



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

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MAY 1 1981

Docket Nos. 50-445
and 50-446

Mr. R. J. Gary
Executive Vice President and
General Manager
Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201



Dear Mr. Gary:

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION FOR COMANCHE PEAK STEAM
ELECTRIC STATION, UNITS 1 & 2

Enclosed is a request for additional information which we require to complete our evaluation of your application for operating licenses for Comanche Peak Steam Electric Station Units 1 and 2. This request for additional information is the result of our continuing review by the Instrumentation & Control Systems Branch, Materials Engineering Branch, Structural Engineering Branch, Reactor Systems Branch and Radiological Assessment Branch.

Please amend your FSAR to include the information requested in the Enclosure.

A timely response to these questions will enable us to proceed with this review. Should you have any questions concerning this request for additional information, please contact us.

Sincerely,

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

Enclosure:
Request for Additional
Information

cc w/encl:
See next page

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Mr. R. J. Gary
Executive Vice President and
General Manager
Texas Utilities Generating Company
2001 Bryan Tower
Dallas, Texas 75201

cc: Nicholas S. Reynolds, Esq.
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Westinghouse Electric Corporation
P. O. Box 355
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Citizens for Fair Utility Regulation
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Resident Inspector/Comanche Peak
Nuclear Power Station
c/o U. S. Nuclear Regulatory Commission
P. O. Box 38
Glen Rose, Texas 76043

ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION

COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 & 2

DOCKET NOS. 50-445 AND 50-446

032.0 INSTRUMENTATION & CONTROL SYSTEMS BRANCH

032.105 Failure Modes and Effects Analysis (FMEA) Interface Requirements

The SAR refers to the Westinghouse topical report WCAP-8584, "Failure Mode and Effects Analysis (FMEA) of the Engineered Safety Features Actuation System," as the supporting document on FMEA for the Comanche Peak station's engineered safety features equipments within the Westinghouse scope of supply. We have reviewed the WCAP-8584, and find the methodology and the general conclusions to be acceptable. However, in the Appendix B and C of the report, Westinghouse specifies the interface requirements for electrical circuit and instrument impulse lines separation involving other plant systems included in the balance of plant. The conformance to these requirements has not been addressed in the Comanche Peak FSAR. Please identify the difference between the Comanche Peak design and the Westinghouse specified interface requirement as described in the WCAP-8584. Justification should be provided for all deviations including the interface requirements to show that the ESF systems in the Comanche Peak design will perform their safety functions acceptably following any single failure.

032.106 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 (name of Plant). 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 (if the FSAR has not already been printed it may be incorporated in the original FSAR) and submitted to the NRC Office of Nuclear Reactor Regulations as a licensing submittal.

See Attachment 1.

032.107 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 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.

See Attachment 2.

032.108 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 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 identified 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.

See Attachment 3.

ATTACHMENT 1

Proposed Transmittal To Applicants Related To The Concern
That Simultaneously Initiated Failures Of Control Systems and Instrumentation
May Inhibit Safe Reactor Shutdown (Bulletin 79-27)

ALL OPERATING LICENSE APPLICANTS

SUBJECT: CONCERNS RELATED TO I&E BULLETIN 79-27

Operating reactors and some near term OL applicants have previously received I&E Bulletin 79-27 which is enclosed. The concerns which prompted the Bulletin apply to all OL applicants. If you have not already responded to the concerns of Bulletin 79-27, you are now requested to do so, but with two exceptions. First, the time for response will be determined on a case by case basis so that the 90 day limit in Item 4 is not applicable. Secondly, your reply should be made in the same way as other responses to requests for additional information by NRR.

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 occurred 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):

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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.

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

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 Time-pieces 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

ATTACHMENT 2

Proposed Transmittal To Applicants Related To The Concern Regarding
High Energy Line Breaks And Consequential Control System Failures

ALL OPERATING LICENSE APPLICANTS

SUBJECT: HIGH ENERGY LINE BREAKS AND CONSEQUENTIAL CONTROL SYSTEM FAILURES

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.

The NRC is 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 the 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. BWR applicants should consider analogous interactions as relevant to the BWR designs.

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 characteristics 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.

POOR ORIGINAL

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&O 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

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 interactions, 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 which in themselves are sets of low probability.

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.

