VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

May 24, 1979

Mr. Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Serial No. 220/040279
PSE&C/CMRjr:vml:wang
Docket Nos.: 50-280
50-281

License Nos.: DPR-32
DPR-37

Dear Mr. Stello:

Your letter of April 2, 1979 requested certain information in connection with the March 13, 1979 Order to Show Cause for Surry Power Station Units 1 and 2. Specifically, three enclosures were attached to your letter delineating information which you must have prior to start-up of the Units. The purpose of this letter is to provide our response to your letter.

Information regarding each of the referenced Enclosures is as follows:

Enclosure 1 - Four items identified in your letter required resolution. Our response is included as an Attachment to this letter with each of the four items addressed as requested. The Attachment is entitled "Response to Enclosure 1 of NRC Letter Dated April 2, 1979." It should be noted that a full response to Item 3 regarding IE Bulletin 79-02 will be provided at a later date. A complete status and schedule together with other information required by the Bulletin will be provided on or about June 1, 1979.

Enclosure 2 - Enclosure 2 of your April 2 letter requested specific information with regard to each pipe stress problem. Each Reanalysis Information Package is being developed in accordance with your instructions and will be maintained by Stone & Webster for Vepco. It is available to the NRC staff on a continuing basis.

Enclosure 3 - The verification of the piping computer codes used by Stone & Webster in the analysis and reanalysis has been discussed in "Plan for Verification of Dynamic Analysis Codes," April 4, 1979, revised April 6, 1979; Status Report on "Plan for Verification of Dynamic Analysis Codes," dated April 13, 1979; Status Report on "Plan for Verification of Dynamic Analysis Codes," dated April 27, 1979;

Add W. Russell

A001
S 1/1

7905250132

letter to Mr. Victor Stello, Jr., dated May 9, 1979; letter to Mr. Victor Stello, Jr., dated May 11, 1979; letter to Mr. Harold Denton dated May 14, 1979. This work has been carried out in conjunction with Brookhaven National Laboratory as directed by the NRC. The Plan and Status Reports referenced above have been submitted separately by Stone & Webster on April 6, 13, and 27, 1979.

If you should have any questions, please contact us.

Very truly yours,

W. C. Spencer

W. C. Spencer
Vice President
Power Station Engineering
and Construction Services

Attachment

ATTACHMENT

RESPONSE TO ENCLOSURE 1 OF NRC LETTER
DATED APRIL 2, 1979

May 24, 1979

SURRY POWER STATIONS UNITS 1 & 2

VIRGINIA ELECTRIC AND POWER COMPANY

Enclosure I

Item 1

Those safety systems, or portions thereof, for which intramodal loads were combined algebraically utilizing the SHOCK II computer program are listed in Table 1 of Item 4 (this table applies to Unit 1 only). This table also lists systems originally analyzed with SHOCK 0/SHOCK I computer codes. SHOCK 0/I does not combine inertial forces algebraically at the intramodal level.

Reanalysis efforts are being concentrated on safety systems which were originally analyzed by the SHOCK II code and encompass approximately 70 separate computer problems. Those problems which were originally analyzed using SHOCK 0 or SHOCK I will be reanalyzed using the NUPIPE computer code. Although reanalysis of these problems is not part of the March 13, 1979 Order to Show Cause and is, therefore, not required prior to start up, we believe it to be expedient to perform these analyses to have all affected systems on a common computer code. We do not consider completion of the reanalysis as a requirement for start up.

For specific information concerning computer code listings, please refer to Stone & Webster's letter of April 6, 1979 to the NRC; Operating License DPR-32, 36, 37, 59, and 66, entitled, "Application for Withholding Proprietary Information from Public Disclosure."

Item 2

Table II of Item 4 lists the systems, or portions thereof, which were not subjected to a computer analysis.

Surry Power Stations Units 1 & 2 were designed to the ANSI B31.1 Code for Power Piping. The Code, at that time, essentially required the designer to consider the effects of earthquake without specifying methods or guidelines for detailed design.

Methods used for design of the piping at the Surry Plant not subjected to computer seismic analysis were based on simple beam formulations which, in essence, controlled seismic stress levels through use of pre-established seismic spans. These simple beam formulations were utilized to calculate maximum allowable spans based upon an assumed acceleration factor of 1.5 times the peak acceleration obtained from the response spectra. In calculating the maximum span lengths, it was assumed that a longitudinal pressure of 4,000 psi and a maximum deadweight stress of 1,500 psi were present in the pipe. This combined value of 5,500 psi was subtracted from the allowable stress (1.2 Sh for pressure and deadweight and seismic) to obtain a seismic allowable stress.

