TECHNICAL EVALUATION REPORT

OPERATING REACTOR PORV REPORTS (F-37)

TMI ACTION PLAN REQUIREMENTS

COMBUSTION ENGINEERING OWNERS GROUP (CEN-145)

NRC DOCKET NO. Various

FRC PROJECT C5506

FRC ASSIGNMENT 7

NRC CONTRACT NO. NRC-03-81-130

FRC TASK 409

Prepared by

Franklin Research Center
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Prepared for

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

Mr. G. J. Overbeck, Mr. S. M. Jenkins, and Mr. T. J. DelGaizo contributed to the technical preparation of this report through a subcontract with WESTEC Services, Inc.

1. INTRODUCTION

1.1 PURPOSE OF REVIEW

This technical evaluation report (TER) documents an independent review of the report of "PORV Failure Reduction Methods" prepared for the Combustion Engineering (CE) Owners Group in response to NUREG-0737 [1], "Clarification of TMI Action Plan Requirements," Item II.K.3.2, "Report on Overall Safety Effect of Power Operated Relief Valve Isolation System," as it pertains to the CE-designed units. This evaluation was performed with the following objectives:

- o to ensure that the CE response is complete and properly documents the information required by NUREG-0737, Item II.K.3.2
- o to ensure that the CE estimated probabilities satisfy the review criteria.

1.2 GENERIC BACKGROUND

In NUREG-0635 [2], "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants," the Nuclear Regulatory Commission's (NRC) Bulletins and Orders Task Force recommended the following:

> "Licensees should provide a system which closes the block valve automatically whenever the reactor coolant system pressure decays to a preset value subsequent to a PORV opening. This system should include an override feature so that pressure relief can be accomplished at lower pressures, as necessary.
>
> Combustion Engineering should prepare a report documenting the actions which have been taken to decrease the probability of a small-break LOCA caused by a stuck-open PORV. The report should include an evaluation describing how the actions taken constitute a significant improvement in reactor safety.
>
> Any future failure of a PORV or safety valve to close should be reported to the NRC promptly. All future challenges of the PORVs and safety valves should be documented in the annual report."

These recommendations were later included in NUREG-0660 [3], "NRC Action Plan Developed as a Result of the TMI-2 Accident." The first recommendation

was incorporated into NUREG-0660 as Item II.K.3.1, "Installation and Testing of Automatic Power-Operated Relief Valve Isolation System," and the second two recommendations were modified and combined to form Item II.K.3.2, "Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System." In Reference 1, the staff delayed implementation of Item II.K.3.1, until the pending PORV reliability analysis of Item II.K.3.2 confirmed the necessity of an automatic isolation system. Specifically, NUREG-0737, Item II.K.3.2 stated:

"(1) The licensee should submit a report for staff review documenting the various actions taken to decrease the probability of a small-break loss-of-coolant accident (LOCA) caused by a stuck-open power-operated relief valve (PORV) and show how those actions constitute sufficient improvements in reactor safety.

(2) Safety-valve failure rates based on past history of the operating plant designed by the specific nuclear steam supply system (NSSS) vendor should be included in the report submitted in response to (1) above."

In addition, Reference 1 further clarified that:

"Modifications to reduce the likelihood of a stuck-open PORV will be considered sufficient improvements in reactor safety if they reduce the probability of a small-break LOCA caused by a stuck-open PORV such that it is not a significant contributor to the probability of a small-break LOCA due to all causes. (According to WASH-1400, the median probability of a small-break LOCA S_2 with a break diameter between 0.5 in. and 2.0 in. is 10^{-3} per reactor-year with a variation ranging from 10^{-2} to 10^{-4} per reactor-year.)

The above-specified report should also include an analysis of safety-valve failures based on the operating experience of the pressurized-water-reactor (PWR) vendor designs. The licensee has the option of preparing and submitting either a plant-specific or a generic report. If a generic report is submitted, each licensee should document the applicability of the generic report to his own plant.

Based on the above guidance and clarification, each licensee should perform an analysis of the probability of a small-break LOCA caused by a stuck-open PORV or safety valve. This analysis should consider modifications which have been made since the TMI-2 accident to improve the probability. This analysis shall evaluate the effect of an automatic PORV isolation system specified in Task Action Plan Item II.K.3.1. In evaluating the automatic PORV isolation system, the potential of causing a subsequent stuck-open safety valve and the overall effect on safety (e.g., effect on other accidents) should be examined.

Actual operational data may be used in this analysis where appropriate. The bases for any assumptions used should be clearly stated and justified.