ROD CONTROL SYSTEM

659

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

Control System Accident	Reactor Control	Pressurizer		Feedwater Control	Steam Generator Pressure Control	Steam Dump System	Turbine Control
		Pressure Control	Level 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 HIS 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

POOR ORIGINAL

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

POOR ORIGINAL

PRESSURIZER PORV CONTROL SYSTEM

POTENTIAL SOLUTIONS

SHORT TERM

- o INVESTIGATE WHETHER PRESSURIZER PORV CONTROL SYSTEM WILL FAIL OR OPERATE NORMALLY WHEN EXPOSED TO ADVERSE ENVIRONMENT.
- o 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

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

POOR ORIGINAL

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

POOR ORIGINAL

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 "TURNDOWN"
- 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

POOR ORIGINAL

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

POOR ORIGINAL

STEAM GENERATOR PORV CONTROL SYSTEM

POTENTIAL SOLUTIONS

SHORT TERM

- INVEST GATE 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

POOR ORIGINAL

ATTACHMENT 3

Proposed Transmittal To Applicants Related To The Concern That Common
Electrical Power Sources Or Sensors May Cause Multiple Control System Failures

ALL OPERATING LICENSE APPLICANTS

SUBJECT: 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 bases" 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 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 identified 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.

124.0 MATERIALS ENGINEERING BRANCH, MATERIALS APPLICATION SECTION

124.1 We have reviewed Section 10.2.3 of the Final Safety Analysis Report. Additional information is required to complete our evaluation relating to the design, assembly and operating conditions of the low pressure turbine discs. Past experience with similar equipment in the United Kingdom and more recently with Westinghouse turbines in the United States has revealed a propensity for stress corrosion cracking in discs which was not predictable. In order for the staff to assess the potential for stress corrosion cracking in the low pressure turbine discs, the following information will be required:

- A- What lubricant was used in the hub area of the discs for assembly.
- B- What are the similarities/differences between the discs in the Comanche Peak turbines and those used by Westinghouse.
- C- What are the operating temperatures in the bore area of the discs.
- D- Which disc or discs are exposed to a moisture level during operation that approximates the level of moisture present in cases of cracking.
- E- What are the calculated critical crack sizes and what is the method used to calculate that size.
- F- What capability for volumetric inspection of the disc hub areas is available to Comanche Peak.

130.0 STRUCTURAL ENGINEERING BRANCH

130.36 In a letter dated April 21, 1980, to all construction permit and operating license applicants, the NRC requested information on the presence of concrete masonry walls within seismic Category 1 structures and the mounting of piping supports and restraints on these walls. Your letter of July 29, 1980, responded there were "several masonry walls located within Category 1 structures, however, none of these walls are being used to support any safety grade components." The staff requires additional information about these masonry walls within the Category 1 structures (size, location, proximity to safety systems and components) and the effect of their postulated failure on safety systems and components during a seismic event.

If there are potential safety concerns created by collapse of any masonry walls, these walls need to be designed for the seismic environmental loads. In that case, describe their conformance with the attached "SEB Interim Criteria for Safety-Related Masonry Wall Evaluation."

SEB INTERIM CRITERIA FOR
SAFETY-RELATED MASONRY WALL EVALUATION

TABLE OF CONTENTS

1. General Requirements
2. Loads and Load Combinations
 - a. Service Load Conditions
 - b. Extreme Environmental, Abnormal, Abnormal/Severe Environmental, and Abnormal/Extreme Environmental Conditions
3. Allowable Stresses
4. Design and Analysis Considerations
5. Revision of Criteria
6. References

1. General Requirements

The materials, testing, analysis, design, construction and inspection related to the design and construction of safety-related concrete masonry walls shall conform to the applicable requirements contained in Uniform Building Code - 1979, unless specified otherwise, by the provisions in this criteria.

The use of other industrial codes, such as ACI-531, ATC-3 or NCMA is also acceptable. However, when the provisions of these codes are less conservative than the corresponding provisions of the interim criteria, their use should be justified on a case-by-case basis.

2. Loads and Load Combinations

The loads and load combinations shall include consideration of normal loads, severe environmental load, extreme environmental load, and abnormal loads. Specifically, for operating plants the load combinations provided in plant's FSAR shall govern. For operating license applications, the following load combinations shall apply (for definition of load terms, see SRP Section 3.8.4.II-3).

(a) Service Load Conditions:

- (1) $D + L$
- (2) $D + L + E$
- (3) $D + L + W$

If thermal stresses due to T_0 and R_0 are present, they should be included in the above combinations, as follows:

- (1a) $D + L + T_o + R_o$
- (2a) $D + L + T_o + R_o + E$
- (3a) $D + L + T_o + R_o + W$

Check load combination for controlling condition for maximum 'L' and for no 'L'.

