safety valves in conjunction with the reactor protection system. In other words, a reactor trip is required to prevent RCS overpressurization. To illustrate this, an analysis was performed taking credit for each of four different reactor trip functions (high pressurizer pressure, overtemperature ΔT, low feedwater flow, and low-low steam generator water level) such that safety valve relief capacities could be confirmed. The results shown in Figure 2-2 and Table 2-3 of WCAP-7769, Revision 1 indicate that any one of these functions is adequate to trip the reactor and prevent an overpressure condition in the RCS. This does not imply that multiple single failures have to be assumed for the protection function to be accomplished or that the first protection function is defeated. The Surry-specific analysis documented in the Surry Core Uprating Licensing Report and discussed below credits the first reactor trip function (excluding a direct reactor trip on turbine trip) to provide adequate protection against RCS overpressurization following a loss of load transient. Surry-specific analyses to justify each of the multiple trip functions as being adequate for overpressure protection are not required to confirm the capacity of the safety valves. The conservative analyses documented in Section 2 of WCAP-7769, Revision 1, show that safety valve sizing on Westinghouse PWRs is adequate to prevent RCS overpressurization.

Complete Loss of External Electrical Load in the Surry Core Uprating Licensing Report

The limiting RCS overpressure event, the Complete Loss of External Electrical Load, is documented in Section 3.5.8 of the Surry Core Uprating Licensing Report. The attached Table 1-1 contains the key assumptions, plant conditions, and features modeled in the analysis, which was performed by Virginia Power using the RETRAN model (VEP-FRD-41A, "Reactor System Transient Analysis Using the RETRAN Computer Code," Virginia Electric and Power Company, May 1985). The analysis assumes operation at 102% of the uprated core power of 2546 MWt and core characteristics applicable to beginning of core life (BOC).

Because of the assumed positive moderator coefficient, the nuclear power increases from its initial value and reaches a peak of 105% at 9.0 seconds. Reactor vessel inlet temperature and pressurizer liquid volume also increase, reaching peak values of 577°F (22.8 seconds) and 896 ft^3 (13.6 seconds), respectively. A reactor trip occurs at 8.76 seconds on a high pressurizer pressure signal. The peak RCS pressure of 2745 psia occurs in the cold leg at 10.2 seconds. The peak flowrate through the pressurizer safety valves occurs at 10.0 seconds and corresponds to 94% of the total valve design capacity. The peak steam generator safety valve flowrate is 42% of the total valve design capacity, occurring at 18.8 seconds.

Table 1-1

Key Assumptions and Plant Model Features
Complete Loss of External Electrical Load Analysis
Surry Unit 1 and 2 - 2546 MWt Core Rated Power

<u>Initial Conditions</u>	
Core Power, MWt	102% of 2546 MWt
Vessel Average Temperature, °F	573.0 + 4°F
Vessel Inlet Temperature, °F	540.4 + 4°F
Pressurizer Pressure, psia	2250 - 30 psia
Main Steam Flow (at 2546 MWt), lbm/hr	11.23×10^6
<u>Core Characteristics</u>	
Moderator Temperature Coefficient, pcm/°F (most positive value)	+6 pcm/°F (0%-50% RTP) ramp to 0 pcm/°F (100% RTP)
Doppler Temperature Coefficient, pcm/°F (least negative value)	-1.0 pcm/°F
Rod Control System	manual mode
<u>Equipment Characteristics</u>	
Pressurizer Safety Valve Configuration	water-filled loop seals
Pressurizer Safety Valve capacity, lbm/hr (per valve)	293330
Pressurizer Safety Valve Lift Model	WCAP-12910, Rev. 1-A model with 1% setpoint shift
Pressurizer Spray	disabled
Pressurizer PORVs	disabled
Steam Generator PORVs	disabled
Steam Dump Valves	disabled

ATTACHMENT 2

Reactor Systems Branch (SRXB) Inquiry
Regarding Locked Rotor Analysis/RCP Trip Assumption - Surry Power Station Units 1 and 2
Proposed Technical Specification Changes to Accommodate Core Uprating

During the March 13, 1995 Virginia Power/NRC conference call, the locked rotor accident analysis was discussed. The following information is provided regarding the RCP trip assumption in the locked rotor accident analysis:

The locked rotor departure from nucleate boiling ratio (DNBR) analysis (which supports the radiological consequences analysis) is described in Section 3.5.7 of the Surry Core Uprating Licensing Report. In this analysis, the unaffected reactor coolant pumps (RCPs) were assumed to trip 2 seconds after generation of the reactor trip signal on low coolant flow. The basis for this assumption is discussed in the following paragraphs.

The trip of the RCPs in the unaffected loops is intended to simulate a localized disruption of power supply or loss of offsite power associated with the locked rotor incident or its initiating event. This power loss is assumed to occur 2 seconds after reactor trip on low loop coolant flow, even though no specific mechanism for such a power loss has been postulated. This assumption is consistent with that made in the uprating analysis for the North Anna locked rotor event. This analysis, which is described in Section 15.4.4 of the North Anna UFSAR, was approved by the NRC by the August 25, 1986 issuance of the Core Uprate License Amendments 84 and 71 for North Anna Units 1 and 2, respectively.

Offsite power system stability studies reported in Section 8.2.2 of the North Anna UFSAR conclude that offsite power sources will remain available for the loss of either or both units or loss of any other units on the Virginia Power system. In applying the RCP trip delay assumption, the reliability of the Surry electrical distribution system has been acknowledged. In the NRC's NUREG-1150 probabilistic risk analysis, it was documented that there have been no loss of offsite power events at Surry Power Station in comparison with 63 events at other domestic nuclear power plants. The Surry locked rotor incident analysis has, thus, accounted for the extremely unlikely occurrence of loss of offsite power in a manner which is consistent with the highly reliable design of the plant's electrical distribution system.