VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

October 5, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Serial No. 776A/091779
PO/FHT:baw
Docket Nos: 50-280
50-281
License Nos: DPR-32
DPR-37

Subject: Surry Power Station Units 1 and 2
Interaction Between Non-Safety Grade Systems
and Safety Grade Systems

Dear Mr. Denton:

We have reviewed your letter of September 17, 1979, on the subject issue and are forwarding the attached response in accordance with your request and 10 CFR 50.54 (f) for Surry Power Station Units 1 and 2.

Very truly yours,

C. M. Stallings

C. M. Stallings
Vice President-Power Supply
and Production Operations

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COMMONWEALTH OF VIRGINIA )
) S. S.
CITY OF RICHMOND )

Before me, Notary Public, in and for the City and Commonwealth aforesaid, today personnally appeared C. M. Stallings, who being duly sworn, made oath and said (1) that he is Vice President-Power Supply and Production Operations, of the Virginia Electric and Power Company, (2) that he is duly authorized to execute and file the foregoing statements in behalf of that Company, and (3) that the statements are true to the best of his knowledge and belief.

Given under my hand and notarial seal this 5th day of October, 1979.

My Commission expires January 20, 1981.

Robert M. Neil
Notary Public

(SEAL)

Attachment

Surry Power Station Units 1 and 2

Environmental Interaction of Safety and Non-Safety Grade Systems

In response to your ORDER pursuant to 10CFR50.54(f) issued on September 17, 1979, information is provided which will enable the staff to make a determination on the subject of potential unreviewed safety question concerning interaction between non-safety grade systems and safety grade systems that were discussed in IE Information Notice 79-22, issued September 14, 1979.

Westinghouse, as a part of its environmental qualification activities for IEEE 323-1974, reviewed original assumptions it made for safety analysis reports. Specifically, could a severe environment cause a failure of a non-protection grade component that was previously assumed to remain "as is" and alter the results of the design basis analysis? Westinghouse addressed the failure of a control system due to an adverse environment inside or outside containment following a high energy line rupture which could negate a protective function performed by a safety grade system. They determined that potential interactions existed for the following systems in conjunction with a feedline rupture event:

1) Steam generator power operated relief valve control system
2) Main feedwater control system
3) Pressurizer power operated relief valve control system

They further determined that a potential interaction existed for the automatic rod control system in conjunction with an intermediate steam-line rupture event.

Since the initial Wesinghouse notification, Westinghouse and VEPCO met with the Staff on September 18, 1979, and discussed the scenarios. An agreement was reached that we would address these scenarios immediately and then continue the review of other items including unaddressed interactions. This continuing review will be done with the assistance of the appropriate vendor. The initial review indicates that Surry has no additional potential interactions and actually has no potential feedwater control problems or automatic rod control problems. The specifics of each case are addressed below:

1. Steam generator power operated relief valve control system:

The three steam generator power operated relief valves (SG-PORV) are postulated to fail open as a result of a main feed line break outside the containment between the penetration and the feedline check valve which would affect the SG-PORV control system.

The SG-PORV's are located in the Main Steam Valve House which is immediately adjacent to the reactor containment. This building is designed to release the energy of a High Energy Line Break to the atmosphere. The main feed line exits the containment within this building and the distance from the main feed line penetration to the main feed line check valve is less than five feet. The SG-PORV's are designed to fail closed on a loss of air or loss of electrical signal to the controls.

The Westinghouse-recommended short term action is to instruct the control room operators that a loss of main feed may cause the SG-PORV's to fail open which would cause a loss of steam driven auxiliary feedwater pump.

As Surry has two half-size motor driven auxiliary feedwater pumps in addition to the full size turbine driven auxiliary feedwater pump, and as the Surry FSAR states that the one motor-driven pump alone is sufficient feed flow in this condition, it is stated that this short-term action is sufficient.

2. Main feedwater control system:

The main feedwater control system is postulated to fail and this reduces the mass in the steam generator to less than analyzed. This failure of the main feed control system is assumed to occur as a result of a small main feedline break (less than 0.2 sq. ft.) between the steam generator nozzle and the feedline check valve.

The main feed control system is located outside the containment in the Service Building. The postulated break would have to occur within the containment as the first check valve is located within containment.

Only protection grade equipment associated with the main feedwater control system would be affected and not the lesser quality control grade equipment as postulated. Therefore, this scenario is not applicable to Surry.

3. Pressurizer power operated relief valve control system:

The pressurizer power operated relief valves (P-PORV) are postulated to fail open upon a feedline break between the steam generator and the containment penetration. The P-PORV's are designed to fail closed upon loss of air or electrical signal to the controlling solenoid valves. The postulated event requires that, for the short term, the Westinghouse recommendation be followed and that the control room operators are instructed that a High Energy Line Break within the containment might cause the P-PORV's to fail open or to fail to close. The action will be to close the block valves in the P-PORV lines,

after the problem has been analyzed.

4. Automatic rod control system:

The automatic rod control system was postulated to receive an erroneous signal from the excore detection system following an inside containment intermediate steamline break (0.1-0.25 sq. ft.) between 70-100 percent power. This erroneous signal must occur within the first two minutes of the accident and will cause the rods to be withdrawn.

Westinghouse has provided VEPCO with a generic intermediate steamline rupture analysis which resulted in a rod withdrawal due to a control system environmental interaction prior to a reactor trip. Westinghouse states that the results of the analysis indicated that no fuel damage occurred, which is consistent with the assumptions made in the Safety Analysis Report.