The results of the probability analysis should then be used to determine whether the modifications already implemented have reduced the probability of a small-break LOCA due to a stuck-open PORV or safety valve a sufficient amount to satisfy the criterion stated above, or whether the automatic PORV isolation system specified in Task Action Item II.K.3.1 is necessary.

In addition to the analysis described above, the licensee should compile operational data regarding pressurizer safety valves for PWR vendor designs. These data should then be used to determine safety-valve failure rates.

The analysis should be documented in a report. If this requirement is implemented on a generic basis, each licensee should review the appropriate generic report and document its applicability to his own plant(s). The report and the documentation of applicability (where appropriate) should be submitted for NRC staff review by the specified date."

1.3 PLANT-SPECIFIC BACKGROUND

In letters to the NRC dated in early 1981 [4], owners of CE-designed units endorsed a report prepared for the Combustion Engineering Owners Group, CEN-145 [5], "PORV Failure Reduction Methods" as the response to NUREG-0737, Items II.K.3.1 and II.K.3.2.

An independent preliminary review of the information presented in Reference 5 resulted in a request for additional information (RAI) being sent to one CE licensee from the NRC on January 20, 1982 [6]. The licensee responded to the staff RAI in letters to the NRC dated April 26, 1982 [7] and June 7, 1982 [8]. This TER is an evaluation of the information presented in References 5, 7, and 8 along with other information pertinent to the topic of a small-break LOCA from a stuck-open PORV or safety valve.

2. REVIEW CRITERIA

The Licensee's response to NUREG-0737, Item II.K.3.2, was evaluated against the acceptance criteria provided by the NRC in a letter dated July 21, 1981 [9], which outlined Tentative Work Assignment F. Specifically, the Licensee's response to NUREG-0737, Item II.K.3.2 was supposed to contain the following information:

"1. The report shall list the actions taken by the licensee to decrease the probability of a small-break LOCA caused by a stuck-open PORV.

2. The report shall include an analysis of safety-valve failure rate based on the past history of the operating plants designed by the licensee's NSSS vendor. This may be a plant-specific report or a generic report showing the applicability to the specific plant.

3. The report shall have an analysis of the probability of a small-break LOCA caused by a stuck-open PORV or a stuck-open safety valve. This analysis shall evaluate the effect of an automatic PORV isolation system. In evaluating this system, the licensee shall evaluate the potential of causing a subsequent stuck-open safety valve and the overall effect on safety.

4. Actual operational data may be used. The basis for any assumption should be clearly stated and justified.

5. The automatic PORV isolation system is not required if the licensee's actions constitute sufficient improvements to reactor safety in reducing the probability of a small-break LOCA due to a stuck-open PORV or a stuck-open safety valve such that it is less than 10^{-3}/reactor-year, the median probability of a small-break LOCA S_2 with a break size between 0.5 in. and 2.0 in. due to all causes."

3. TECHNICAL EVALUATION

The following tasks were to be performed under contract to the NRC [9]:

1. Review the licensee's report required by NUREG-0737, Item II.K.3.2 to determine (1) if a licensee proposes to provide an automatic PORV isolation system and (2) if all the data required in the report have been provided by the licensee. Review the licensee's analysis for completeness in identifying all transients that lead to PORV challenges. The analysis should include failure in the integrated control system (ICS), applicable to Babcock & Wilcox (B&W) plants only, operator error, reliability of PORV block valve, and other initiating events. Review the licensee's analysis of safety valve challenge rate and failure rate to reseat. The analysis should include consideration of the PORV being blocked as a result of leakage, operator action closing the PORV block valve and actuating high pressure injection (HPI) during the recovery from depressurization events.

2. Evaluate the licensee's reports required by NUREG-0737, Item II.K.3.2 against the review criteria in Section 2. If generic reports are submitted, the applicability of the generic reports to the specific plants, should be evaluated. Priority should be given to determining if any of the PWR licensees is required to propose an automatic PORV isolation system. If necessary, a letter was to be provided requesting these PWR licensees to propose such systems and the plant-specific technical basis for this request.

3. Prepare a TER for each plant. The TER will discuss the evaluation of the licensee's reports and, if needed, the proposed automatic PORV isolation system. The TER shall include a discussion of the assumptions made by the licensee in his reports.

This report constitutes a TER in satisfaction of Task 3. Section 3.1 addresses the completeness of the Licensee's report, while Section 3.2 provides an evaluation of the Licensee's analysis. In Section 3.3, additional items relevant to the subject of a small-break LOCA from a stuck-open PORV or safety valve, but not specifically addressed by the Licensee, are considered.

