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October 26, 1982

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

Subject: Byron Station Units 1 and 2
Braidwood Station Units 1 and 2
Pressurizer Safety and Relief Valves
NRC Docket Nos. 50-454, 50-455, 50-456, and 50-457

Dear Mr. Denton:

This is to provide plant-specific information regarding the adequacy of relief and safety valves at Byron and Braidwood. This information is provided to satisfy NRC requirements delineated in item II.D.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements." Review of this information should close the confirmatory issue discussed in section 3.9.3.3 of the Byron SER.

Attachment A to this letter contains the plant-specific valve testing information necessary to satisfy Clarification A of item II.D.1 in NUREG-0737. It summarizes the test data which demonstrate that the Byron/Braidwood reactor coolant system safety and relief valves can be expected to function as designed over the range of expected operating and accident conditions. It also outlines the manner in which the adequacy of piping and supports will be assured.

Attachment B to this letter contains an advance copy of a revised FSAR page containing updated information on item II.D.1. This information will be included in the FSAR in the next amendment.

One signed original and fifteen copies of this letter and the attachments are provided for NRC review. Please address questions regarding this matter to this office.

Very truly yours,

T. R. Tramm

T. R. Tramm
Nuclear Licensing Administrator

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Attachments

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ATTACHMENT A

Byron/Braidwood

Plant Specific Safety and Relief Valve

Functionability Evaluation

I. Relevant Previous Correspondence

Ref. 1: Letter from D. P. Hoffman (Consumers Power to H. R. Denton (NRC) dated April 1, 1982, transmitting the following EPRI Safety and Relief Valve Test Program Reports:

a) Safety & Relief Valve Test Report

b) Valve Selection/Justification Report

c) Test Condition Justification Report

d) Plant Conditions Justification Report
   1) Westinghouse
   2) Combustion Engineering Plants
   3) Babcock & Wilcox Plants

e) Evaluation of RELAP 5 MOD 1 for Calculation of Safety & Relief Valve Discharge Piping Hydrodynamic Loads

Ref. 2: Letter from R. C. Youngdahl (Consumers Power) to H. R. Denton (NRC) dated June 1, 1982, transmitting the following EPRI Test Program:

a) EPRI Safety & Relief Valve Test Program PORV Block Valve Information Package

Ref. 3: Letter from O. Kingsley (Alabama Power) to S. Chilk (NRC) dated July 27, 1982, transmitting WCAP-10105, a report performed for the Westinghouse Owners' Group entitled:

a) Review of Pressurizer Safety Valve Performance as Observed in the EPRI Safety & Relief Valve Test Program

II. Summary

Commonwealth Edison is a participant in the generic EPRI/PWR Safety & Relief Valve Test Program. The reports referenced above document the Test Program results. Commonwealth Edison has performed a final review of the results with regard to valve operability & S/RV piping and support adequacy. We have concluded that the EPRI tests represent the safety, relief, and block valves designs to be used at Byron/Braidwood, and that the piping and support load data provided is sufficient to perform a plant unique assessment of the S/RV piping and supports planned for use at

II. Summary (Cont'd)

Byron/Braidwood. We have also concluded that the conditions tested envelop and conservatively exceed the range of expected operating and accident conditions that we anticipate for Byron/Braidwood.

III. Safety Valve Evaluation

The Byron/Braidwood pressurizer safety valves are Crosby model HP-BP-86 6M6 with loop seal internals. This valve model was tested in the EPRI reported in reference 1. The following sections of the reports in reference 1 are relative specifically to Byron/Braidwood:

1. Safety & Relief Valve Test Report - Section 3.5

2. Valve Selection/Justification Report - Section III-A1

3. Test Condition Justification Report - Westinghouse Plants - All Sections related to 4 loop plants.

Our assessment of the Crosby 6M6 is based on its test performance in response to fluid inlet conditions anticipated for Byron/Braidwood in transient and accident events. Those conditions consist of discharge of the loop seal water immediately upstream of the valve followed by saturated steam for all such events that may result in safety valve actuation. The technical justification for this inlet condition determination was derived by means of a probabilistic risk study. Documentation of this study constitues Appendix A to this evaluation.

This safety valve operability function considered pertinent to Byron/Braidwood plant safety is mitigation of excessive RCS transient pressure increases. The criteria indicative of the 6M6 valve's capability to perform this function are: 1) opening to a position where sufficient flow is achieved to prevent exceeding the 2750 PSIA primary system design limit under the highest anticipated transient over-pressure ramp rate, 2) closure to a position sufficient to curtail redction of RCS pressure below 88% of the safety valve actuation pressure, and 3) internal and external structural integrity of the safety valve sufficient to allow subsequent actuations in the event of repeated overpressure transients.

These criteria are considered to be very conservative with regard to plant safety. The express purpose for defining the criteria parameters is to establish a comparison basis by which it can be defensibly stated that performance within the criteria clearly represents no challenge to the original safety margins. As an illustration, the safety valve closure criteria was judged to be a conservative lower limit, above which safety valve liquid challenges
are not expected. The absence of such challenges is viewed as a clear indication that no encroachments on Byron/Braidwood safety analysis margins are anticipated with respect to safety valve closure.

Conclusions

Quantitative review of the EPRI 6M6 tests with fluid conditions applicable to Byron/Braidwood have led to the conclusion that this valve is in fact capable of performing the operability function required within the criteria identified. It is therefore also concluded that, since the 6M6 test performance was within the anticipated plant safety analysis margins, no equipment or analytical corrective actions are warranted in the interest of plant safety. The factors providing techincal support for these conclusions with respect to each criteria are as follows:

Criteria 1 - Opening

1. Regardless of stem lift delays or upstream piping pressure surges observed in the SPRI test data as a result of loop seal water discharge, in no instance did the tank (pressurizer) pressure exceed the 2750 PSIA criteria. Equally favorable results are anticipated for the Byron/Braidwood plant specific case because:

   1.1 As shown in Section 4.1 of WCAP-10105, (Reference 3) stem lift delays of up to 2 seconds can be accommodated. In the EPRI tests, the stem lift delay in no case exceeded 0.9 seconds. In comparing the Byron/ Braidwood plant loop seal volume (approx. 0.65 ft^3) to the EPRI test facility volume (approx. 1.05 ft^3), it can be seen that the Byron/ Braidwood stem lift delay would in no case exceed the 2 second limit established by Westinghouse.

   1.2 As shown in Section 3.1.1 of WCAP-10105, the transient overpressure ramp rates tested by EPRI were in excess of those for Westinghouse plants. In comparing cases with equal loop seal volumes, the EPRI overpressure ramp rates would clearly result in higher peak pressures than those anticipated for Byron/Braidwood since it is apparent in the test data that the stem lift delay is not significantly affected by the ramp rate.

   1.3 The EPRI test data clearly demonstrates that the tank (pressurizer) pressure is independent of the observed valve inlet pressure surges during loop seal discharges.

   Based on comparison of the EPRI test faciity and Byron/Braidwood plant configurations with respect to tank (pressurizer) volumes, safety valve inlet piping geometry, and downstream backpressure, it is clear that the Byron/ Braidwood pressurizer peak pressure would be even less sensitive to safety valve inlet pressure surges than any responses that were observed in the EPRI tests. The

1.3 (Cont'd)
Byron/Braidwood pressurizer liquid and steam volumes are much larger than the EPRI facility volumes, and the EPRI backpressures were significantly higher than the Byron/Braidwood backpressures (assuming all 3 safety valves and both PORVs input maximum flow). Concurrently, the 6M6 valve tested by EPRI and those planned for use at Byron/Braidwood are virtually identical. It can therefore be seen that this is a conservative assessment.

<u>Criteria 2 - Closure</u>

2. Other than two exceptions in the apparent test results, the 6M6 always closed within 10% of the design actuation pressure. Equally favorable results are expected for the Byron/Braidwood plant specific cases because:

   2.1 The manufacturer's (Crosby) original blowdown ring adjustments provided 10% or less blowdown. The Byron/Braidwood plant specific blowdown ring adjustments have also been established by manufacturer's recommendations. They are to be verified correct by procedure during set-point verification testing conducted during each refueling outage. From the viewpoint of the safety valve itself therefore, fundamentally the same blowdown performance is expected for Byron/Braidwood as that experienced by EPRI.

   2.2 Comparison of the Byron/Braidwood plant configurations and the EPRI test facility regarding upstream piping geometry (length, flow diameter, bend radii) and the downstream backpressures indicate that no significant flow dynamics differences are anticipated between the EPRI and Byron/Braidwood cases that impact blowdown.

   2.3 The two exceptions noted (EPRI tests 920 and 1419) are not considered typical of valve closure performance expected at Byron/Braidwood. In these tests, the 6M6 reopened at a pressure lower than the design actuation pressure and subsequently exhibited a rapid cyclic characteristic. The reopening phenomena was the apparent result of an acoustic wave propogated pressure surge in the upstream pipe of sufficient magnitude to exceed what was at that point a distincly degraded valve actuation pressure. The actuation pressure degradation was precipitated by a significant increase in seat leakage developed by the time of initial closure in these tests. The increased seat leakage effect was pressurization of the huddle chamber (secondary area)

2.3 (Cont'd)
within the valve, thus generating a lifting force at an inlet pressure considerably below the design actuation pressure. The subsequent acoustic pressure surge amplitude was then sufficient to apply the necessary additional force to the valve disc and huddle chamber areas to cause full lift at that lower pressure. At that degraded acuation pressure, but clearly not at the design actuation pressure, the acoustic pressure surge also had sufficient amplitude to produce the cyclic characteristic of the valve. This was due to its wave transit time in relation to the particular upstream piping length, such that the surge impact frequency on the valve disc was applied after valve closure had occurred.

Key elements in the determination that such a phenomena would not take place in the Byron/Braidwood plant specific cases are: 1) seat leakage that could be conservatively anticipated after initial reclosure of the valve, and 2) valve fluid inlet conditions expected during subsequent actuations for repeated overpressure transients. The assumption that the anticipated acoustic wave pressure surge dynamics are as severe in the plant as those present in the tests with respect to upstream piping length and flow area, valve closure time, and blowdown percentage is appropriately conservative. Tests 920 and 1419 were warm loop seal tests (approximately 350°F), and both were conducted immediately following similar warm loop seal test (Nos. 917 and 1415, respectively), without benefit of valve seat refurbishment prior to test initiation. Consequently, at the time of initial valve closure in Tests 920 and 1419, the valve seats had been subjected to more structural distress from full actuations and cycling during loop seal discharges, as well as more severe seat sealing fluid conditions (350°F water as opposed to saturated steam or cold water), than any other points during testing. The seat leakage at these points was of apparent greater magnitude than what was present at corresponding times in any other tests, particularly Tests 917 and 1415. Conservatively assuming similar loop seal water average temperatures, Tests 917 and 1415 are representative of the Byron/Braidwood safety valve configurations that have anticipated acoustic wave pressure surge dynamics significantly less severe than those of the tests, in terms of initial actuation fluid inlet conditions.

This is an appropriate statement because all of the three Byron/Braidwood safety valve configurations for each unit have considerably shorter inlet pipe lengths than the EPRI test facility (approximately 7 feet versus approximately 16 feet). Initial actuation performance by virtue of the fact that the valve seat condition is expected to be at least comparable to the condition present at the outset of Tests 917 and 1415 (determined during required refueling outage setpoint verification testing), is anticipated to be essentially the same as the 917 and 1415 test results.

2.3 (Cont'd)
For Byron/Braidwood however, it is not appropriate to expect that inlet fluid conditions for subsequent actuations from repeated overpressure transients would correspond with initial fluid inlet conditions of Tests 920 and 1419. Such subsequent actuations would, in view of the overpressure ramp rates applicable, take place in saturated steam without a preceding loop seal discharge. Seat sealing with steam is expected to be better than with warm water and no cyclic behavior from loop seal discharge would take place to further degrade seat condition, consequently no significant lowering of the valve actuation pressure is anticipated following closure of the valve from subsequent actuations. In the absence of a degraded actuation pressure, the acoustic pressure surge is of insufficient amplitude to cause the reopening phenomena or any associated cyclic valve characteristic. For Byron/Braidwood plant specific purposes, it can therefore be distinguished that should an initial safety valve actuation similar to Tests 917 or 1415 occur, subsequent actuation closure performance would closely resemble Test 903 rather than Tests 920 and 1419. Since the amount of seat leakage present after Tests 917 and 1415 was negligible and the original valve actuation pressure in Tests 920 and 1419 showed no sign of degradation, it is justifiable to conclude that no initial or subsequent actuation case at Byron/Braidwood would produce the reopening or cycling phenomena observed in these tests.

Criteria 3 - Structural Integrity

3. There were no observations in the EPRI tests of structural damage to the 6M6 that impeded its ability to function repeatedly. The same or getter results are expected at Byron/ Braidwood because:

   3.1 As noted in the preceding discussion, subsequent actuations are expected to take place with saturated steam fluid inlet conditions for Byron/Braidwood plant specific situations. This is clearly less severe than repeated loop seal discharges, such as those that EPRI experienced by conducting multiple loop seal tests without overhauling the 6M6 between tests.

   3.2 Despite high frequency cycling (other than loop seal discharge) in subcooled water tests and tests such as 920 and 1419, no damage sufficient to freeze the 6M6 in a stuck-open or stuck-closed position occurred during the duration of the tests. Although these tests are not actually applicable to Byron/Braidwood, they do demonstrate that the 6M6 is capable of withstanding a highly significant amount of structural distress beyond what is anticipated in the Byron/Braidwood application.

In summation, this assessment was conducted against criteria that are considered conservative with respect to Byron/Braidwood plant safety margins. In turn, the Crosby 6M6 test performance was clearly conservative with respect to these criteria. This perfor- mance, obtained in response to test conditions considered more severe than comparable conditions anticipated for Byron/Braidwood, has convinced us that the Byron/Braidwood plant specific needs for primary system safety valve operability have been demonstrated.

IV. Power Operated Relief Valve Evaluation

The Byron/Braidwood pressurizer power operated relief valves are Copes-Vulcan 3 inch Model D-100-160 type, with 17-4PH cage and 316SS stellite clad plug.

We have determined that the EPRI tests of the 3 inch Copes-Vulcan 17-4PH/316SS stellite clad version of this valve are applicable to the Byron/Braidwood PORVs. The referenced EPRI report sections that apply to Byron/Braidwood are as follows:

1. Safety and Relief Valve Test Report - Section 4.6
2. Valve Selection/Justification Report - Section III-B2
3. Test Condition Justification Report - Sections 2.2 and 3.8
4. Plant Conditions Justification Report - Westinghouse Plants - All sections related to 4 loop plants.

The technical approach used to perform the Byron/Braidwood PORV operability final assessment was fundamentally the same as that used for the Crosby 6M6 safety valve. The only differences were; 1) operability was assessed in terms of the most severe bounding fluid inlet conditions achievable at Byron/Braidwood rather than for expected fluid inlet conditions only as shown in Appendix A for safety valves, and 2) the criteria used to make the assessment are considered to be significantly more conservative than those used for safety valves, in the interest of mitigating unnecessary safety valve challenges.

The PORV operability function considered pertinent to Byron/Braidwood is mitigation of primary system transient pressure increases during cold overpressurization events. The criteria indicative of the Copes-Vulcan valve's capability to perform this function are; 1) full opening on command regardless of fluid inlet pressure or state, within the opening time assumed in the Byron/Braidwood cold overpressurization system design, 2) full closure on command regardless of fluid inlet pressure or state, within the closure time assumed in the Byron/Braidwood cold overpressurization system design, and 3) internal and external strutural integrity of the PORV sufficient to allow subsequent actuations in the event of repeated overpressure transients.

The purpose of defining these criteria and the application of them are consistent with criteria used for the Crosby 6M6 safety valve. In summary, performance within the criteria indicates that no challenge to Byron/Braidwood safety margins exists.

Conclusions

Quantitative review of the EPRI Codes-Vulcan tests which bound the range of fluid conditions achievable at Byron/Braidwood have led to the conclusion that this valve is capable of performing the operability function required within the criteria identified. It is also concluded that, since the Copes-Vulcan PORV test performance was within the present Byron/Braidwood cold overpressurization system design margins, that no equipment or analytical corrective actions are warranted in the interest of plant safety. Evidence providing technical support for these conclusions with respect to each criteria is as follows:

Criteria 1 - Opening

1. In all the EPRI tests of this PORV, it always achieved full disc opening within a period of 0.66 seconds, regardless of the fluid inlet conditions tested. Equally favorable results are anticipated for the Byron/Braidwood plant specific case because:

   1.1 The fluid inlet conditions tested envelop the range of conditions achievable at Byron/Braidwood in cold overpressurization events, as shown in Plant Conditions Justification Report - Westinghouse Plants (Reference 1d1).

