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VIRGINIA ELECTRIC AND POWER COMPANY, RICHMOND, VIRGINIA 23261

October 24, 1979

Mr. James P. O'Reilly, Director
Office of Inspection & Enforcement
U. S. Nuclear Regulatory Commission
Region II
101 Marietta Street, Suite 3100
Atlanta, Georgia 30303

Serial No. 792A
PSE&C/GLP:mac:wang

Docket No. 50-339

Dear Mr. O'Reilly:

On September 24, 1979, a report was made under the provisions of 10CFR50.55(e) concerning interaction between non-safety grade systems and safety grade systems.

We have reviewed these concerns and have made the following evaluations.

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. Pressurizer power operated relief valve control system
3. Main feedwater 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.

These four consequential failures were found to violate internal Westinghouse safety analysis criteria. Specifically, hot leg boiling could occur following a feedline rupture with a consequential failure and minimum DNBR could fall below 1.30 prior to a reactor trip following an intermediate steam line rupture with a consequential failure.

A review of the North Anna Final Safety Analysis Report was performed to determine if this new information was outside the bounds of the FSAR. Hot leg boiling was permitted by the analysis of a feedline rupture event (FSAR section 15.4.2.2). As previously stated, recent Westinghouse criteria did not permit this result. Therefore, while a NSSS vendor criterion has not been met, this new analysis does not exceed the existing FSAR analysis. For North

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Anna, an intermediate steam line rupture event is directly comparable to an excessive load increase incident (FSAR Section 15.2.11). This new information does not exceed the existing FSAR analysis. Specifically, three reactor protection circuits are provided (the first is assumed to fall in this new analysis):

Over temperature Delta-T
NIS Power Range (High Flux)
Overpower Delta-T

Further clarification of the envelope of protection provided by overtemperature Delta-T and overpower Delta-T is shown in FSAR Figure 15.1.1. (Note that this new analysis by the NSSS vendor did not consider other protection circuits which would serve to mitigate this event, i.e., Safety Injection Trip from Steam Line Differential Pressure or Containment High Pressure.)

Information is provided below for North Anna control systems with respect to the specific concerns of I.E. Information Notice 79-22, issued September 14, 1979.

Steam generator power operated relief valve control system: These valves are located in the Main Steam Valve House. They could be subject to an adverse environment in the event of a main feedwater line rupture in this building. (The new Westinghouse analysis assumes a break between the containment penetration and the first upstream check valve. This run of pipe is very short and is ANSI B31.7 piping.)

The steam generator PORV's fail to a closed position upon a loss of air or electrical signal. These valves are air operated with an electrical to pneumatic interface device which converts the electrical control signal to a pneumatic control signal. FSAR Supplement S10.19 states that the auxiliary feedwater system meets the guidelines of Branch Technical Position APCSB No. 10-1. Additionally, FSAR Supplement response 10.19b.2, in effect, already addressed the new Westinghouse analysis in that it discusses the loss of the turbine-driven pump and one motor-driven pump. Additionally, FSAR Supplement S15.18 indicates that one steam generator is sufficient to provide cooling and that FSAR 15.4.2.2. is the bounding analysis for the one-on-one auxiliary feedwater system.

Pressurizer power operated relief valve control system: These valves are located inside the containment. They could be subject to an adverse environment in the event of a main feedwater line rupture in this building. The pressurizer PORV's fail to a closed position upon a loss of air or electrical signal. These valves are air operated with solenoid valves in the air line.

An additional design feature is provided that could mitigate the effects of a RCS depressurization if a PORV were open. Air is removed from the control air system for these valves by a signal derived from protection grade equipment upon RCS pressure (sensed in the pressurizer) falling below a preset value.

Main feedwater control system: These valves are located in the Service Building and are upstream of the containment penetrations and check valves. They could be subject to an adverse environment in the event of a main feedwater line rupture in this building. (Note that a break in this area would be upstream of the check valve for each line. FSAR 15.4.2.2 states that this break location would be treated as a loss of normal feedwater which is covered by FSAR 15.2.8. This section assumes that the steam generators are at a nominal 0% indicated level. This is, then, already consistent with the new Westinghouse analysis for this consequential failure.)

The feedwater regulating valves fail to a closed position upon a loss of air or electrical signal. These valves are air operated with solenoid valves in the air line and an electrical to pneumatic interface device which converts the electrical control signal to a pneumatic control signal.

Automatic rod control system: The excore power range detectors are assumed to fail in the new NSSS vendor analysis. These detectors are located within the neutron shield tank. The detectors are lifted into dry spaces within the tank. These spaces are "dead-ended" and, thus, the detectors would not be exposed to a severe environment. Additionally, the largest opening into the area under the vessel is at the RHR mezzanine, which is above the bottom of the shield tank. This further reduces the potential for an adverse environment to affect the detectors and cable connectors. Incore detector guide tubes (and thimbles) pass through much of this piping. NIS cable is run in conduit from the detector to the containment electrical penetration. The detectors themselves do not have an environmental qualification. However, they can operate at 175° for eight (8) hours and probably could operate at en elevated temperature for shorter durations. The cable has been qualified for 300°F for 15 minutes followed by 252°F for the period of 15 minutes to 13 days following the event. (The cable is radiation qualified to 2 x $10^8$ rads and is qualified for a spray solution of boric acid and sodium hydroxide.)

Additionally, FSAR 6.2.2 states that the containment design temperature is exceeded for a major steam line break for only one (1) minute. The break size assumed by Westinghouse in this new analysis is 0.1 to 0.25 square feet at a reactor power level of 70 to 100%. This break is small compared to the large Main Steam line break. Thus, the peak containment temperature would be reached later in the event since the blowdown rate is lower. Consequently, the NIS power range detectors should remain operable to provide the protection for excessive core power.

## Long Term Action

Licensed reactor operators and licensed senior reactor operators will review this response so that they will be informed of new information generated by Westinghouse.

This review is intended to provide an understanding of recent NSSS vendor work and to develop an awareness of system interactions.

Based on this review, we have concluded that this does not constitute a significant safety concern.

We consider this our final report on these four system interaction concerns; if further information is required, please advise.

Very truly yours,

Sam C. Brown, Jr.
Senior Vice President - Power Station
Engineering and Construction

cc: Mr. Victor Stello, Director
Office of Inspection & Enforcement

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation

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