Calculating maximum spans by this procedure results in maximum allowable spans greater than the deadweight spans recommended in ANSI B31.1. Thus deadweight governs and provides a greater number of supports resulting in closely spaced restraints. To minimize effects of concentrated weights, restraints were placed as required at valves and other concentrated masses.

For Surry Units 1 & 2, this piping was generally analyzed as follows, with the option of utilizing more rigorous methods available to the analyst:

Nominal Pipe Size, Inches	Method	Where Performed
8" and above	Rigorous (Computer)	Engineering Office
2 1/2" to 6"	Simplified	QEngineering Office
2" and below	Simplified	Field

Piping 2 in. and below was shown on the piping drawings "diagrammatically" (i.e., without detailed dimensions). The stress engineers located supports during the installation process working at the site with erection isometric sketches.

As described above, the stress analysis was performed by assuming many simply supported straight beams, the spans of which are governed by dead load spacing requirements of ANSI B31.1. The piping fundamental frequencies associated with these maximum allowable spans (9.7 to 13.6 cycles per second) are not in resonance with the resonant range acceleration value of the building in which they are located (2 to 8 cycles per second).

Item 3

IE Bulletin No. 79-02 entitled "Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts" will be addressed in separate Vepco correspondence to be submitted on or about June 1, 1979. Status and schedule of the work in progress, flexibility considerations, and other information pertinent to the Bulletin will be contained in that submittal. If the pipe stress reanalysis results in increased support load such that a redesign of a support or the addition of supports are required, the base plate(s) of those supports will be designed so that the intent of the IE Bulletin 79-02 is satisfied.

Item 4

Table I

The following is a table of the piping addressed in Item 1 (i.e., systems, or portions thereof, which used a computer code for seismic analysis). This includes all safety-related systems and non-safety systems (or portions thereof) which could affect the operation of safety systems.

Item	System	Analysis	Re-Analysis
1.	Reactor Coolant System	SHOCK O/I	NUPIPE **
	(S&W Portion Only)	SHOCK II	
2.	Pressurizer Spray and Relief	SHOCK II	NUPIPE
3.	Safety Injection	SHOCK O/I	
		SHOCK II	NUPIPE
	Containment Spray	SHOCK O/I, II	NUPIPE
5.	Recirculation Spray	SHOCK O/I	NUPIPE
6.	Containment Vacuum System	SHOCK II	NUPIPE
7.	Chemical and Volume Control System	SHOCK II	NUPIPE
8.	Residual Heat Removal	SHOCK O/I	NUPIPE
		SHOCK II	
9.	Component Cooling	SHOCK O/I, II	NUPIPE
10	Fuel Pit Cooling	SHOCK O/I	NUPIPE
11.	Service Water System	SHOCK II	NUPIPE
12.	Main Steam System	SHOCK II	NUPIPE
13.	Auxiliary Feedwater System	SHOCK O/I, II	NUPIPE
14.	Steam Generator Feedwater	SHOCK II	NUPIPE
15.	Fire Protection System ***	SHOCK II	NUPIPE
16.	Diesel Muffler *	SHOCK II	NUPIPE

* Non-safety system.

** Reactor Coolant system was re-analyzed using NUPIPE as part of Steam Generator replacement program.

*** Part of system safety related.

Item 4

Table II

The following is a table of piping systems addressed in Item 2 (i.e., systems, or portions thereof, which were evaluated by methods other than computer codes, as discussed in Item 2).

Item	System
1.	Reactor Coolant Systemm
2.	Pressurizer Spray and Relief
3.	Safety Injection
4.	Containment Spray
5.	Recirculation Spray
6.	Containment Vacuum System
7.	Chemical and Volume Control System
8.	Residual Heat Removal
9.	Component Cooling
10.	Fuel Pit Cooling
11.	Compressed Air System
12.	Service Water System
13.	Fire Protection System *
14.	Main Steam System
15.	Auxiliary Feedwater System
16.	Steam Generator Feedwater System
17.	Primary Vent and Drain System Piping
18.	Secondary Vent and Drain
19.	Gaseous Waste Disposal System
20.	Boron Recovery System

* Part of system safety related