3.1 REVIEW OF THE CE REPORT FOR COMPLETENESS

The review and evaluation of the information presented in Reference 5, as supplemented by the additional information presented in References 7 and 8, forms the basis of this report. Reference 5 was prepared for the Combustion

Engineering Owners Group by Combustion Engineering, Inc. (CE) for the purpose of generically addressing the requirements of NUREG-0737, Item II.K.3.2. (See Section 1.2 of this report for more detailed information pertaining to the requirements of NUREG-0737, Item II.K.3.2.) In Reference 5, CE describes the various modifications that have been incorporated into CE-designed plants since the Three Mile Island (TMI) accident and presents a probabilistic analysis of the likelihood of a small-break LOCA from a stuck-open PORV. Included in the probabilistic analysis is the evaluation of a pre-TMI, CE-designed baseline plant, the effect on the plant of the post-TMI modifications as implemented, and the effect of a conceptually designed automatic PORV isolation system as identified in NUREG-0737, Item II.K.3.1. In Reference 8, the Licensee presents a probabilistic analysis of a small-break LOCA from a stuck-open safety valve.

3.1.1 CE's Technical Approach

Several methodologies presently exist for determining the frequency of a small-break LOCA caused by a stuck-open PORV or safety valve. Inherent in all of these methodologies is the requirement to determine the frequency and number of PORV or safety valve challenges (demands to open) and the probability of the PORV or safety valve failing to close once it has opened. The probabilistic analysis tool chosen in Reference 5 for determining the expected frequency of a small-break LOCA from a stuck-open PORV is the fault tree. The probabilistic analysis method chosen in Reference 8 for determining the expected frequency of a small-break LOCA from a stuck-open safety valve is the event tree. As demonstrated in WASH-1400 [10], "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," the use of event trees and fault trees as probablistic analysis tools is an acceptable technical approach for analyzing reactor incidents such as a stuck-open PORV or safety valve.

Since well-documented probablistic analysis techniques have been used, a detailed discussion of the technical approach is not required. The following subsections, which describe the analyses as presented in References 5, 7, and 8, are provided for clarity.

3.1.2 CE's Fault Tree Transient Initiator Event Frequencies

A survey was conducted in early 1980 to compile the operating experience and PORV initiating transient history of CE-designed operating plants. The survey results indicated that, during 29 reactor-years of operation, only three PORV transient-related openings were reported. In addition, the survey indicated that 16 high pressurizer pressure reactor trips had occurred. Since the PORV opening setpoint pressure of CE-designed plants is the same as the high pressurizer pressure reactor trip setpoint, it can be concluded that an additional 16 PORV transient-related opening events had occurred. Based on these historical data, the PORV opening transient-related event frequency for CE-designed plants was 0.66 per reactor-year.

In addition, CE assigned a value of 2.8×10^{-3} per reactor-year for the expected frequency of a spurious PORV opening, taken from "Post TMI Evaluation Task 3 Follow-up Report, Pressurizer Systems and Emergency Power Supplies" [11].

3.1.3 CE's Fault Tree Branches

In Reference 5, CE developed a fault tree that was used with the transient initiator frequencies identified in Section 3.1.2 of this report to evaluate the frequency of a small-break LOCA from a stuck-open PORV. The fault tree is based on the premise that each initiator event results in a single PORV challenge event (i.e., the PORV actuation setpoint is exceeded only once per initiator event). A CE licensee justifies this assumption in Reference 7 as follows:

> "Only one PORV opening is expected during a pressurization event in which the PORV's are actuated. As described in Section 3.9 of CEN-145, the coincidence of the PORV opening setpoint and the high pressure reactor trip at approximately 2400 psia on the Calvert Cliffs Nuclear Power Plant insures that the reactor is shutting down as the PORV's are opening, if not before. By the time the PORV's blow down to the reset pressure, the typical post-reactor trip pressure reduction is noted in the licensing and analyses of FSAR pressurization events. It should be noted that a more realistic best estimate analysis of the pressurization event, described in CEN-128, 'Response of CE NSSS of Transients and Accidents,'

indicates that PORV's are not challenged when the pressure reduction due to systems such as pressurizer spray and turbine bypass are considered."

The fault tree starts with the desired outcome of a "LOCA due to a PORV path left open." The fault tree then branches into two identical sequences to accurately model the fact that CE-designed operating plants, with the exception of Arkansas Power and Light Company's Arkansas Nuclear One Unit 2, have two PORVs which open simultaneously at the same pressure setpoint. The two sequences that model the PORVs branch into two paths. These paths are:

1. PORV opens and fails to reclose, and
2. block valve fails to close.

The first path, "PORV opens and fails to reclose," branches into two paths. These are:

1. PORV opens, and
2. PORV fails to reclose.

The "PORV opens" path uses the frequencies discussed in Section 3.1.1 of this report. The "PORV fails to reclose" path uses the PORV failure rate discussed in Section 3.1.4 of this report.