   1.2 The PORV opening time assumed in the Byron/Braidwood cold overpressurization system design is 2 seconds, which conservatively exceeds the 0.66 second opening time observed as a maximum in all the EPRI tests.

Criteria 2 - Closure

2. In all the EPRI tests of this PORV, it always achieved full disc closure within a period of 1.24 seconds, regardless of the fluid inlet conditions tested. Equally favorable results are anticipated for the Byron/Braidwood plant specific cases because:

   2.1 As stated in Item 1.1, the fluid inlet conditions tested envelop the range of conditions achievable at Byron/Braidwood in cold overpressurization events.

   2.2 The PORV closure time assumed in the Byron/Braidwood cold overpressurization system is 2 seconds, which conservatively exceeds the 1.24 second opening time observed as a maximum in the EPRI tests.

Criteria 3 - Structural Integrity

3. Upon disassembly inspection of the Copes-Vulcan PORV, conducted only after all of the EPRI tests had been completed, no evidence of structural damage was observed that could in any way be interpretated as a challenge to operability for repeated overpressure transients. Equally favorable results are anticipated for the Byron/Braidwood plant specific case because:

   3.1 EPRI test performance regarding seat leakage, considered to be a conservative indicator of internal structural distress, never exceeded 0.0042 GPM. This is clearly insignificant in comparison to the Byron/Braidwood unidentified RCS leakage limit of 1 GPM.

   3.2 No performance differences were observed in the EPRI tests, regardless of any applied bending moment preloads.

In summation, this assessment was conducted against criteria that are considered conservative with respect to Byron/Braidwood plant safety margins. In turn, the Copes-Vulcan PORV test performance was clearly conservative with respect to these criteria. This performance, obtained in response to test conditions considered more severe than comparable conditions anticipated for Byron/Braidwood, has convinced us the Byron/Braidwood plant specific needs for primary system PORV operability has been demonstrated. We are also convinced that this demonstration extends beyond the considerations of cold overpressurization events only, such that the Byron/Braidwood PORV operability for all overpressure transient functions is assured. This assurance provides an added measure of confidence that the objective of mitigating unnecessary safety valve challenges has been attained.

V. PORV Block Valve Operability

We are in full agreement with the generic PWR utilities' position identified in reference 2. As noted in the documentation associated with the generic response, the Byron/Braidwood PORV block valves are 3 inch valves.

Having reviewed the EPRI/Marshall Velan tests, documented in reference 2, we have determined that the test results are applicable to Byron/Braidwood. We have therefore concluded that the Byron/Braidwood plant specific requirements relative to PROV block valve operability have been satisfied.

VI. S/RV Piping and Support Adequacy

Installation of the S/RV piping and supports at Byron and Braidwood is presently incomplete. The analysis necessary to demonstrate the adequacy of these components is expected to be complete in early 1983.

The ongoing assessment is based in comparison of the thermal-hydraulic forcing function predictions used in the system design to new predictions that are being generated through the use of the RELAP 5 MOD 1 computer code. The adequacy of this computer code was demonstrated as part of the EPRI test program documented in reference le.

The comparison criteria that is being used in the assessment is straightforward. If the original design forcing function predictions equal or are more severe than the RELAP 5 MOD 1 predictions, it can be concluded that the original design basis adequacy has been verified. If the RELAP 5 MOD 1 predictions are found to be more severe than the original predictions, this will be considered as sufficient justification to review the impact of the differences noted on the system structural analysis.

The elements of our assessment action plan are as follows:

1. New thermal-hydraulic forcing functions predictions for this system are being generated through use of the RELAP 5 MOD 1 computer code. An element of this effort will be a sensitivity study to determine the effects of evaluating the loop seal temperature. Based on a comparison of the EPRI cold loop seal versus warm loop seal test results, it is anticipated that force reduction benefits are likely to be very significant, by as much as a factor of 10 to 1. Safety valve inlet temperatures of up to 300°F will be considered. Higher temperatures are specifically excluded in order to avoid potential seat leakage challenges.

2. The RELAP 5 MOD 1 predictions will be compared to the original system design predictions. The comparison will initially exclude predictions generated in the elevated loop seal temperature studies.

   2.1 If the initial comparison reveals that the RELAP 5 MOD 1 predictions equal or are less severe than the original predictions, the conclusion will be drawn that the original design basis adequacy has been verified.

   2.2 If the initial comparison reveals that the RELAP 5 MOD 1 predictions are more severe than the original predictions, the impact of the differences noted on the system structural analysis will be addressed.

      2.2.1 If the determination is made that the differences noted will not have any significant adverse impact on the system structural analysis, the technical justification for this conclusion will be documented.

2.2.2 If it is determined that the differences noted are likely to increase the structural loadings on the system to the extent that additional actions are warranted, options will be considered to either; 1) Upgrade the structural supports, or 2) extend the forcing function comparison to include the results of the elevated loop seal temperature studies. The objective in selecting this last option would be to evaluate the effectiveness and feasibility of prospective elevated loop seal temperature hardware changes, in terms of the anticipated structural loadings to the current analysis envelope.

ATTACHMENT B

E.23 Relief and Safety Valve Test Requirements (II.D.1)

POSITION

By letter dated April 1, 1982, D. P. Hoffman (Consumers Power) transmitted the Safety and Relief Valve Test Report for the EPRI PWR Safety and Relief Valve Test Program. This report summarizes all the operability test data collected on relief and safety valves. Byron/ Braidwood plants have Copes-Vulcan Model D-100-160 3-inch air-operated globe relief valves (316SS w/stellite clad plug and 17-4PH cage) and Crosby Model HP-BP-86, size 6M6 safety valves. Specific results in Sections 3.5 and 4.6 of the EPRI safety and relief valves test report are applicable to Commonwealth Edison plants. Final evaluation of the data indicates that the relief and safety valves will perform their intended functions for all expected fluid inlet conditions. Commonwealth Edison submitted the plant specific final evaluation confirming the adequacy of the relief and safety valves, and othe plant specific data for all relief and safety valve inlet conditions, by letter from T. R. Tramm dated October 26, 1982.

Regarding verification of Block functionability, this topic was discussed between the PWR utilities and the NRC staff. Commonwealth Edison concurs with the final conclusions reached between the PWR owners and the NRC staff.

APPENDIX A

PROBABILISTIC EVALUATION OF HIGH PRESSURE LIQUID CHALLENGES TO SAFETY/RELIEF VALVES IN THE ZION, BYRON/BRAIDWOOD PWR PLANTS

Submitted to:

COMMONWEALTH EDISON COMPANY
Reliability & Design Engineering Group

Report Prepared by:

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June 25, 1982

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1. INTRODUCTION

Based upon the requirements of NUREG-0737[1], owners of nuclear power plants must perform plant-specific evaluations to ensure that the Power Operated Relief Valves (PORVs) and spring-loaded Safety/Relief Valves (S/RVs) are operable and provide effective pressure relief under the possible range of discharge conditions. The Electrical Power Research Institute (EPRI) has conducted the PWR Safety and Relief Valve Test Program[2] to provide a generic basis for addressing these requirements. This program is nearly completed, and data is available to provide the starting point for evaluations by individual utilities.

EPRI experimental results, as well as various independent analyses[3,4], have shown that saturated liquid and subcooled liquid discharge through the safety valves constitute the most severe challenges to such valves and associated piping networks. There are two major reasons for concern about S/RV liquid discharge.

First, S/RVs open rather quickly, with stroke times of 50 ms or less, so that the discharge piping may experience large dynamic loads as a wave front of liquid enters pipe segments. PORVs open more slowly over periods of 1/2s to 1s, and thus give rise to smaller dynamic loads. Further, the loads are proportional to mass flow rate, which can be up to 6 times higher for liquid flow compared to steam flow. Thus, from the point of view of discharge piping stresses, the combination of liquid discharge through an S/RV represents the greatest challenge.

Second, it has been shown by EPRI test results and by independent analyses[5] that plants with long SRV inlet piping may be subject to chatter oscillations of the spring loaded valve. Such oscillatory behavior is most likely for subcooled liquid discharge, because the high mass flow rate generates water-hammer pressure waves of very large amplitude. Oscillatory behavior may also be observed during the expulsion of subcooled liquid loop seals.

For each of these two concerns, the situation is most severe for far-subcooled liquid discharge, which gives the maximum mass flow rates. Saturated or "slightly-subcooled" liquid discharge is somewhat less troublesome, because there may be sufficient flashing at the valve so that two-phase flow effects can substantially reduce the mass flow rate and thus mitigate the event.

It is believed that liquid discharge can be tolerated both from the perspectives of discharge piping stresses and of valve oscillations. However, piping analysis is under way to verify the ability of discharge piping to accomodate the required loads. High pressure liquid discharge, while admittedly a conceivable event, is extremely unlikely when considered in the context of available systems, operating procedures and time for operator action, and infrequency of the initiating events. It is desirable to understand the frequency of occurrence of SRV liquid discharge in order 1) to provide perspective and recognition of the relative unimportance of subcooled discharge, in view of relatively high stresses which may be computed for such discharges, and 2) to provide a more rational basis for defining realistic inlet conditions for SRV discharge by eliminating those conditions which are shown to be very unlikely.

The present study uses techniques[6] from Probablistic Risk Assessment to evaluate the frequency at which S/RV liquid discharge may be encountered in the Zion, Byron, and Braidwood plants of Commonwealth Edison Company. SAI has performed similar analyses to evaluate liquid discharge risks in Boiling Water Reactors. Based upon previous generic analysis by Westinghouse[7], and upon additional plant-specific hand calculations, event trees are developed to qualitatively describe the event sequences which may cause S/RV liquid discharge and to identify the system functions and operator actions which may favorably or unfavorably affect the outcome. Fault tree analysis is then used to quantitatively evaluate the failure probabilities for the required system and operator responses. Extensive use of previous research results[8,9,10,11] for event initiation frequences, component failure rates, and human error probabilities are used where applicable.

Results show that liquid discharge from the pressurizer safety-reliev valves is extremely unlikely for the Zion, Byron and Braidwood plants. Subcooled liquid discharge may occur at the rate of 9.6E-8 events/reactor unit-year at

Zion and 5.8E-8 events/reactor-year at Byron/Braidwood. Saturated liquid discharge may occur with frequences of 3.65E-8 and 1.05E-8 events/reactor-year at Zion and Byron/Braidwood respectively.

2. EVENT TREE ANALYSIS

2.1 General Discussion

In support of the EPRI/PWR Safety and Relief Valve Test Program, Westinghouse has performed a generic evaluation of the expected range of fluid inlet conditions for pressurizer safety and relief valves for plants designed by Westinghouse. The resulting report[7] provides a comprehensive discussion of all transients with the potential for high pressure discharge as well as bounding calculations for the actual conditions to be expected by 2-loop, 3-loop and 4-loop plants.

That report provides the starting point for our analysis. However, as a generic bounding analysis the Westinghouse report quite properly 1) asssumes multiple system failures without evaluating their likelihood, 2) ignores the helpful effects of any operator action, and 3) fails to take the credit for plant-specific characteristics which mitigate the events. The present report 1) modifies the important Westinghouse reported sequences in accordance with the plant-specific characteristics of Zion and of Byron-Braidwood and 2) evaluates the frequency of the important sequence by incorporating results of fault tree analysis described in Section 3.

The basic conditions for liquid discharge (of any kind) is that the pressurizer pressure reach the S/RV set point of 2485 psig and that the pressurizer water level rises to the top at the same time. Because of the different levels of interest in these scenarios, as discussed in Section 1 we will distinguish between:

- i) opening discharge of subcooled liquid
- ii) opening discharge of saturated liquid
- iii) opening discharge of saturated steam followed by delayed discharge of saturated liquid.

The Westinghouse report identified the transients potentially leading to liquid discharge as:

i) FSAR events - a) Feedwater Pipe Rupture*

b) Accidental Depressurization

ii) Extended High Pressure Injection Event (Spurious Safety Injection)*

iii) Cold Overpressurization events - a) mass input event*

b) heat input event

The starred events will be analyzed in detail in this report. FSARs for Zion[12] and for Byron/Braidwood[13] make it clear that the Feedwater Pipe Rupture is the only FSAR event of concern for these plants. The "heat input event" for cold overpressurization was included under the mass input event, because the latter is, 1) more likely to occur, 2) much easier to characterize quantitatively (without extensive system transient analysis), and 3) less easily mitigated because it is a very fast acting event.

2.2 Extended High Pressure Injection at Power

Spurious actuation of the safety injection system can be caused by operator error or by a false actuating signal. Should the operator fail indefinitely to recover from safety injection and in particular fail to trip the centrifugal charging pumps, the primary system may fill with subcooled water. Following an initial drop in pressure due to primary coolant contraction, the system would begin to repressurize after the pressurizer becomes solid due to continued charging pump operation. This event has a fairly high frequency of occurrence, but it is also very easy to detect and terminate. Generic data[8] for PWRs leads to a frequency of 1.6×10^{-1} events/reactor-year for Byron-Braidwood, which is a relatively new design and thus representative of the general PWR population. Plant specific data[8] obtained for Zion dictates a frequency of 6.0×10^{-1} for the Zion plant.

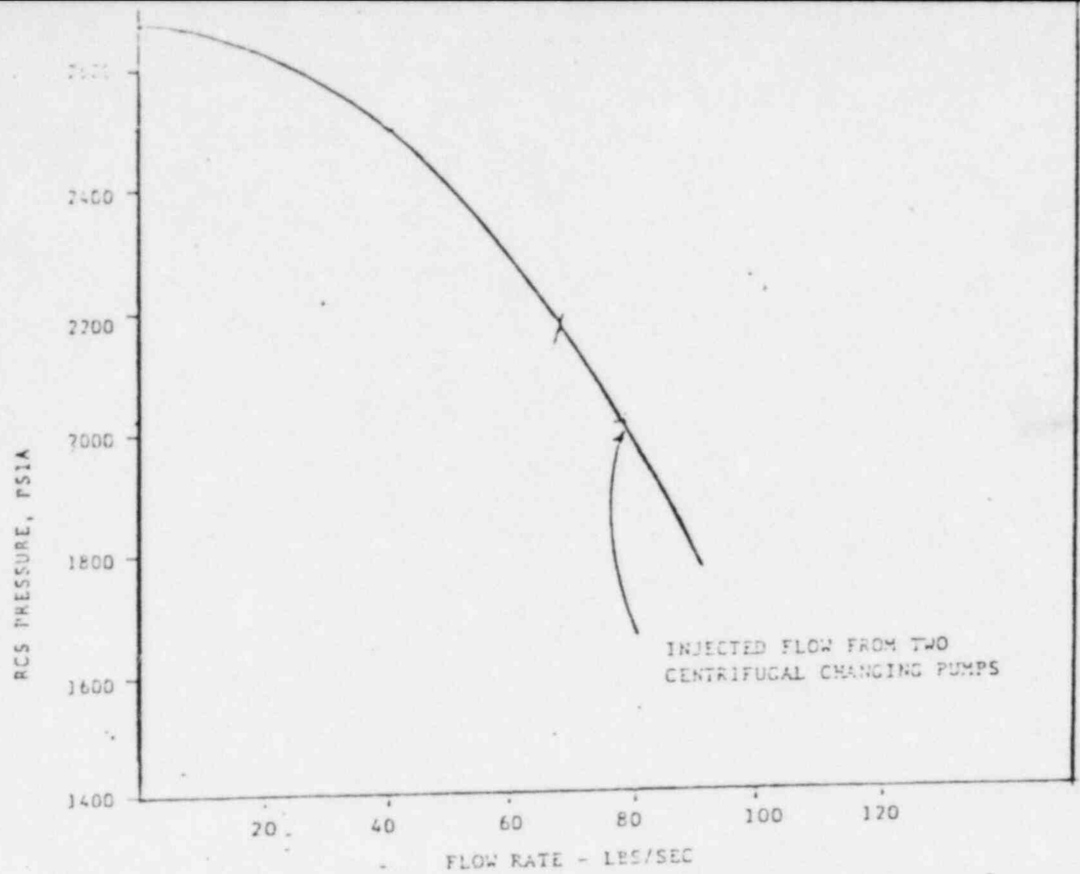
A Safety Injection Signal (SIS) results in a reactor trip followed by a turbine trip. The letdown is automatically isolated and is therefore unavailable for pressure relief. The centrifugal charging pumps force ECCS water into two primary cold legs. Since there is no letdown (which in any case does not have sufficient capacity for mitigation) the primary loop water inventory steadily increases. At first, the pressure drops due to the coolant contraction, until the pressurizer water level increases to the top. With pressurizer control lost, the system would then pressurize until high pressure liquid is discharged through the PORVs or safety relief valves, unless there is appropriate operator action or successful discharge through PORVs.