(b) Extreme Environmental, Abnormal, Abnormal/Severe Environmental and Abnormal/Extreme Environmental Conditions

- (4) $D + L + T_o + R_o + E'$
- (5) $D + L + T_o + R_o + W_t$
- (6) $D + L + T_a + R_a + 1.5 P_a$
- (7) $D + L + T_a + 1.25 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.25 E + R_a$
- (8) $D + L + T_a + R_a + 1.0 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.0 E'$

In combinations (6), (7), and (8), the maximum values of P_a , T_a , R_a , Y_j , Y_r , and Y_m , including an appropriate dynamic load factor, should be used unless a time-history analysis is performed to justify otherwise. Combinations (5), (7) and (8) and the corresponding structural acceptance criteria should be satisfied first without the tornado missile load in (5) and without Y_r , Y_j , and Y_m in (7) and (8). When considering these loads, local section strength capacities may be exceeded under these concentrated loads, provided there will be no loss of function of any safety-related system.

Both cases of L having its full value or being completely absent should be checked.

3. Allowable Stresses

Allowable stresses provided in Chapter 24 of UBC-79, as supplemented by the following modifications/exceptions shall apply.

- (a) When wind or seismic loads (OBE) are considered in the loading combinations, no increase in the allowable stresses is permitted.
- (b) Use of allowable stresses corresponding to special inspection category shall be substantiated by demonstration of compliance with the inspection requirements of the NRC criteria.
- (c) No tension perpendicular to bed joints of either reinforced or unreinforced masonry walls is allowed, except in the evaluation of unreinforced masonry walls of operating plants. In such cases, the allowable values of UBC-79 can be used, if justified by test program or other means.
- (d) For load conditions, which represent extreme environmental, abnormal, abnormal/severe environmental, and abnormal/extreme environmental conditions the allowable working stresses may be multiplied by the factors shown in the following table:

<u>TYPE OF STRESS</u>	(1)	FACTOR
Axial or Flexural Compression		2.5
Bearing		2.5
Reinforcement stress except shear		2.0 but not to exceed 0.9 fy
Shear reinforcement and/or bolts		1.5
Masonry tension parallel to bed joint		1.5
Shear carried by masonry		1.0
Masonry tension perpendicular to bed joint		
for reinforced masonry		0
for unreinforced masonry(2)		1.0

Notes

(1) When anchor bolts are used, design should prevent facial spalling of masonry unit.

(2) See 3 (c).

4. Design and Analysis Considerations

(a) The analysis should follow established principles of engineering mechanics and take into account sound engineering practices.

(b) Assumptions and modeling techniques used shall give proper considerations to boundary conditions, cracking of sections, if any, and the dynamic behavior of masonry walls.

(c) Damping values to be used for dynamic analysis shall be those for reinforced concrete given in Regulatory Guide 1.61.

- (d) In general, for operating plants, the seismic analysis and Category I structural requirements of FSAR shall apply. For other plants, corresponding SRP requirements shall apply.
- (e) The analysis should consider both in-plane and out-of-plane loads.
- (f) Interstory drift effects should be considered.
- (g) In new construction, no unreinforced masonry wall is permitted, also all grout in concrete masonry walls shall be compacted by vibration.
- (h) For masonry shear walls, the minimum reinforcement requirements of ACI-531 or ATC-3 shall apply.
- (i) Special constructions (e.g. multiwythe, composite) or other items not covered by the code shall be reviewed on a case-by-case basis for their acceptance.
- (j) Licensees or applicants shall submit QA/QC information, if available, for staff's review.

In the event, QA/QC information is not available, a field survey and a test program reviewed and approved by the staff shall be implemented to ascertain the conformance of masonry construction to design drawings and specifications (e.g. rebar and grouting).

- (k) For masonry walls requiring protection from spalling and scabbing due to accident pipe reaction (Y_p), jet impingement (Y_j) and missile impact (Y_m), the requirements of SRP 3.5.3 shall apply. Any deviation from the SRP 3.5.3 shall be reviewed and approved on a case-by-case basis.

5. Revision of Criteria

The criteria will be revised, as appropriate, based on:

- (a) Design review meetings with the selected licensees and their A/E's.
- (b) Experience gained during review.
- (c) Additional information developed through testing and researches.