The other major path discussed above, "block valve fails to close," branches into two paths. These paths are:

1. operator fails to close valve, and
2. equipment failure.

The "operator fails to close valve" path simulates the failure of the operator to recognize and take the appropriate action of shutting the manual block valve in the case of a stuck-open PORV. The "equipment failure" path accounts for the various failure modes of the PORV block valve, including the failure of the automatic closure signal if the automatic closure system is postulated to be installed.

By modifying the basic fault tree described above and shown in Figure 1, CE performed small-break LOCA from a stuck-open PORV expected frequency calculations for five cases. The cases were for a CE-designed plant with:

1. a turbine runback feature and no operator action

2. no turbine runback feature and no operator action

3. no turbine runback feature and operator action

4. no turbine runback feature and no operator action, but automatic closure of the block valve

5. no turbine runback feature and no operator action, but automatic closure of series-redundant block valves.

3.1.4 CE's Probability Data

In order for CE to quantify the fault tree that was developed, probability data had to be gathered for each path at each node.

As detailed in Section 3.1.2 of this report, CE used historical operating data collected from a survey of the CE-designed operating plants to determine the expected frequency of the transient initiator events.

The probability data assigned to the other fault tree branches do not deal with the expected transient frequency of the plant. Instead, the remaining probability data deal with operator and equipment reliability. Specifically, operator and components data are necessary for:

1. failure of the PORV to reclose on demand once it has opened

2. failure of an operator to block the stuck-open PORV after it should have closed

3. failure of the PORV block valve to close (both manually and automatically).

For the failure rate of the PORV to reclose on demand once it has opened, CE used a value of 2×10^{-2} failure per demand. This failure rate was based on the operating history of Babcock & Wilcox (B&W) plants which use PORVs similar to those of CE-designed plants. This failure rate did not incorporate the CE operating history of no failures in 38 operational openings. It used only the B&W history of three failures in 150 operational openings.

For the failure rate of an operator to block the stuck-open PORV after it should have closed, it was stated in Reference 7:

"The value is based on data in Wash 1400 Table III 6-1, "General Error Rate Estimates." The value used is the mean of the error rate at the operator's normal stress level and the error rate at severe stress levels.

The error rate at the operator's normal stress level was taken to be the sum of the estimated error rates where display is available in the control room and no arithmetic calculation is required. This result was rounded conservativity to 1.0×10^{-2}.

Severe stress level probability was taken as the upper value of the general error rate given very high stress levels where dangerous activities are occurring rapidly, or 3×10^{-1}. The computed (mean) error rate was $(3 \times 10^{-1} + 1 \times 10^{-2})/2$ or 1.55×10^{-1}."

To calculate the failure rate of the PORV block valve to close, CE postulated the failure modes of the block valve. These were:

1. mechanical malfunction of 6.59×10^{-6} per demand based on the CE interim data base
2. block valve motor failure of 2.02×10^{-4} per demand based on the CE interim data base
3. block valve breaker failure of 1×10^{-6} per demand based on IEEE Std 500-1977
4. automatic signal not received of 1.2×10^{-2} based on WASH-1400.

3.1.5 Method of Reducing PORV System Failure

In Reference 5, CE provided a discussion of possible methods for reducing PORV system failures by reducing the frequency of challenges to the PORVs. Of the six possible methods, only one (elimination of turbine runback) was considered not to adversely affect the plant. A review of the turbine runback feature indicated that its elimination would not adversely affect plant operation while reducing PORV challenges to a significant degree. The methods rejected as causing adverse impact, along with a very brief summary of the impact, are provided below:

Method	Impact
Raise PORV Setpoint	High pressurizer pressure reactor trip would also be raised. This would invalidate the safety analysis and increase primary safety valve challenges.

Method	Impact
Lower High Pressurizer Pressure Trip Setpoint	This would also lower PORV setpoint, thereby increasing PORV challenges.
Raise PORV Setpoint and Add Another High Pressurizer Pressure Reactor Trip at 2400 psig	Very small number of PORV openings would be avoided by difficult and impractical circuitry changes and and bistable addition.
Block Out and/or Deactivate PORV During Operation	PORVs should be used to preclude safety valve challenges. If a safety valve sticks open, there is no block valve to mitigate this failure.
Reduce Operating Pressure	Operating DNB ratio would be decreased. Also, load rejection pressure overshoot would be increased due to delay in reaching high pressure reactor trip.

