Docket
50-3

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

GL-79-49

October 5, 1979

TO ALL POWER REACTOR LICENSEES

SUBJECT: SUMMARY OF MEETINGS HELD ON SEPTEMBER 18-20, 1979 TO DISCUSS A POTENTIAL UNREVIEWED SAFETY QUESTION ON INTERACTION BETWEEN NON-SAFETY GRADE SYSTEMS AND NSSS SUPPLIED SAFETY GRADE SYSTEMS (I&E INFORMATION NOTICE 79-22)

I. Introduction

A series of meetings was held with all four light water reactor vendors and the corresponding utilities to discuss the effect of I&E Information Notice 79-22 on nuclear power plant owners. I&E Information Notice 79-22, issued on September 14, 1979, notified the nuclear industry of a potential unreviewed safety question at Public Service Electric and Gas Company's Salem Unit 1 nuclear facility. The meetings were held in the Bethesda offices of the NRC according to the following schedule:

Westinghouse - September 18, 1979
Combustion Engineering - September 19, 1979
Babcock and Wilcox - September 20, 1979; a.m.
General Electric - September 20, 1979; p.m.

The Nuclear Regulatory Commission staff was seeking additional information from operators of all nuclear power plants on a potential unreviewed safety question involving malfunctions of control equipment under accident conditions. This equipment consists of electrical components used for reactor and plant control under normal operating conditions.

Some of this equipment could be adversely affected by steam or water from certain pipe breaks, such as in the main steam line inside or outside plant containment buildings. The consequences of a control system malfunction could result in conditions more or less severe than those previously analyzed. The NRC staff intends to determine the degree to which the validity of previous safety reviews are affected and whether changes in design or operating procedures will be required.

II. Background

As part of the Westinghouse Environmental Qualification Program, IEEE 323-74 has been reviewed, in particular, sections dealing with environmental

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interactions. Westinghouse design philosophy is that if a component is necessary to function in order to protect the public, it is "protection" grade. Should a non-protection grade component perform normal action in response to system conditions, it must be shown to have no adverse impact on protection grade component response. If a component did not receive a signal to change state, it was assumed to remain "as is". Part of the environmental qualifications require the demonstration that severe environments will not cause common failure of "protection" grade components. An outgrowth of the environmental qualification program review was a determination if the severe environment can cause a failure of a non-protection grade component that was previously assumed to remain "as is" and alter the results of the design basis analysis.

Westinghouse formed an Environmental Interaction Committee whose charter was to identify, 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 malfunction or failure could exceed the safety limits previously satisfied by accident analyses presented in Westinghouse plants' SARs. The Committee was also to establish, for any adverse interactions identified, recommendations to resolve the issue. The assumed ground rules for the investigations performed by Westinghouse are enumerated on page five of Enclosure 2. The investigation resulted in a compilation of potential control system consequential failures (due to environmental considerations) which affected plant safety analyses. The investigation considered seven accident scenarios and seven control systems interactions in a matrix form, as shown on page 6 of Enclosure 2. The accidents are: 1) small steam line rupture; 2) large steam line rupture; 3) small feedline rupture; 4) large feedline rupture; 5) small LOCA; 6) large LOCA; and, 7) rod ejection. The control systems are: 1) reactor control; 2) pressurizer pressure control; 3) pressurizer level control; 4) feedwater control; 5) steam generator pressure control; 6) steam dump system control; and 7) turbine control.

The investigations identified potential significant system response interactions in the:

a. steam generator power operated relief valve control system;

b. pressurizer pressure control system;

c. main feedwater control system; and,

d. rod control system.

III. Discussion

A. The first in the series of meetings was with Westinghouse and utilities that own Westinghouse reactors. The meeting was attended by seventy (70) persons representing the NRC, PSE&G along with nine other utilities, Westinghouse and the other three light water reactor vendors, utility owner groups, four A/E consultants, the ACRS, AIF and EPRI. The list of attendees is presented as Enclosure 1.

Westinghouse's presentation is included as Enclosure 2.

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 2, page 6. 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 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; and,

e. prior commitments made by utility to the NRC.

The Westinghouse analysis did not consider plant operators as part of the control systems nor was the time allotted for operator "inaction" considered. The assumed operator action times, as stipulated in plant analysis, were used without modification. Equipment in a control system or part of a control system was assumed to fail as a system in the most adverse direction for conservatism. Westinghouse stated that the possible matrix interactions will remain for all of their plants and the interactions may be removed 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 application; and,

e. prior commitments made by utility to the NRC.

It should be noted that Westinghouse only analyzed accidents and not transients.

Further, long-term investigations may be required to analyze the transient cases. Initial conditions and assumptions are shown on pages 5, 7, 9, 14, 15, 22, 23, 27, 28, 33, 37 and 38.

Westinghouse presented their analyses for the four control systems identified as follows:

A. Steam Generator Power Operated Relief Valve Control System

The areas of concern for this system are:

1. multiple steam generator blowdown in an uncontrolled manner;

2. loss of turbine driven auxiliary feedwater pump; and,

3. primary hot leg boiling following feedline rupture.

The assumptions used are presented on page 15 of Enclosure 2. Potential solutions to the Steam Generator PORV Control System interaction problems were presented as both short term and long term. The short-term solutions are to:

1. investigate whether the SG PORV Control System will operate normally or fail in a closed position when exposed to an adverse environment; and,

2. modify the operating instructions to alert operators to the possibility of a consequential failure in the SG PORV Control System caused by an adverse environment.

If evident, close block valves in the relief lines.

The long-term solutions are:

1. redesign the SG PORV Control System to withstand the anticipated environment;

2. relocate the SG PORVs and controls to an area not exposed to the environment resulting from ruptures in the other loops;

3. install two safety grade solenoid valves in each PORV to vent air on a signal from the protection system, thereby ensuring that the valve will remain closed initially or will close after opening; and,

4. install two safety grade MOVs in each relief line to block venting on signal from the protection system.

Westinghouse presented similar analyses for the other three control systems along with the assumptions, areas of concern and potential solutions. These are presented in Enclosure 2.

a. Steam Generator PORV Control System pp. 14-21, Enclosure 2.

b. Main Feedwater Control System pp. 22-26, Enclosure 2.

c. Pressurizer PORV Control System pp. 27-32, Enclosure 2.

d. Rod Control System pp. 37-42, Enclosure 2.

At the end of Westinghouse's presentation, the NRC staff caucused to discuss their future plans and actions. When all attendees reconvened the meeting was opened to discussions of the impact of the NRC 10 CFR 50.54(f) letter, vendor and utility plans, and staff plans.

Westinghouse stated that they would establish an action plan along the guidelines of NUREG-0578. Westinghouse also stated that their investigations were carried further than FSAR analyses and they would need to evaluate consequential failures on a realistic basis; this evaluation may eliminate some problems. Westinghouse stated that their investigations are lower probability subsets of SAR analyses which in themselves are sets of low probability. Westinghouse expressed doubts that a conclusive determination can be made of the qualification status of all of the involved equipment in 20 days.

Robinson plant representatives noted that their secondaries are open and therefore breaks outside of containment present no problem. They indicated their basic approach to answering the 20-day letter will be to follow the short-term Westinghouse recommendations.

Representatives of Salem also stated that their intent is to follow the short-term Westinghouse recommendations to satisfy the request of the 20-day letter.

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 established guidelines stating their position on the problem along with their recommendations.

B. The second in the series of meetings was held with Combustion Engineering and utilities that own CE's reactors. The meetings were attended by 52 persons representing the NRC, all four light water reactor vendors, five utilities, various consultants, the ACRS, AIF and EPRI. The list of meeting attendees is presented as Enclosure 3.

They explained the concerns presented by Westinghouse and the four control systems that could be affected as a result of the adverse environment of a high energy pipe break and whose consequential failure could invalidate the accident analysis of plant SARs.

Previous analyses did not specifically take control systems into account in accident scenarios and the systems were considered passive in the analyses. The staff explained its earlier understanding regarding control systems interaction in accidents as one in which the accidents were expected to be quick and the control systems did not have the time to contribute significantly to the consequences. If most of industry reviewed their accident analyses according to the staff position on control system contribution, then a need does, in fact, exist to further the scope of accident analyses to include potential consequential failure modes of the control systems,

Industry representatives stated that in the allotted 20 days, they could only skim the surface in accident review with the inclusion of control system interactions. An initial approach would be of a mechanistic nature to determine what control system would be involved and what type of hardware would be necessary to initiate fixes rather than using an analytical approach to determine the contribution of control systems on accident consequences.

Combustion Engineering's plans are to identify the control systems that could cause interactions and then look at resolutions to the problem on a per plant basis since some solutions are plant dependent. The action process to be followed is presented as Enclosure 4 and is as follows:

1. Identify those non-safety related control systems, inside and outside containment, whose malfunction could adversely affect the accident or transient when subjected to an adverse environment caused by a high energy pipe break.

2. Determine the limiting malfunctions and their impact during high energy pipe breaks for those control systems.

3. Determine the short term and long term corrective actions.

Combustion Engineering stated that in their plants, operaton of control systems is not required in order to mitigate the consequences of the transients analyzed in Chapter 15. The analyses in Chapter 15 include the assumption that these control systems respond normally to each transient and that their operational mode is that which would be most adverse for the transient under consideration. The consequences produced by any credible malfunction of these control systems would be less severe than any which would be produced by the mechanisms considered as causes of the transients analyzed in Chapter 15.

Some discussion followed dealing with the failure modes of control system and whether the failure mode is in the most adverse direction or in the design direction. Resolution of this topic was not obtained but will be addressed on a plant-by-plant basis.

Again utilities presented their concerns over the 20-day letter and what is expected of them in this time frame. They stated that in order to follow the directions of the letter all components would have to be reviewed to determine if the non-safety grade system failure mode would aggrevate the accident consequences.

C. The third in the series of meetings was held with Babcock and Wilcox and utilities that own B&W reactors. The meetings were attended by fifty-six (56) persons representing the NRC, reactor vendors, seven utilities, various consultants, the AIF and EPRI along with the Union of Concerned Scientists.

The NRC staff explained the background history leading up to the "20-day" letter and the fact that they consider the problem a generic one common to all LWRs.

The utility representatives stated that they will answer the letter themselves without specific participation of the owners group, which they consider germane only to TMI-2 related subejct. Most of the work, the detailed action plans of which have not yet been established, will be performed by the various utilities and their architect engineers and consultants, with generic material supplied by the reactor vendor.

The utility representatives understand the environment to be plant specific and will use that environment in their analyses for control system failure. The system failure will include not only component failure but also failure of transducers, wires, and hot and cold shorts. The adequacy of fixes for the long-term and the combination of consequential failures is not expected to be considered in the allotted 20 days.

Babcock and Wilcox representatives stated that in the past, evaluations were performed for the sequence of events leading up to the trip, a time of about 5 to 10 seconds. Prior to that time the control systems have no effect on the accident sequence or consequence. Failure of control systems will be investigated in view of the severity of the possible accident; if the control system failure increases the consequences, then that system will be considered.

The approach proposed by B&W and the utilities is outlined in Enclosure 6 and is as follows:

1. Evaluate the impact of IE 79-22 on licensing basis accident analyses.

2. Identify accidents which will yield the adverse environment.

3. Define inputs and responses used.

4. Verify conclusions and justify continued operation.

The utilities will alert the plant operators to the potential failure of the plant control systems in total or in providing correct information. The abnormal and emergency procedures will be reviewed to determine how failure of non-safety grade systems or improper information will affect the prescribed operator action.

D. The fourth and final in the series of meetings was with General Electric and utilities that own GE reactors. The meeting was attended by 52 people representing the NRC, three reactor vendors, nine utilities, architect engineers, consultants, and the AIF. The list of attendees is presented as Enclosure 7.

The NRC staff presented highlights of the previous meetings and the concerns identified by Westinghouse. The staff stated that a more sophisticated evaluation of the accident analysis is required to see if the course and consequences of the accident are altered by consequential failure of non-safety grade control systems.

General Electric representatives stated that their analyses have considered high energy pipe breaks in many locations and are more detailed since BWRs have a larger number of pipes inside and outside containment carrying radioactive liquids. The BWR leak detection capabilities are correspondingly greater. Special attention is given to separation criteria viz., various systems are in separate cubicles and inside a class 1 secondary as well as primary containment.

The high energy line break is not considered a problem. In 1970, Dresden 2 experienced opening of a safety valve and a resulting 10 psi and 340 F environment. The equipment was examined and the qualifications were subsequently upgraded.

GE representatives stated that they performed sensitivity studies on their non-safety grade systems to determine if they are heavily relied upon during an accident. The studies revealed that there was no heavy dependence upon those systems.

It must be noted that the GE non-safety grade system and components comprise only approximately 25% of a typical plant total. The utilities will perform their own analyses on BOP systems to satisfy the requirements of the "20-day" letter.

IV. NRC Comments

The NRC staff stated that they understood the requests by the nuclear industry regarding position and direction on the request found in the NRC 10 CFR 50.54(f) letter dated September 17, 1979 but would wait until the conclusion of the scheduled meetins with all four light water reactor vendors. The staff further stated a Commission Information paper would be prepared discussing the staff's judgment regarding the magnitude of the concern and the appropriateness of industry's response for resolution of the problem.

