UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

APR 30 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos.: 50-454/455
and 50-456/457

Commonwealth Edison Company
ATTN: Mr. J. S. Abel
Director of Nuclear Licensing
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. Abel:

Subject: Request for Information Regarding Four Potential Problem Areas in the Design and Analysis of the Instrumentation and Control Systems at Byron/Braidwood

The staff has identified four potential areas in the instrumentation and control systems that must be addressed in your OL license application. The first two are related to events that have occurred at operating plants, the third, generic questions regarding environmental qualification of Westinghouse NSSS equipment, and the fourth, seeks to ensure that all credible design basis events which can lead to failure of power or control systems are considered in Chapter 15 of the FSAR. These four items of concern are listed as questions 30.040 through 30.043 in the Enclosure.

We request that you amend your FSAR to reflect your response to these questions and any other outstanding items by August 1, 1981.

If you need further clarification on these items, contact the Licensing Project Manager for Byron, Larry White (301/492-8285) or the Licensing Project Manager for Braidwood, Kenneth Kiper (301/492-7318).

Sincerely,

P. S. Youngblood for

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As stated

cc: See next page

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THIS DOCUMENT CONTAINS
POOR QUALITY PAGES

Mr. J. S. Abel
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

ccs:
Mr. William Kortier
Atomic Power Distribution
Westinghouse Electric Corporation
P. O. Box 355
Pittsburgh, Pennsylvania 15230

Paul M. Murphy, Esq.
Isham, Lincoln & Beale
One First National Plaza
42nd Floor
Chicago, Illinois 60603

Mrs. Phillip B. Johnson
1907 Stratford Lane
Rockford, Illinois 61107

Ms. Julianne Mahler
Center for Governmental Studies
Northern Illinois University
DeKalb, Illinois 60115

C. Allen Bock, Esq.
P. O. Box 342
Urbanan, Illinois 61820

Thomas J. Gordon, Esq.
Waaler, Evans & Gordon
2503 S. Neil
Champaign, Illinois 61820

Ms. Bridget Little Rorem
Appleseed Coordinator
117 North Linden Street
Essex, Illinois 60935

Kenneth F. Levin, Esq.
Beatty, Levin, Holland,
Basofin & Sarsany
11 South LaSalle Street
Suite 2200
Chicago, Illinois 60603

Mr. Edward R. Crass
Nuclear Safeguards and Licensing Division
Sargent & Lundy Engineers
55 East Monroe Street
Chicago, Illinois 60603

Nuclear Regulatory Commission, Region III
Office of Inspection and Enforcement
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Myron Cherry, Esq.
Cherry, Flynn and Kanter
1 IBM Plaza, Suite 4501
Chicago, Illinois 60611

ENCLOSURE
BYRON/BRAIDWOOD STATIONS

Instrumentation & Control Systems Branch

30.040 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 (Attachment 1). 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 Byron/Braidwood. 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 this issue 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.

30.041 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 (Attachment 2). 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

of corrective actions taken or planned as stated in Action 1 thru 3 and submit the report called for in Action Item 4. The report should be submitted to the NRC Office of Nuclear Reactor Regulation as a licensing submittal in the form of an FSAR amendment.

30.042 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. Attachment 3 is a copy of IE Information Notice 79-22, and reprinted copies of an August 20, 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. You should consider analogous interactions relevant to the designs of the Byron and Braidwood plants.

30.043 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 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 malfunction could lead to failure or malfuction 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.

ATTACHMENT 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

SSINS No.: 6820
Accession No.:
7910250499

November 30, 1979

IE Bulletin No. 79-27

LOSS OF NON-CLASS-1-E INSTRUMENTATION AND CONTROL POWER SYSTEM BUS DURING OPERATION

Description of Circumstances:

On November 10, 1979, an event occurred at the Oconee Power Station, Unit 3, that resulted in loss of power to a non-class-1-E 120 Vac single phase power panel that supplied power to the Integrated Control System (ICS) and the Non-Nuclear Instrumentation (NNI) System. This loss of power resulted in control system malfunctions and significant loss of information to the control room operator.

Specifically, at 3:16 p.m., with Unit 3 at 100 percent power, the main condensate pumps tripped, apparently as a result of a technician performing maintenance on the hotwell level control system. This led to reduced feedwater flow to the steam generators, which resulted in a reactor trip due to high coolant system pressure and simultaneous turbine trip at 3:16:57 p.m. At 3:17:15 p.m., the non-class-1-E inverter power supply feeding all power to the integrated control system (which provides proper coordination of the reactor, steam generator feedwater control, and turbine) and to one NNI channel tripped and failed to automatically transfer its loads from the DC power source to the regulated AC power source. The inverter tripped due to blown fuses. Loss of power to the NNI rendered control room indicators and recorders for the reactor coolant system (except for one wide-range RCS pressure recorder) and most of the secondary plant systems inoperable, causing loss of indication for systems used for decay heat removal and water addition to the reactor vessel and steam generators. Upon loss of power, all valves controlled by the ICS assumed their respective failure positions. The loss of power existed for approximately three minutes, until an operator could reach the equipment room and manually switch the inverter to the regulated AC source.

