APR 16 1981

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U.S. NUCLEAR REGULATORY COMMISSION

Docket Nos.: 50-361/362

Mr. Robert Dietch
Vice President
Southern California Edison Company
2244 Walnut Grove Avenue
P. O. Box 800
Rosemead, California 91770

Mr. D. W. Gilman
Vice President - Power Supply
San Diego Gas & Electric Company
101 Ash Street
P. O. Box 1831
San Diego, California 92112

Gentlemen:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION - SAN ONOFRE NUCLEAR GENERATING STATION, UNITS 2 AND 3

The Instrumentation and Control Systems Branch has identified four concerns that must be addressed prior to completion of its review of operating license applications. The specific conerns are delineated in the enclosure. We request that you amend your Final Safety Analysis Report to reflect your responses as soon as possible. Should you have any questions, contact the Licensing Project Manager, Harry Rood.

Sincerely,

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

Enclosure:
As stated

cc: See next page.

DISTRIBUTION: SEE NEXT PAGE.

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SURNAME	IJLee:jb	HRood	FJMiraglia	RLTedesco			
DATE	4/14/81	4/14/81	4/ /81	4/15/81			

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

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R Tedesco

Robert L. Tedesco, Assistant Director
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Division of Licensing

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Vice President
Southern California Edison Company
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P. O. Box 800
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Mr. D. W. Gilman
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Chickering & Gregory
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Mt. La Jolla, California 92037

Resident Inspector, San Onofre/NPS
c/o U. S. Nuclear Regulatory Commission
P. O. Box AA
Oceanside, California 92054

ENCLOSURE

222.0 Instrumentation & Control Systems Branch

222. 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 concerns 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 San Onofre. 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 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 and submitted to the NRC Office of Nuclear Reactor Regulation as a licensing submittal.

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

222. 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 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 reviews 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 kind of interactions which might occur. Your review should include those scenarios, where appiciable, but should not necessarily be limited to them. Applicants with other LWR designs should consider analogous interactions as relevant to their designs.

222. 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 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.

BACKGROUND

INFORMATION

Enclosure 2

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.

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

IE Bulletin No. 79-27
Novebmer 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
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

Enclosure 3

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.

ENCLOSURE 4

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.

DUPE 7908220124

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

DUPE 7911050066 (2pp.)

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

DUPE 8007150020 (1p.)

MEETING HIGHLIGHTS

POTENTIAL UNREVIEWED SAFETY QUESTION ON INTERACTION BETWEEN NON-SAFETY GRADE SYSTEMS AND SAFETY GRADE SYSTEMS

I&E Information Notice 79-22, dated September 14, 1979, was issued informing the nuclear industry of a potential unreviewed safety question at Salem, Unit 1 of Public Service Electric and Gas Company, based on a Westinghouse review of the environmental qualification of equipment. Certain non-safety grade equipment, if subjected to an adverse environment such as results from a high-energy line break inside or outside of containment, could impact the safety analyses and the protective functions performed by safety grade equipment.

Meetings were arranged with all four light water reactor vendors according to the following schedule:

Westinghouse - Tuesday, September 18

Combustion Engineering - Wednesday, September 19

Babcock and Wilcox - Thursday a.m., September 20

General Electric - Thursday, p.m., September 20

During the Westinghouse meeting, they identified, for all high-energy line breaks and possible locations, the control systems that could be affected as a result of the adverse environment and whose consequential failure could invalidate the accident analyses presented in Westinghouse plants' SARs. Recommendations were also presented for resolving the adverse interactions identified.

Westinghouse's investigation identified seven accidents and seven control systems that could possibly interact and presented them in a matrix form as shown in Enclosure 1. As can be seen the potential interactions that could degrade the accident analyses are in the:

a. Automatic Rod Control System

b. Pressurizer PORV Control System

c. Main Feedwater Control System

d. Steam Generator PORV Control System

Westinghouse presented their recommended short-term and long-term solutions, presented as Enclosure 2.

Westinghouse stated that the possible matrix interactions may increase as more detailed analyses are performed but the interactions will remain for all of their plants and the interactions may be eliminated only if conditions are such that plant specific designs mitigate the interactions because of:

a. system layout

b. type of equipment used

c. qualification status of equipment utilized

d. design basis events considered for license applications

e. prior commitments made by utility to the NRC.

Westinghouse stated that their investigations were carried further than FSAR analysis and they would need to evaluate consequential failures on a realistic basis; this evaluation may eliminate some problems. Westinghouse also stated that their investigations are lower probability subsets of FSAR analyses whicy in themselves are sets of low probability.