More specific staff statements were made in terms of generating a plant specific matrix of potential environmental interactions of control system for each plant. The NRC requested that they be notified of further analyses and the individuals that will perform them, either reactor vendors, the owners groups, or the individual utilities.

The NRC noted that at this time, it is not evident which utilities are faced with what environmental interaction problems. The effects of implementing all of the Westinghouse recommended short-term "fixes" may be contradicted by other sequences. Multiple failure analyses could be performed but this would take months and could not possibly be ready in 20 days.

The NRC recommended that utilities check if qualified equipment is in place to determine the magnitude of a total qualification program.

The staff advised the utilities to check the validity of their operating procedures in light of the steam environment around various components and the reliability of certain control valves in question; also, use should be made of all information available in files of vendors, A/Es, and consultants dealing with the problem.

The staff is aware that sufficient time is not available to identify all of the potential interactions but some of the more obvious ones must be reviewed. For example, some utilities might choose to operate their plants in the interim period using a manual rod mode instead of the preferred automatic mode; also, the PORV block valves may be operated in the closed position. The determination of what systems are suspect and the possible 20-day solutions must be answered by each individual utility according to their plant design. Operator training would have to be stressed to make the operators aware that potential consequential failures may exist that would mask the real failure and give erroneous readings.

The staff stated that for the "20-day" letter response, the utilities should use engineering judgment and evaluations instead of detailed analyses that would be time consuming and might limit the utility response to a limited number of evaluations.

V. Conclusions

The staff indicated that there were three possible options that could be followed in providing a short-term response.

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. It must be noted that in this situation some components that are relied upon to work properly might be wiped out by consequential failures under certain conditions and accident sequences.

3. The "worst case" plant should be selected and a bounding analysis performed to determine the time frame available for qualification of equipment.

The staff reiterated the presented recommendations, possible interim solutions that are plant specific, and in addition, requested the following:

1. Identify equipment and control systems which are either needed to mitigate the consequences of a high energy pipe break or could adversely affect the consequences of these events.

2. Check the locations, expected environment, and environmental qualifications of the equipment and control system identified in part 1.

3. If some of these are found not be qualified for the environmental conditions, propose an appropriate fix, i.e., design change, change in operating procedures, acceptable consequences argument based on your evaluation, etc. Provide a schedule for the proposed fix.

George Kuzmycz

George Kuzmycz, Project Manager
Division of Project Management

Mr. William J. Cahill, Jr.
Consolidated Edison Company of New York, Inc.

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cc: White Plains Public Library
100 Martine Avenue
White Plains, New York 10601

Joseph D. Block, Esquire
Executive Vice President
Administrative
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Edward J. Sack, Esquire
Law Department
Consolidated Edison Company
of New York, Inc.
4 Irving Place
New York, New York 10003

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N.W.
Washington, D. C. 20005

Dr. Lawrence R. Quarles
Apartment 51
Kendal at Longwood
Kennett Square, Pennsylvania 19348

Theodore A. Rebelowski
U. S. Nuclear Regulatory Commission
P. O. Box 38
Buchanan, New York 10511

ENCLOSURE 1
MEETING ATTENDEES

NRC
D. Ross
D. Eisenhut
J. Heltemes
G. Kuzmycz
J. Guttmann
W. Jensen
S. Israel
G. Lainas
V. Benaroya
R. Woodruff
A. Dromerick
B. Smith
M. Grotenhuis
~~A.~~ Schwencer
P. Norian
F. Orr
F. Odar
T. Dunning
W. Gammill
S. Salah
J. Stolz
Z. Rosztoczy
T. Novak
J. Beard
M. Cliramak
D. Tondi
C. Berlinger
L. Kintner
J. Mazetis
K. Mahan
D. Thatcher
J. Burdoin
P. Mathews
M. Lynch
R. Scholl

WESTINGHOUSE
K. Jordan
R. Sero
R. Steitler
G. Lang
G. Butterworth
V. Sluss
F. Noon

PSE&G Co.
F. Librizzi
R. Mittl
J. Wroblewski
J. Gogliardi
P. Moeller
R. Fryling

VENDORS
N. Shirley - G.E.
W. Lindblad - G.E. Portland
R. Borsun - B&W
C. Brinkman - C.E.

UTILITIES
D. Waters - CP&L
M. Scott - Con. Ed.
G. Copp - Duke Power
N. Mathur - PASNY
J. Barnsberry - S. Cal. Ed.
K. Vehstedt - AEPSC
R. Shoberg - AEPSC
E. Smith - VEPCO
T. Peebles - VEPCO
P. Herrmann - Southern Co. Services

W. House - Bechtel
T. Martin - Nutech
J. McEment - Stafeo
M. Wetterhahn - Conner, Moore & Corber
K. Layer - BBR
E. Igne - ACRS
P. Higgins - AIF
R. Leyse - EPRI

ENCLOSURE 2

W ENVIRONMENTAL QUALIFICATION ACTIVITIES (IEEE 323-74)

- SEISMIC TESTS
- AGING PROGRAM
- ENVIRONMENTAL ENVELOPES
- INSTRUMENT ACCURACIES
- ENVIRONMENTAL INTERACTIONS

- HISTORY

  ACRS CONCERNS

  NRC ACTIONS/QUESTIONS

  AREAS: SYSTEMS INTERACTIONS
  INTERFACE CRITERIA (STANDARDIZATION)
  HELB PROTECTION

INDUSTRY DESIGN PHILOSOPHY

IF A COMPONENT IS NECESSARY TO FUNCTION IN ORDER TO PROTECT THE PUBLIC, IT IS "PROTECTION" GRADE. SHOULD A NON-PROTECTION GRADE COMPONENT PERFORM NORMAL ACTION IN RESPONSE TO SYSTEM CONDITIONS, IT MUST BE SHOWN TO HAVE NO ADVERSE IMPACT ON PROTECTION GRADE COMPONENT RESPONSE. IF A COMPONENT DID NOT RECEIVE A SIGNAL TO CHANGE STATE, IT WAS ASSUMED TO REMAIN "AS IS".

- ENVIRONMENTAL QUALIFICATION

  DEMONSTRATE THAT SEVERE ENVIRONMENT WILL NOT CAUSE COMMON FAILURE OF "PROTECTION" GRADE COMPONENTS

- NEW QUESTION TO BE ADDRESSED

  CAN THE SEVERE ENVIRONMENT CAUSE A FAILURE OF A NON-PROTECTION GRADE COMPONENT THAT WAS PREVIOUSLY ASSUMED TO REMAIN "AS IS" AND ALTER THE RESULTS OF THE DESIGN BASIS ANALYSES?

- REGULATORY ENVIRONMENT TODAY

  - POST-TMI/2 REACTION
  - NUREG-0578
  - ACRS PRESENTATIONS BY NRC

ENVIRONMENTAL INTERACTION COMMITTEE

INTERACTION TO BE ADDRESSED:

A CONSEQUENTIAL FAILURE OF A CONTROL SYSTEM DUE TO AN ADVERSE ENVIRONMENT INSIDE OR OUTSIDE CONTAINMENT FOLLOWING A HIGH ENERGY RUPTURE WHICH NEGATES A PROTECTIVE FUNCTION PERFORMED BY A SAFETY GRADE SYSTEM.

COMMITTEE CHARTER:

FOR ALL HIGH ENERGY LINE BREAKS AND POSSIBLE LOCATIONS, IDENTIFY CONTROL SYSTEMS THAT COULD BE AFFECTED AS A RESULT OF THE ADVERSE ENVIRONMENT AND WHOSE CONSEQUENTIAL MALFUNCTION OR FAILURE COULD INVALIDATE THE ACCIDENT ANALYSIS PRESENTED IN THE PLANT SAR. FOR ANY ADVERSE INTERACTIONS IDENTIFIED, ESTABLISH RECOMMENDATIONS TO RESOLVE THE ISSUE.

ASSUMED GROUNDRULES FOR INVESTIGATION

- CONTROL SYSTEMS (OR PARTS) NOT SUBJECT TO HIGH ENERGY LINE BREAK ENVIRONMENT
  - EQUIPMENT ASSUMED TO REMAIN 'AS IS' OR OPERATE WITHIN SPECIFIED ACCURACY, WHICHEVER IS MORE SEVERE
- RANDOM FAILURES IN THE CONTROL SYSTEM ARE NOT POSTULATED TO OCCUR COINCIDENT WITH THE STUDIED EVENT
- PROTECTION SYSTEMS ARE ASSUMED TO FUNCTION CONSISTENT WITH REQUIREMENTS OF IEEE-279-1971 (INCLUDING SINGLE FAILURE IN PROTECTION SYSTEM).
- OPERATOR ACTION TIME ASSUMED CONSISTENT WITH SAR ASSUMPTIONS
- CONTROL SYSTEMS (OR PARTS) SUBJECT TO HIGH ENERGY LINE BREAK ENVIRONMENT
  - UNQUALIFIED EQUIPMENT ASSUMED TO FAIL IN MOST ADVERSE DIRECTION
  - QUALIFIED EQUIPMENT ASSUMED TO REMAIN 'AS IS' OR OPERATE WITHIN SPECIFIED ACCURACY.

(QUALIFIED = DESIGN CAN BE SHOWN TO BE COMPATIBLE WITH POSTULATED ENVIRONMENT)

Control System / Accident	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

IDENTIFIED POTENTIAL CONCERNS

- SYSTEMATIC INVESTIGATION IDENTIFIED POTENTIAL ENVIRONMENTAL INTERACTION IN:

  - STEAM GENERATOR POWER OPERATED RELIEF VALVE CONTROL SYSTEM
  - PRESSURIZER PRESSURE CONTROL SYSTEM
  - MAIN FEEDWATER CONTROL SYSTEM
  - ROD CONTROL SYSTEM

  INTERACTION MODE AND POSSIBLE FIXES IDENTIFIED

- INVESTIGATION TO DATE LIMITED TO IMPACT OF ADVERSE ENVIRONMENT ON CONTROL SYSTEMS AND POTENTIAL CONSEQUENTIAL EFFECTS

- REMAINING AREA UNDER INVESTIGATION BY COMMITTEE IS THE EFFECT OF ADVERSE ENVIRONMENTS ON VALVE OPERATORS ASSOCIATED WITH 'INACTIVE' VALVES LOCATED IN PROTECTION SYSTEMS

  - NO OPERABILITY REQUIREMENT ON VALVE THEREFORE NO QUALIFICATION SPECIFIED FOR VALVE OR OPERATOR
  - HOWEVER, ACCIDENT ANALYSIS ASSUMES VALVE STAYS 'AS IS'

PLANT APPLICABILITY OF CONCERNS & RECOMMENDATIONS

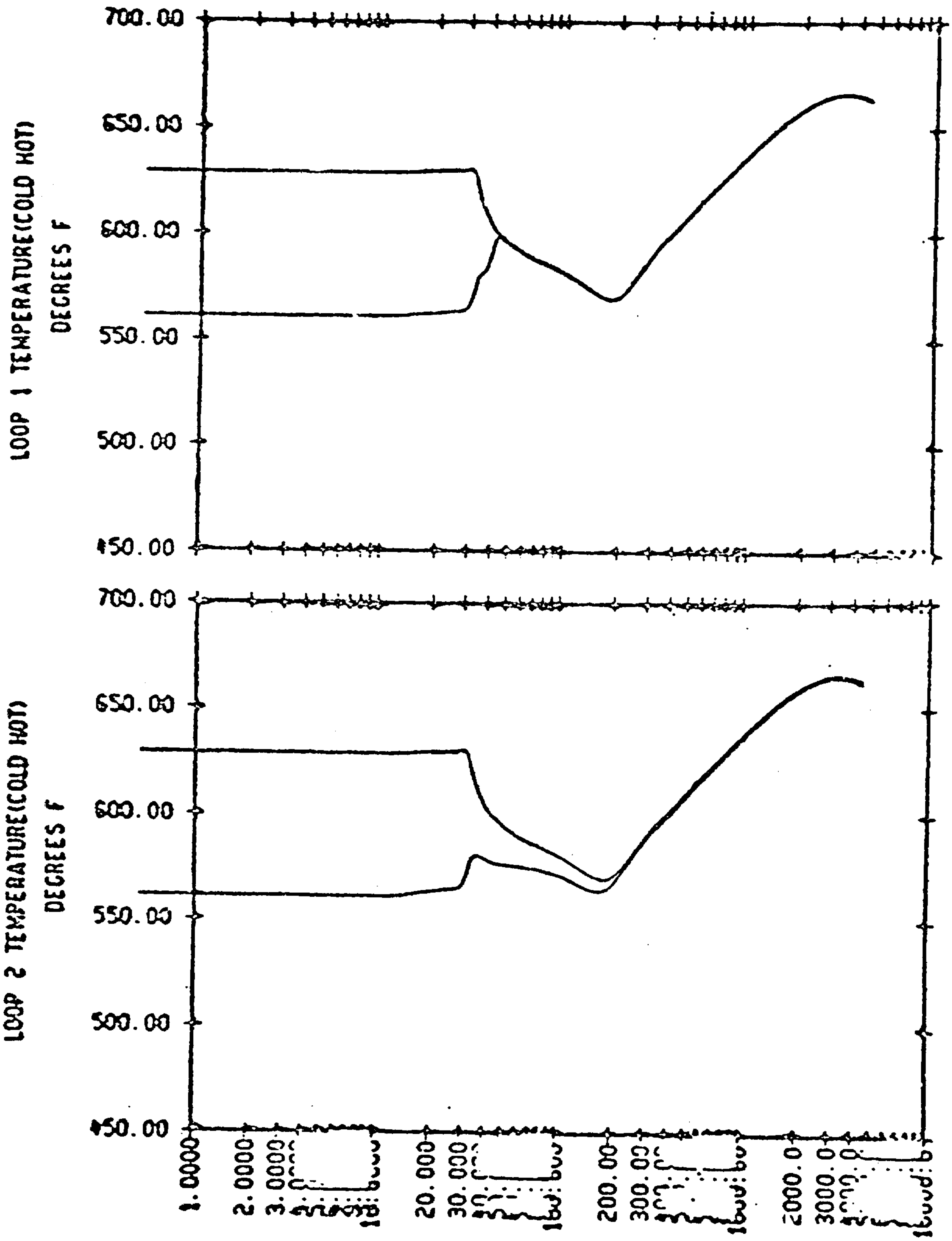
- IDENTIFIED CONCERNS ARE NOT GENERIC SINCE IMPACTED BY MANY PLANT SPECIFIC DESIGNS:
  - SYSTEM LAYOUT
  - TYPE OF EQUIPMENT UTILIZED
  - QUALIFICATION STATUS OF EQUIPMENT UTILIZED
  - DESIGN BASIS EVENTS CONSIDERED FOR LICENSE APPLICATION
  - COMMITMENTS MADE BY UTILITY TO NRC
- RECOMMENDATIONS
  - UTILITY REVIEW OF IDENTIFIED CONCERNS WITH RESPECT TO PLANT CHARACTERISTICS AND LICENSING COMMITMENTS
  - FOLLOW-UP BY UTILITIES TO CONSIDER POTENTIAL FOR ADVERSE ENVIRONMENTAL INTERACTION FROM CONTROL SYSTEMS AS YET UN-REVIEWED BY WESTINGHOUSE

