SNUPPS
Standardized Nuclear Unit
Power Plant System

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Executive Director

May 18, 1981

SLNRC 81-031 FILE: 0541
SUBJ: SNUPPS FSAR - NRC Request for Additional Information

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

Docket Numbers: STN 50-482, STN 50-483, STN 50-486

Reference: NRC (Tedesco) letter to Union Electric (Bryan) and Kansas Gas and Electric (Koester), dated April 16, 1981: Same subject

Dear Mr. Denton:

The referenced letter requested additional information in the area of instrumentation and controls. The enclosure to this letter provides the requested information and will be incorporated into the SNUPPS FSAR in a future revision.

Very truly yours,

N A Petrick

Nicholas A. Petrick

RLS/mtk

Enclosure

cc: J. K. Bryan UE
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Q420.1 Loss of Non-Class IE Instrumentation and Control Power System Bus During Power Operation (IE Bulletin 79-27)

If reactor controls and vital instruments derive power from common electrical distribution systems, the failure of such electrical distribution systems may result in an event requiring operator action concurrent with failure of important instrumentation upon which these operator actions should be based. This concern was addressed in IE Bulletin 79-27. On November 30, 1979, IE Bulletin 79-27 was sent to operating license (OL) holders, the near term OL applicants (North Anna 2, Diablo Canyon, McGuire, Salem 2, Sequoyah, and Zimmer), and other holders of construction permits (CP), including Callaway 1 and Wolf Creek. Of these recipients, the CP holders were not given explicit direction for making a submittal as part of the licensing review. However, they were informed that the issue would be addressed later.

You are requested to address these issues by taking IE Bulletin 79-27 Actions 1 thru 3 under "Actions to be Taken by Licensees". Within the response time called for in the attached transmittal letter, complete the review and evaluation required by Actions 1 thru 3 and provide a written response describing your reviews and actions. This report should be in the form of an amendment to your FSAR and submitted to the NRC Office of Nuclear Reactor Regulations as a licensing submittal.

RESPONSE

Power for the vital reactor instrumentation and protection systems is provided by the Class 1E instrument ac power system. This system is composed of four independent 120-volt ac power supplies to provide power to the four channels of the vital reactor protection and instrumentation systems. With one channel inoperable, the remaining three channels are capable of monitoring the vital reactor parameters continuously and safely shutting down the reactor.

Each essential power panel is fed from a dedicated Class 1E inverter, which, in turn, is fed from one of four independent Class 1E batteries. In the event of loss of power as a result of an inverter failure, a backup supply is provided from a Class 1E MCC, through a regulating transformer. A spare inverter is provided for long-term inverter repairs. Each battery has an associated charger that is fed from a diesel-generator backed bus.

Power for the four non-Class 1E reactor process control channels is provided by the non-Class 1E ac power system through two non-Class 1E UPS*s. Each power supply train supplies a dedicat[illegible] UPS* that, in turn, supplies two process control cab[illegible] backup dc supply is provided to the UPS* in the event [illegible] primary source is not available.

The backup dc power source is the non-Class 1E dc system. This system is composed of two station batteries and two battery chargers. Both of the chargers are powered from a diesel generator-backed bus.

In the event of loss of power as a result of an inverter failure, two trains of backup power to the process cabinets are provided by manual switches from the non-Class 1E ac system. These trains of ac power are provided with a cross tie for additional reliability.

Power for miscellaneous non-Class 1E instrument loads is provided by the non-Class 1E instrument ac power system. This system is powered from the Class 1E power system through a qualified isolating regulating transformer. One transformer is provided for each train of instrument ac. No cross ties are provided.

The Class 1E instrument ac power system is provided with the following alarms in the control room:

a. Loss of inverter dc input

b. Loss of inverter ac output

c. Loss of switchboard voltage

The non-Class 1E dc system is provided with the following alarms in the control room:

a. System ground

b. Battery imbalance

c. Charger dc overvoltage

d. Charger ac undervoltage

e. Charger dc undervoltage

f. Charger ac and dc breakers open

g. Charger failure

h. Loss of distribution board voltage

i. Loss of switchboard voltage

* uninterruptable power supply

The non-Class 1E instrument ac system is provided with a loss of bus voltage alarm in the control room.