ROD CONTROL SYSTEM

AREAS OF CONCERN

- CONTROL ROD WITHDRAWAL DUE TO CONTROL SYSTEM ENVIRONMENTAL CONSEQUENTIAL FAILURE (POWER RANGE EXCORE DETECTOR AND ASSOCIATED CABLING)

- MINIMUM DNBR FALLS BELOW 1.30 PRIOR TO REACTOR TRIP

Westinghouse and the utility representatives all doubt that they can conclusively determine the qualification status of all of the involved equipment in 20 days. Both Westinghouse and the utility representatives stated that they will respond to the 20-day letter by addressing the four control systems identified in a manner suggested by the Westinghouse recommendations unless the NRC staff provides directions to the contrary and further establishes guidelines stating their position on the problem along with their recommendations.

The NRC staff stated that they are sympathetic to the requests by the nuclear industry regarding position and direction but this can be formulated only at the conclusion of the scheduled meetings with all four light water reactor vendors. At that time the staff will present their results, magnitude and direction to industry for resolution of the problem.

At this time, it is not evident which utilities are faced with what environmental interaction problem. The effects of implementing all of the Westinghouse recommended short-term "fixes" may be contradicted by other sequences. There are three parts to the problem dealing with the basis of short-term operation:

1. qualify equipment to the appropriate environment; this would take longer than 20 days and would, more likely, for most utilities, be a long-term partial solution.
2. short-term "fixes" should be in place pending long-term solutions such as the above. It must be noted that in this situation, some components that are relied upon to operate might possibly be wiped out by consequential failures under certain conditions and accident sequences if the postulated adverse environment is established.
3. the "worst case" plant should be selected and a bounding analysis performed to determine the time frame available for qualification of equipment.

Accident \ Control System	Reactor Control	Pressurizer: Pressure Control	Pressurizer: Level Control	Feedwater Control	Steam Generator Pressure Control	Steam Dump System	Turbine Control
Small Steamline Rupture	X	X			X		
Large Steamline Rupture		X			X		
Small Feedline Rupture	X	X		X	X		
Large Feedline Rupture	X	X			X		
Small LOCA	X	X		X			
Large LOCA							
Rod Ejection							

PROTECTION SYSTEM-CONTROL SYSTEM POTENTIAL ENVIRONMENTAL INTERACTION

X - POTENTIAL INTERACTION IDENTIFIED THAT COULD DEGRADE ACCIDENT ANALYSIS

☐ - NO SUCH INTERACTION MECHANISM IDENTIFIED

ROD CONTROL SYSTEM

POTENTIAL SOLUTIONS

SHORT TERM

DETERMINE IF THE ADVERSE ENVIRONMENT CAN IMPACT EXCORE DETECTORS AND ASSOCIATED CABLING PRIOR TO REACTOR TRIP FOLLOWING INTERMEDIATE STEAMLINE RUPTURE.

- REMOVE NIS SIGNAL FROM POWER MISMATCH CIRCUIT IN ROD CONTROL SYSTEM (PROCESS CONTROL CABINET)
- EMPLOY MANUAL ROD CONTROL

LONG TERM

- USE CONTAINMENT PRESSURE TRIP AND QUALIFY EXCORE DETECTOR TO LESS SEVERE ENVIRONMENT (ALSO REQUIRES QUALIFYING CABLING FROM DETECTOR TO PENETRATION)
- QUALIFY EXCORE DETECTOR TO STEAMLINE BREAK ENVIRONMENT 420°F CURVE ALSO REQUIRES QUALIFYING CONNECTION AND CABLING FROM EXCORE DETECTOR TO PENETRATION

PRESSURIZER POWER OPERATED RELIEF VALVE CONTROL SYSTEM

AREAS OF CONCERN

- CONTROL SYSTEM ENVIRONMENTAL FAILURE CAUSES SMALL LOCA IN STEAM SPACE OF PRESSURIZER DUE TO SECONDARY HIGH ENERGY LINE RUPTURE
- HOT LEG BOILING OCCURS FOLLOWING FEEDLINE RUPTURE

PRESSURIZER PORV CONTROL SYSTEM

POTENTIAL SOLUTIONS

SHORT TERM

- INVESTIGATE WHETHER PRESSURIZER PORV CONTROL SYSTEM WILL FAIL OR OPERATE NORMALLY WHEN EXPOSED TO ADVERSE ENVIRONMENT.
- MODIFY OPERATING INSTRUCTIONS TO ALERT OPERATOR TO THE POSSIBILITY OF A CONSEQUENTIAL FAILURE IN THE PRESSURIZER PORV CONTROL SYSTEM CAUSED BY ADVERSE ENVIRONMENT. IF EVIDENT, CLOSE BLOCK VALVES IN RELIEF LINES.