6. References

- (a) Uniform Building Code - 1979 Edition
- (b) Building Code Requirements for Concrete Masonry Structures
ACI-531 - 79 and Commentary ACI-531R - 79.
- (c) Tentative Provisions for the Development of Seismic Regulations
for Buildings - Applied Technology Council ATC 3-06.
- (d) Specification for the Design and Construction of Load-bearing Concrete
Masonry - NCMA August, 1979.
- (e) Trojan Nuclear Plant Concrete Masonry Design Criteria Safety
Evaluation Report Supplement - November, 1980.

331.0 RADIOLOGICAL ASSESSMENT BRANCH

331.22 Based on information contained in the draft document "Criteria for Utility Management and Technical Competence" it is our position that your station organization chain (Figure 13.1-4) should show that the radiation protection group is a separate organization from the Chemistry group and that the radiation protection manager report directly to the General Superintendent. Regulatory Guide 8.8 states that the RPM should be independent of the technical support division and should have direct recourse to the plant manager to resolve questions relating to the conduct of the radiation protection program.

Concurrent to the change request described above, Figure 13.1-4 should also show that Health Physics technicians and Chemistry technicians become separate groups and each report directly to their respective Radiation Protection and Chemistry group managers. This change request is also in accordance with the staff position in the aforementioned draft document. Alternatively, you should describe your methods for qualifying and maintaining qualification for technicians in both specialties, including training and experience requirements in ANSI 18.1, if combined specialties are to be maintained. Your FSAR and proposed Technical Specifications should therefore be revised accordingly.

212.0 REACTOR SYSTEMS BRANCH

212.137 For the low temperature overpressure protection (LTOP) address the following concerns:

- a. If the failure of a DC bus causes the failure of a letdown valve in the closed position and also fails one of the RCS PORVs in the closed position (i.e., not being able to open on demand), the letdown isolation will cause an overpressurization event. The only means available to mitigate this transient is the opening of the available PORV. However, if a single failure disables this PORV, the overpressurization event may exceed Appendix G limits. Discuss the applicability of the above scenario to Comanche Peak Steam Electric Station (CPSES).
- b. Discuss the capability of the PORVs operators to withstand an OBE and still be operable so that an overpressure protection for low temperature operation will be afforded.
- c. The response to question 212.16 indicates that LTOP system electronics will be tested during each refueling. However, our position is that LTOP system electronics should be tested prior to each shutdown. Please revise your response to reflect our position.

212.138 For the RHRS/SRP section 5.4.7 please discuss the following:

- a. Steam side PORVs operators are not qualified to operate following a seismic event. However, their operation is necessary to cool the RCS down to the RHR design conditions. Therefore, CPSES should demonstrate by operational testing that controlled plant cooldown can be accomplished by manual operation of these PORVs. The

criteria for this operational testing are: (i) a person should be dedicated for PORV operation, (ii) accessibility of the PORVs should be demonstrated, (iii) the ability to establish communications with the control room should be demonstrated, and (iv) the licensee should include in the plant test program a test that verifies the ability to achieve plant cooldown using manual operation of steam PORVs under natural circulation conditions.

- b. To depressurize the RCS down to the RHR design conditions the use of the PZR auxiliary spray and/or the PZR PORVs may be required. Please either (i) upgrade valve operators, air and power supplies for the above valves so that they can be remotely manipulated following an SSE, or (ii) ascertain that limited manual actions or repairs can be accomplished to correct the situation while the plant remains at hot-standby.
- c. Verify that CPSES meets R.G. 1.68 for PWRs, test and analysis for cooldown and boron mixing under natural circulation.
- d. Verify that CPSES meets R.G. 1.33 for PWRs, include specific procedure and information for cooldown under natural circulation.

212.139 Section 10.3 in the FSAR describes the isolation function of MSIVs. It is not clear that all MSIVs will close during or after a seismic event. Please explain in detail the operation of those valves and the qualification of the gas supply, solenoid power supply and valves operators.

212.140 In reference to section 6.3 in the FSAR:

- (a) A check valve in series with a MOV are in line between the RWST on one hand and the RHR pumps, the charging pumps, and the SI pumps on the other hand. Following automatic switchover of the RHR pumps suction to the sump, the operator is relied on to close those open suction MOVs. Table 6.3-5 (sh 3/4) takes credit for the isolation capability of the check valves in series with the MOVs. Explain how would the check valves prevent forward air flow through them (after the RWST is emptied) if one MOV failed to close.
- (b) Analyze and provide the sequence of operator actions as a function of time after the automatic switchover signal (i) with successful RWST isolation, and (ii) with one RHR suction MOV failing to close, and therefore, one RHR pump continues to take suction from the RWST at some flow rate. The operator may be distracted by trying to correct the failed valve position, hence, potentially compromise the integrity of the containment spray pumps.
- (c) Discuss the uncertainties in RWST level measurement (i.e., Lo, Lo-Lo, etc.) and the minimum amount of RWST water required for (i) maximum ESF pumps NPSH, and (ii) completion of manual switchover assuming continued RHR suction from the RWST.