SAR FEEDLINE RUPTURE EVENT

- MAIN FEEDLINE RUPTURE OCCURS DOWNSTREAM OF FEEDLINE CHECK VALVE
- MAIN FEEDWATER SPILLS OUT RUPTURE
- SECONDARY INVENTORY SPILLS THROUGH RUPTURED FEEDLINE
- PRIMARY BEGINS HEATUP DUE TO PARTIAL LOSS OF LOAD
- REACTOR TRIP OCCURS ON LOW LOW STEAM GENERATOR WATER LEVEL IN RUPTURED STEAM GENERATOR
- AUXILIARY FEEDWATER PUMPS INITIATED ON LOW LOW STEAM GENERATOR WATER LEVEL. TURBINE TRIP OCCURS ON REACTOR TRIP
- PRIMARY BEGINS COOLDOWN WHILE HEAT REMOVAL CAPABILITY OF SECONDARY INITIALLY EXCEEDS DECAY HEAT GENERATED IN CORE
- PRIMARY BEGINS HEATUP WHEN SECONDARY INVENTORY NOT CAPABLE TO REMOVE DECAY HEAT
- STEAM GENERATORS IN INTACT LOOPS BEGIN REPRESSURIZING DUE TO AUTOMATIC OR MANUAL MAIN STEAMLINE ISOLATION
- STEAM DRIVEN AUXILIARY FEEDWATER PUMP OBTAINS STEAM FROM AT LEAST TWO MAIN STEAMLINES. STEAMLINE ISOLATION INSURES SOURCE OF STEAM SUPPLY
- PRIMARY CONTINUES TO HEATUP UNTIL AUXILIARY FEEDWATER BEING INJECTED INTO INTACT STEAM GENERATORS IS SUFFICIENT TO REMOVAL DECAY HEAT

WESTINGHOUSE PROPRIETARY CLASS 2

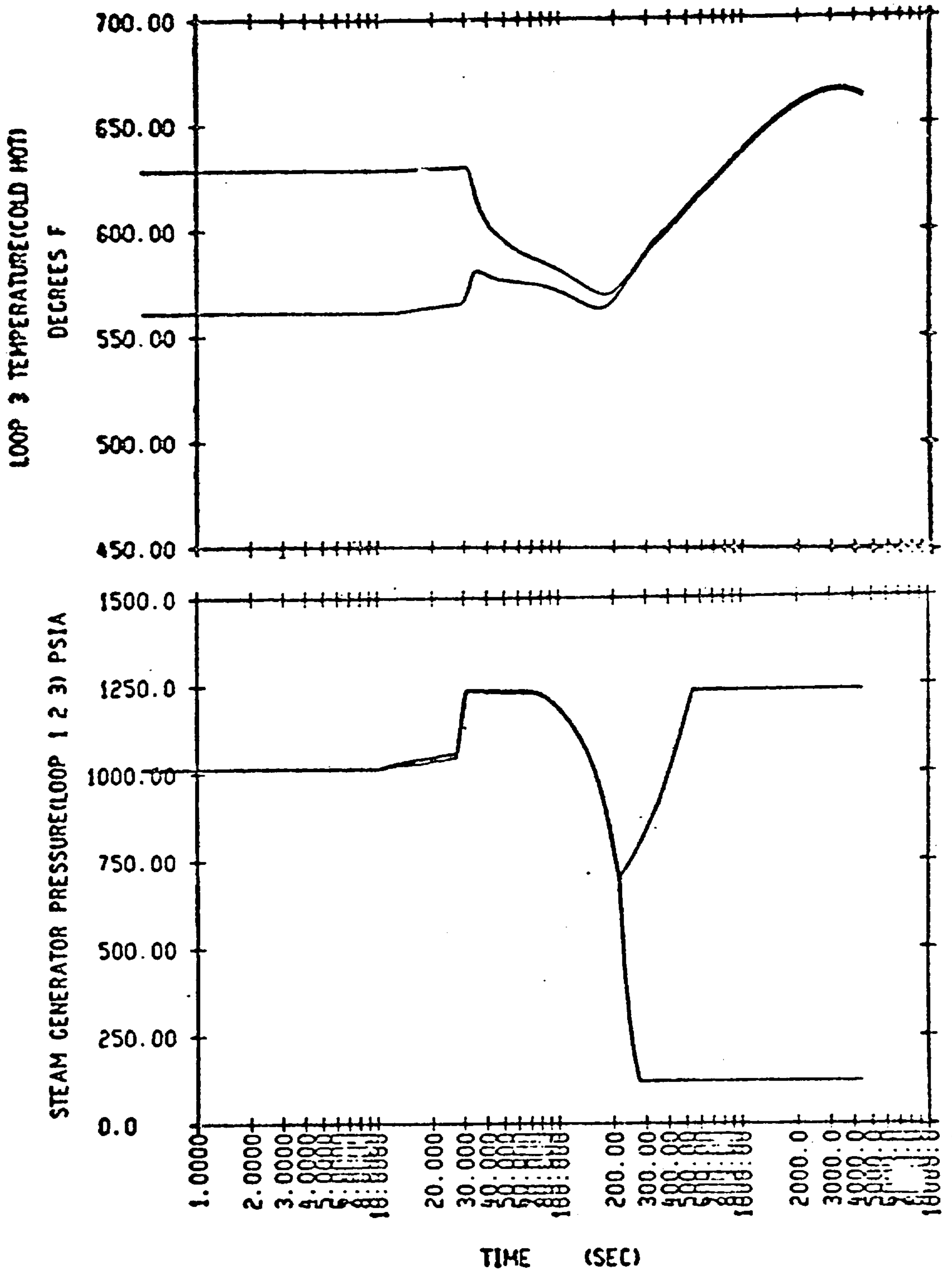
WCAP- 9230



5-10 Primary Temperature Transients Following a Feedline Rupture Assuming Worst Case Initial Conditions and Assumptions for a 3-Loop Plant

WESTINGHOUSE PROPRIETARY CLASS 2

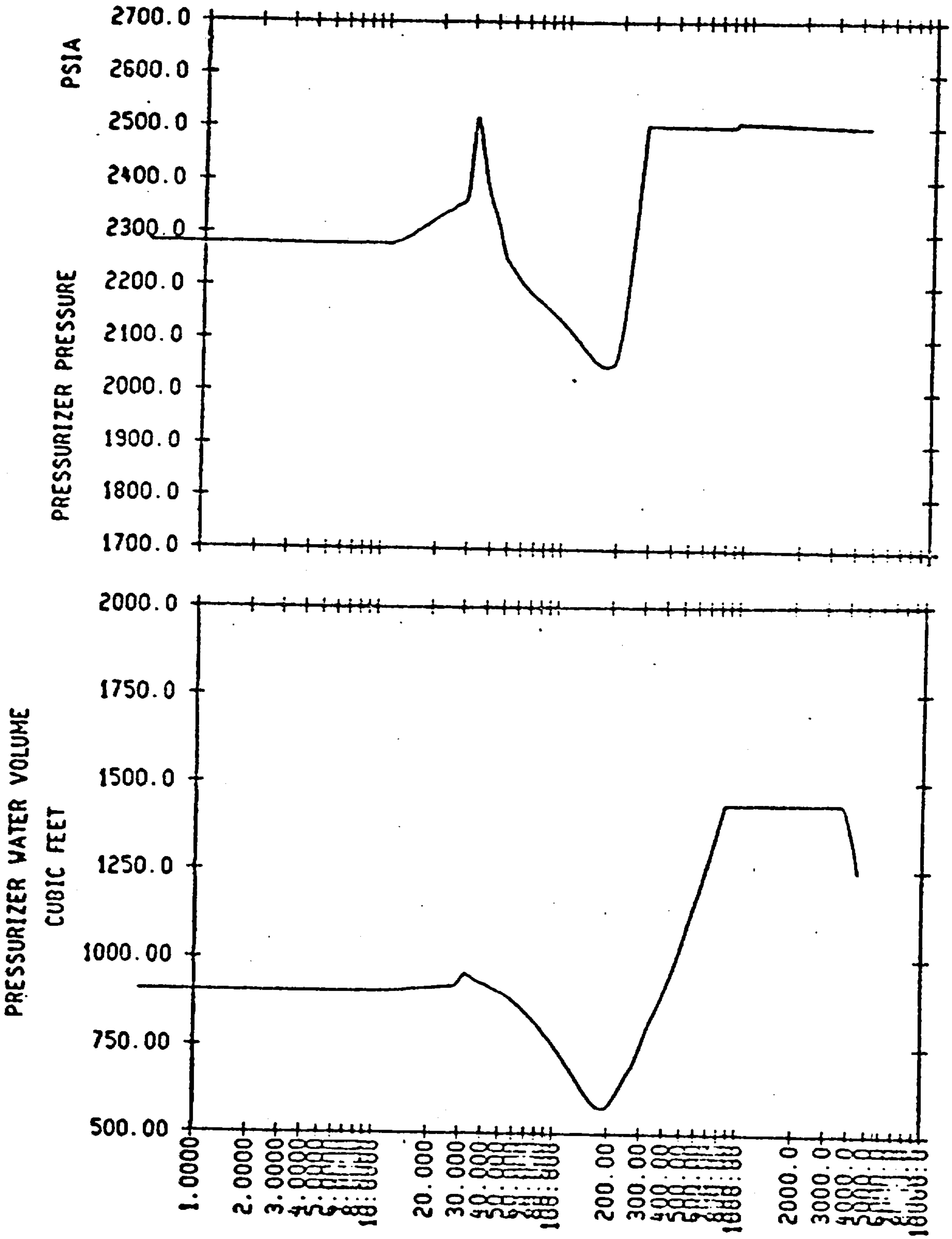
WCAP-9230



5-11 Primary Temperature and Steam Generator Pressure Following a Feedline Rupture Assuming Worst Case Initial Conditions and Assumptions for a 3-Loop Plant