In addition to reducing PORV challenges, improved PORV system failure countermeasures were discussed. Three of the proposed methods were judged to have positive effects on mitigating the consequences of PORV system failure: improved PORV indication, PORV power from emergency power supplies, and improved operator capability. The fourth method, providing automatic closure of the block valve whenever a PORV failed to close on demand, was determined to be a complex alternative with its own failure modes and therefore required further evaluation of positive and negative effects.

In summary, CE identified a failure reduction program to be implemented at all CE-designed operating plants. The failure reduction program described in Reference 5 is as follows:

1. The turbine runback feature to be eliminated.
2. The motor operators for the PORV block valves and the pilot solenoids for the PORVs to be provided with emergency power supplies to permit them to function upon the loss of all non-emergency power.
3. Ultrasonic flowmeters to be installed on the PORV discharge piping to provide a direct measurement of steam flow and, therefore, of PORV position, with indication and alarm in the control room.
4. Operator training programs to be initiated to provide the operator with a more comprehensive understanding of plant operation under

emergency conditions. Guidelines and detailed emergency operating procedures to be developed to aid the operator to cope with a spectrum of emergency conditions. This includes the conditioning of the operator to recognize and respond promptly to PORV failure to prevent escalation of the failure to a small-break LOCA.

3.1.6 Analysis and Result of Failure Reduction Program

In Appendix A to Reference 5, CE provided an analysis to estimate the reliability of the PORV system, as well as an estimate of reliability expected following implementation of the PORV Failure Reduction Program. As noted earlier, CE used the B&W plant PORV demand failure rate of 0.02 failures-to-close per opening. The CE failure rate data base was not used as it represented a small statistical sample. A combined CE and B&W data base, which would reduce the failure rate by 20% to 0.016 failures-to-close per opening, was not used by CE. Westinghouse data were not used because of the different PORV vendor used by Westinghouse.

A value of 0.155 was used for the probability of failure of the operator to isolate the failed-open PORV. This value was based on the data in WASH-1400 and was taken as the mean between the operator's normal stress level and severe stress level failure probabilities.

The frequencies for a PORV loss-of-coolant accident for the various conditions analyzed are summarized below:

Case No.	Description	Frequency per Year
1	Turbine Runback Feature and No Operator Action	2.6×10^{-2}
2	No Turbine Runback Feature and No Operator Action	1.1×10^{-2}
3	No Turbine Runback Feature and Operator Action	1.8×10^{-3}
4	No Turbine Runback Feature and Automatic Closure of the Block Valve	1.4×10^{-4}
5	No Turbine Runback Feature and Automatic Closure of Series Redundant Block Valves	1.7×10^{-6}

3.1.7 Primary Safety Valves

With regard to the primary safety valves, CE made the following statement in Reference 5:

"No primary safety valve lifts have been reported for CE operating plants during approximately 30 reactor-years of operation. Westinghouse plants also have not reported any primary safety valve lifts. One primary safety valve lift has been noted in a B&W plant, but no details were given. In view of the lack of challenges to the primary safety valves, a direct quantitative estimate of their reliability based on experience cannot be made."

CE then proceeded to discuss the similarities between the primary safety valves and the main steam safety valves (MSSVs). In concluding the discussion, CE stated:

"Based on the seven reported MSSV failures and the 5650 estimated MSSV demands, a failure rate of 1.24×10^{-3} per demand is estimated. This failure rate is lower than the value of 2×10^{-2} estimated for power operated relief valves in NUREG-0560. Assuming that the MSSV reliability data are to some degree applicable to the primary safety valves, the data suggests that the primary safety valves may be more reliable than the PORVs. More definite conclusions must await development of operational and/or test data on primary safety valves."

3.1.8 Comparison With Other PWRs

In Reference 5, CE described a basic difference in the design function of the PORVs in a CE-designed plant as opposed to those in B&W- and Westinghouse-designed plants. The distinction is significant in that there is an inherent incremental margin to PORV challenges of the CE design as compared to those of B&W and Westinghouse designs. CE's statement is provided below:

"On CE plants, the initial design function of the PORVs was solely to reduce the challenges to the primary safety valves during power operation. The PORVs on B&W and W plants had an additional function, namely, to reduce the frequency of reactor trips due to high pressure. The PORV actuation set point on CE plants coincides with the high pressure reactor trip setpoint, whereas, the other PWR vendors required that the PORV actuation pressure be below the high pressure reactor trip setpoint in order to reduce the number of high pressure trips. The CE design allows the specification of a higher PORV actuation pressure, and

therefore a greater margin above the normal plant operating prssure than do the other PWR designs. Typically, the margin between normal operating pressure and the PORV actuation setpoint was about 150 psi for CE plants, 100 psi for W plants, and 70 psi for B&W plants. This difference provided an incremental margin to PORV challenges in CE plants compared with those of the other PWR vendors."