The above event was discussed in IE Information Notice No. 79-29, issued November 16, 1979.

NUREG 0600 "Investigation into the March 28, 1979 TMI Accident" also discusses TMI LER 78-021-03L whereby the RCS depressurized and Safety Injection occured on loss of a vital bus due to inverter failure.

Actions to Be Taken by Licensees

For all power reactor facilities with an operating license and for those nearing completion of construction (North Anna 2, Diablo Canyon, McGuire, Salem 2, Sequoyah, and Zimmer):

1. Review the class-1-E and non-class 1-E buses supplying power to safety and non-safety related instrumentation and control systems which could affect the ability to achieve a cold shutdown condition using existing procedures or procedures developed under item 2 below. For each bus:

   a) identify and review the alarm and/or indication provided in the control room to alert the operator to the loss of power to the bus.
   b) identify the instrument and control system loads connected to the bus and evaluate the effects of loss of power to these loads including the ability to achieve a cold shutdown condition.
   c) describe any proposed design modifications resulting from these reviews and evaluations, and your proposed schedule for implementing those modifications.

2. Prepare emergency procedures or review existing ones that will be used by control room operators, including procedures required to achieve a cold shutdown condition, upon loss of power to each class 1-E and non-class 1-E bus supplying power to safety and non-safety related instrument and control systems. The emergency procedures should include:

   a) the diagnostics/alarms/indicators/symptom resulting from the review and evaluation conducted per item 1 above.

   b) the use of alternate indication and/or control circuits which may be powered from other non-class 1-E or class 1-E instrumentation and control buses.

   c) methods for restoring power to the bus.

   Describe any proposed design modification or administrative controls to be implemented resulting from these procedures, and your proposed schedule for implementing the changes.

3. Re-review IE Circular No. 79-02, Failure of 120 Volt Vital AC Power Supplies, dated January 11, 1979, to include both class 1-E and non-class 1-E safety related power supply inverters. Based on a review of operating experience and your re-review of IE Circular No. 79-02, describe any proposed design modifications or administrative controls to be implemented as a result of the re-review.

4. Within 90 days of the date of this Bulletin, complete the review and evaluation required by this Bulletin and provide a written response describing your reviews and actions taken in response to each item.

Reports should be submitted to the Director of the appropriate NRC Regional Office and a copy should be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

If you desire additional information regarding this matter, please contact the IE Regional Office.

IE Bulletin No. 79-27
November 30, 1979

Enclosure

RECENTLY ISSUED
IE BULLETINS

Bulletin No.	Subject	Date Issued	Issued To
79-26	Boron Loss From BWR Control Blades	11/20/79	All BWR power reactor facilities with an OL
79-25	Failures of Westinghouse BFD Relays In Safety-Related Systems	11/2/79	All power reactor facilities with an OL or CP
79-17 (Rev. 1)	Pipe Cracks In Stagnant Borated Water System At PWR Plants	10/29/79	All PWR's with an OL and for information to other power reactors
79-24	Frozen Lines	9/27/79	All power reactor facilities which have either OLs or CPs and are in the late stage of construction
79-23	Potential Failure of Emergency Diesel Generator Field Exciter Transformer	9/12/79	All Power Reactor Facilities with an Operating License or a construction permit
79-14 (Supplement 2)	Seismic Analyses For As-Built Safety-Related Piping Systems	9/7/79	All Power Reactor Facilities with an OL or a CP
79-22	Possible Leakage of Tubes of Tritium Gas in Timepieces for Luminosity	9/5/79	To Each Licensee Who Receives Tubes of Tritium Gas Used in Timepieces for Luminosity
79-13 (Rev. 1)	Cracking in Feedwater System Piping	8/30/79	All Designated Applicants for OLs
79-02 (Rev. 1) (Supplement 1)	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	8/20/79	All power Reactor Facilities with an OL or a CP
79-14 (Supplement)	Seismic Analyses For As-Built Safety-Related Piping Systems	8/15/79	All Power Reactor Facilities with an OL or a CP

SSINS: 6820
Accession No.:
8002280639

ATTACHMENT 2
UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

March 13, 1980

IE Bulletin No. 80-06

ENGINEERED SAFETY FEATURE (ESF) RESET CONTROLS

Description of Circumstances:

On November 7, 1979, Virginia Electric and Power Company (VEPCO) reported that following initiation of Safety Injection (SI) at North Anna Power Station Unit 1, the use of the SI Reset pushbuttons alone resulted in certain ventilation dampers changing position from their safety or emergency mode to their normal mode. Further investigation by VEPCO and the architect-engineer resulted in discovery of circuitry which similarly affected components actuated by a Containment Depressurization Actuation (CDA, activated on Hi-Hi Containment Pressure). The circuits in question are listed below:

Component/System	Problem
Outside/Inside Recirculation Spray Pump Motors	Pump motors will not start after actuation if CDA Reset is depressed prior to starting timer running out (approx. 3 minutes)
Pressurized Control Room Ventilation Isolation Dampers	Dampers will open on SI Reset
Safeguards Area Filter Dampers	Dampers reposition to bypass filters when CDA Reset is depressed
Containment Recirculation Cooler Fans	Fans will restart when CDA Reset is depressed
Service Water Supply and Discharge Valves to Containment	If service water is being used as the cooling medium prior to CDA actuation, valves will reopen upon depressing CDA reset
Service Water Radiation Monitoring Sample Pumps	Pumps will not start after actuation if CDA reset is depressed prior to motor starting timers running out
Main Condenser Air Ejector Exhaust Isolation Valves to the Containment	After receiving a high radiation monitor alarm on the air ejector exhaust, SI actuation would shut these valves and depressing SI Reset would reopen them

Review of circuitry for ventilation dampers, motors, and valves reported by VEPCO resulted in discovery of similar designs in ESF-actuated components at Surry Unit 1 and Beaver Valley; where it has been found that certain equipment would return to its normal mode following the reset of an ESF signal; thus, protective actions of the affected systems could be compromised once the associated actuation signal is reset. These two plants had Stone and Webster Engineering Corporation for the architect-engineer as did the North Anna Units.

The Stone and Webster Engineering Corporation and VEPCO are preparing design changes to preclude safety-related equipment from moving out of its emergency mode upon reset of an Engineered Safety Features Actuation Signal (ESFAS). This corrective action has been found acceptable by the NRC, in that, upon reset of ESFAS, all affected equipment remains in its emergency mode.

The NRC has performed reviews of selected areas of ESFAS reset action on PWR facilities and, in some cases, this review was limited to examination of logic diagrams and procedures. It has been determined that logic diagrams may not adequately reflect as-built conditions; therefore, the requested review of drawings must be done at the schematic/elementary diagram level.

There have been several communications to licensees from the NRC on ESF reset actions. For example, some of these communications have been in the form of Generic Letters issued in November, 1978 and October, 1979 on containment venting and purging during normal operation. Inspection and Enforcement Bulletins Nos. 79-05, 05A, 05B, 06A, 06B and 08 that addressed the events at TMI-2 and NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations. However, each of these communications has addressed only a limited area of the ESF's. We are requesting that the reviews undertaken for this Bulletin address all of the ESF's.

Actions To Be Taken By Licensees:

For all PWR and BWR facilities with operating licenses:

1. Review the drawings for all systems serving safety-related functions at the schematic level to determine whether or not upon the reset of an ESF actuation signal, all associated safety-related equipment remains in its emergency mode.

2. Verify the actual installed instrumentation and controls at the facility are consistent with the schematics reviewed in Item 1 above by conducting a test to demonstrate that all equipment remains in its emergency mode upon removal of the actuating signal and/or manual resetting of the various isolating or actuation signals. Provide a schedule for the performance of the testing in your response to this Bulletin.

3. If any safety-related equipment does not remain in its emergency mode upon reset of an ESF signal at your facility, describe proposed system modification, design change, or other corrective action planned to resolve the problem.

4. Report in writing within 90 days, the results of your review and include a list of all devices which respond as discussed in item 3 above, actions taken or planned to assure adequate equipment control, and a schedule for implementation of corrective action. This information is requested under the provisions of 10 CFR 50.54(f). Accordingly, you are requested to provide within the time period specified above, written statements of the above information, signed under oath or affirmation. Reports shall be submitted to the Director of the appropriate NRC Regional Office and a copy shall be forwarded to the NRC Office of Inspection and Enforcement, Division of Reactor Operations Inspection, Washington, D.C. 20555.

For all power reactor facilities with a construction permit, this Bulletin is for information only and no written response is required.