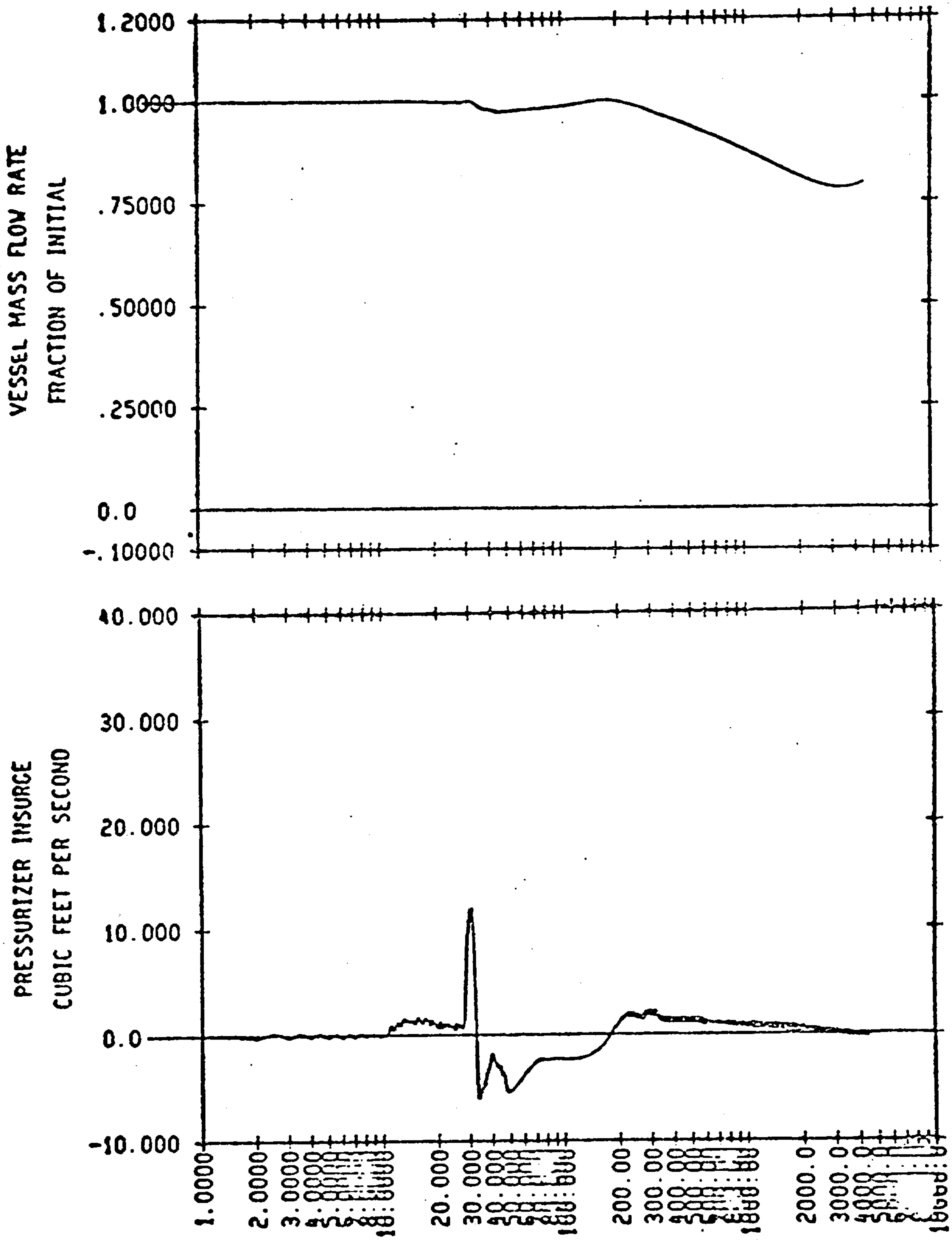
WESTINGHOUSE PROPRIETARY CLASS 2

WCAP- 9230



5-12 Pressurizer Pressure and Water Volume Following a Feedline Rupture Assuming Worst Case Initial Conditions and Assumptions for a 3-Loop Plant

WESTINGHOUSE PROPRIETARY CLASS 2
WCAP-9230



5-13 Vessel Mass Flow Rate and Pressurizer Insurge Following a Feedline Rupture Assuming Worst Case Initial Conditions and Assumptions for a 3-Loop Plant

STEAM GENERATOR POWER OPERATED RELIEF VALVE (PORV) CONTROL SYSTEM

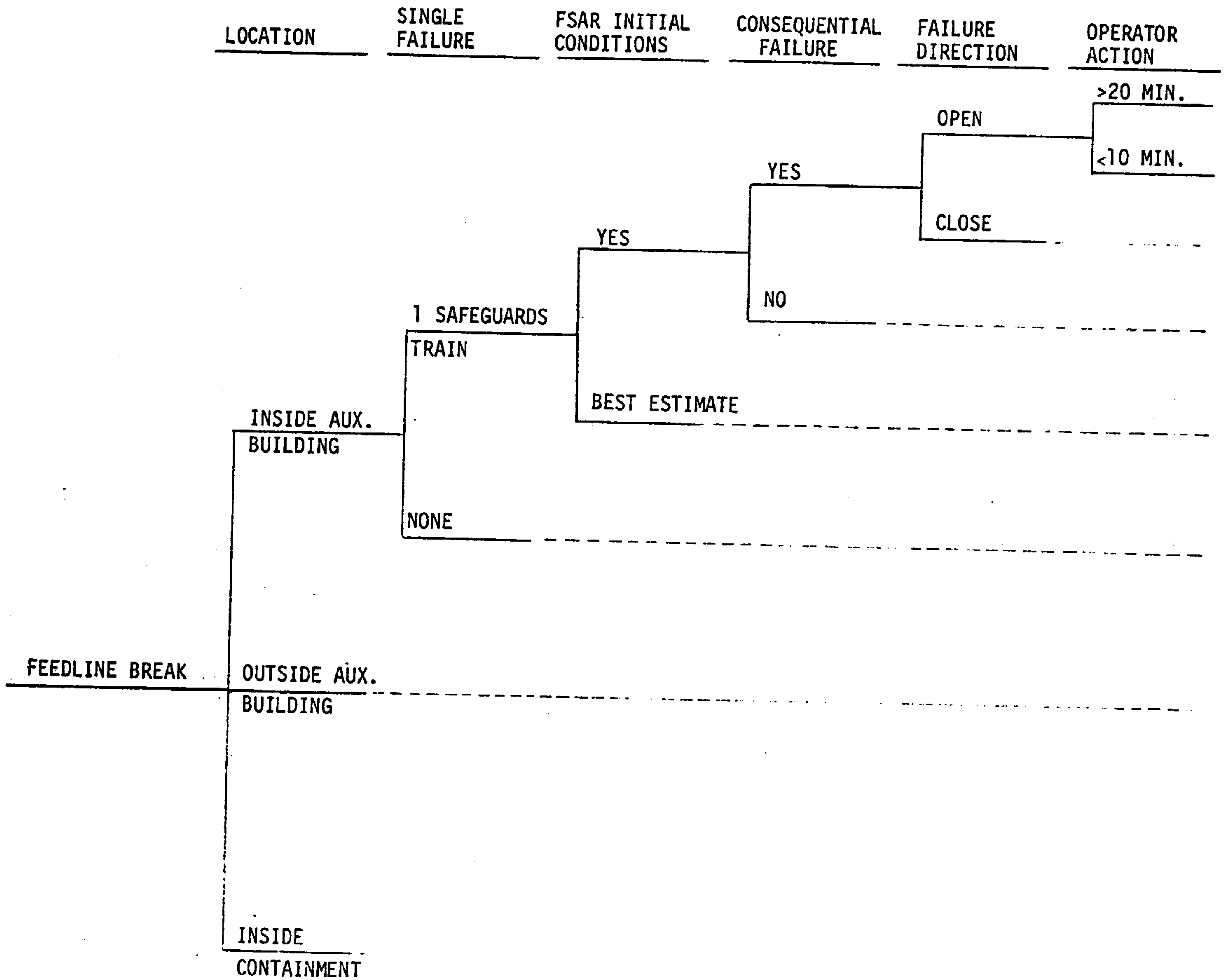
- FEEDLINE RUPTURE OCCURS IN MAIN OR AUXILIARY FEEDWATER LINES IN AUXILIARY BUILDING BETWEEN CONTAINMENT PENETRATION AND CHECK VALVES
- MAIN FEEDWATER SPILLS OUT RUPTURE
- SECONDARY INVENTORY SPILLS INTO AUXILIARY BUILDING THROUGH RUPTURED FEEDLINE
- REACTOR TRIP OCCURS ON LOW LOW STEAM GENERATOR WATER LEVEL IN RUPTURED STEAM GENERATOR
- AUXILIARY FEEDWATER PUMPS INITIATED ON LOW LOW STEAM GENERATOR WATER LEVEL. TURBINE TRIP OCCURS ON REACTOR TRIP.
- STEAM GENERATORS IN INTACT LOOPS BEGIN REPRESSURIZING DUE TO AUTOMATIC OR MANUAL MAIN STEAMLINE ISOLATION
- ADVERSE ENVIRONMENT INSIDE AUXILIARY BUILDING IMPACTS STEAM GENERATOR PORV CONTROL SYSTEM POTENTIALLY CAUSING THE VALVES TO INADVERTENTLY OPEN OR FAIL TO CLOSE DUE TO AN ENVIRONMENTAL CONSEQUENTIAL FAILURE
- STEAM GENERATORS THAT SUPPLY STEAM TO TURBINE DRIVEN AUXILIARY FEEDWATER PUMP DEPRESSURIZE TO ATMOSPHERIC PRESSURE VIA FAILED OPEN STEAM GENERATOR PORV'S, CAUSING TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS TO STOP
- IF SINGLE ACTIVE FAILURE ASSUMED IS A MOTOR DRIVEN AUXILIARY FEEDWATER PUMP, ALL AUXILIARY FEEDWATER IS LOST TO ALL STEAM GENERATORS
- PRIMARY BEGINS TO HEATUP RAPIDLY DUE TO LOSS OF SECONDARY HEAT SINK AND HOT LEG BOILING COMMENCES
- TIME OF OPERATOR ACTION TO MANUALLY CLOSE VALVES IN AUXILIARY FEEDWATER LINE TO RUPTURED STEAM GENERATOR OR TO MANUALLY BLOCK STUCK OPEN STEAM GENERATOR PORV'S DETERMINES SEVERITY OF ACCIDENT RESULTS

STEAM GENERATOR PORV CONTROL SYSTEM

ASSUMPTIONS:

- FEEDLINE RUPTURE OUTSIDE CONTAINMENT
- WORST SINGLE ACTIVE FAILURE ASSUMED IN SAFEGUARDS TRAIN
- FSAR INITIAL CONDITIONS
- ADVERSE ENVIRONMENT IMPACTS SG PORV CONTROL SYSTEM RESULTING IN CONSEQUENTIAL FAILURE
- STEAM GENERATOR PORV CONTROL SYSTEM DIRECTS VALVES TO MOVE TO OPEN POSITION
- OPERATOR ACTION NOT ASSUMED FOR AT LEAST 20 MINUTES

STEAM GENERATOR PORV



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

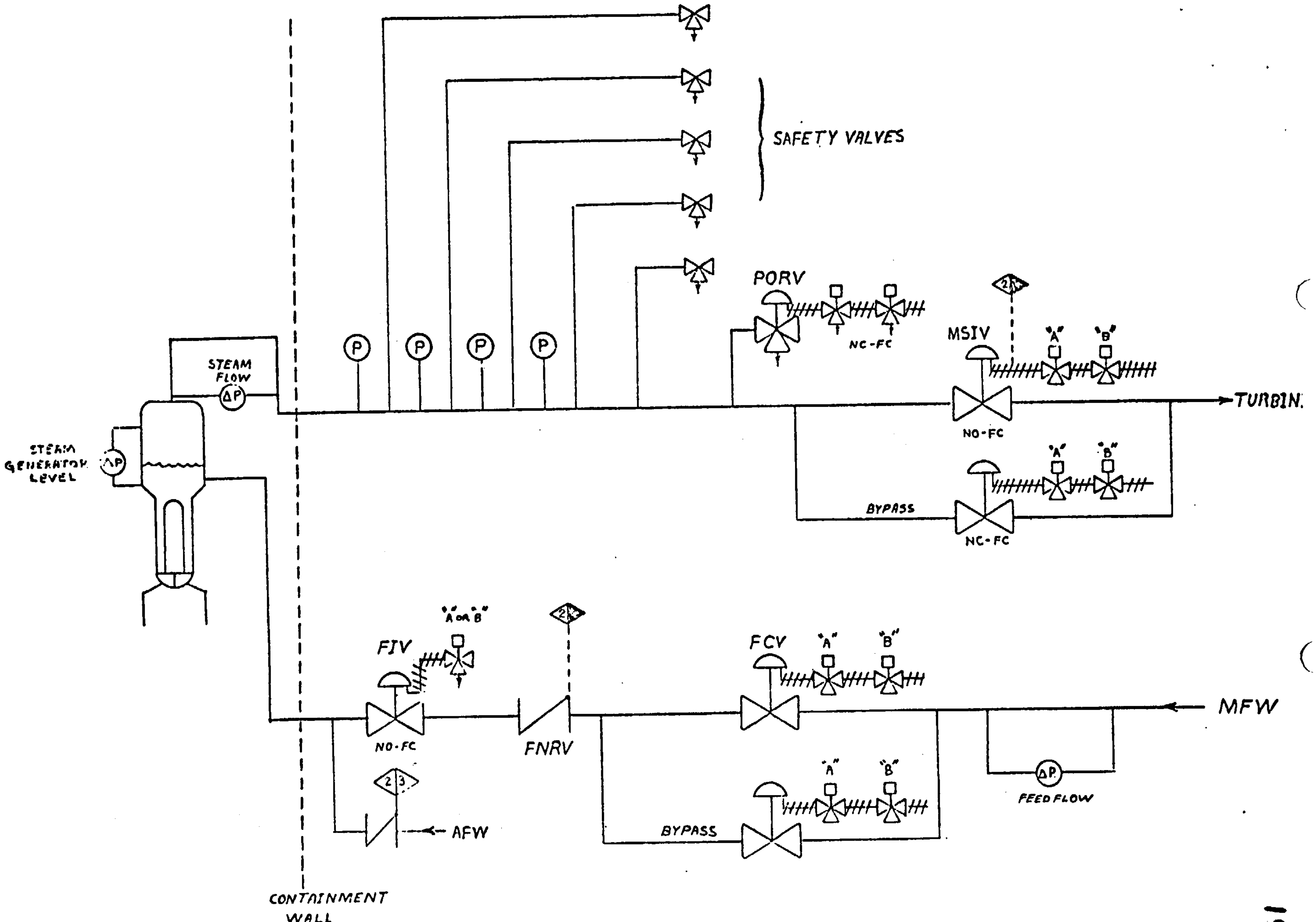
LONG TERM

- 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



SAFETY VALVES
PORV
NC-FC
MSIV
"A"
"B"
TURBIN.
NO-FC
BYPASS
NC-FC
STEAM FLOW
ΔP
P
STEAM GENERATOR LEVEL
FIV
"A" OR "B"
NO-FC
FNRV
FCV
MFW
AFW
BYPASS
FEED FLOW
CONTAINMENT WALL