3.1.9 Conclusion

CE has submitted a report which shows that the frequency of a small-break LOCA due to PORV failure has been reduced to the range of recurrence frequencies for small pipe rupture estimated in WASH-1400. The report shows that incorporation of an automatic block valve feature would further improve PORV system reliability; however, the Licensee states further evaluation of positive and negative impacts on overall plant safety require further study.

The report documents the various actions taken to decrease the probability of a small-break LOCA due to a stuck-open PORV or safety valve. The analysis considered operator error, reliability of the PORV block valve, and initiating events that result in an overpressurization. The analysis did not consider depressurization events that actuate high pressure injection and require operator action to prevent challenges to the PORV during recovery. CE has provided data to support the quantification of the fault tree paths at each node. The CE report includes a discussion of the safety valve challenge rate. However, instead of compiling operational data regarding safety valves for use in determining safety-valve failure rates, CE cited a lack of historical data to permit quantification. Where appropriate, CE has used operational data, including operational data from B&W-designed plants.

In summary, CE has submitted a report which is complete with the following exceptions:

- Depressurization events were not considered as initiating events
- Operational historical data were not used to compile safety-valve failure rates.

3.2 EVALUATION OF THE CE REPORT SUBMITTED IN RESPONSE TO NUREG-0737, ITEM II.K.3.2

The evaluation of the information reviewed in Section 3.1 of this report, as well as other information pertinent to the stuck-open PORV or safety relief valve topic, is provided in this section.

3.2.1 Evaluation of CE's Fault Tree Transient Initiator Event Frequencies

In Reference 5, CE determined a PORV initiator event frequency based on a survey taken in 1980 of 29 years of operating history. The frequency of 0.66 events per reactor-year for CE plants was based on a total of 19 events occurring in the 29-year period.

CE noted that recording of all PORV actuations had not previously been a requirement. Consequently, only three PORV actuations during power operations had been recorded. Sixteen additional actuation events, however, could be inferred from the recorded number of high pressurizer pressure reactor trip events. The inference was possible because the high pressure trip signal is generated by the same bistable which actuates the PORV. CE went on to note that 11 of the 16 high pressure reactor trips were caused by the turbine runback feature of the protection system. Since this feature has reportedly been eliminated from all CE plants, these actuation events were eliminated from the data base leaving a total of 8 (3 + 16 - 11) in 29 years for an initiator event frequency (with no turbine runback feature) of 0.276 per reactor-year.

In evaluating this approach, three items require further discussion:

1. elimination of the 11 turbine-runback-initiated events from the data
2. the possibility that a significant number of unrecorded PORV actuation events were not included in the 1980 survey
3. the possibiity of multiple PORV cycles per initiator event.

Each of these items is discussed separately below.

Elimination of the turbine runback events from the data is problematic in that some plant transient initiated the turbine runback. From the data

presented, it is not clear whether a PORV actuation would have occurred, had the turbine runback feature not been available during these transients. The turbine runback feature was installed to anticipate load changes which otherwise could lead to a turbine trip. With the runback feature removed, it is likely that several of the recorded events would have initiated a turbine trip.

As documented in CE-plant FSARs, a turbine trip results in a challenge of the PORV when no credit is taken for the coincident reactor trip. However, a turbine trip does, by design, initiate a reactor trip on CE-designed plants and, therefore, the primary pressure transient will be terminated before reaching the PORV setpoint. Consequently, even with the conservative assumption that all 11 turbine runback events would result in a turbine trip, less than 0.11 PORV challenges would result assuming the reactor trip feature to be at least 99% reliable.

In summary, CE's elimination of the 11 turbine runback initiator events from the data bases because of the removal of the runback feature from the plant is considered technically valid, because this action inserts a reactor trip coincident with a potential turbine trip. CE's use of a 0.276 initiator event frequency, with the turbine runback feature removed, is considered valid.

Regarding the possibility that unrecorded initiator events have bypassed the data survey, it is believed that the survey provides an adequate reflection of initiator events for the following reasons:

1. The requirements to record the high pressurizer pressure reactor trips ensures that the majority of potential PORV actuations, because of the design of CE plants is such that a pressure transient sufficient to cause a reactor trip must challenge the PORVs.
2. Reactor trips other than overpressure events are unlikely to challenge the PORVs because of the prior insertion of control rods. There are no PORV challenges of this nature in the data base.
3. Independent data taken from EPRI Report NP-2230 of January 1982 (ATWS: A Reappraisal [12]) tend to confirm the survey data.