LONG TERM

- REDESIGN PRESENT CONTROL SYSTEM TO WITHSTAND ANTICIPATED ENVIRONMENT
- INSTALL MOV IN SERIES WITH EXISTING MOV BLOCK VALVE. INSTALL PROTECTION GRADE CIRCUITRY TO CLOSE VALVES FOLLOWING ADVERSE CONTAINMENT ENVIRONMENT.
- INSTALL TWO SAFETY GRADE SOLENOID VALVES ON EACH PORV TO VENT AIR ON SIGNAL FROM PROTECTION SYSTEM.
- UPGRADE CONTROL LOGIC, MOV BLOCK VALVE AND SOLENOID OPERATOR TO CLOSE FOLLOWING ADVERSE CONTAINMENT ENVIRONMENT.

MAIN FEEDWATER CONTROL SYSTEM

AREAS OF CONCERN

- ALL MAIN FEEDWATER LOST TO INTACT STEAM GENERATORS FOLLOWING SMALL FEEDLINE RUPTURE

- PRIMARY HOT LEG BOILING FOLLOWING FEEDLINE RUPTURE

MAIN FEEDWATER CONTROL SYSTEM

POTENTIAL SOLUTIONS

SHORT TERM

- INVESTIGATE WHETHER MAIN FEEDWATER CONTROL SYSTEM WILL FAIL OR OPERATE NORMALLY WHEN EXPOSED TO ADVERSE ENVIRONMENT

- TAKE CREDIT FOR OPERATOR ACTION PRIOR TO ALL SG'S REACHING LOW-LOW LEVEL TRIP SETPOINT FOLLOWING SMALL FEEDLINE RUPTURE

LONG TERM

- ISOLATE FEEDWATER CONTROL SYSTEM FROM THE ADVERSE ENVIRONMENT RESULTING FROM PIPE RUPTURES IN OTHER LOOPS

- REVISE LICENSING CRITERIA TO PERMIT BULK BOILING IN THE RCS PRIOR TO TRANSIENT "TURNAROUND"

- INSTALL NON-RETURN VALVE IN MAIN FEEDWATER LINE INSIDE CONTAINMENT. POSSIBILITY OF A SMALL FEEDLINE RUPTURE INSIDE CONTAINMENT BETWEEN CHECK VALVE AND STEAM GENERATOR REQUIRES QUALIFICATION OF STEAM FLOW TRANSMITTER TO PREVENT MALFUNCTION OF FEEDWATER CONTROL SYSTEM

STEAM GENERATOR POWER OPERATED RELIEF VALVE

CONTROL SYSTEM

AREAS OF CONCERN:

- MULTIPLE STEAM GENERATOR BLOWDOWN IN AN UNCONTROLLED MANNER

- LOSS OF TURBINE DRIVEN AUXILIARY FEEDWATER PUMP

- PRIMARY HOT LEG BOILING FOLLOWING FEEDLINE RUPTURE

STEAM GENERATOR PORV CONTROL SYSTEM

POTENTIAL SOLUTIONS

SHORT TERM

- INVESTIGATE WHETHER SG PORV CONTROL SYSTEM WILL OPERATE NORMALLY OR FAIL IN CLOSED POSITION WHEN EXPOSED TO ADVERSE ENVIRONMENT
- MODIFY OPERATING INSTRUCTIONS TO ALERT OPERATOR TO THE POSSIBILITY OF A CONSEQUENTIAL FAILURE IN THE SG PORV CONTROL SYSTEM CAUSED BY ADVERSE ENVIRONMENT. IF EVIDENT, CLOSE BLOCK VALVES IN RELIEF LINES

LONG TERM

- REDESIGN SG PORV CONTROL SYSTEM TO WITHSTAND ANTICIPATED ENVIRONMENT
- RELOCATE SG PORV'S AND CONTROLS TO AN AREA NOT EXPOSED TO THE ENVIRONMENT RESULTING FROM RUPTURES IN OTHER LOOPS
- INSTALL TWO SAFETY GRADE SOLENOID VALVES ON EACH PORV TO VENT AIR ON SIGNAL FROM THE PROTECTION SYSTEM, THEREBY ENSURING THAT THE VALVE WILL REMAIN CLOSED INITIALLY OR CLOSE AFTER OPENING
- INSTALL TWO SAFETY GRADE MOV'S IN EACH RELIEF LINE TO BLOCK VENTING ON SIGNAL FROM PROTECTION SYSTEM