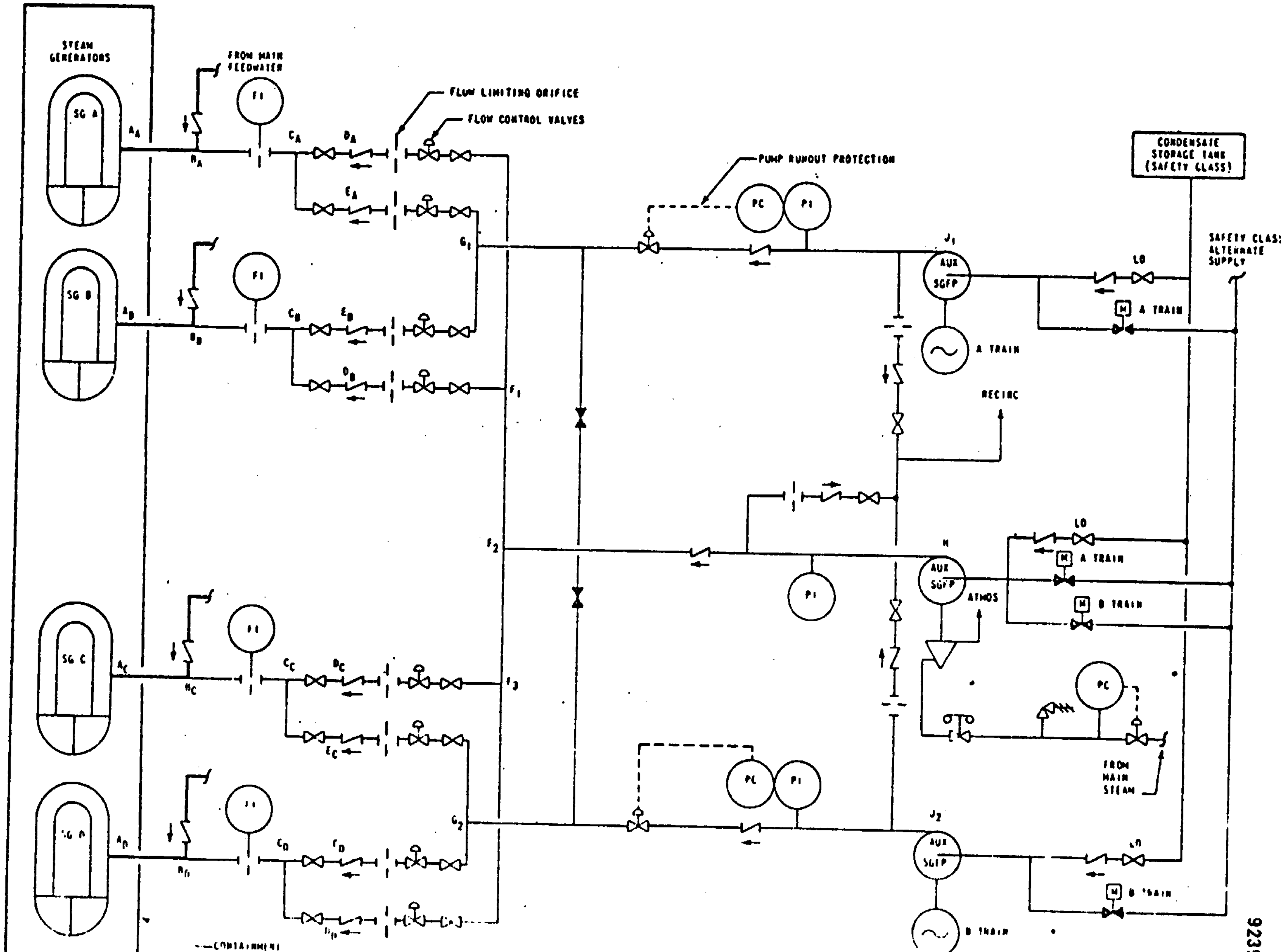


Figure 6. Auxiliary Feedwater System (Four-Loop Plant)

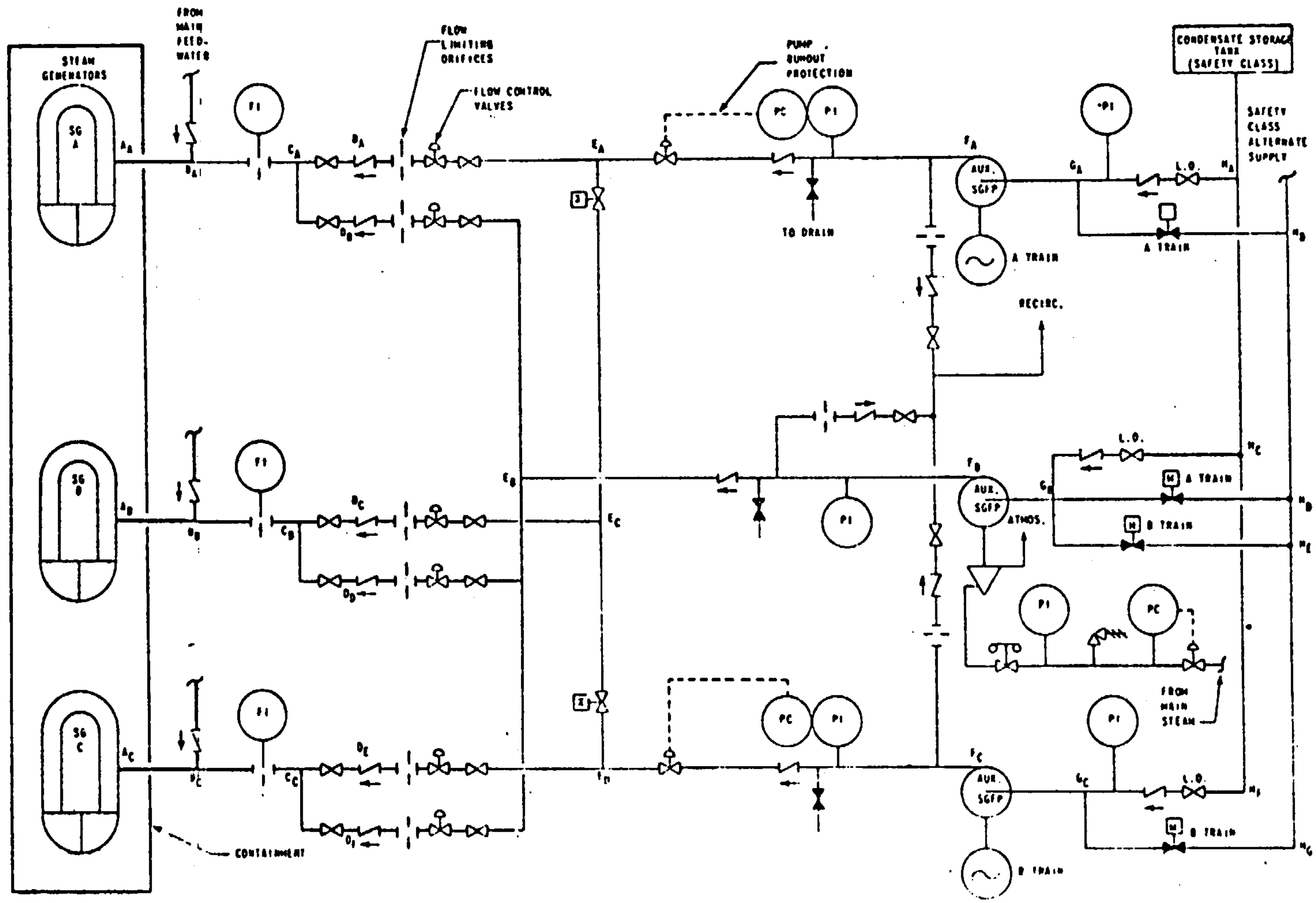


Figure 7. Auxiliary Feedwater System (Three-Loop Plant)

9239-5

MAIN FEEDWATER CONTROL SYSTEM

- SMALL FEEDLINE RUPTURE OCCURS IN MAIN OR AUXILIARY FEEDWATER LINES IN AUXILIARY BUILDING BETWEEN CONTAINMENT PENETRATION AND CHECK VALVES

- MAIN FEEDWATER AND POSSIBLY SECONDARY INVENTORY SPILLS INTO AUXILIARY BUILDING THROUGH SMALL FEEDLINE RUPTURE

- ADVERSE ENVIRONMENT CAUSED BY RUPTURE IN FEEDLINE IMPACTS MAIN FEED-WATER CONTROL SYSTEM LOCATED IN AUXILIARY BUILDING

- FEEDWATER CONTROL SYSTEM MALFUNCTIONS SUCH THAT ALL STEAM GENERATORS AT LOW LOW STEAM GENERATOR WATER LEVEL AT TIME OF REACTOR TRIP

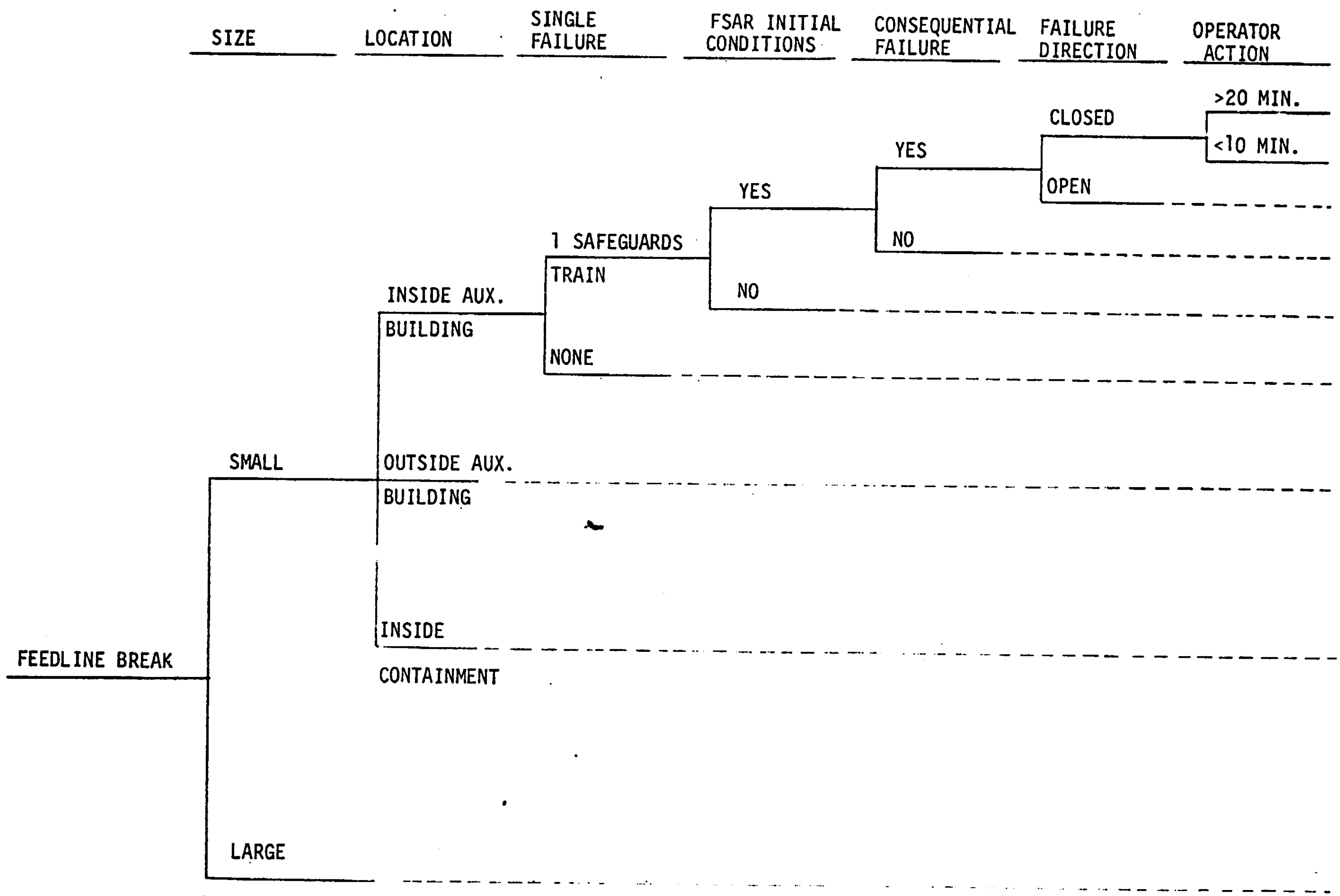
- RESULTS OF ACCIDENT WITH ABOVE CONDITIONS AT TIME OF REACTOR TRIP MORE SEVERE THAN THOSE PRESENTED IN MANY SAFETY ANALYSIS REPORTS

FEEDWATER CONTROL SYSTEM

ASSUMPTIONS:

- SMALL FEEDLINE RUPTURE OUTSIDE CONTAINMENT IN AUXILIARY BUILDING
- WORST SINGLE ACTIVE FAILURE ASSUMED IS SAFEGUARDS TRAIN
- FSAR INITIAL CONDITIONS
- ADVERSE ENVIRONMENT IMPACTS MAIN FEEDWATER CONTROL SYSTEM RESULTING IN CONSEQUENTIAL FAILURE
- MAIN FEEDWATER CONTROL SYSTEM DIRECTS FCV's IN INTACT LOOPS TO MOVE TO THE CLOSED POSITION
- OPERATOR ACTION NOT ASSUMED FOR AT LEAST 20 MINUTES

FEEDWATER CONTROL



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

PRESSURIZER POWER OPERATED RELIEF VALVE (PORV) CONTROL SYSTEM

- FEEDLINE RUPTURE OCCURS IN MAIN FEEDLINE INSIDE CONTAINMENT BETWEEN STEAM GENERATOR NOZZLE AND CONTAINMENT PENETRATION

- MAIN FEEDWATER SPILLS OUT RUPTURE

- SECONDARY INVENTORY SPILLS INTO CONTAINMENT THROUGH RUPTURED FEEDLINE

- REACTOR TRIP OCCURS ON LOW LOW STEAM GENERATOR WATER LEVEL IN RUPTURED STEAM GENERATOR

- AUXILIARY FEEDWATER PUMPS INITIATED ON LOW LOW STEAM GENERATOR WATER LEVEL. TURBINE TRIP OCCURS ON REACTOR TRIP

- ADVERSE ENVIRONMENT INSIDE CONTAINMENT IMPACTS PRESSURIZER PORV CONTROL SYSTEM POTENTIALLY CAUSING THE VALVES TO INADVERTENTLY OPEN OR FAIL TO CLOSE DUE TO AN ENVIRONMENT CONSEQUENTIAL FAILURE

- PRIMARY PRESSURE DECREASES DUE TO STUCK OPEN PRESSURIZER PORV'S

- HOT LEG BOILING COMMENCES

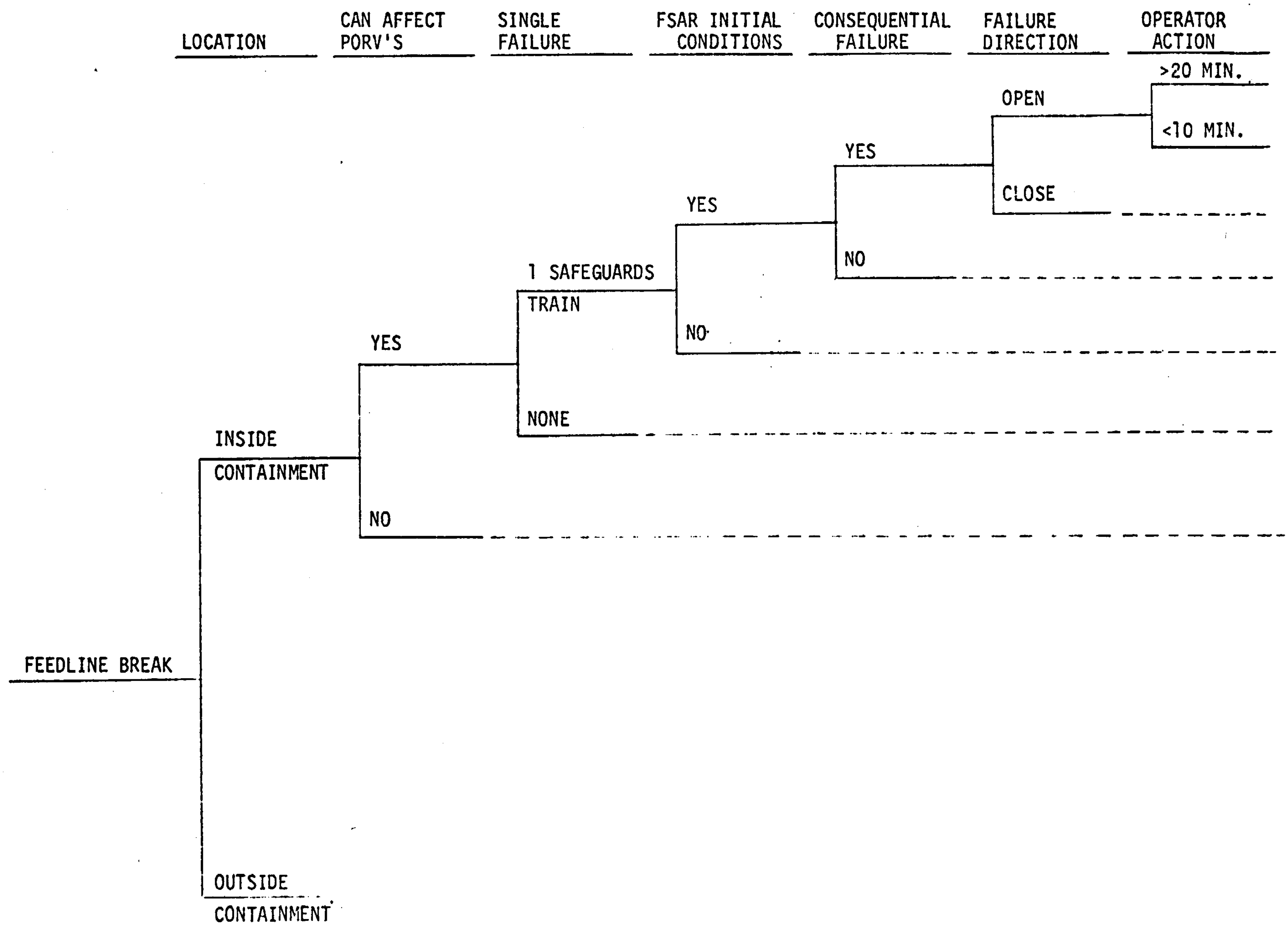
- TIME OF OPERATOR ACTION TO MANUALLY CLOSE BLOCK VALVES IN PRESSURIZER PORV RELIEF LINES DETERMINES SEVERITY OF ACCIDENT RESULTS

PRESSURIZER PORV CONTROL SYSTEM

ASSUMPTIONS:

- FEEDLINE RUPTURE OCCURS INSIDE CONTAINMENT
- WORST SINGLE ACTIVE FAILURE ASSUMED IS SAFEGUARDS TRAIN
- FSAR INITIAL CONDITIONS
- ADVERSE ENVIRONMENT IMPACTS PRESSURIZER PORV CONTROL SYSTEM RESULTING IN CONSEQUENTIAL FAILURE
- PRESSURIZER PORV CONTROL SYSTEM DIRECTS RELIEF VALVES TO MOVE TO OPEN POSITION
- OPERATOR ACTION NOT ASSUMED FOR AT LEAST 20 MINUTES

PRESSURIZER PORV



29

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

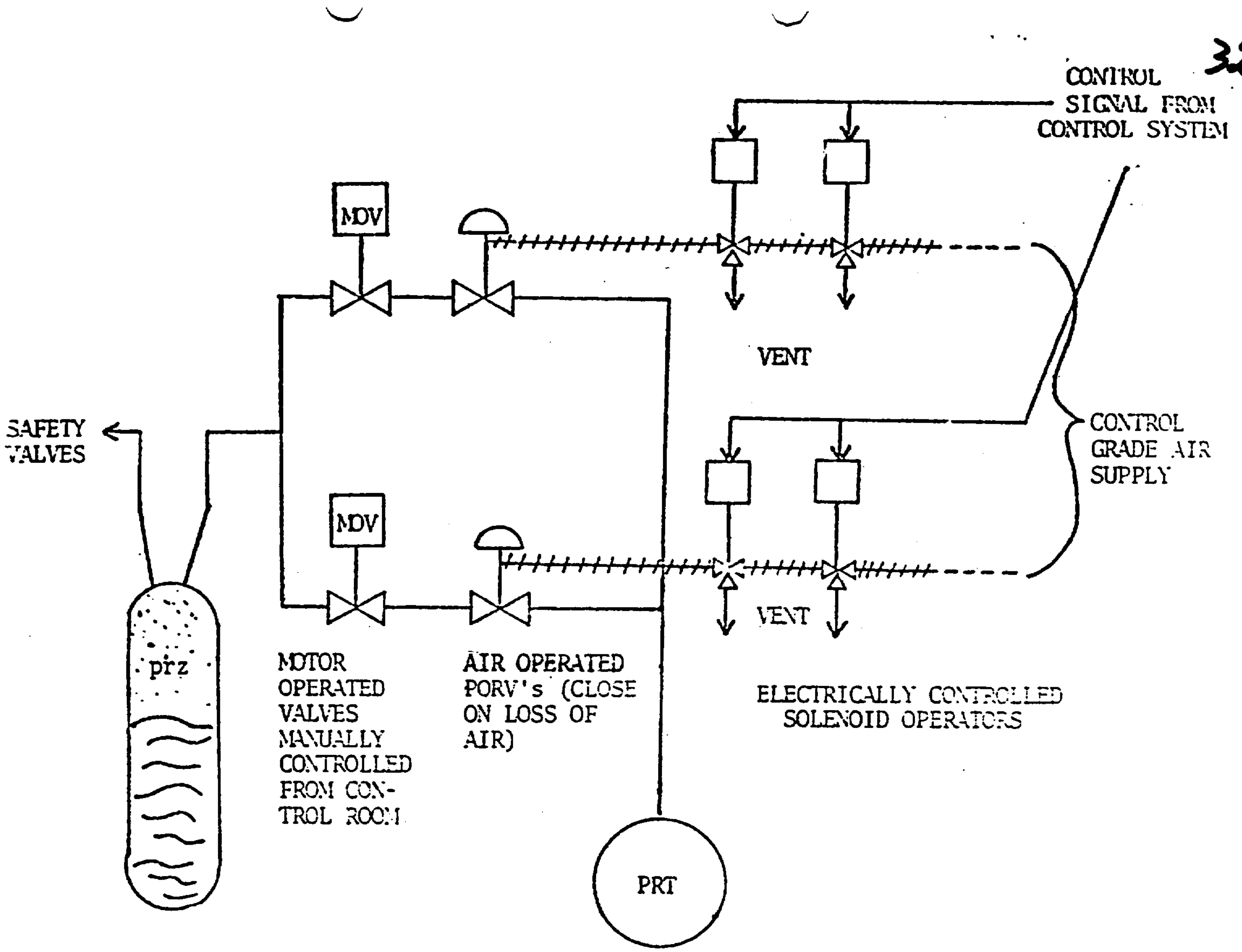
POTENTIAL SOLUTIONS

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.


CONTROL SIGNAL FROM CONTROL SYSTEM
MOV
MOV
VENT
SAFETY VALVES
CONTROL GRADE AIR SUPPLY
VENT
prz
MOTOR OPERATED VALVES MANUALLY CONTROLLED FROM CON-TROL ROOM
AIR OPERATED PORV's (CLOSE ON LOSS OF AIR)
ELECTRICALLY CONTROLLED SOLENOID OPERATORS
PRT

SAR INTERMEDIATE STEAMLINE RUPTURE EVENT

- INTERMEDIATE STEAMLINE RUPTURE OCCURS UPSTREAM OF MAIN STEAMLINE ISOLATION VALVES
- COLD LEG TEMPERATURE GRADUALLY DECREASES DUE TO APPARENT EXCESSIVE LOAD INCREASE
- NUCLEAR POWER INCREASES DUE TO MODERATOR FEEDBACK COEFFICIENTS (ASSUMES EOL CORE CONDITIONS)
- REACTOR TRIP OCCURS ON OVERPOWER DELTA-T FUNCTION
- TURBINE TRIP OCCURS DUE TO REACTOR TRIP
- STEAMLINE ISOLATION OCCURS AUTOMATICALLY OR MANUALLY CLOSED
- RUPTURED STEAMLINE BLOWS DOWN TO CONTAINMENT PRESSURE. STEAMLINES IN ISOLATED LOOPS EXPERIENCE SLIGHT INCREASE IN PRESSURE