In NUREG-0653 [2], the NRC stated:

"The vast majority of transients that actually occur in power plants are not as severe as those postulated in FSARs (e.g., the initial conditions are less limiting, system failures are not as extensive, the heat transfer coefficients are not as biased). CE indicates that of all the transients analyzed in FSARs, only loss-of-load, uncontrolled rod withdrawal, or loss of all non-emergency ac power could actually result in lifting a PORV. Based upon plant operating experience, the only event observed which had caused PORVs to open is the loss of load or turbine runback event."

Using the data from Reference 12 (ATWS), the following event frequencies for CE plants are derived:

Event No.	Event	Total Events	No. Years	Event/Year
2	Uncontrolled Rod Withdrawal	0	15.42	0
33	Turbine Trip	30	15.42	1.94
34	Generator Trip	6	15.42	0.39
35	Total Loss of Offsite Power	2	15.42	0.13
		38	15.42	2.46

Applying the conservative assumption that 10% of these events would activate a PORV, the initiator event frequency would be 0.246, which is nearly identical to CE's frequency of 0.276 for non-turbine-runback plants.

With regard to the possibility of multiple PORV cycles per initiator event, it is stated in Reference 7 that only one PORV challenge occurs per initiator event because a reactor trip occurs simultaneously with reaching the PORV setpoint; therefore, by the time the PORV blowdown is complete, a post-reactor shutdown pressure reduction is in progress. This assumption is considered to be technically valid, and the consideration of multiple cycles per initiator event does not appear to be warranted where PORV actuation is automatic and not the result of operator action.

In summary, an initiator event frequency of 0.276 is considered a satisfactory initiator event frequency.

3.2.2 Evaluation of CE's Probability Data

Review of the probability data submitted in Reference 5 indicates two areas which require further discussion. These areas are:

1. the probability of a PORV to fail to close on demand once it has opened

2. the probability that the operator will fail to block the stuck-open PORV by manually shutting the block valve.

The probability of a PORV failing to close on demand once it has opened was determined by CE to be 2×10^{-2} failures per demand. The operation history of PORVs at CE plants is that there have been no failures in 38 valve openings. At B&W-designed plants, in which the same Dresser electromatic solenoid pilot-operated valves are used, there have been three failures during power operations in 150 openings. Since the B&W information provided a larger data base, CE chose to use the B&W information to derive the 2×10^{-2} failure rate. In determining the failure probability of a specific valve design, however, there is no reason why the data from CE and B&W plants should not be combined. In this case, there have been three failures in 188 openings for a 1.6×10^{-2} failures per demand rate.

With regard to the probability that the operator will terminate a potential LOCA by shutting the block valve, CE assigned an error rate of 0.155 per event, based upon WASH-1400 data. Additional information on the probability that an operator will take a certain action under emergency conditions can be found in the Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications, NUREG-CR/1278 [13]. An independent calculation of the expected operator error rate using the techniques of NUREG-CR/1278 yields a rate of 1.5×10^{-2} per demand, which is considerably below the error rate used by CE.

3.2.3 Evaluation of CE's Conclusions on PORV Reliability

In Reference 5, CE provided the following conclusions:

"The C-E operating plants after approximately 29 reactor-years of operation have experienced no PORV failures during power operation. The

elimination of the turbine runback feature and the provision of a direct reliable means for indicating PORV position to the operator provided significant improvements in system reliability. The recurrence frequency of a small break LOCA due to PORV failure has been reduced by an estimated factor of about 15 to a value of about 1.8×10^{-3} per reactor-year. This recurrence frequency is well within the 90% confidence range of the recurrence frequencies of 10^{-2} to 10^{-4} per reactor-year for a LOCA due to a small pipe rupture estimated in WASH-1400. Improved operator training programs and emergency procedures, as well as the provision of emergency power to the PORVs and to their block valves, though not quantified, has reduced the small break LOCA recurrence frequency even further. The incorporation of the feature of automatic block valve closure upon PORV failure would further increase PORV system reliability."

Figure 1 shows the calculation of CE's recurrence frequency of 1.8×10^{-3} per reactor-year for a small-break LOCA due to a stuck-open PORV (turbine runback feature eliminated). Figure 2 shows the same calculation with the following exceptions: (1) a PORV failure of 1.6×10^{-2} has been used (combines CE data and B&W data), (2) an operator error rate of 1.5×10^{-2} has been used (from NUREG-CR/1278), and (3) accounts for the possibility that a PORV which spuriously opens will not reseat (i.e., failure probability of 1.0). This calculation yields a recurrence frequency of 2.2×10^{-4} per year, which is below both the CE determination and the WASH-1400 median probability of 1×10^{-3} per year.