For Zion and Byron/Braidwood, successful operation of only one PORV is sufficient to remove liquid supplied by both charging pumps and thus to eliminate the possibility of SRV liquid discharge. Figure 2.1 shows the characteristic curves for the charging pumps at Zion and Byron/Braidwood. At 2335 psig and saturated steam conditions, the single-PORV flow rates comparable to these curves are 58.3 lb/s for Zion and 457 gpm for Byron/Braidwood. For saturated liquid PORV discharge, mass flow rates are even higher. It is apparent that one PORV provides adequate pressure relief for extended safety injection. The first branch of the event tree of Figure 2.2 reflects this fact. In the fault tree evaluation of PORV, only automatic actuation is considered; no credit is taken for operator action of the PORVs.

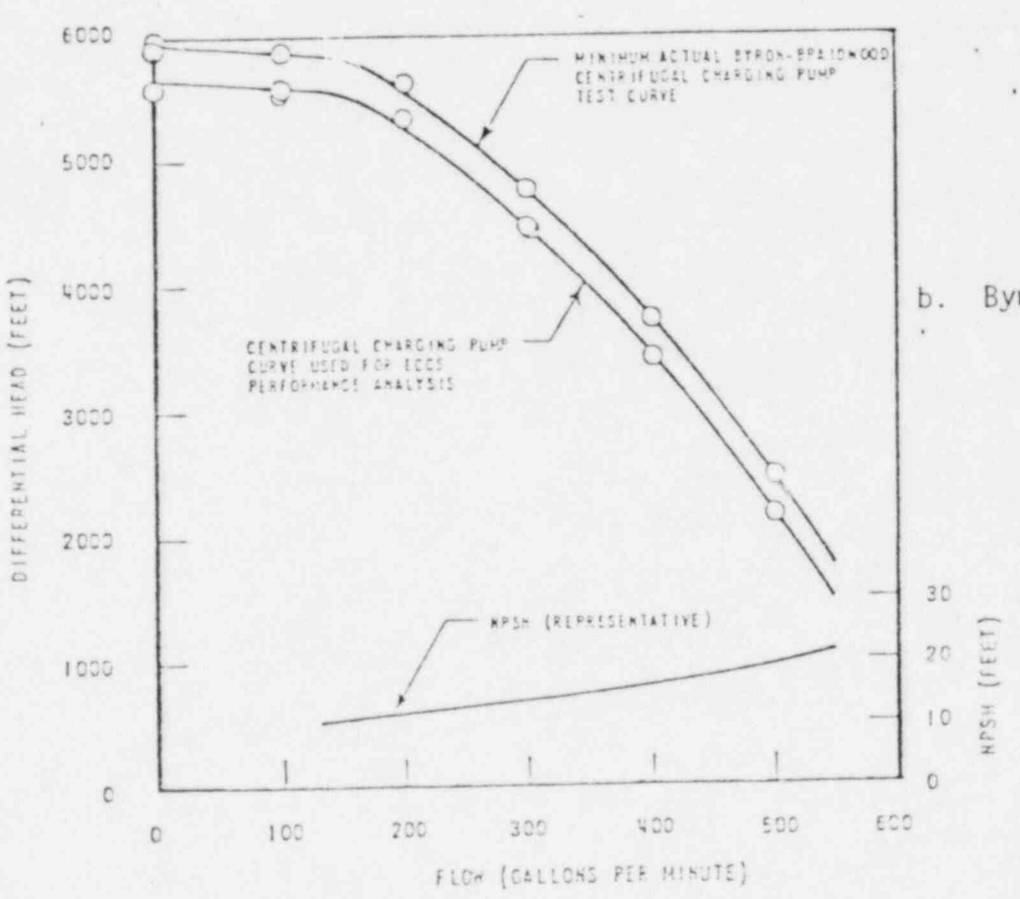
Given that both PORVs fail, a simple mass balance shows that at least 20 minutes is required for the pressurizer bubble to collapse and for liquid discharge to occur. However, there are clear cut operating procedures[14] for recovery from safety injection which will require the operator to reset the SIS within a few minutes. Further, this is an event which is neither extremely rare nor difficult to interpret, so there is a high likelihood that the event will be successfully terminated by the operator. The human response is analyzed in Section 3.3. The computed human error probabilities are expected to be very conservative, since the 20 minute response time obtained from the simple mass balance is expected to be a very short and thus conservative estimate of response time, which may be on the order of hours.

When the numerical results are obtained from section 3, it is found that the frequency for SRV liquid discharge following a spurious safety injection is 3.6E-8 events/reactor-year for Zion and 9.9E-9 events/reactor-year for Byron/Braidwood.

Further, the S/RVs would in any case be first challenged by steam discharge followed by a transition to saturated or slightly subcooled liquid discharge,

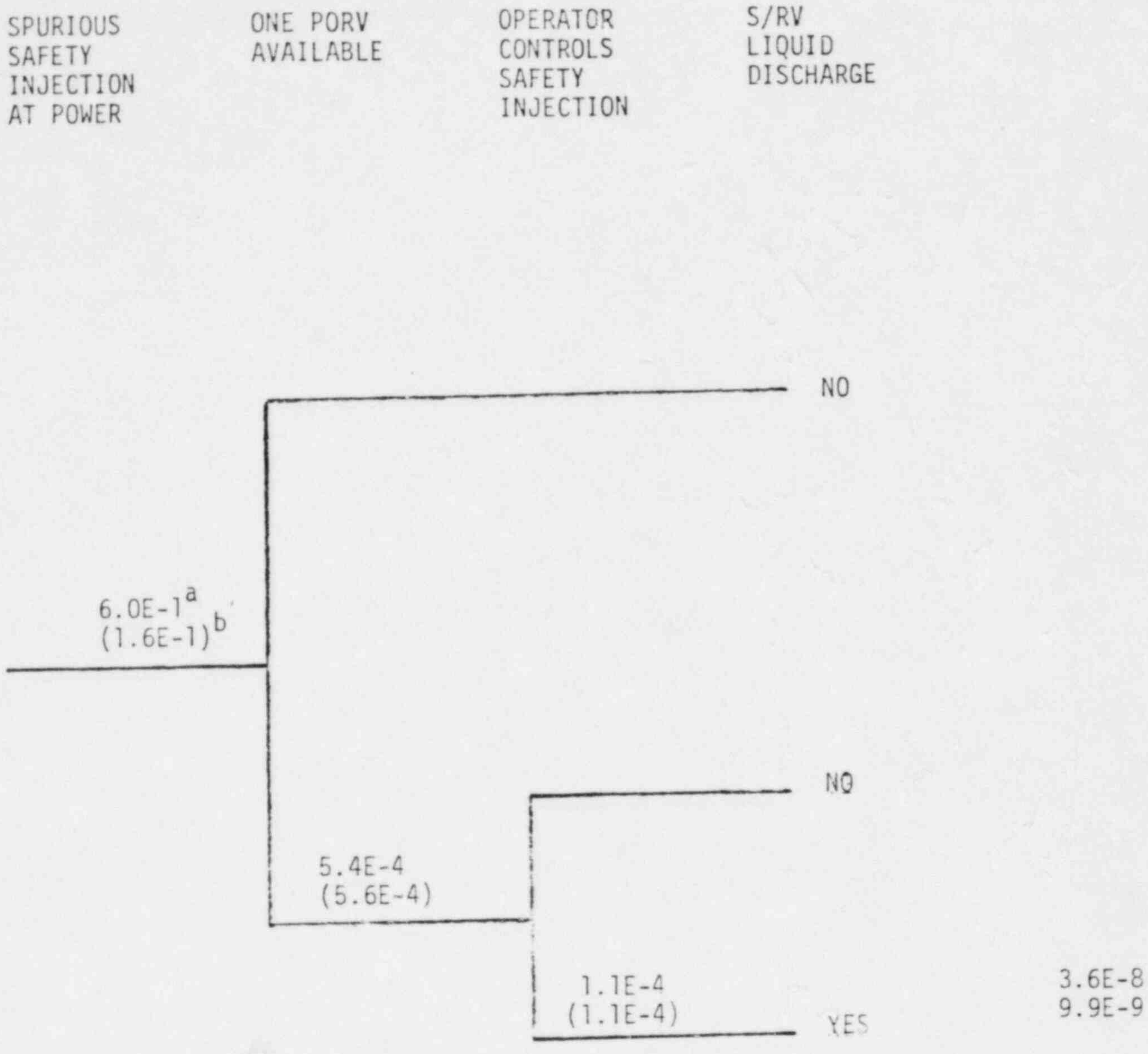


a. Zion



b. Byron/Braidwood

Fig. 2.1
CENTRIFUGAL CHARGING PUMP CHARACTERISTIC CURVES



[a]Zion

[b]Byron/Braidwood

Fig. 2.2

EVENT TREE FOR EXTENDED HIGH PRESSURE INJECTION AT POWER reducing the potential for chatter instability as well as the amplitude of dynamic loads on discharge piping.

2.3 Cold Overpressurization Event

A cold overpressurization event represents the most likely source for liquid S/RV discharge for which the liquid may not be preceded by any steam discharge. The entire system begins from cold shutdown conditions, so that 1) subcooled liquid is present throughout the primary loop so that S/RV discharge will assume maximum mass flow rates and thus face the greatest problems with respect to waterhammer instability and downstream piping loads, and 2) pressure control is inadequate to prevent rather rapid pressure excursions from occurring, since no steam bubble exists in the pressurizer.

An overpressurzation event from cold shutdown conditions can be mitigated by enhanced mass input to the primary loop. Only the centrifugal charging pumps are capable of raising the primary loop pressure as far as 2485 psig for S/RV discharge. The initiators essentially involve spurious and uncontrolled opening of a flow path through the centrifugal charging pumps. The most credible events causing cold overpressurization are:

a) Failure of the Air Supply System in the Zion Plant would cause the charging flow control valve and the letdown valve to fail closed. This in turn causes a net injection of mass by the centrifugal charging pump and a very high rate of primary loop pressurization. A value[15] of 8×10^{-4} failures/demand was used for this analysis; this number is thought to be very large and thus extremely conservative.

   [This scenario does not not apply to Byron/Braidwood; a loss of instrument air pressure would also close the valves (CV8324A and CV8324B) upstream of the Regenerative Heat Exchanger. Therefore, there would be no mass addition to the RCS in Byron/Braidwood.]

b) Failure of the charging flow control valve to operate as required; this could be caused by local valve failure or local failure in the air supply to the flow control valve. Based upon data from reference 11, the rate at which this initator occurs was determined to be 1.2×10^{-3} failures/year. Since the reactor is at low pressure and the centrifugal charging pumps flow rate increase with decreasing RCS pressure, a large mass injection causing a high rate of pressurization would occur. Although the letdown path is available throughout the transient the letdown relief rate is inadequate to effectively mitigate the event, and no credit is taken for letdown in the probabilistic analysis.

Further, no credit is taken for rapid operator action, because very high pressurization rates (up to 100 psi/s)[7] have been predicted for such events.

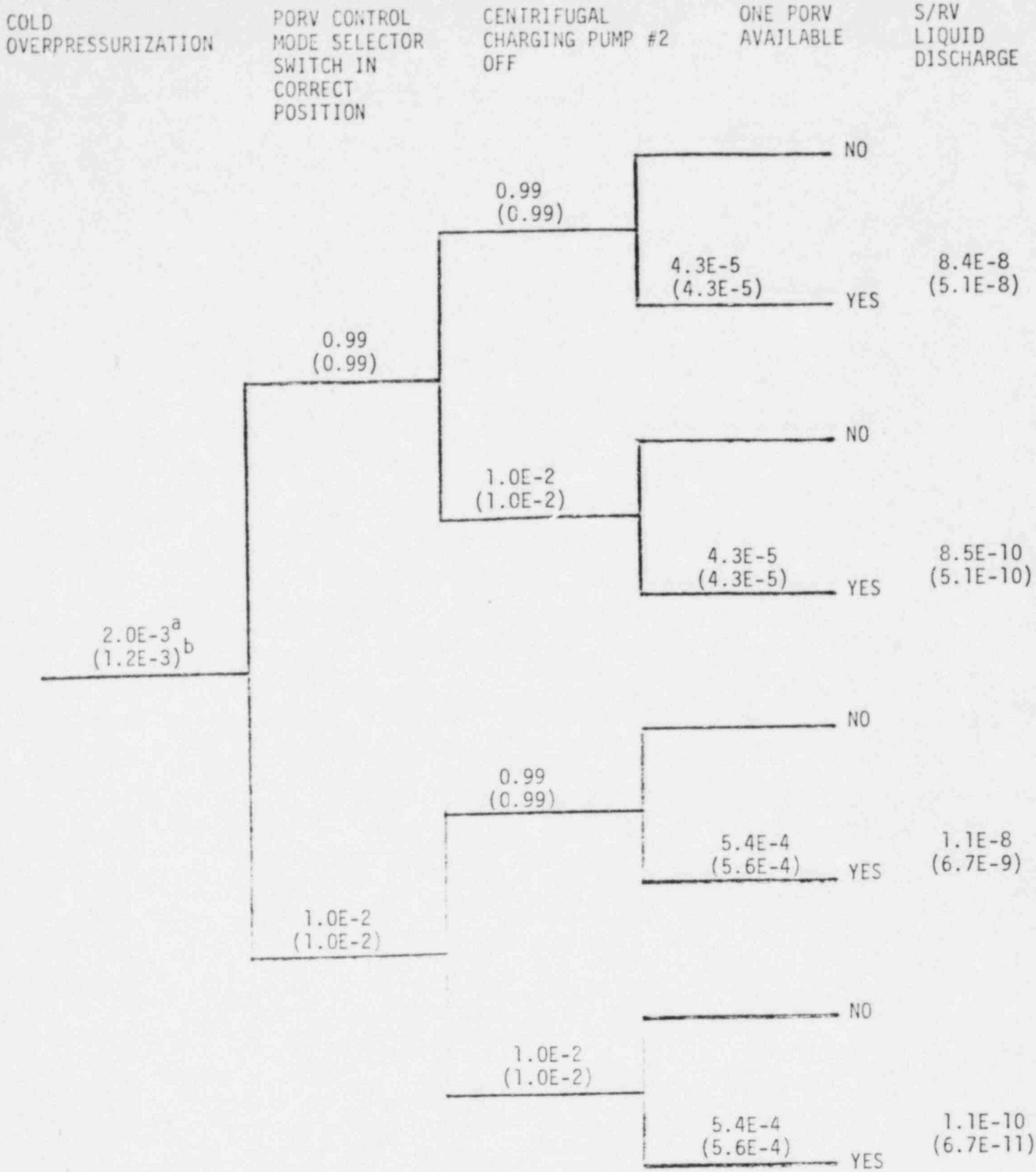
For Zion, the PORV set points during cold shutdown mode are set to 500 psig, as long as the PORV control mode Selector Switch is correctly placed at the "AUTO LOW TEMP POSITION." Zion operating procedure GOP-2 dictates this switch placement so that procedure violation would be necessary for the set points to remain at their at-power position.

For Byron/Braidwood, the operator must correctly re-set two switches (PORV Control Selector Switch to "AUTO" and cold overpressure control switch to"ON") in order to ensure that the correct PORV setpoint is chosen.

The first branch in the event tree in 2.3 involves failure of the operator to follow procedures for mode selection. According to NUREG/CR-1278, the human error probability (HEP) for failure to follow this type of procedure is .01.

The second branch concerns an operator error of turning on a second centrifugal charging pump in violation of procedure GOP-2.

The PORV failure rates depend upon whether the PORVs have been set in the cold shutdown mode (correct) or in the at-power mode (incorrect). In the latter



[a]Zion

[b]Byron/Braidwood

Fig. 2.3
EVENT TREE FOR COLD OVERPRESSURIZATION

case the failure rates are identical to those developed for the at-power case (see Section 3.1.1). However, if the PORVs are in the correct mode, then failure rates are those developed in Section 3.1.3.

When numerical results are obtained from Section 3, it is found that the frequency for SRV discharge due to a cold overpressurization event is 9.6E-8 events/reactor year for Zion and 5.8E-8 events/reactor year for Byron/Braidwood.