WESTINGHOUSE PROPRIETARY CLASS 2

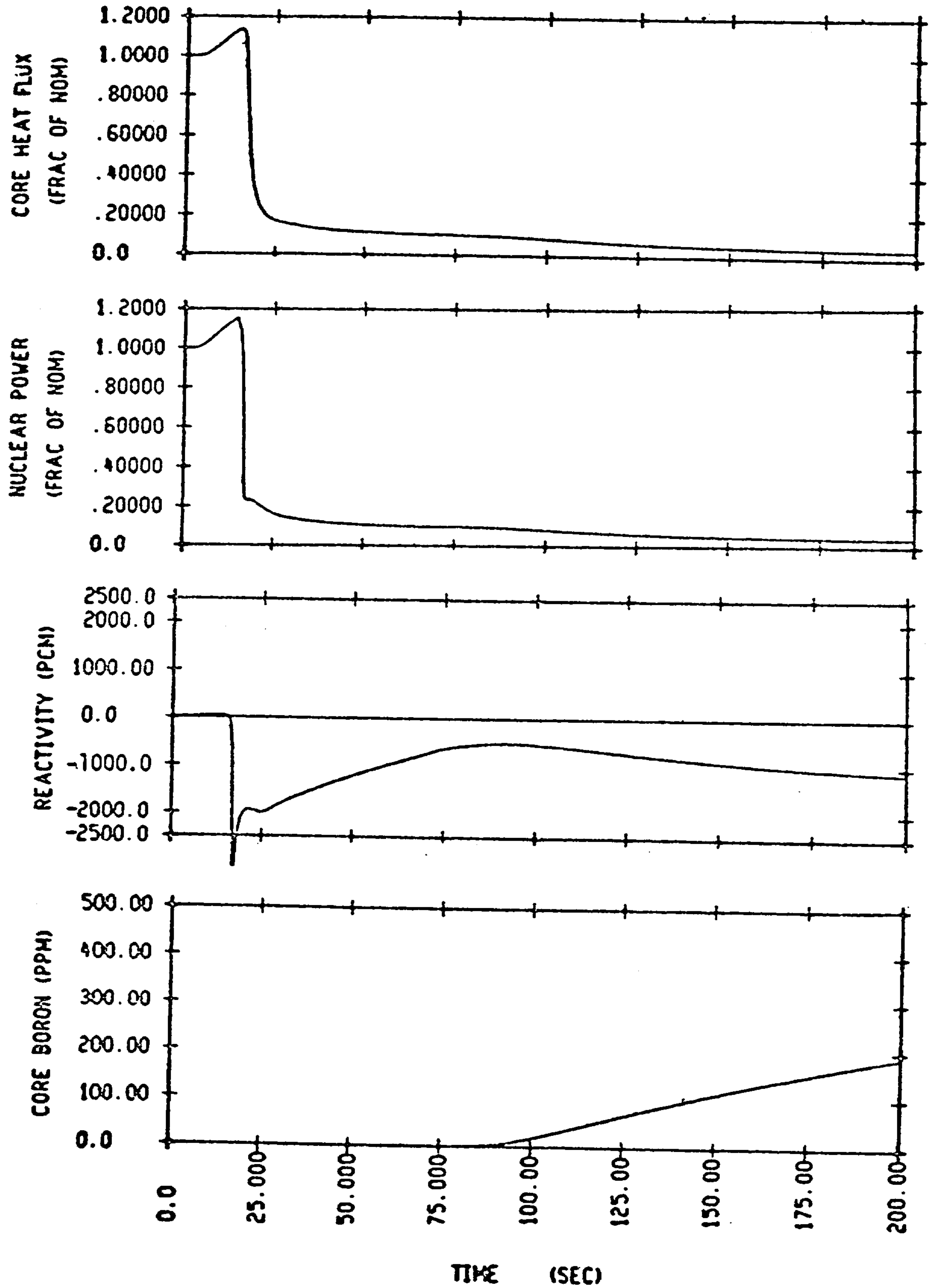


FIGURE 3.2-4 - TIME DEPENDENT PARAMETERS 3 LOOP, 100% POWER BREAK AREA - 0.22 FT^2

WESTINGHOUSE PROPRIETARY CLASS 2
WCAP-7226

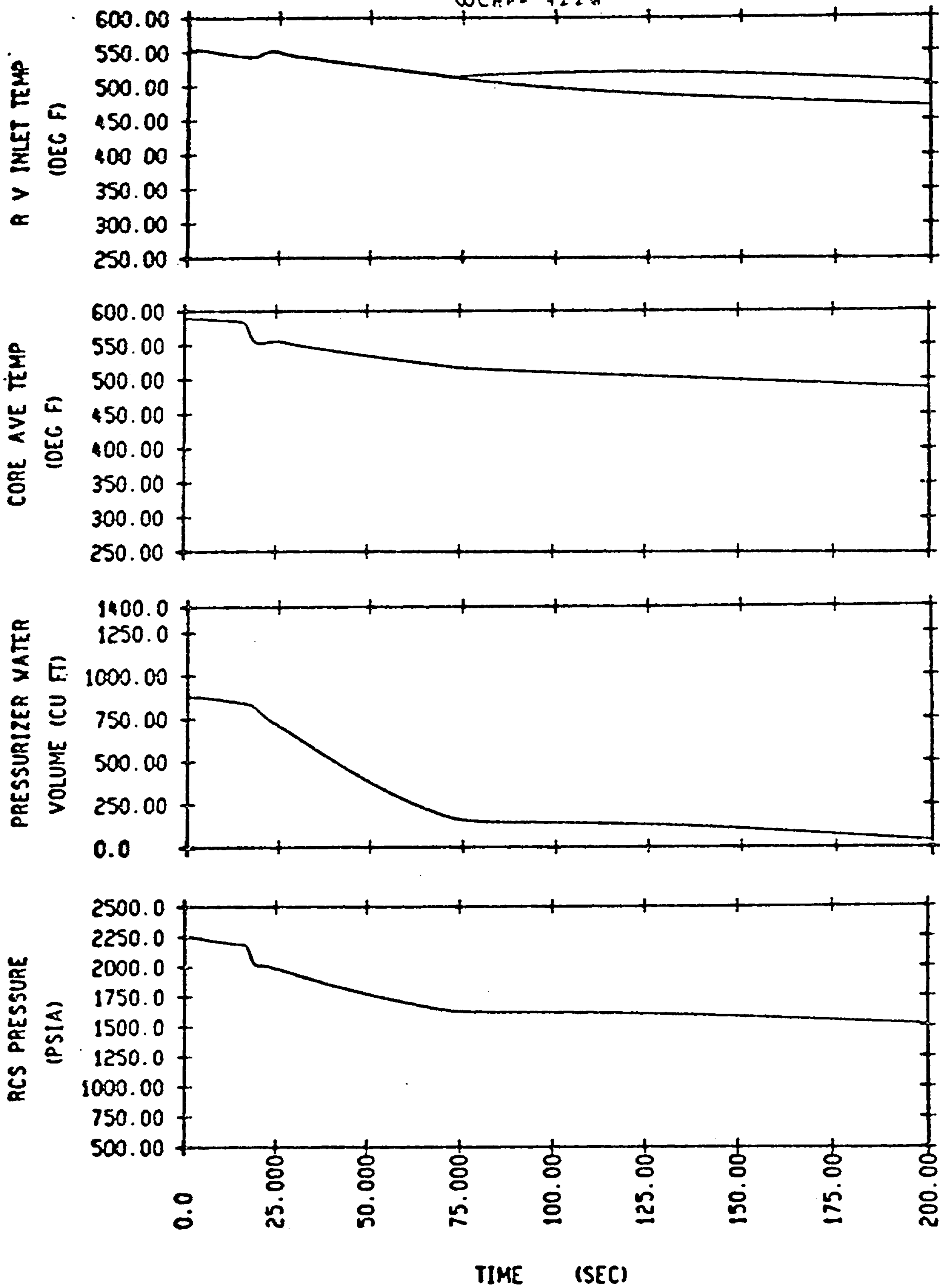


FIGURE 3.2-5 - TIME DEPENDENT PARAMETERS 3 LOOP, 100% POWER BREAK AREA = 0.22 FT^2

WESTINGHOUSE PROPRIETARY CLASS 2

WCAP-9226

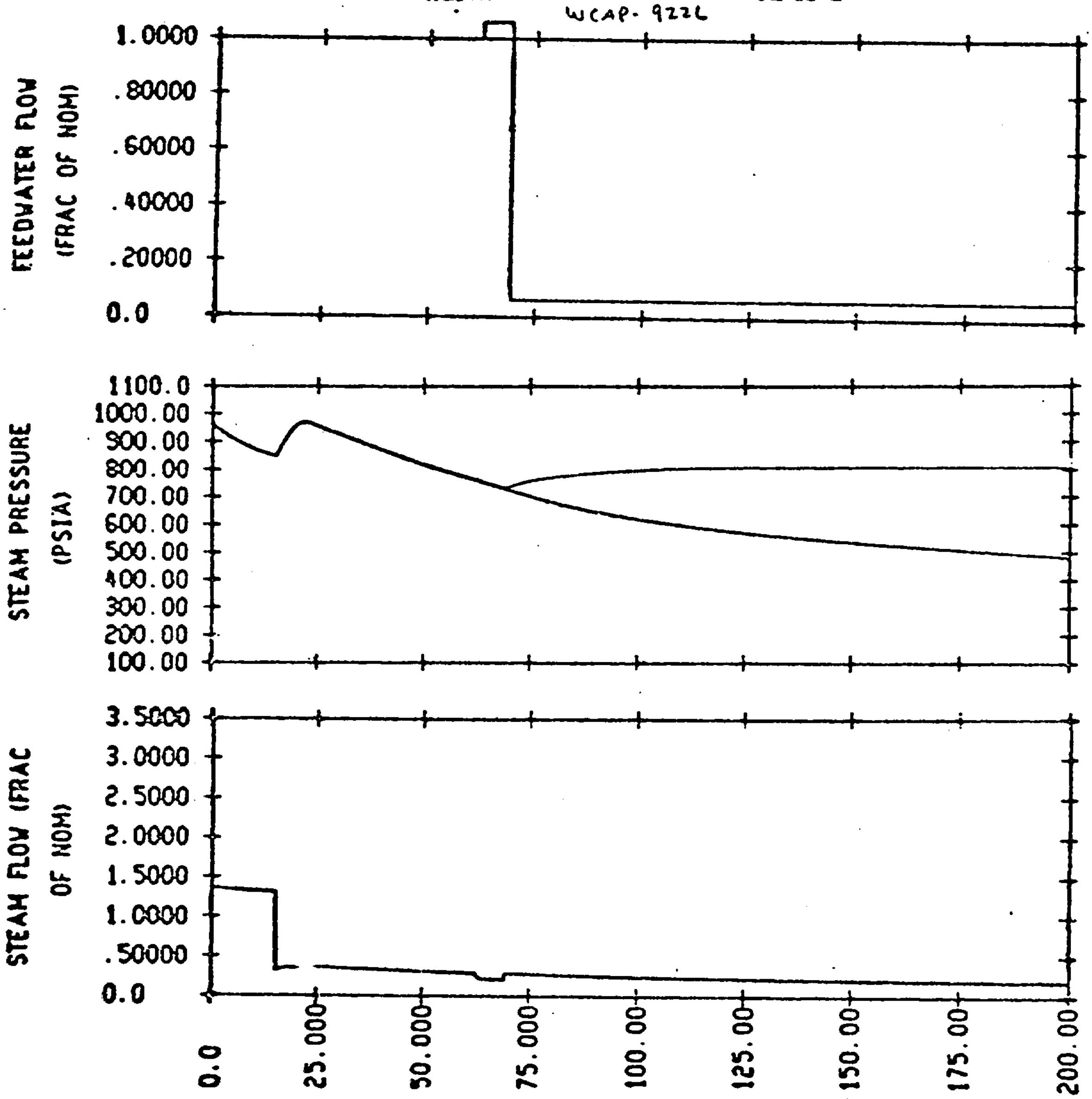


FIGURE 3.2-6 - TIME DEPENDENT PARAMETERS 3 LOOP, 100% POWER BREAK AREA = 0.22 FT^2

ROD CONTROL SYSTEM

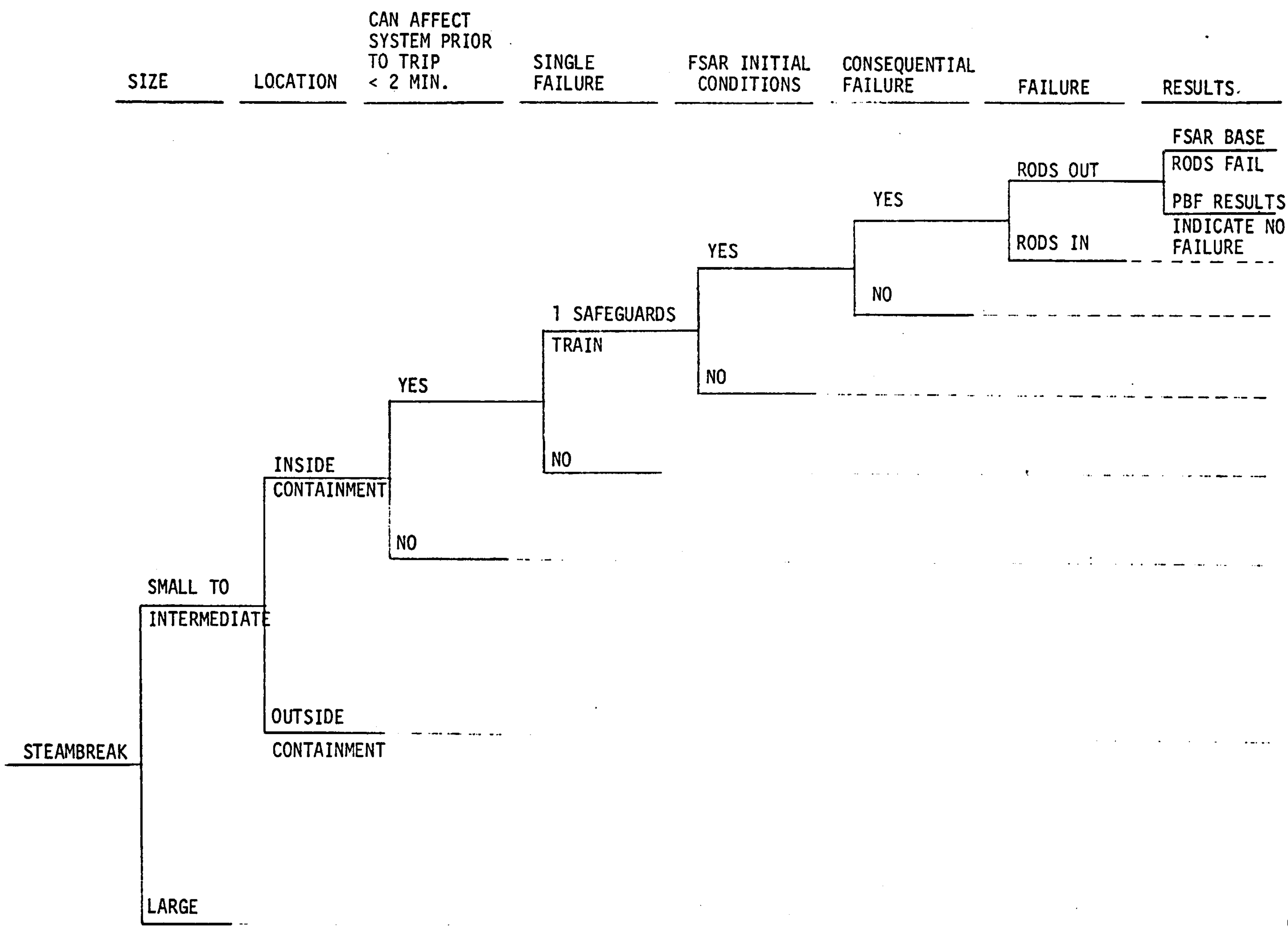
- INTERMEDIATE STEAMLINE RUPTURE (0.1 TO 0.25 SQUARE FEET PER LOOP FROM 70 TO 100 PERCENT POWER) OCCURS INSIDE CONTAINMENT
- ROD CONTROL SYSTEM IN AUTOMATIC MODE
- ADVERSE ENVIRONMENT FROM STEAMLINE RUPTURE IMPACTS EXCORE DETECTORS AND ASSOCIATED CABLING
- ENVIRONMENTAL CONSEQUENTIAL FAILURE OCCURS IN ROD CONTROL SYSTEM WHICH CAUSES CONTROL RODS TO BEGIN STEPPING OUT PRIOR TO REACTOR TRIP
- MINIMUM DNBR FALLS BELOW 1.30 (GREATER THAN 1.1) PRIOR TO A REACTOR TRIP ON OVERPOWER DELTA-T FUNCTION WHICH EXCEEDS LICENSING CRITERIA IN MANY SAFETY ANALYSIS REPORTS

ROD CONTROL SYSTEM

ASSUMPTIONS

- INTERMEDIATE STEAMLINE RUPTURE OCCURS INSIDE CONTAINMENT
- ADVERSE ENVIRONMENT IMPACTS ROD CONTROL SYSTEM COMPONENTS PRIOR TO REACTOR TRIP
- WORST SINGLE ACTIVE FAILURE ASSUMED IS SAFEGUARDS TRAIN
- FSAR INITIAL CONDITIONS
- ADVERSE ENVIRONMENT IMPACTS ROD CONTROL SYSTEM RESULTING IN CONSEQUENTIAL FAILURE
- ROD CONTROL SYSTEM DIRECTS CONTROL RODS TO WITHDRAWAL

ROD CONTROL SYSTEM



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

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.

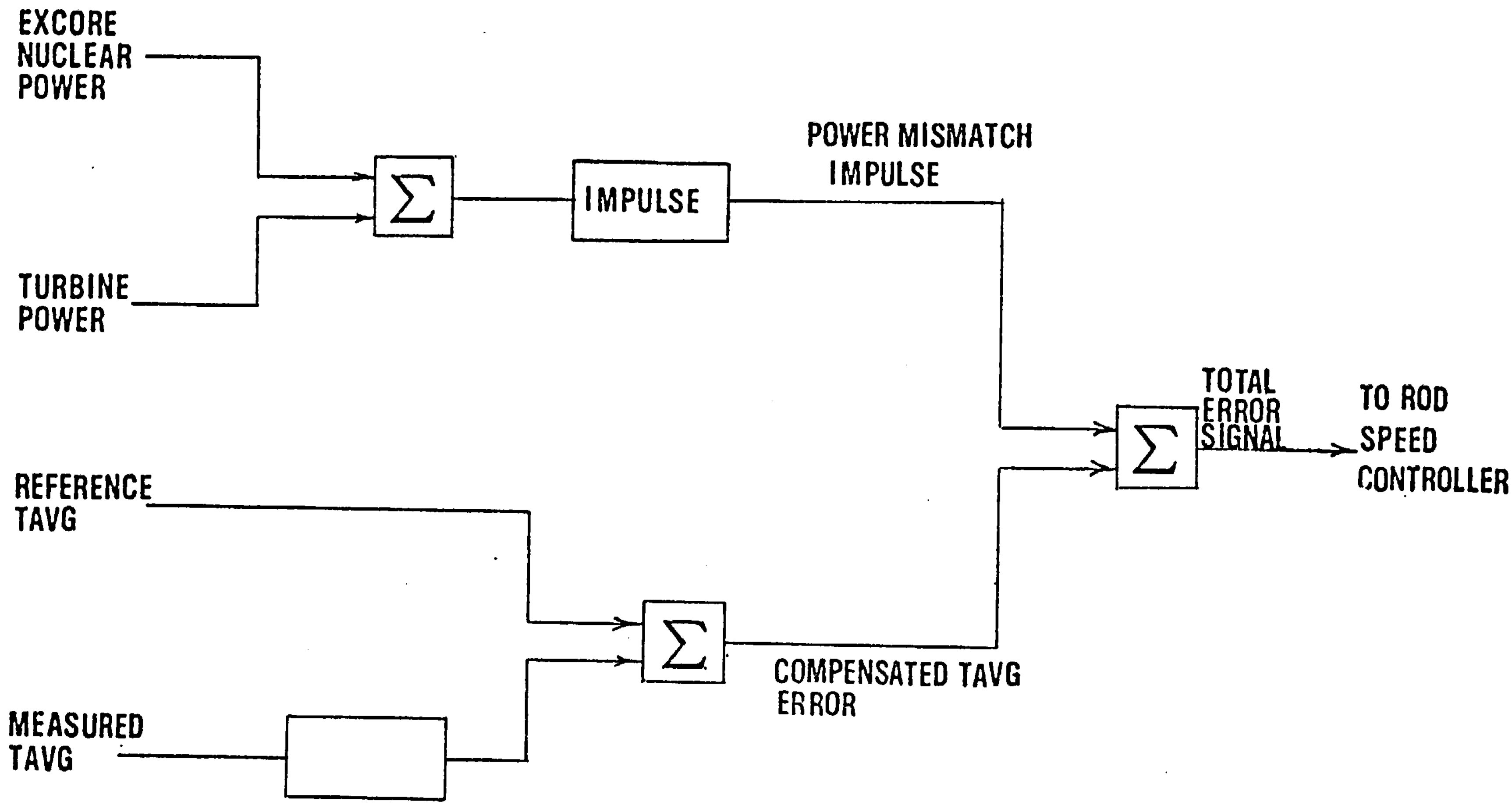
- REMOVE NIS SIGNAL FROM POWER MISMATCH CIRCUIT IN ROD CONTROL SYSTEM (PROCESS CONTROL CABINET)

- 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



ROD CONTROL SYSTEM
SIMPLIFIED SCHEMATIC

42

ENCLOSURE 3

MEETING ATTENDEES

NRC

D. Ross
T. Novak
G. Kuzmycz
S. Lewis
D. Tondi
W. Jensen
J. Guttmann
J. Mazetis
S. Israel
C. Berlinger
Z. Rosztoczy
F. Orr
J. Heltemes
J. Rosenthal
M. Cliramal
J. Joyce
R. Scholl
T. Dunning
J. Burdoin
R. Woodruff
S. Salah
K. Mahan
H. Rood
D. Thatcher
B. Morris
S. Sands
T. Houghton
D. Tibbitts
R. Reil
G. Lainas
E. Conner
P. Norian

CE

R. Daigle
C. Brinkman
W. Burchill
J. Westhaven
C. Kling
P. Delozier

C. Faust - Westinghouse
R. Borsum - B&W
N. Shirley - GE

G. Liebler - Fla. P&L Co.
R. Marusich - Consumers Power Co.
R. Kacich - Northeast Utilities
J. Regan - Northeast Utilities
R. Olson - Baltimore G&E Co.
H. O'Brien - TVA

R. Harris - NUSCO
G. Falibota - Bechtel
E. Inge - ACRS
P. Higgins - AIF
R. Leyse - EPRI

ENCLOSURE 4

ACTION PROCESS FOR I&E INFORMATION NOTICE NO. 79-02