212.141 Response to Q212.125 states that for a SLB the emergency procedures call for stopping the SI-on (among other signals) a 20% PZR level. However, the response to Q212.124 states that PZR level error can be as high as 25%. Please explain.

- 212.142 What auxiliary feedwater temperature is used in the Chapter 15 accidents analyses? Show that the temperatures are conservative, and demonstrate how will the conservatism be maintained (i.e., heaters in the condensate storage tank, qualifications of the heaters, etc.).
- 212.143 For the proposed sump testing for CPSES please:
- (a) Prior to conducting the test we recommend that you provide for our review a test proposal outlining the objectives of the test and the main characteristics of each general part of the test.
 - (b) After test completion, provide the test results and your proposals to modify the sumps characteristics, if necessary.
- 212.144 During our reviews of license applications we have identified concerns related to the containment sump design and its effect on long term cooling following a Loss of Coolant Accident (LOCA).
- These concerns are related to (1) creation of debris which could potentially block the sump screens and flow passages in the ECCS and the core, (2) inadequate NPSH of the pumps taking suction from the containment sump, (3) air entrainment from streams of water or steam which can cause loss of adequate NPSH, (4) formation of vortices which can cause loss of adequate NPSH, air entrainment and suction of floating debris into the ECCS and (5) inadequate emergency procedures and operator training to enable a correct response to these problems.

It is worthy to note that preoperational recirculation tests performed by utilities have consistently identified the need for plant modifications. The NRC has begun a generic program to resolve this issue. However, more immediate actions are required to assure greater reliability of safety system operation. We therefore require you take the following actions to provide additional assurance that long term cooling of the reactor core can be achieved and maintained following a postulated LOCA:

- a. Establish a procedure to perform an inspection of the containment, and the containment sump area in particular, to identify any materials which have the potential for becoming debris capable of blocking the containment sump when required for recirculation of coolant water. Typically, these materials consist of: plastic bags, step-off pads, health physics instrumentation, welding equipment, scaffolding, metal chips and screws, portable inspection lights, unsecured wood, construction materials and tools as well as other miscellaneous loose equipment. "As licensed" cleanliness should be assured prior to each startup.

This inspection shall be performed at the end of each shutdown as soon as practical before containment isolation.

- b. Institute an inspection program according to the requirements of Regulatory Guide 1.82, item 14. This item addresses inspection of the containment sump components including screens and intake structures.
- c. Develop and implement procedures for the operator which address both a possible vortexing problem (with consequent pump cavitation) and sump blockage due to debris. These procedures should address all likely scenarios and should list all instrumentation available to the operator (and its location) to aid in detecting problems which may arise, indications the operator should look for, and operator actions to mitigate these problems.

- d. Pipe breaks, drain flow and channeling of spray flow released below or impinging on the containment water surface in the area of the sump can cause a variety of problems; for example, air entrainment, cavitation and vortex formation.

Describe any changes you plan to make to reduce vortical flow in the neighborhood of the sump. Ideally, flow should approach uniformly from all directions.

- e. Evaluate the extent to which the containment sumps in your plant meet the requirements for each of the items previously identified; namely debris, inadequate NPSH, air entrainment, vortex formation, and operator actions.

The following additional guidance is provided for performing this evaluation:

- (1) Refer to the recommendations in Regulatory Guide 1.82 (Section C) which may be of assistance in performing this evaluation.
- (2) Provide a drawing showing the location of the drain sump relative to the containment sumps.
- (3) Provide the following information with your evaluation of debris:
 - (a) Provide the size of openings in the fine screens and compare this with the minimum dimensions in the pumps which take suction from the sump, the minimum dimension in any spray nozzles and in the fuel assemblies in the reactor core or any other line in the recirculation flow path whose size is comparable to or smaller than the sump screen mesh size in order to show that no flow blockage will occur at any point past the screen.

- (b) Estimate the extent to which debris could block the trash rack or screens (50 percent limit). If a blockage problem is identified, describe the corrective actions you plan to take (replace insulation, enlarge cages, etc.).