2.4 Main Feedwater Pipe Rupture Event

A main feedwater pipe rupture, if large enough, can prevent the addition of sufficient feedwater into the steam generators to sustain shell-side fluid inventory. Should the large break occur between the check valve and the steam generator, the water can quickly discharge through the break causing a rapid loss of heat sink in the affected loop.

The FSAR transient response for Byron/Braidwood is included as Figure 2.4. This event is not analyzed in the Zion FSAR, but the results are not expected to be significantly different. Following the injection of cold ECCS water, the pressurizer pressure and level both initially drop due to the negative surge rates caused by primary loop water shrinkage. After about 5 minutes, the pressure again rises due to reduced heat removal through the steam generators. Eventually saturated steam is discharged through the S/RVs at about 7 minutes into the transient followed by a transition to saturated liquid discharge after 13 minutes into the event. The safety relief valves thus serve as a heat (and mass) sink to stabilize the transient until their final closure at 20 minutes.

In the FSAR, no credit is taken for the operation of the PORVs, which would open at 2335 psig and discharge saturated steam for pressure relief. As discussed in Section 2.2 by Figure 2.1, one PORV would be adequate to balance the cold-water input from both safety injection pumps and thus control the primary loop pressure to 2335 psig, so that the spring-loaded safety valves would never open. It is possible, however, that the PORVs may eventually

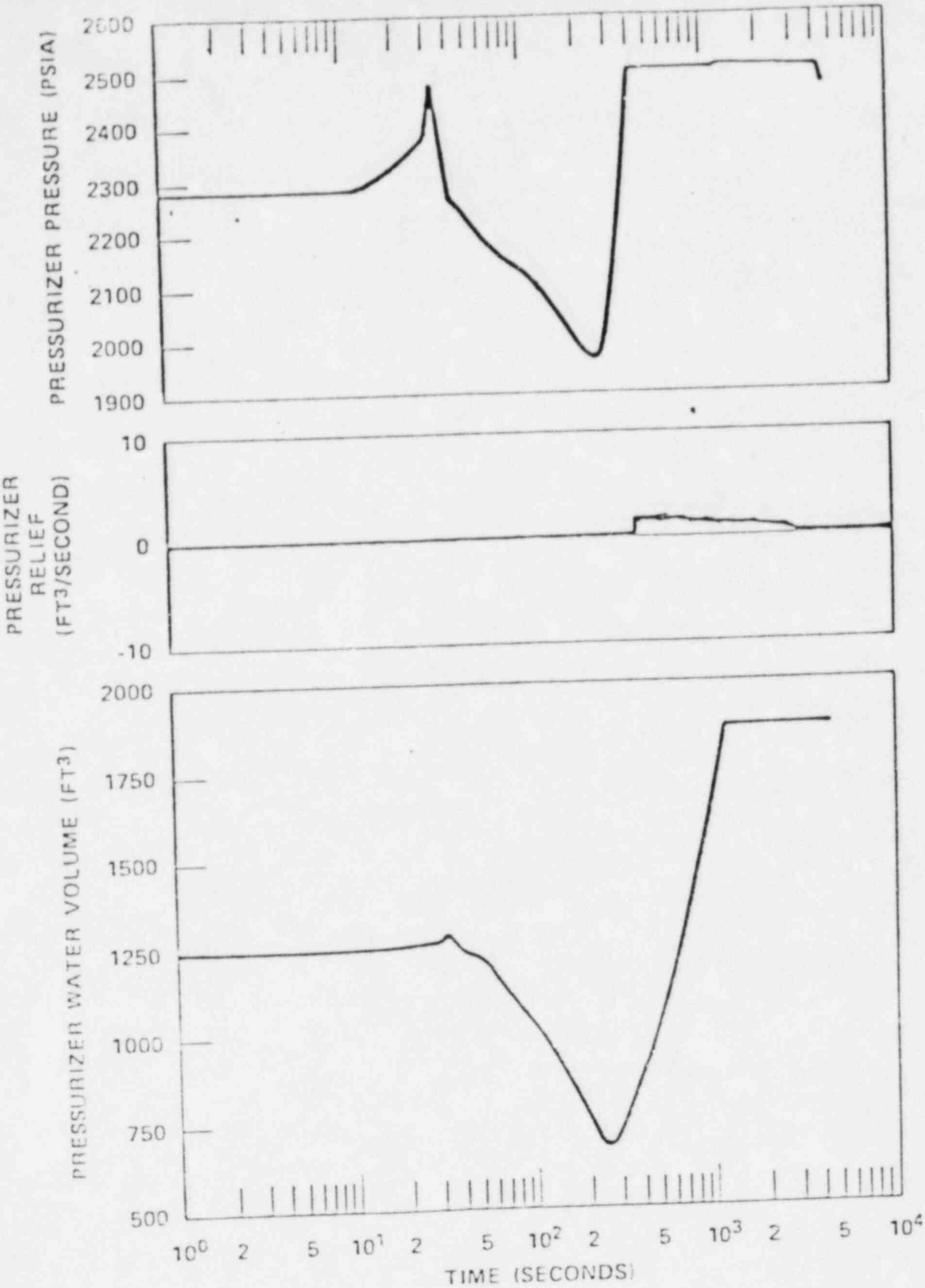


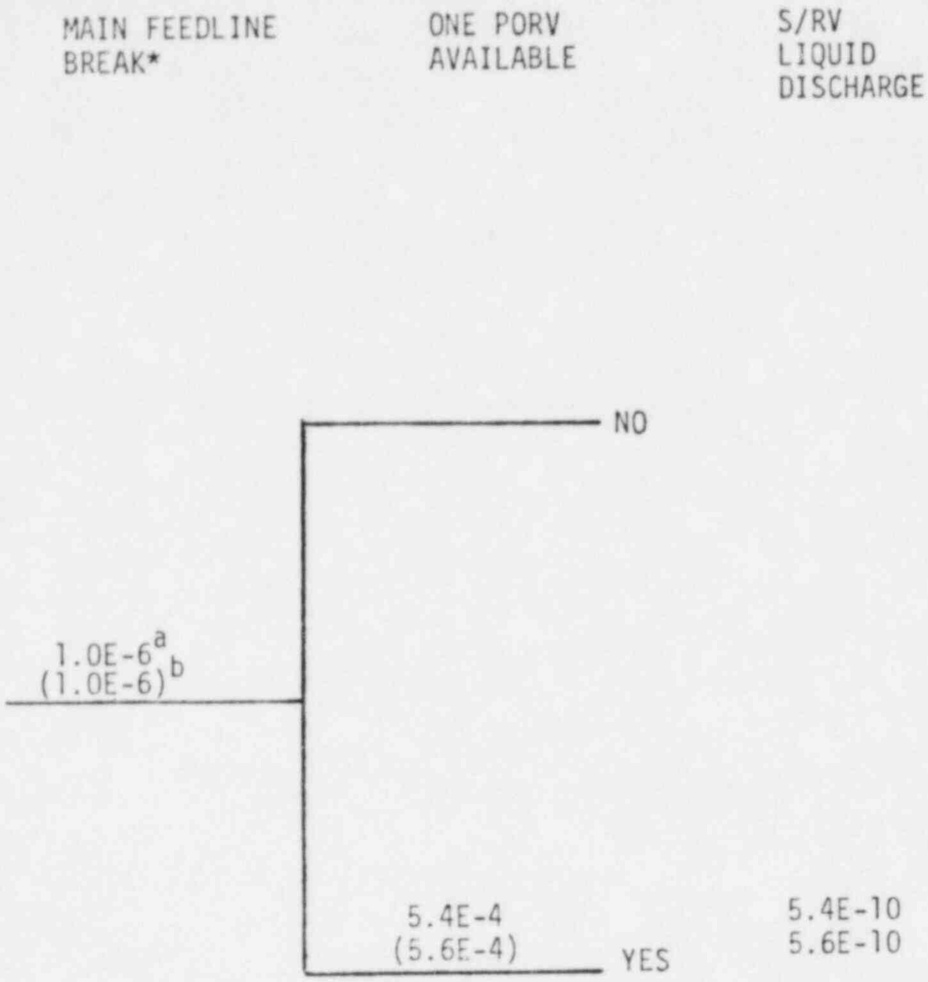
Fig. 2.4

BYRON/BRAIDWOOD STATIONS: PRESSURIZER PRESSURE, WATER VOLUME AND PRESSURIZER RELIEF FOR MAIN FEEDLINE BREAK WITH OFFSITE POWER (Fig. 15-2.4 - B/B FSAR) themselves discharge some saturated liquid after a transition from saturated steam discharge.

No credit is taken for operator action in the very simple event tree of Figure 2.5, although shutting down the charging pumps would at any time effectively terminate the pressure up-transient leading to high pressure discharge. Operating procedures exist[14] which require such actions to be taken once the pressurizer pressure has stabilized and begun its increasing trend.

The initiation frequency of this event is very small, because it involves a large break in a relatively short stretch of piping between the check valve and steam generator. Based upon the WASH-1400 pipe failure probability of 1.0E-6/yr-section, the initiation frequency is chosen to be 10^{-6} events/reactor-year.

The PORV system failure probability is computed in Section 3.1 to be 5.4E-4 and 5.6E-4 for Zion and Byron/Braidwood, respectively, resulting in a very small event frequency which would be even smaller should operator action be included. Compared to the other events of Sections 2.2 and 2.3, the feedwater line break is thus insignificant as an initiator for high pressure liquid discharge through safety relief valves. Further, such liquid discharge is always preceded by saturated steam discharge, so that chatter instability is precluded, and the transient loads to discharge piping would be substantially mitigated.



[a]Zion

[b]Byron/Braidwood

*Break between check valve and SG

Fig. 2.5
EVENT TREE FOR MAIN FEEDLINE BREAK

3. FAULT TREE ANALYSIS

The analysis of the event trees described in the previous section shows that the Power Operated Relief Valves (PORVs) constitute the only equipment capable of eliminating the possible liquid challenges to the Safety/Relief Valves (S/RVs) in the Zion and Byron/Braidwood (B/B) plants. Fault tree analysis techniques were used to quantify the unavailability of the PORVs for the Zion and Byron/Braidwood plants. Note, from the event trees that the availability of one PORV is sufficient for mitigation of the posible incidents. The Zion plant is discussed in detail, and for Byron/Braidwood only the system differences and modifications are discussed.

3.1 Fault Tree Analysis of PORV's for Zion

Two pressurizer power operated relief valves (PORVs) exist in each unit of the Zion plant. Figure 3.1 shows a sketch of one of those PORVs with its actuation air supply. The PORVs are actuated on a signal from the Overpressurization Detection Instrumentation (Fig. 3.2) through its actuation circuit shown in fig. 3.3. As directed by operating procedures,[14] "the operator adjusts the position of the PORV Control Mode Selector Switch according to the Reactor mode of operation. If the Reactor is operating at power the switch is on "AUTO", and if the Reactor is in Cold Shutdown mode the switch is on "AUTO-LOW TEMP." As seen in Fig. 3.2, these different switch positions dictate different PORV actuation signals. Therefore, the unavailability of the PORVs is dependent upon the mode of operation of the Reactor. The following sections discuss the fault-tree analysis for the Reactor "At Power" and "Cold Shutdown" modes of operation.

3.1.1 Reactor at Power

A fault tree for failure of both PORVs to open when the reactor is at power was constructed and quantified. The fault tree is shown in Fig. 3.4. Note that in Fig. 3.4 the details are given only for one of the PORVs; the other is identical to the first one. The results of this analysis show an overall median unavailability (for both PORVs) equal to 5.4E-4/demand.