Approved by GAO, B180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

ATTACHMENT 3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

September 14, 1979

IE Information Notice No. 79-22

QUALIFICATION OF CONTROL SYSTEMS

Public Service Electric and Gas Company notified the NRC of a potential unreviewed safety question at their Salem Unit 1 facility. This notification was based on a continuing review by Westinghouse of the environmental qualifications of equipment that they supply for nuclear steam supply systems. Based on the present status of this effort, Westinghouse has informed their customers that the performance of non-safety grade equipment subjected to an adverse environment could impact the protective functions performed by safety grade equipment. These non-safety grade systems include:

Steam generator power operated relief valve control system

Pressurizer power operated relief valve control system

Main feedwater control system

Automatic rod control system

These systems could potentially malfunction due to a high energy line break inside or outside of containment. NRC is also concerned that the adverse environment could also give erroneous information to the plant operators. Westinghouse states that the consequences of such an event could possibly be more limiting than results presented in Safety Analysis Reports, however, Westinghouse also states that the severity of the results can be limited by operator actions together with operating characterisitics of the safety systems. Further, Westinghouse has recommended to their customers that they review their systems to determine whether any unreviewed safety questions exist.

This Information Notice is provided as an early notification of a possibly significant matter. It is expected that recipients will review the information for possible applicability to their facilities. No specific action or response is requested at this time. If NRC evaluations so indicate, further licensee actions may be requested or required. If you have questions regarding this matter, please contact the Director of the appropriate NRC Regional Office.

No written response to this Information Notice is required.

REPRINT

Westinghouse Electric Corporation
Water Reactor Division
Nuclear Service Division
Box 2728
Pittsburgh, Pennsylvania 15230

August 30, 1979
PSE-79-21

Mr. F. P. Librizzi, General Manager
Electric Production
Public Service Electric and Gas Company
80 Park Place
Newark, New Jersey 07101

Dear Mr. Librizzi:

Public Service Electric and Gas Co.
Salem Unit No. 1
QUALIFICATION OF CONTROL SYSTEMS

As part of a continuing review of the environmental qualifications of Westinghouse supplied NSSS equipment, Westinghouse has also found it necessary to consider the interaction with non-safety grade systems. This investigation has been conducted to determine if the performance of non-safety grade systems which may not be protected from an adverse environment could impact the protective functions performed by NSSS safety grade equipment. The NSSS control and protection systems were included in this review to assess the adequacy of the present environmental qualification requirements.

As a result of this review, several systems were identified which, if subjected to an adverse environment, could potentially lead to control system operation which may impact protective functions. These systems are:

- Steam generator power operated relief valve control system
- Pressurizer power operated relief valve control system
- Main feedwater control system
- Automatic rod control system

Each of the above mentioned systems could potentially malfunction if impacted by adverse environments due to a high energy line break inside or outside containment. In each case, a limited set of breaks, coupled with possible consequential control malfunction in an adverse direction, of the above events could yield results which are more limiting than those presented in the plant Safety Analysis Reports. In all cases, however, the severity of the results can be limited by operator actions together with operating characteristics of the safety systems.

We believe these systems identified do not constitute a substantial safety hazard. However, Westinghouse recommends you review them to determine if any unreviewed safety questions or significant deficiencies exist in your plant(s).

To assist you in understanding these concerns, Westinghouse will hold a seminar in Pittsburgh on Thursday, September 6 at Westinghouse R&D Center, Building 701, with all our operating plant customers. The seminar will address the potential impact of these concerns for various plant designs and various licensing bases.

Please contact your WNSD Regional Service office to confirm your attendance at the seminar. We will provide additional details concerning the agenda and other meeting arrangements as they become available.

Very truly yours,

ORIGINAL SIGNED BY

F. Noon, Manager
Eastern Regional & WNI Support

SR4/CC13&14

cc: H. J. Midura
H. J. Heller
R. D. Rippe
T. N. Taylor
R. A. Uderitz
C. F. Barclay W

REPRINT

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
Salem Nuclear Generating Station
P. O. Box 56
Hancocks Bridge, New Jersey 08038

September 10, 1979

Mr. Boyce H. Grier
Director of USNRC
Office of Inspection and Enforcement
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Sir:

REPORTABLE OCCURRENCE 79-58/01P
SALEM NO. 1 UNIT LER

This letter will serve to confirm our telephone report to Mr. Gary Schneider of the Regional NRC office on Friday, September 6, 1979, advising of a potential reportable occurrence in accordance with Technical Specification 6.9.1.8.

We have been notified by our Engineering Department that a Westinghouse conducted review of the environmental qualifications of Westinghouse supplied NSSS equipment has identified that conditions associated with high energy line breaks inside or outside containment and their impact on non-safety control systems may constitute an unreviewed safety question. The control systems concerned are steam generator power operated relief valve control, pressurizer power operated relief valve control, main feedwater control and automatic rod control systems.

A detailed report will be submitted in the time period specified by the Technical Specifications.

Very truly yours,

Original Signed By

H. J. Midura
Manager - Salem Generating Station

AWK:jds

CC: General Manager - Electric Production
Manager - Quality Assurance