- IDENTIFY THOSE NON-SAFETY RELATED CONTROL SYSTEMS (BOTH INSIDE & OUTSIDE CONTAINMENT) WHOSE MALFUNCTION COULD ADVERSELY AFFECT THE ACCIDENT OR TRANSIENT WHEN SUBJECTED TO ADVERSE ENVIRONMENT CAUSED BY A HIGH ENERGY PIPE BREAK!
- DETERMINE THE LIMITING MALFUNCTIONS DURING HIGH ENERGY PIPE BREAKS FOR THOSE CONTROL SYSTEMS.
- DETERMINE THE IMPACT OF THE MALFUNCTION OF THOSE SYSTEMS.
- DETERMINE SHORT TERM ACTIONS IF NECESSARY.
- DETERMINE LONG TERM ACTIONS IF NECESSARY.

ENCLOSURE 5

MEETING ATTENDEES 9/20/79AM

NRC

D. Ross
T. Novak
G. Kuzmycz
R. Capra
S. Lewis
D. Tondi
T. Dunning
Z. Rosztoczy
W. Jensen
J. Mazetis
S. Israel
J. Rosenthal
M. Fairtile
J. S. Ckesumal
M. Cleramal
R. Scholl
J. Beard
J. Joyce
D. Thatcher
D. DiIanni
G. Lainas
B. Morris
S. Diab

R. Leipe -EPRI
P. Higgins - AIF
T. Martin - NUTECH
E. Roy - Bechtel
T. Reitz - G/C Inc.
E. Weiss - Union Concerned Scientists
R. Pollard - UCS

B&W

R. Borsum
J. Taylor
H. Roy
E. Kane
S. Eschbach
B. Short
M. Bonaea
G. Brazill
B. Karrasel
R. Wright
D. Hallman

B. Day - Brown Boveri Reaktorbau
C. Faust - Westinghouse

L. Stalter - Toledo Edison
F. Miller - Toledo Edison
T. Myers - Toledo Edison
R. Gill - Duke Power
T. McMeekin - Duke Power
P. Abraham - Duke Power
K. Canady - Duke Power
R. Dieterich - SMUD
E. Good - FPC
B. Simpson - FPC
C. Hartman - Met Ed
P. Trimble - Arkansas P&L
R. Hamm - Consumer P. Co.

ENCLOSURE 6

UTILITY/B&W PROGRAM

EVALUATE IMPACT ON LICENSING BASIS ACCIDENT ANALYSES DUE TO CONSEQUENTIAL ENVIRONMENTAL EFFECTS ON NON-SAFETY GRADE CONTROL SYSTEMS.

- IDENTIFY LICENSING BASIS ACCIDENTS WHICH CAUSE AN ADVERSE ENVIRONMENT FOR EACH PLANT.
- DEFINE SAFETY ANALYSIS INPUTS AND RESPONSES USED DURING LICENSING BASIS ACCIDENTS.
- VERIFY SAFETY ANALYSIS CONCLUSIONS OR RECOMMEND ACTIONS JUSTIFYING CONTINUED OPERATION.

ENCLOSURE 7

MEETING ATTENDEES 9/20/79PM

NRC

D. Ross
T. Novak
G. Kuzmycz
R. Frahm
D. Tondi
T. Dunning
D. Lynch
J. Joyce
C. DeBevec
D. Thatcher
R. Scholl
W. Hodges
T. Ippolito
V. Rooney
J. Rosenthal
W. Jensen
J. Guttman
J. Hannon
T. Keven
G. Lainas
P. Norian

C. Feltman - Bechtel
M. David - Bechtel
T. Martin - NUTECH
P. Higging - AIF

GE

N. Shirley
L. Youngborg
J. Cleveland
C. Sawyer
P. Marriott
L. Gifford

D. Rawlins - W
C. Faust - W
R. Borsum - B&W

T. Rogers - Pacific Gas & Elec.
W. Mindich - Phil. El. Col
C. Cowan - Phil. El. Co.
G. Edwards - Phil. El. Co.
T. Scull - Phil.El. Co.
J. Knubel - JCP&L Co.
T. Tipton - JCP & L Co.
L. Rucker - Boston Ed.
J. Vorees - Boston Ed.
S. Maloary - Boston Ed.
J. Sheppard - CPCo.
R. Hoston - CPCo.
L. Mathews - Southern Co. Services
C. Verprek - PSE&G
R. Rajoram - PASNY
R. Rogers - TVA
M. Wiesburg - TVA
V. Bgnum - TVA

cc
Mr. Lawrence E. Minnick, President
Yankee Atomic Electric Company
20 Turnpike Road
Westboro, Massachusetts 01581

Greenfield Community College
1 College Drive
Greenfield, Massachusetts 01301