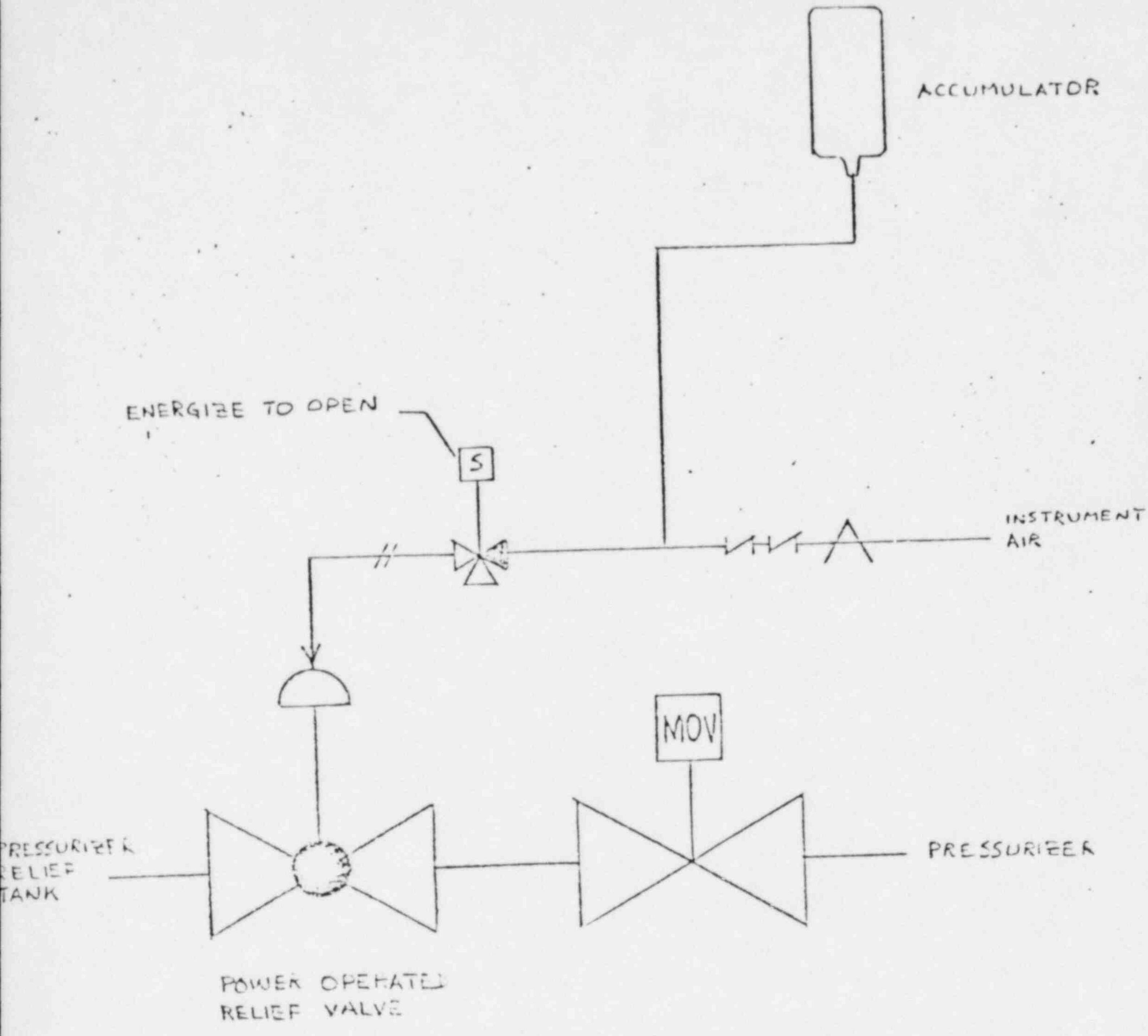


Fig. 3.1

POWER OPERATED RELIEF VALVE SKETCH WITH ACTUATION AIR SUPPLY

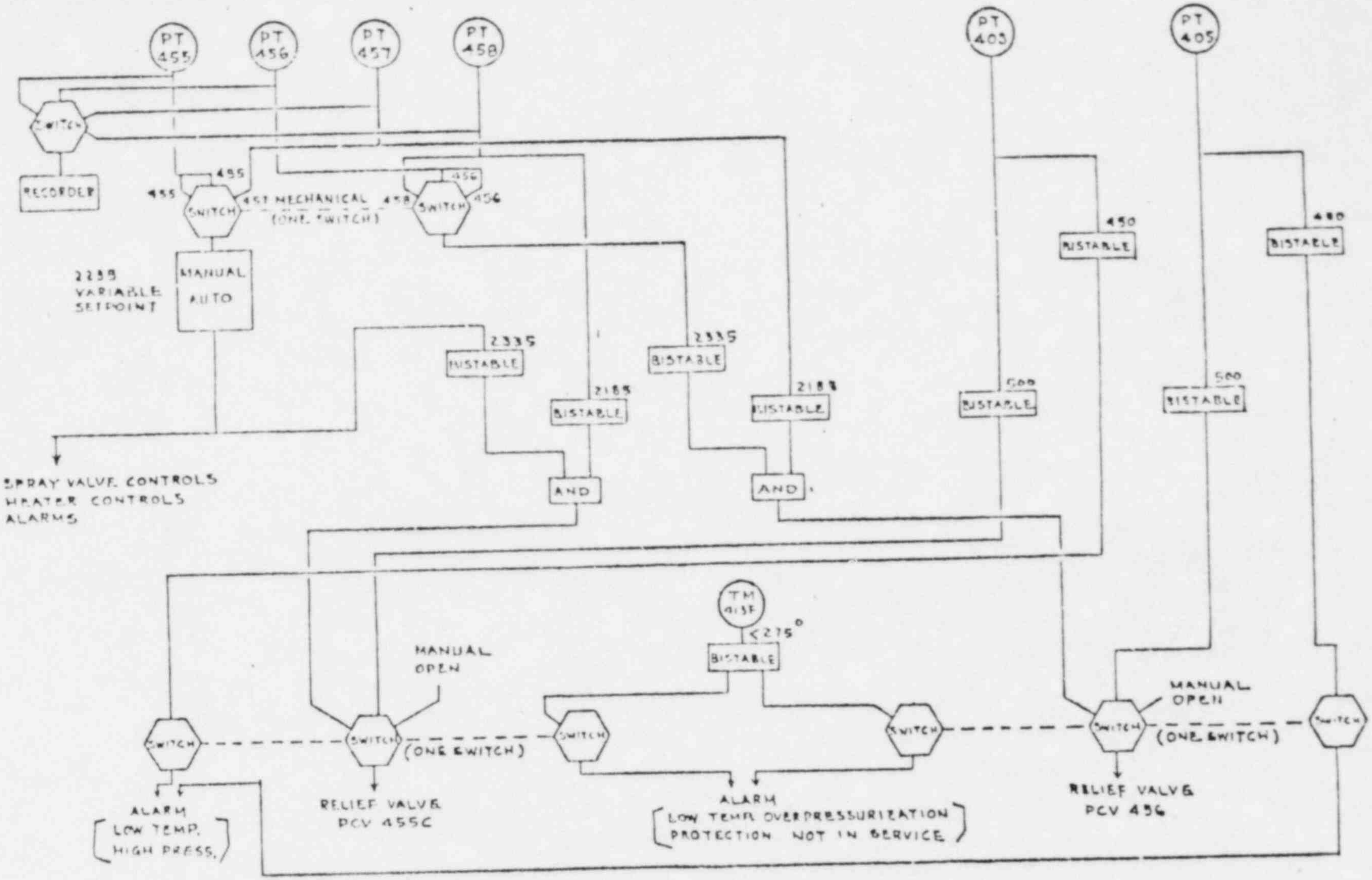


Fig. 3.2

ZION OVERPRESSURIZATION DETECTION INSTRUMENTATION

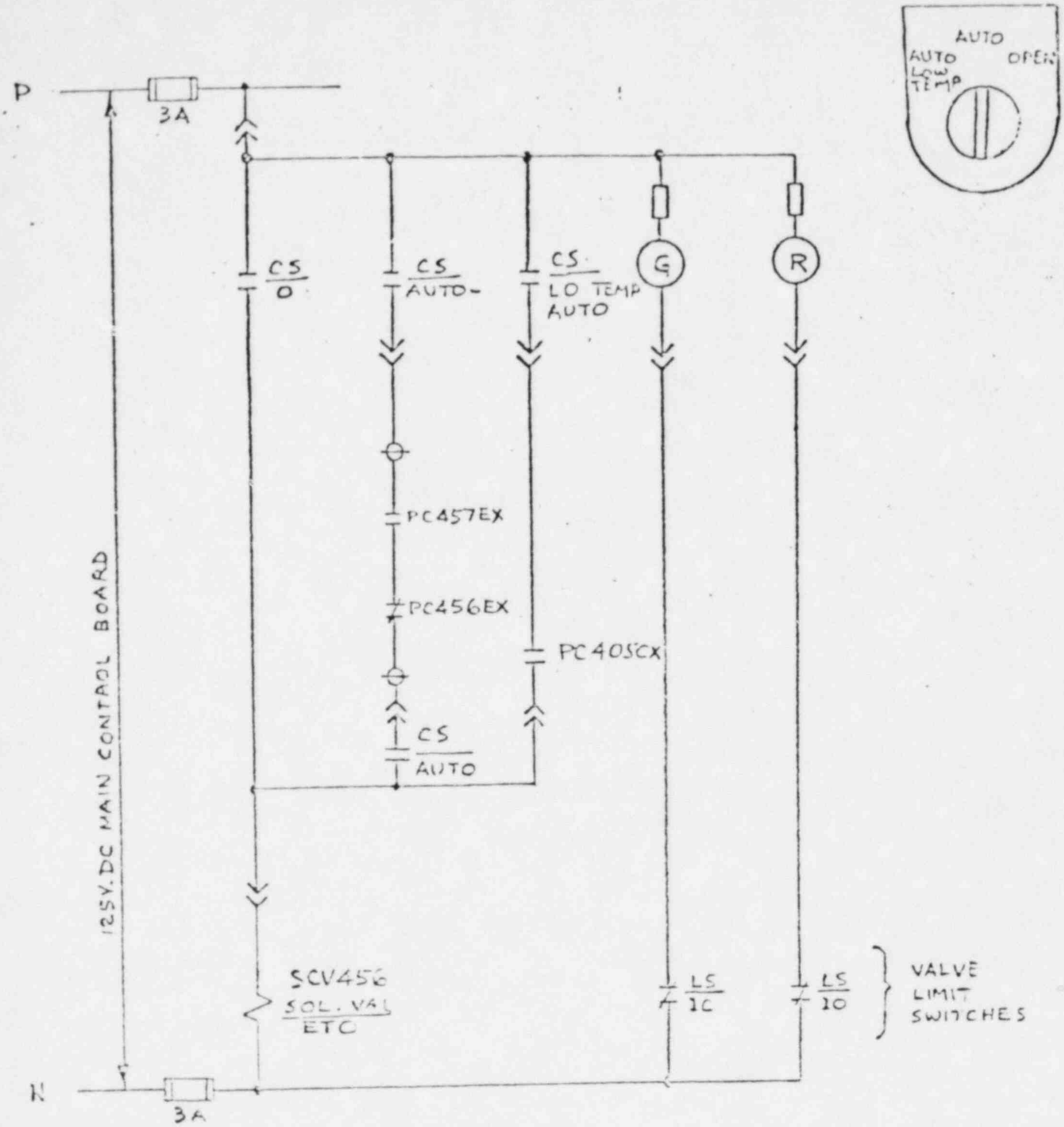


Fig. 3.3

ZION POWER OPERATED RELIEF VALVE PCV456
ACTUATION CIRCUIT (TYPICAL)

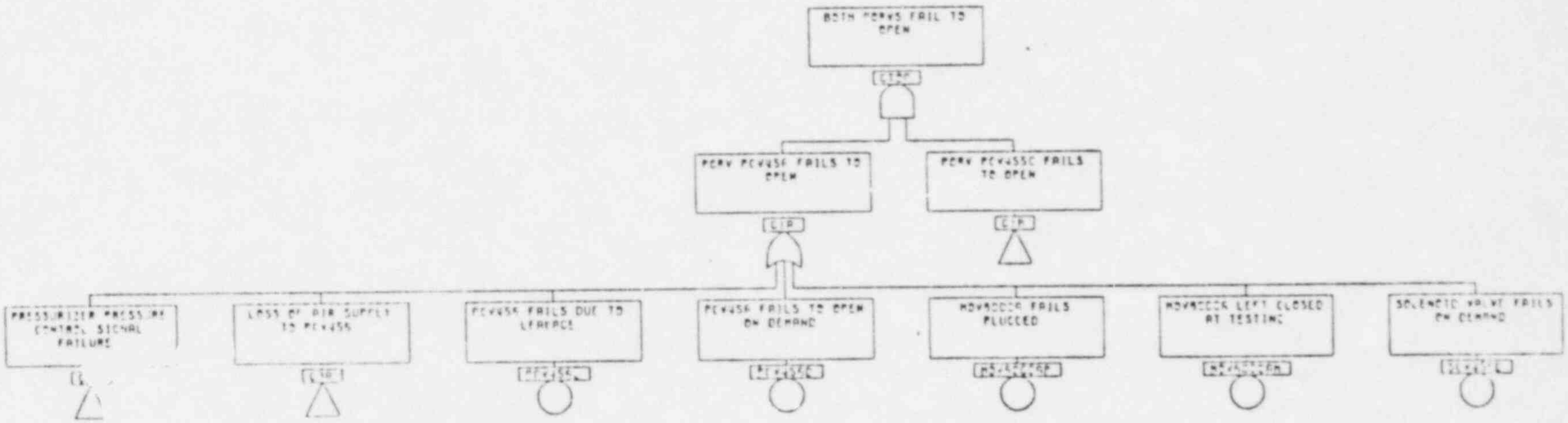


Fig. 3.4

REACTOR AT POWER - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - ZION

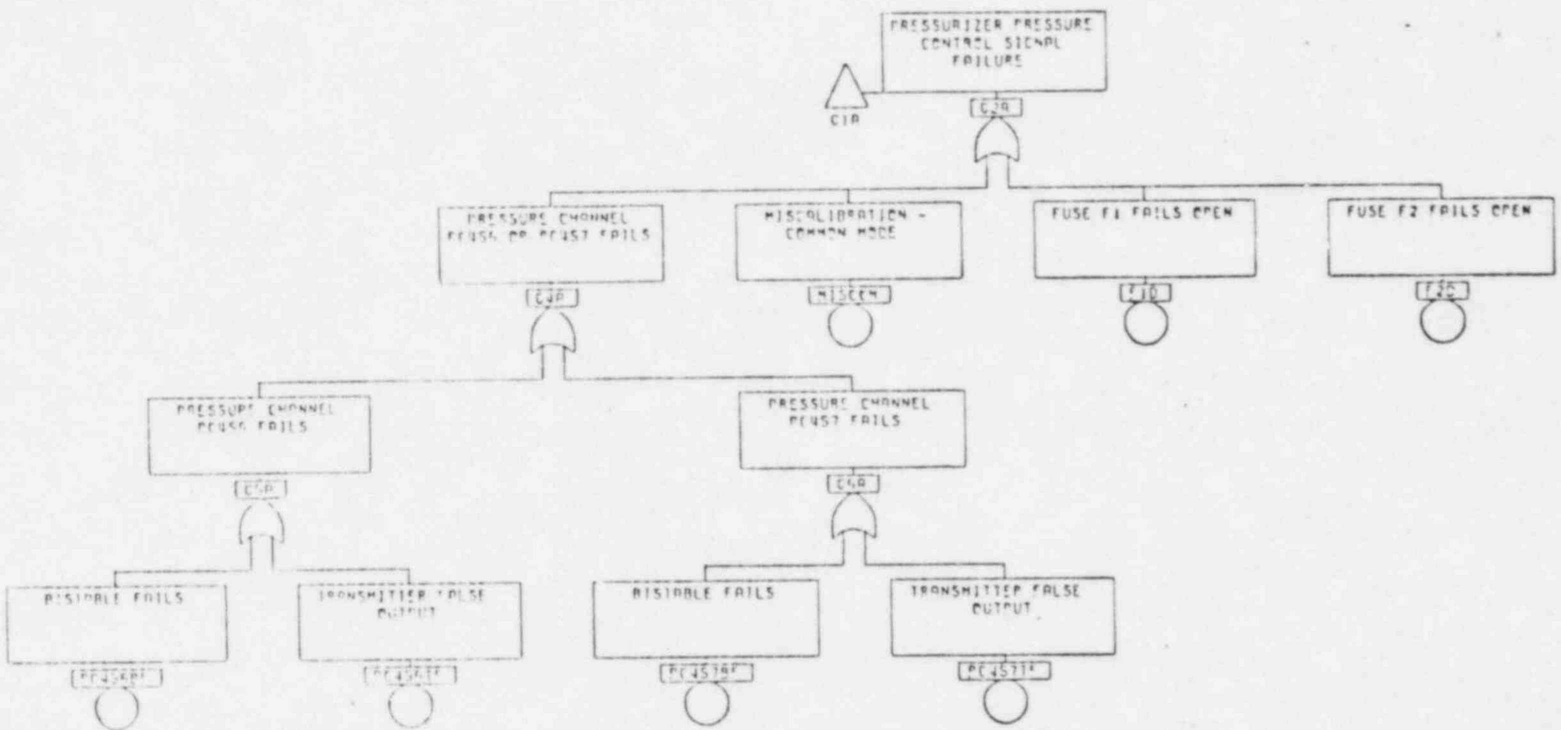


Fig. 3.4 (Continued)

REACTOR AT POWER - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - ZION

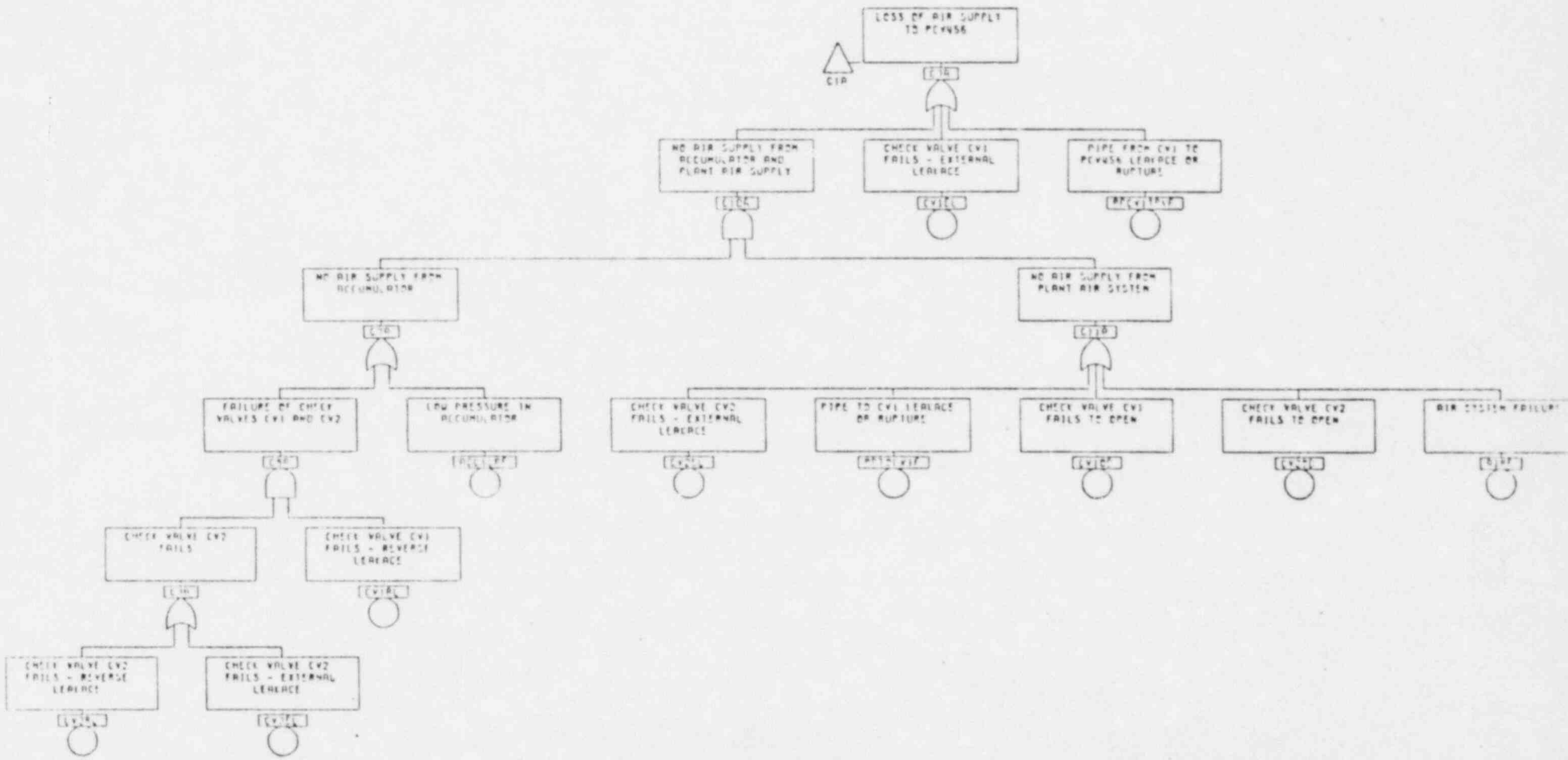


Fig. 3.4 (Continued)

REACTOR AT POWER - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - ZION

The main contributor to the unavailability of both PORVs is a common mode miscalibration of two or more comparators in the pressurizer pressure control system. This failure accounts for more than 90% of the total unavailability, and it is discussed in the next section.