Procedures (i.e. emergency procedures, administrative procedures, and/or alarm procedures) will be developed that address action item number two of IE Bulletin 79-27. As a result of the review of IE Bulletin 79-27 and IE Circular 79-02, no design modifications are required. However, the ongoing development of procedures and administrative controls will consider these IE issuances.

Q420.2 Engineered Safety Features (ESF) Reset Controls (IE Bulletin 80-06)

If safety equipment does not remain in its emergency mode upon reset of an engineered safeguards actuation signal, system modification, design change or other corrective action should be planned to assure that protective action of the affected equipment is not compromised once the associated actuation signal is reset. This issue was addressed in IE Bulletin 80-06 (enclosed). For facilities with operating licenses as of March 13, 1980, IE Bulletin 80-06 required that reviews be conducted by the licensees to determine which, if any, safety functions might be unavailable after reset, and what changes could be implemented to correct the problem.

For facilities with a construction permit including OL applicants Bulletin 80-06 was issued for information only.

The NRC staff has determined that all CP holders, as a part of the OL review process, are to be requested to address this issue. Accordingly, you are requested to take the actions called for in Bulletin 80-06 Actions 1 thru 4 under "Actions to be Taken by Licensees". Within the response time called for in the attached transmittal letter, complete the review verifications and description.

RESPONSE

A review has been conducted of the drawings for all systems serving safety-related functions at the schematic level to determine whether or not, upon reset of an ESF actuation signal, all associated safety-related equipment remains in its emergency mode. The review revealed that certain equipment would, in particular circumstances, change state upon ESF reset. The affected equipment included the control room and electrical equipment room air-conditioning units, the containment air coolers, the hydrogen mixing fans, and the component cooling water heat exchanger temperature control valves. The control circuits for this equipment were revised to provide seal-in features so that an ESF reset would not change the safeguards state of the equipment.

There are two preoperational tests (see Sections 14.2.12.1.71 and 14.2.12.1.72) which require that equipment alignment be verified after reset of ESF actuation signals. These tests will verify that the installed controls are consistent with the schematics reviewed and that all equipment remains in its emergency mode upon ESF reset.

Q420.3 Qualification of Control Systems (IE Information Notice 79-22)

Operating reactor licensees were informed by IE Information Notice 79-22, issued September 19, 1979, that certain non-safety grade or control equipment, if subjected to the adverse environment of a high energy line break, could impact the safety analyses and the adequacy of the protection functions performed by the safety grade equipment. Enclosed is a copy of IE Information Notice 79-22, and reprinted copies of an August 30, 1979 Westinghouse letter, and a September 10, 1979 Public Service Electric and Gas Company letter which address this matter. Operating Reactor licensees conducted reviews to determine whether such problems could exist at operating facilities.

We are concerned that a similar potential may exist at light water facilities now under construction. You are, therefore, requested to perform a review to determine what, if any, design changes or operator actions would be necessary to assure that high energy line breaks will not cause control system failures to complicate the event beyond your FSAR analysis. Provide the results of your review, including all identified problems and the manner in which you have resolved them to NRR.

The specific "scenarios" discussed in the above referenced Westinghouse letter are to be considered as examples of the kinds of interactions which might occur. Your review should include those scenarios, where applicable, but should not necessarily be limited to them. Applicants with other LWR designs should consider analogous interactions as relevant to their designs.

RESPONSE

SNUPPS reported on the matters addressed in IE Information Notice 79-22 to the NRC in two letters (SLNRC 79-15 dated September 28, 1979 and SLNRC 80-6 dated February 5, 1980). These reports were submitted to NRC Inspection and Enforcement pursuant to 10 CFR 50.55(e). The latter report stated that final resolution would be provided in revisions to the SNUPPS FSAR. The resolution and/or current status is provided below.