The failure data used for this analysis is shown in Table 3.1.

3.1.2 Miscalibration

The probability of miscalibrating two or more comparators which actuate the signal to open the PORVs is determined using technqiues described in NUREG/CR-1278[10]. These techniques for the "Direct Estimation of Conditional at Probabilities" were developed by the Nuclear Regulatory Commission for evaluation of human reliability in reactor operations. The evaulation is done in detail by considering both small and large miscalibrations. A large change is defined as one that is so extreme so as to be not normally expected, while a small change is one that can be expected to occur occasionally because of variations in equipment or other conditions.

To check the calibration the technician must first set up the test equipment. An error in this initial setup is the initiating event for miscalibration. Figure 3.5 presents the Probability Tree Diagram for this calibration task. It is necessary to point out here that the checking of the calibration of all pressure channel comparators is done by the same technician once per refueling shutdown.

From Figure 3.5, it is seen that the probability of a large miscalibration of two or more comparators (F_2) is equal to 5.0E-6/act, the probability of a small miscalibration of two or more comparators (F_1) is 5.0E-4/act, and the probability of a small or large miscalibration ($F_1 + F_2$) is equal to 5.05E-4/ act.

The following comments are necessary for a better understanding of the Probability Tree Diagram in Fig. 3.5:

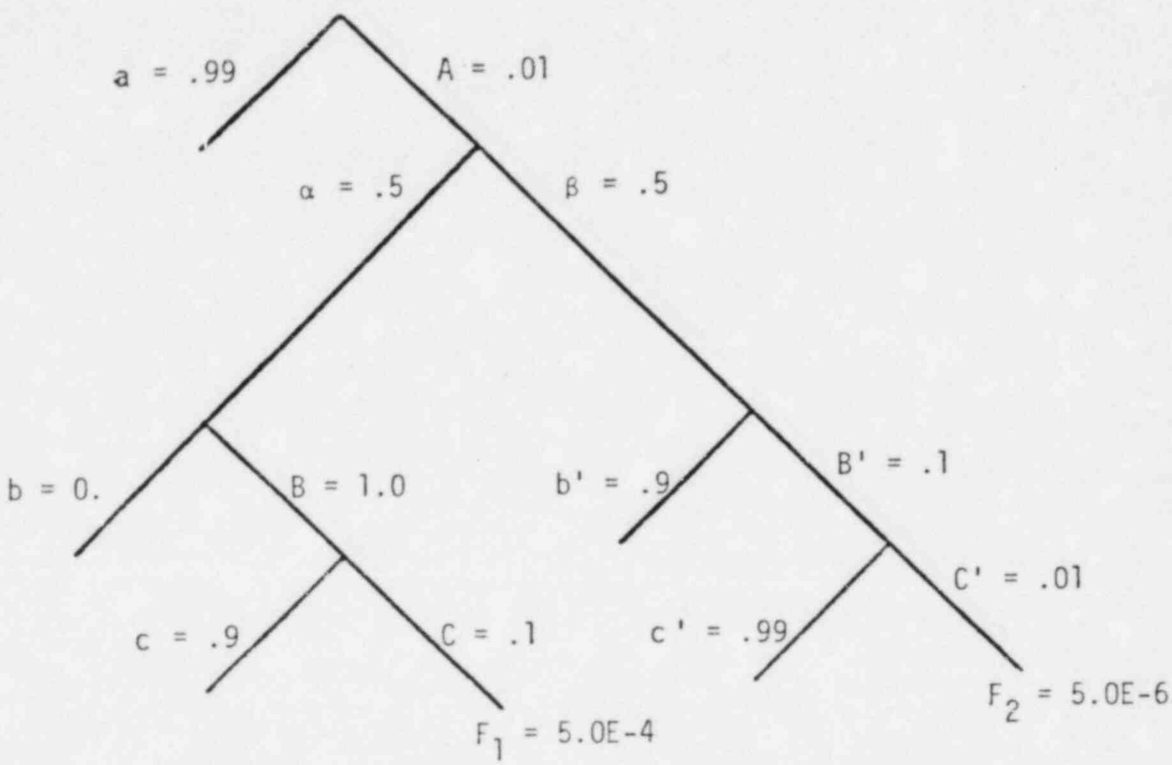
TABLE 3.1

FAILURE DATA FOR FAULT TREE ANALYSIS

COMPONENT	FAILURE MODE	FAILURE RATE (1/hr)	EXPOSURE TIME (hr)	UNAVAILABILITY	REF
Air Operated Valve	Failure to Open on Demand			2.0E-3/d	LER
Air Operated Valve	Leakage	2.0E-7	4380[a]	8.8E-4	LER
Motor Operated Valve	Plugged	6.0E-8	4380[a]	2.6E-4	LER
Check Valve	External Leakage	5.0E-8	4380[a]	2.2E-4	LER
Check Valve	Reverse Leakage	7.0E-7	4380[a]	3.1E-3	LER
Check Valve	Fails to Open on Demand			1.0E-4/d	LER
Solenoid Operated Valve	Fails on Demand			1.0E-3/d	WASH-1400
Pipe ($\phi < 3$in.)	Leakage or Rupture	1.0E-9/hr/ section	4380[a]	1/3E-5/ section	WASH-1400
Accumulator	Low Pressure in Accumulator			1.0E-6	Zion PSS
Bistable (Includes Bistable & Logic Relays)	Fails on Demand			6.7E-6/d	Zion PSS
Transmitter (Includes Sensor & Transmitter)	Fails to Provide Proper Output	1.66E-6	4[b]	6.6E-6	Zion PSS

[a]Assumes test every year

[b]Mean time to detection for these transmitters (Zion PSS)



A - FAILURE TO SET UP TEST EQUIPMENT CORRECTLY

α - Small Miscalibration of Test Equipment
B - For a Small Miscalibration Failure to Detect Miscalibration for First Setpoint
C - For a Small Miscalibration Failure to Detect Miscalibration for Second Setpoint
β - Large Miscalibration of Test Equipment
B'- For a Large Miscalibration Failure to Detect Miscalibration for First Setpoint
C'- For a Large Miscalibration Failure to Detect Miscalibration for Second Setpoint

Fig. 3.5

PROBABILITY TREE DIAGRAM FOR CALIBRATION TASK

1. The c mplete notation for the conditional probabilities ever cs is not employed but should be understood, e.g., is w itten instead of $\alpha|A$, i.e. probability of α "given A."
2. As suggested by NUREG/CR-1278, it is estimated that a miscalibration would be equally likely to result in a large change or in a small change. This assumption is conservative since the total probability (i.e. the summation of the probabilities of small and large miscalibration) is used in this analysis. A more realistic analysis would include only the large miscalibration, because the miscalibration error will cause a PORV failure (prior to an S/RV challenge at 2485 psig) only if the setpoint is calibrated to a value greater than 2485 psig. The differences between calibrations at 2485 and 2335 should certainly be considered a large error.
3. It is conservatively assumed that if the technician does not detect the instrument error by the time he calibrates the second setpoint, 100% of the time he will continue the erroneous calibration through the third and subsequent setpoints.

3.1.3 Reactor in Cold Shutdown Mode

There are several differences in PORV operations during at Power and Cold Shutdown modes of operations. Since these differences affect the PORV failure probabilities, they are discussed below:

i) Pressurizer Pressure Control Signal: In the cold shutdown mode, as can be seen in Fig. 3.3, the PORV control mode selector switch is in the position "AUTO LOW TEMP" and the corresponding actuation circuit has a signal from only one pressure comparator. This modification to the tree given in Fig. 3.4 is presented in Fig. 3.6.

ii) Miscalibration: With the reactor in Cold Shutdown mode the PORVs are set to open at 500 psig. Therefore, only a very large miscalibration will cause the actuation of the S/RVs, whose setpoint is at 2485 psig, before the actuation of the PORVs.

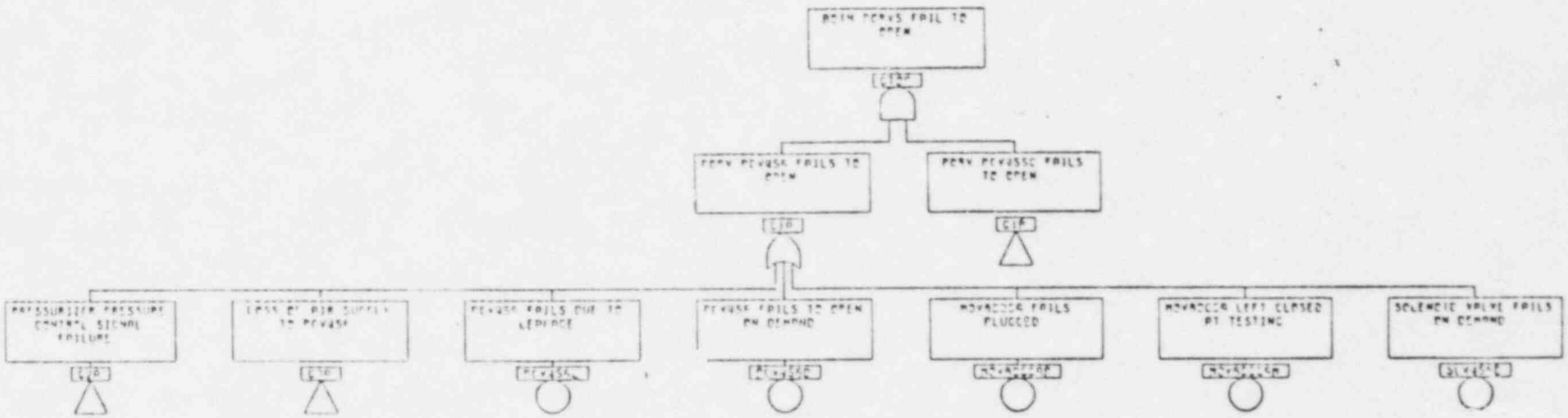


Fig. 3.6

REACTOR AT COLD SHUTDOWN - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - ZION

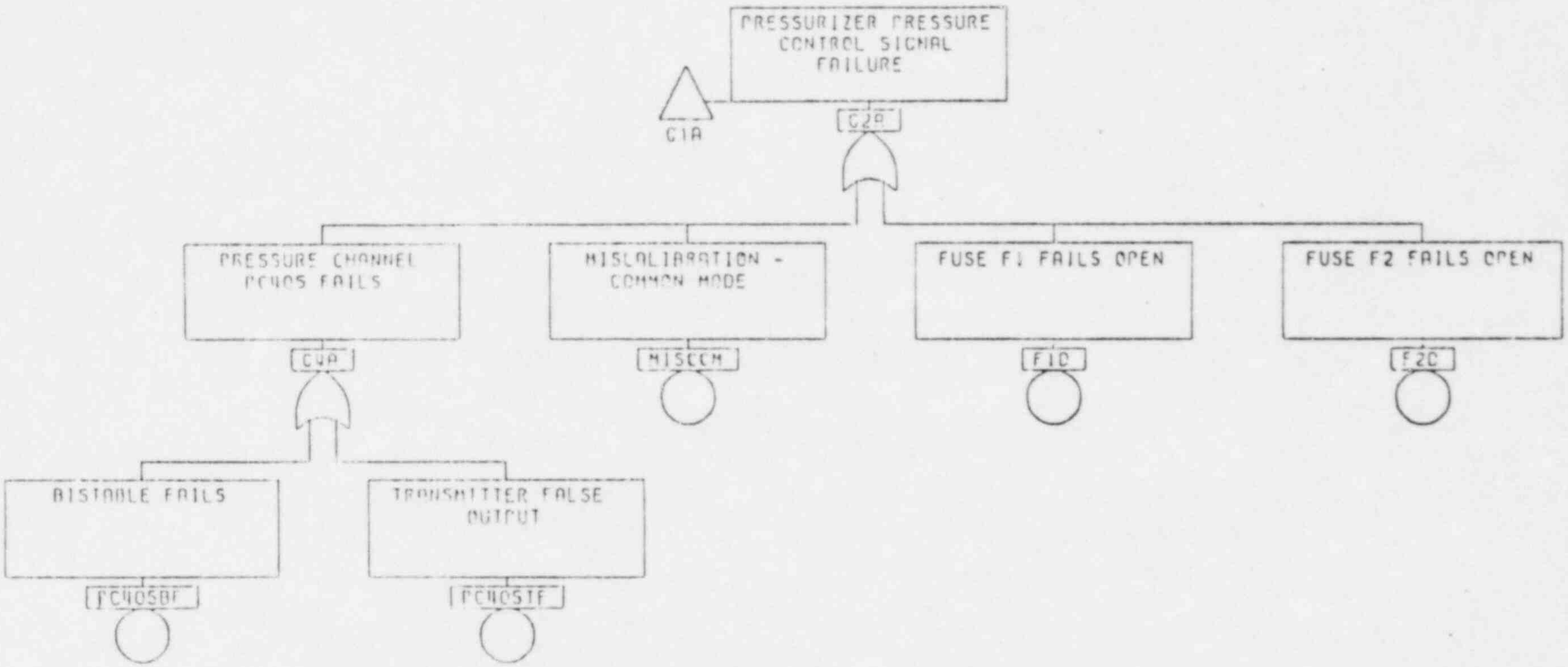


Fig. 3.6 (Continued)

REACTOR AT COLD SHUTDOWN - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - ZION

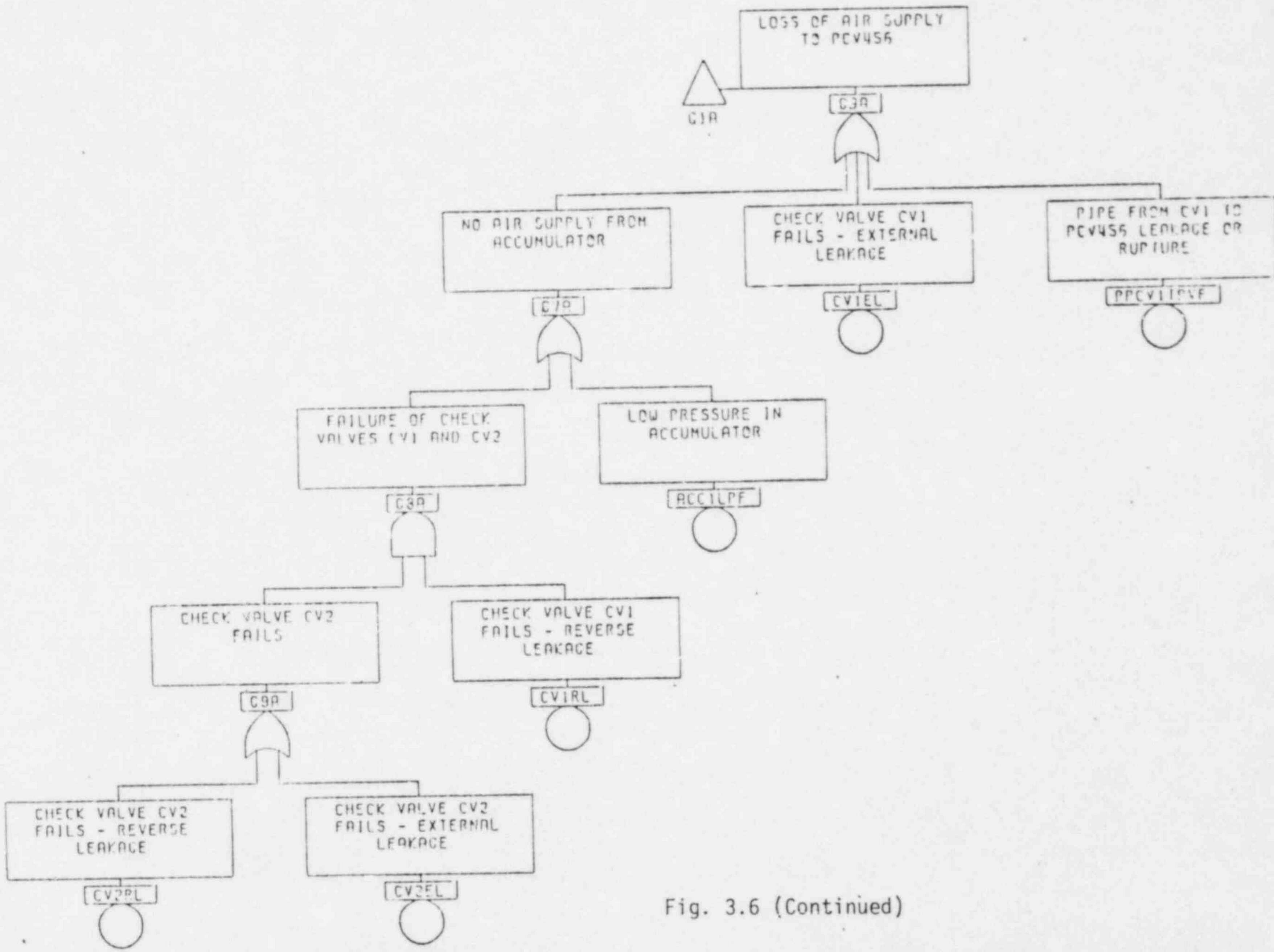


Fig. 3.6 (Continued)

REACTOR AT COLD SHUTDOWN - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - ZION

Based on this, and using the probability tree diagram in Fig. 3.5, the miscalibration error was taken to be 5.0E-6/act, corresponding to a large miscalibration only.

iii) Air Supply From Instrument Air System: As discussed in Section 2.3, the failure of this system is one of the initiators for a Cold Overpressurzation in the Zion plant. Therefore, it is assumed that the instrument air system fails, so that the loss of air supply for the PORVs is represented in the fault tree (Fig. 3.6) simply by failure of the air accumulator or failure of check valves and piping.