Westinghouse identified four control systems for generic consideration of nonsafety grade/safety grade interface interactions.

a. Steam generator power-operated relief valve control system - a piping failure in the vicinity of the steam generator relief valves could be assumed to cause the valves to stick open. The combination of the pipe failure, an assumed single failure, and the stuck open valve(s) may result in inadequate auxiliary feedwater flow.

The SNUPPS main steam atmospheric relief valves and the associated pressure transmitters have been procured as Class 1E devices which are environmentally qualified for the effects of high energy line breaks. Therefore, this scenario does not present a safety problem for the SNUPPS design.

b. Pressurizer power-operated relief valves control system - A failure of secondary system piping inside the containment is assumed to cause pressurizer power-operated relief valves (PORV) to open. The resultant secondary break coincident with PORV opening may have more severe consequences than those accidents previously analyzed.

The SNUPPS pressurizer PORV and associated pressure transmitters meet Class 1E requirements and are qualified to the postulated accident environments inside the containment. Therefore, this scenario does not present a safety problem for the SNUPPS design.

c. Main feedwater control system - A small secondary system break could affect normal feedwater flow control, causing low steam generator levels prior to protective actions for the break.

The SNUPPS secondary line break accident has been reanalyzed, assuming the control and protection grade system interaction. The analysis shows that this scenario can be accommodated without violating design conditions and acceptance criteria. A summary of the analysis will be submitted in a future revision to the FSAR. The summary will include an identification of the analysis assumptions that are different from those used in Westinghouse Topical Report WCAP-9230.

d. Automatic rod control system - An intermediate size high energy line break is assumed to affect the rod control system, such that the

initial conditions previously assumed for the break may not be valid.

The SNUPPS power range ex-core detectors and associated in-containment equipment are currently being environmentally qualified. Completion of the qualification program is scheduled for the end of 1981. Successful qualification will eliminate this scenario as a safety concern for the SNUPPS design.

In addition to the above-mentioned generic Westinghouse interactions, other scenarios that could potentially produce adverse control and protection system interactions are being considered by industry generic groups in association with Westinghouse, such as, through the sub-committee on systems interaction sponsored by the Atomic Industrial Forum.

For additional evaluation of control grade system failures, refer to the response to Question #420.4.

Q420.4 Control System Failures

The analyses reported in Chapter 15 of the FSAR are intended to demonstrate the adequacy of safety systems in mitigating anticipated operational occurrences and accidents.

Based on the conservative assumptions made in defining these design-basis events and the detailed review of the analyses by the staff, it is likely that they adequately bound the consequences of single control system failures.

To provide assurance that the design basis event analyses adequately bound other more fundamental credible failures you are requested to provide the following information:

(1) Identify those control systems whose failure or malfunction could seriously impact plant safety.

(2) Indicate which, if any, of the control systems identified in (1) receive power from common power sources. The power sources considered should include all power sources whose failure or malfuction could lead to failure or malfunction of more than one control system and should extend to the effects of cascading power losses due to the failure of higher level distribution panels and load centers.

(3) Indicate which, if any, of the control systems identified in (1) receive input signals from common sensors. The sensors considered should include, but should not necessarily be limited to, common hydraulic headers or impulse lines feeding pressure, temperature, level or other signals to two or more control systems.

(4) Provide justification that any simultaneous malfunctions of the control systems indentified in (2) and (3) resulting from failures or malfunctions of the applicable common power source or sensor are bounded by the analyses in Chapter 15 and would not require action or response beyond the capability of operators or safety systems.

RESPONSE

This question requires extensive evaluation as discussed with the NRC (ICSB) during a meeting on April 28, 1981. The evaluation consists of postulating credible failures which affect the major nuclear steam supply system control systems and demonstrating that for each failure the resulting event would be within the bounds of existing accident analyses. The events which will be considered credible are:

a. Loss of any single instrument

b. Loss of any single instrument line

c. Loss of power within a single control system

d. Loss of power to all systems powered by a single separation group

The analysis will be limited to the reactor control system, steam dump system, pressurizer pressure control system, pressurizer level control system, and feedwater control system.

The results of this analysis/evaluation will be submitted to the NRC by June 15, 1981.