The results of this analysis show an overall median unavailability equal to 4.3E-5/demand for the cold shutdown mode. The main contributors are combinations of single failures in both valves.

3.2 Modifications to PORV Fault Trees for Byron-Braidwood Plants

The fault trees for failure of both PORVs to open when the reactor is at Power and on Cold Shutdown modes of operation for Byron-Braidwood plants are shown in Figures 3.7 and 3.8, respectively. The only differences between the Byron/Braidwood and Zion plants are:

i) When the reactor is operating at power, the PORV control mode selector switch is required to be on "AUTO" (see Fig. 3.9). However, there is a probability that the operator leaves that switch on "CLOSE" and this failure, by itself will result in an unavailability of that PORV. By comparison, in the Zion plant the PORV control mode selector switch does not have position "CLOSE" as seen in Fig. 3.3.

ii) When the reactor is in Cold Shutdown Mode of operation, the accident that might lead to liquid challenge to the S/RVs is a Cold Overpressurization Event. As discussed in Section 2.3, the initiating events for Zion includes a failure of the air supply system, and thus this event is not present in the fault tree for the opening of the PORVs (Fig. 3.6). However, for the B/B plants a failure of the air supply system is not

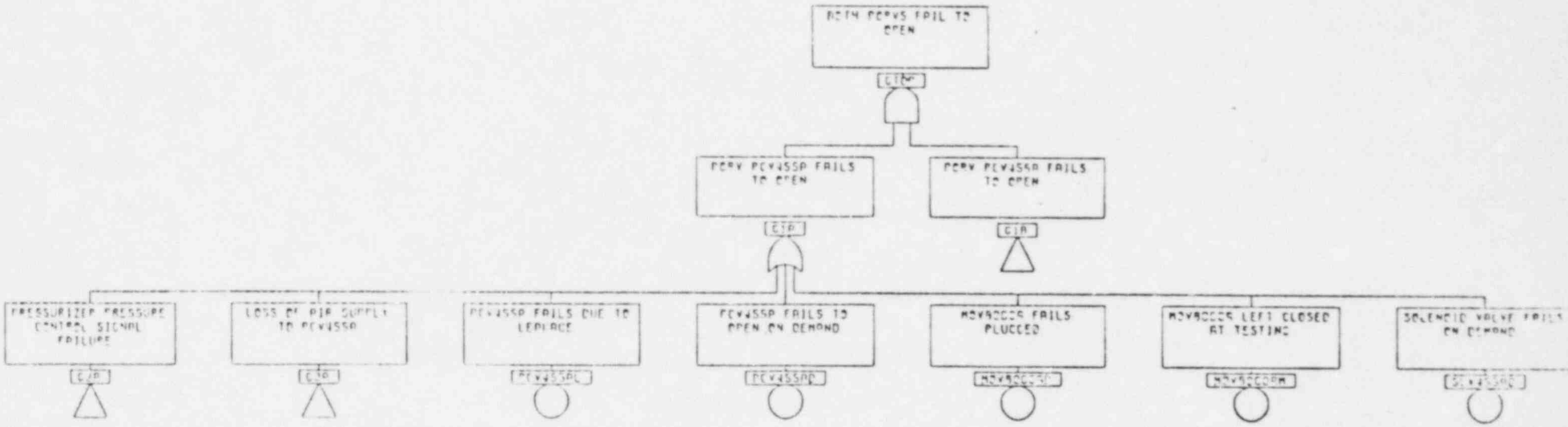


Fig. 3.7

REACTOR AT POWER - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - BYRON/BRAIDWOOD

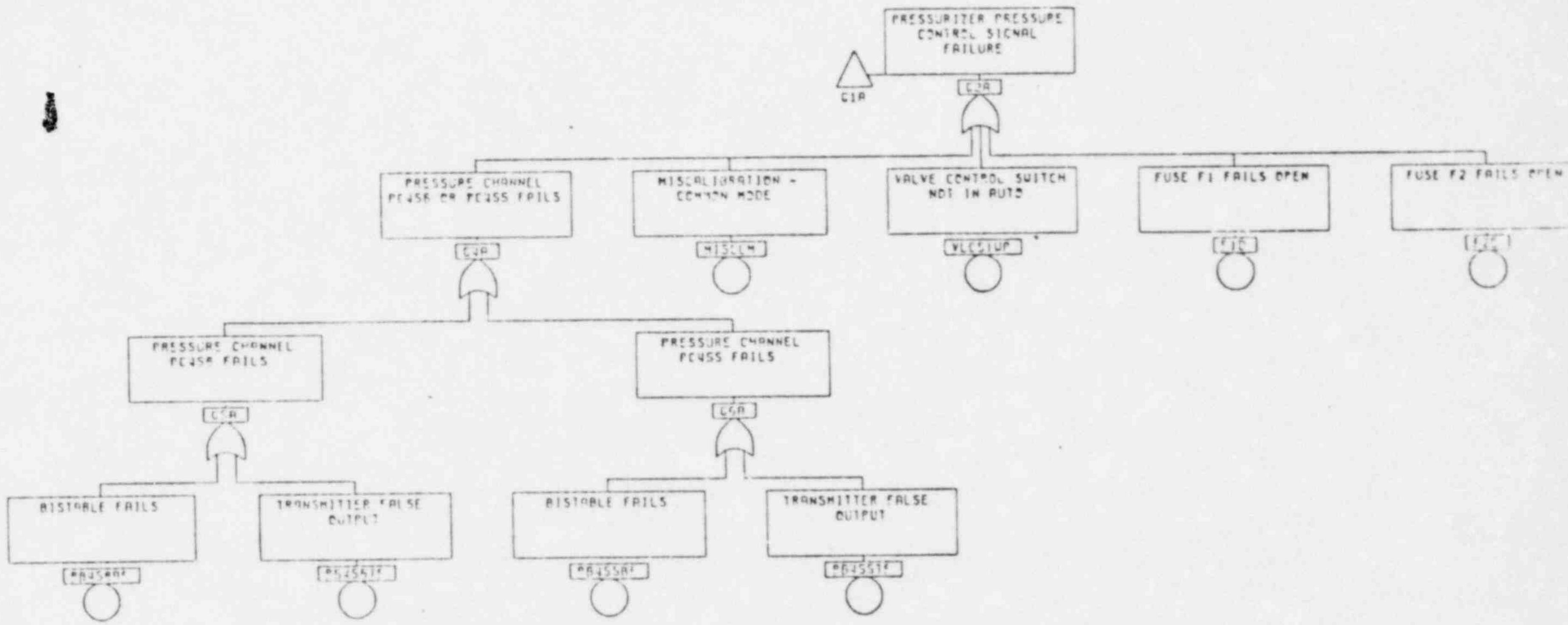


Fig. 3.7 (Continued)

REACTOR AT POWER - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - BYRON/BRAIDWOOD

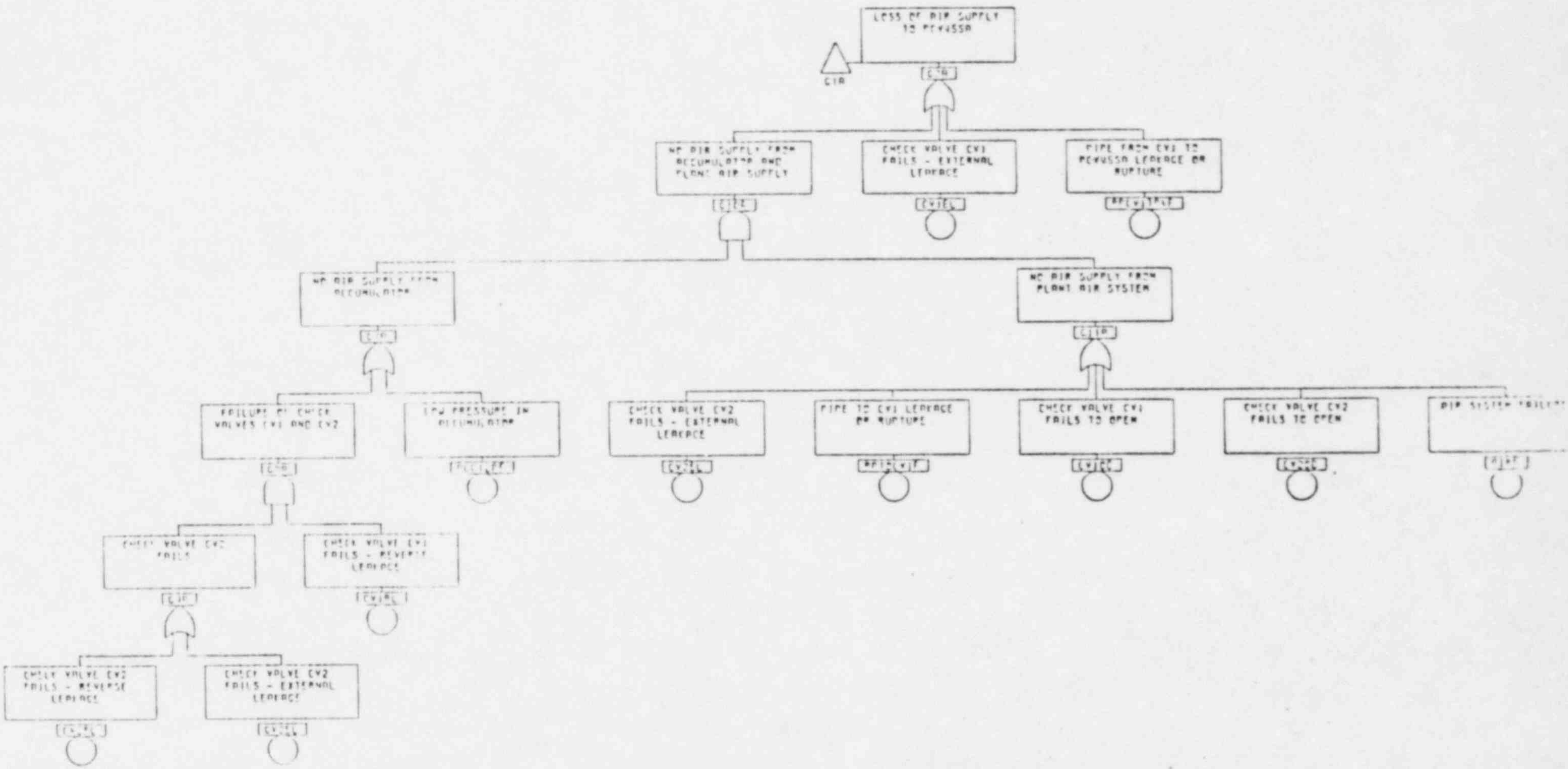


Fig. 3.7 (Continued)

REACTOR AT POWER - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - BYRON/BRAIDWOOD

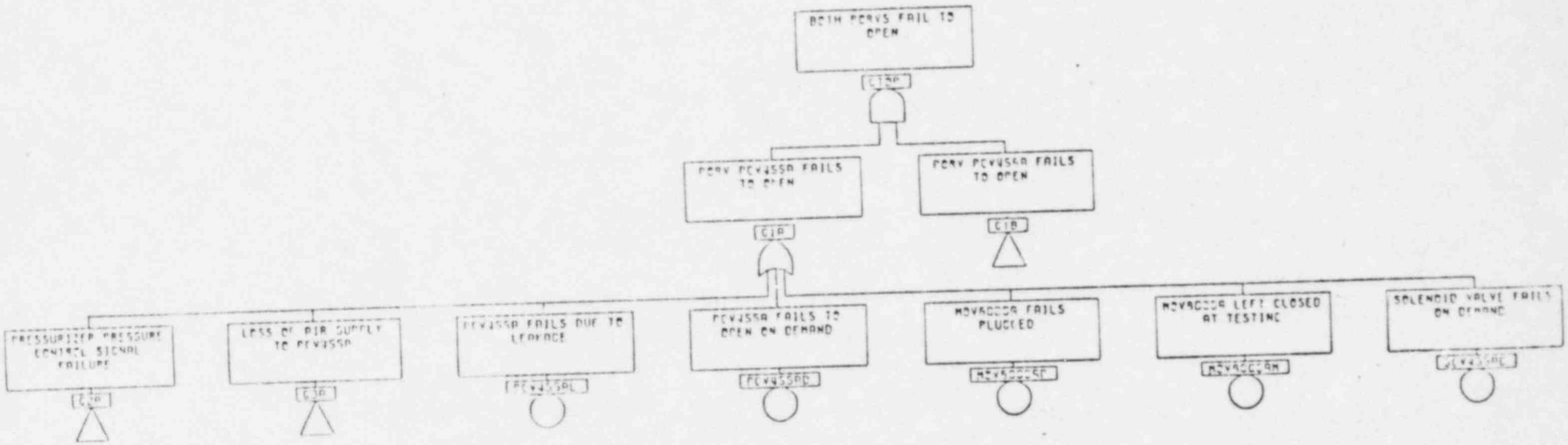


Fig. 3.8

REACTOR AT COLD SHUTDOWN - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - BYRON/BRAIDWOOD

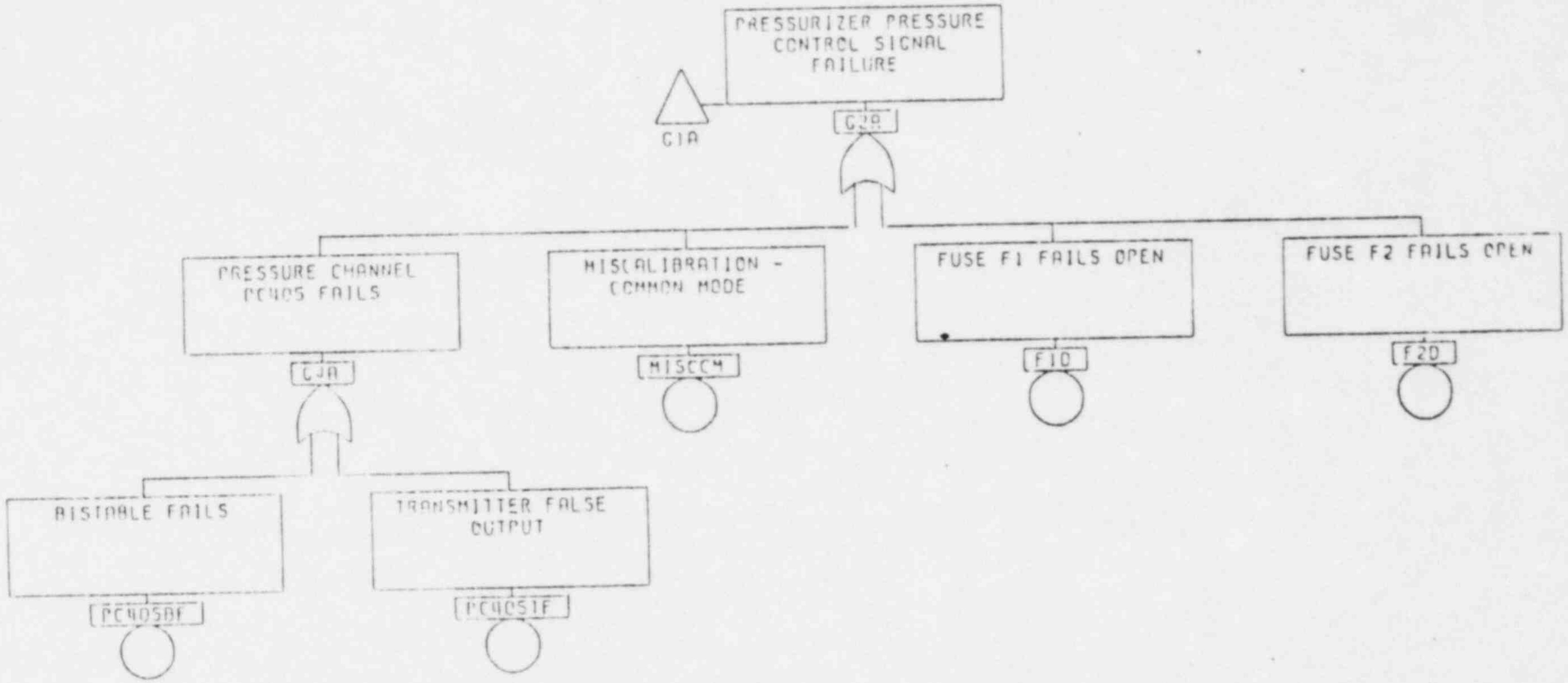


Fig. 3.8 (Continued)

REACTOR AT COLD SHUTDOWN - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - BYRON/BRAIDWOOD

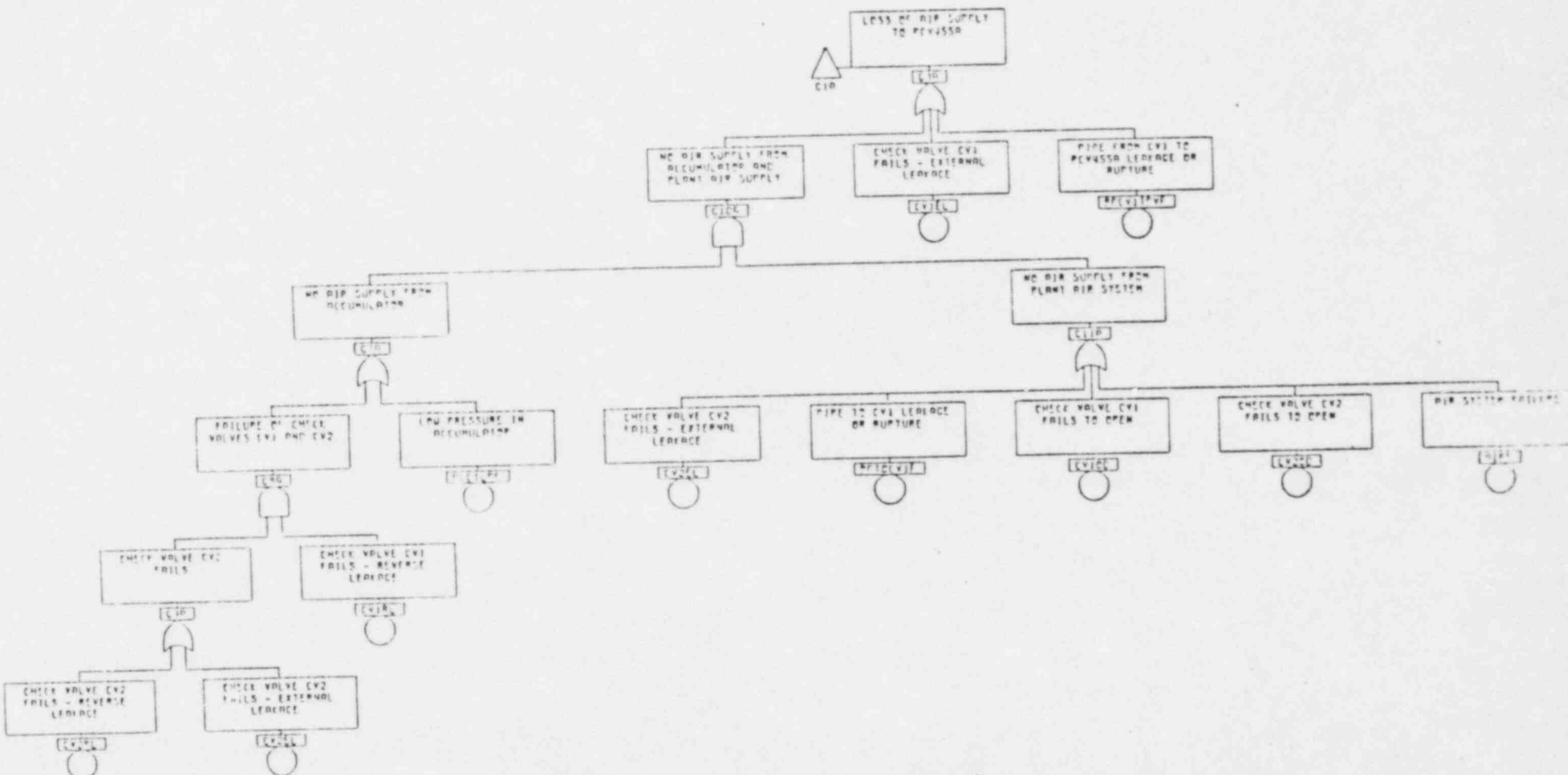


Fig. 3.8 (Continued)

REACTOR AT COLD SHUTDOWN - FAULT TREE FOR UNAVAILABILITY OF BOTH PORVs - BYRON/BRAIDWOOD

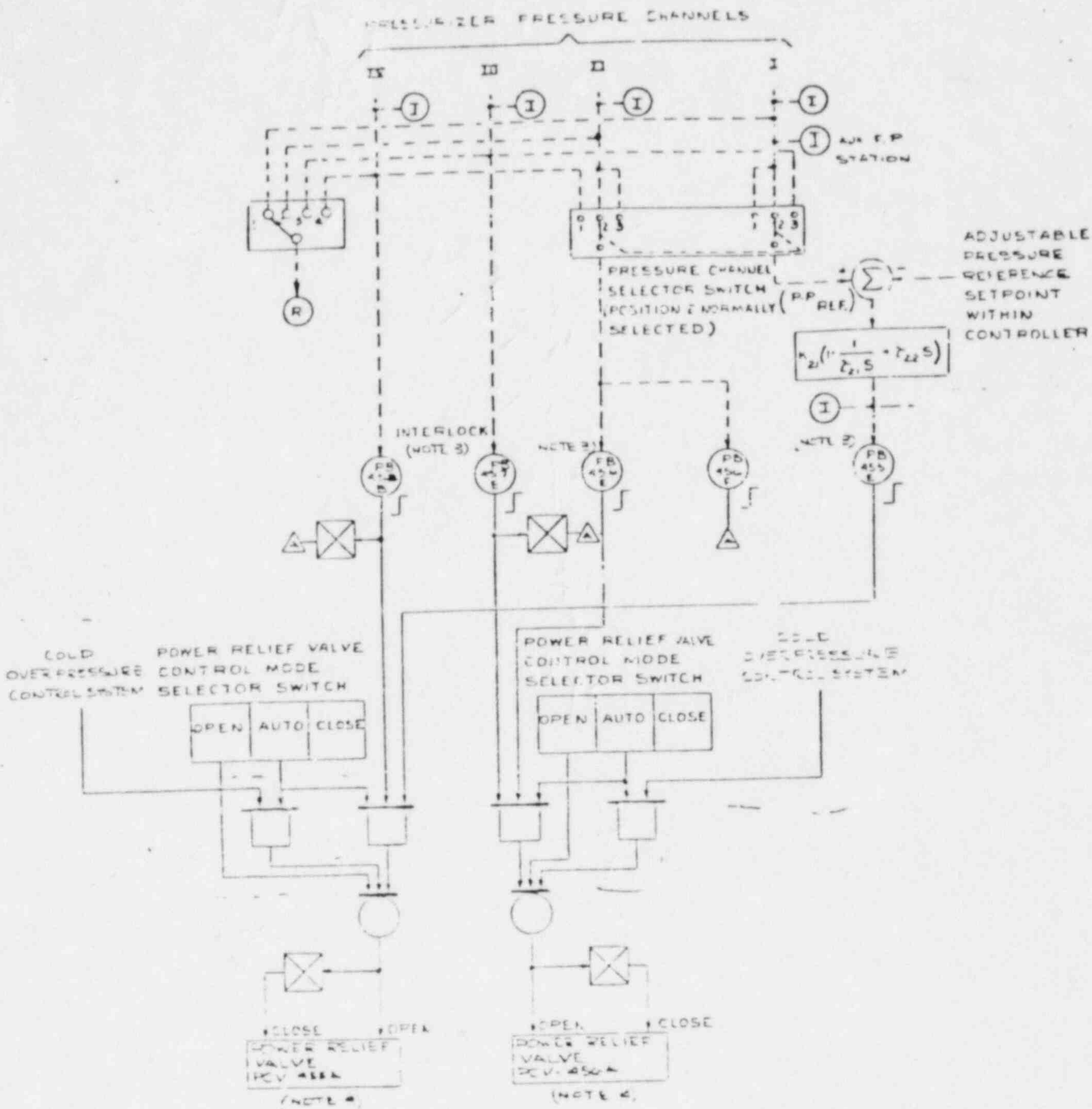


Fig. 3.9

BYRON/BRAIDWOOD - PORVs LOGIC DIAGRAM

an initiating event for Cold overpressurization; therefore, the impact of this failure is accounted for in the fault tree for both PORVs to open (Fig. 3.8).

3.3 Failure to Recover from Spurious Safety Injection

The failure to recover from a spurious safety injection appears in the event tree for Extended High Pressure Injection at Power (see Sec. 2.2 and Fig. 2.2). As discussed in Section 2.2 this is an event that is neither extremely rare nor difficult to control, and there are procedures[14] for that recovery. Furthermore, since the operator has at least 20 minutes for recovery (as discussed in Section 2.2) this event is considered in this analysis as only a moderately high stress level event.

According to NUREG/CR-1278 the basic human error probability for this event is 0.003 and a multiplier of 2 is recommended for moderately high stress level. In this analysis a value of 0.02 is used for the basic human error probability, and to be conservative a multiplier of 6 (instead of 2) was used.

Five people would be in the control room[16]. Three of the five are reactor operators (RO) and at least one of them has a senior reactor operator's (SRO) license. The remaining two are the shift engineers (SE, who has an SRO license) and shift technical advisor (STA, who also has an SRO license). All of them except for one reactor operator would be totally involved with the affected unit. The remaining operator would be running the other unit. To compute the human error probabilities, one uses the formulas recommended by NUREG/CR-1278 with the following dependencies among operators: high dependence (HD) between the two reactor operators; moderate dependence (MD) between the SE and the first two; low dependence between the STA and the rest. The error frequency of the four-person team for this task (Recovery from Spurious Safety Injection) would be:

$$2.0E\text{-}2 \times \frac{1 + 2.0E\text{-}2}{2} \times \frac{1 + 6 \times 2.0E\text{-}2}{7} \times \frac{1 + 19 \times 2.0E\text{-}2}{20} = 1.1E\text{-}4$$

This value, 1.1E-4, is used in the event tree for Extended High Pressure Injection at Power as shown in Fig. 2.2.

4. SUMMARY OF RESULTS

4.1 Zion Plant

The median estimate for the frequency of liquid discharge from safety/relief valve in the Zion plant is 1.3×10^{-7} events/reactor year. Table 4.1 breaks down this result according to the three initiating events. The frequency is dominated by contributions from the cold overpressurization tree event. S/RV liquid discharge is thus an extremely unlikely event.

4.2 Byron and Braidwood Plants

The median estimate for the frequency of liquid discharge from a safety/relief valve is the Byron-Braidwood plant is 6.9×10^{-8} events/reactor-year. Table 4.2 breaks down this result according to the three initiating events. The frequency is dominated by contributions from the cold overpressurization event. S/RV liquid discharge is thus an extremely unlikely event.

4.3 General Conclusions

The discharge of liquid from safety relief valves in the Zion, Bryon, and Braidwood plants has been shown to be a possible but extremely unlikely event. The estimated frequencies are based upon conservative data and assumptions and are sufficiently low that even order-of-magnitude errors would not effect the qualitative conclusions.

Further, the scenarios of S/RV liquid discharge have been predicted to occur less frequently than a small break LOCA, while the consequences (e.g. hypothetical overstressing of S/RV discharge piping) are certainly much less severe. From the point of view of safety risks, S/RV liquid discharge appears to be an insignificant concern compared with LOCA events or FSAR transient events. While it is of course advantageous for S/RVs and associated piping to be verified operable for a wide range of inlet conditions, it is equally important that engineering resources not be diverted from more realistic pursuits such as improved S/RV discharge under expected saturated steam inlet conditions.

TABLE 4.1

SUMMARY OF RESULTS FOR ZION PLANT FREQUENCY OF LIQUID DISCHARGE FROM SAFETY/RELIEF VALVES

Initiating Event	Calculated Frequency Of Occurrence (Events/Reactor Year)	Type of Discharge
Extended High Pressure Injection	3.6×10^{-8}	Steam followed by saturated or slightly subcooled liquid; possible valve cycling
Cold Overpressurization	9.6×10^{-8}	Far subcooled liquid
Main Feedwater Pipe Rupture	5.4×10^{-10}	Steam followed by saturated liquid
TOTAL	1.3×10^{-7}	

TABLE 4.2

SUMMARY OF RESULTS FOR BYRON/BRAIDWOOD PLANTS
FREQUENCY OF LIQUID DISCHARGE FROM SAFETY/RELIEF VALVES

Initiating Event	Calculated Frequency Of Occurrence (Events/Reactor Year)	Type of Discharge
Extended High Pressure Injection	9.9×10^{-9}	Steam followed by saturated or slightly subcooled liquid; possible valve cycling
Cold Overpressurization	5.8×10^{-8}	Far subcooled liquid
Main Feedwter Pipe Rupture	5.6×10^{-10}	Steam followed by saturated liquid
TOTAL	6.9×10^{-8}	

REFERENCES

1. NUREG-0737, "Clarification of TMI Action Plan Requirements," Item II.d.1, November 1980.
2. W.R. Hocking, et. al., "EPRI/CE PWR Safety Valve Test Program," EPRI Research Project V102-2, Final Report, July 1982.
3. D. F. Streinz, "EPRI/CE PWR Safety and Relief Valve Test Program, Upstream Pressure Oscillations," Combustion Engineering Letter PE-81-415, December 18, 1981.
4. B. R. Strong, et.al., "Steam Hammer Design Loads for Safety/Relief Valve Discharge Piping," from Safety Relief Valves, ASME, 1979.
5. B.S. Singer and R. S. May, "Analysis of Safety/Reliev Valve Chatter and Transient Problems," SAI report to Commonwealth Edison, June 1982.
6. D. Harris, et.al., "Probabilistic Evaluation of High Pressure Liquid Challenge of Safety/Relief Valve Piping," SAI report SAI-245-81-PA submitted to BWR Owner's group and General Electric Company, April 1981.
7. A. Meliksetian and A.M. Sulencar, "Valve Inlet Fluid Conditions for Pressurized Safety and Relief Valves in Westinghouse-Designed Plants", EPRI research project V102-19, January 29, 1982.
8. Zion Probabilistic Safety Study. Commonwealth Edison Company.
9. Reactor Safety Study: An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants, "NUREG-75/014, October 1975.
10. A. D. Swain and H. E. Guttman, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," NUREG/CR-1278, October 1980.
11. W. H. Hubble and C. F. Miller, "Data Summaries of Licensee Event Reports of Valves at U.S. Commercial Nuclear Power Plants," NUREG/CR-1363, June 1980.
12. Zion Station, Units 1 and 2, Final Safety Analysis Report. Commonwealth Edison Company, December 1970.
13. Byron/Braidwood Final Safety Analyses Report. Commonwealth Edison Company.
14. Zion Emergency and Operations Procedures.
    EOP-0: Safety Injection/Accident Diagnostics
    GOP-1: Plant Startup
    GOP-2: Plant Shutdown

15. E. W. Hagen, "Compressed Air and Backup in Nuclear Power Plants," Report by ORNL on a contract for NRC, to be published.
16. Private Communications with G. Klopp and F. Highland (Commonwealth Edison Company).