With regard to installation of an automatically operated block valve feature, CE's analysis indicates that this feature would reduce the frequency of a LOCA from a stuck-open PORV to 1.4×10^{-4} events per year, while an automatic closure feature employing series-redundant block valves would reduce the frequency even further to 1.7×10^{-6} events per year.

The recurrence frequency of a small-break LOCA from a stuck-open PORV, however, is already well within the 90% confidence range of 10^{-2} to 10^{-4} given in WASH-1400 (conservatively 1.8×10^{-3} and probably more realistically 1.4×10^{-4}).

3.2.4 Evaluation of Primary Safety Valves

Section 14.5 (Loss of Load Event) of the FSAR for one CE-designed plant (Calvert Cliffs plant) discusses the situation in which a turbine trip

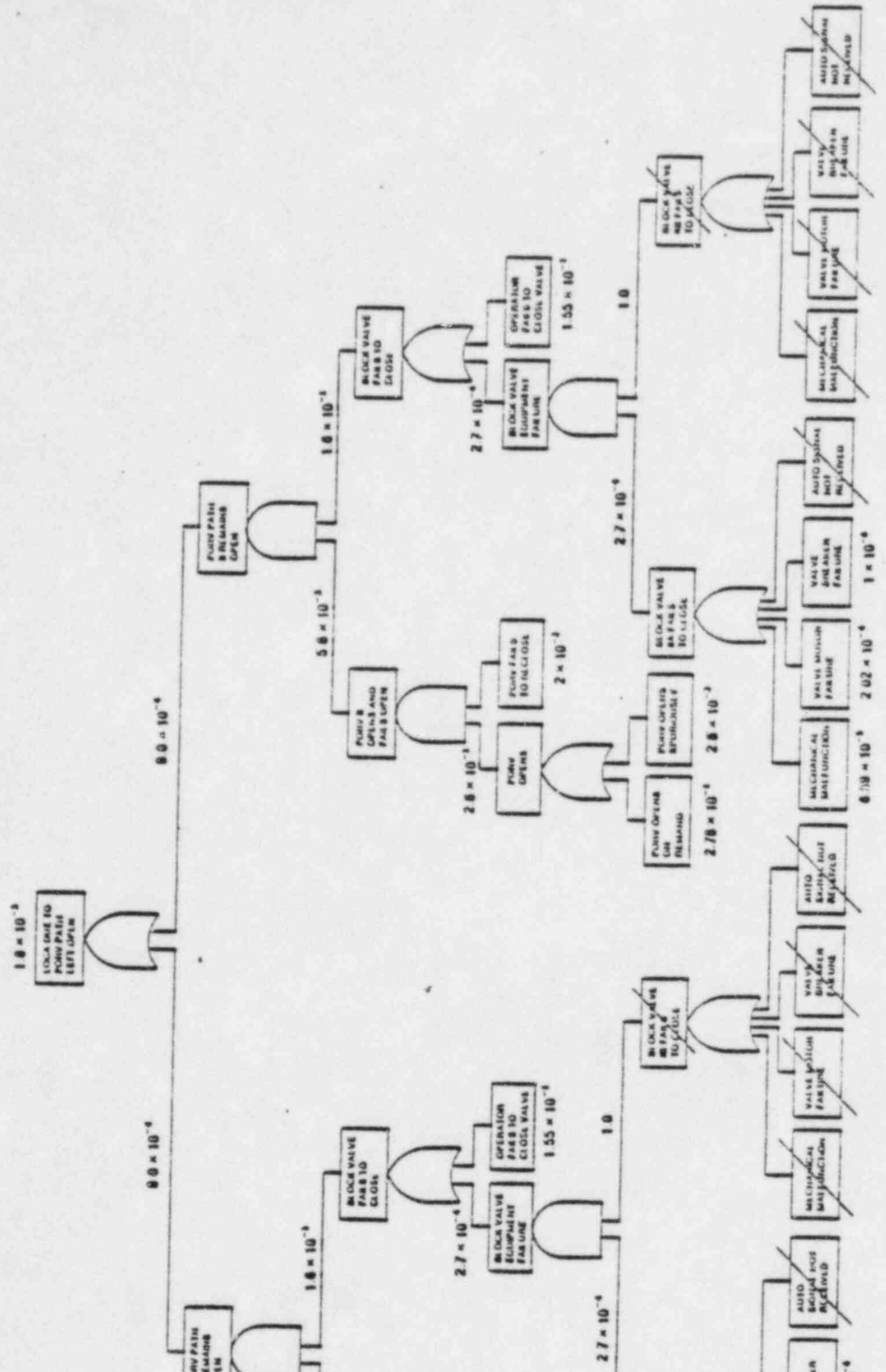

Figure 1. Combustion Engineering Basic Fault Tree

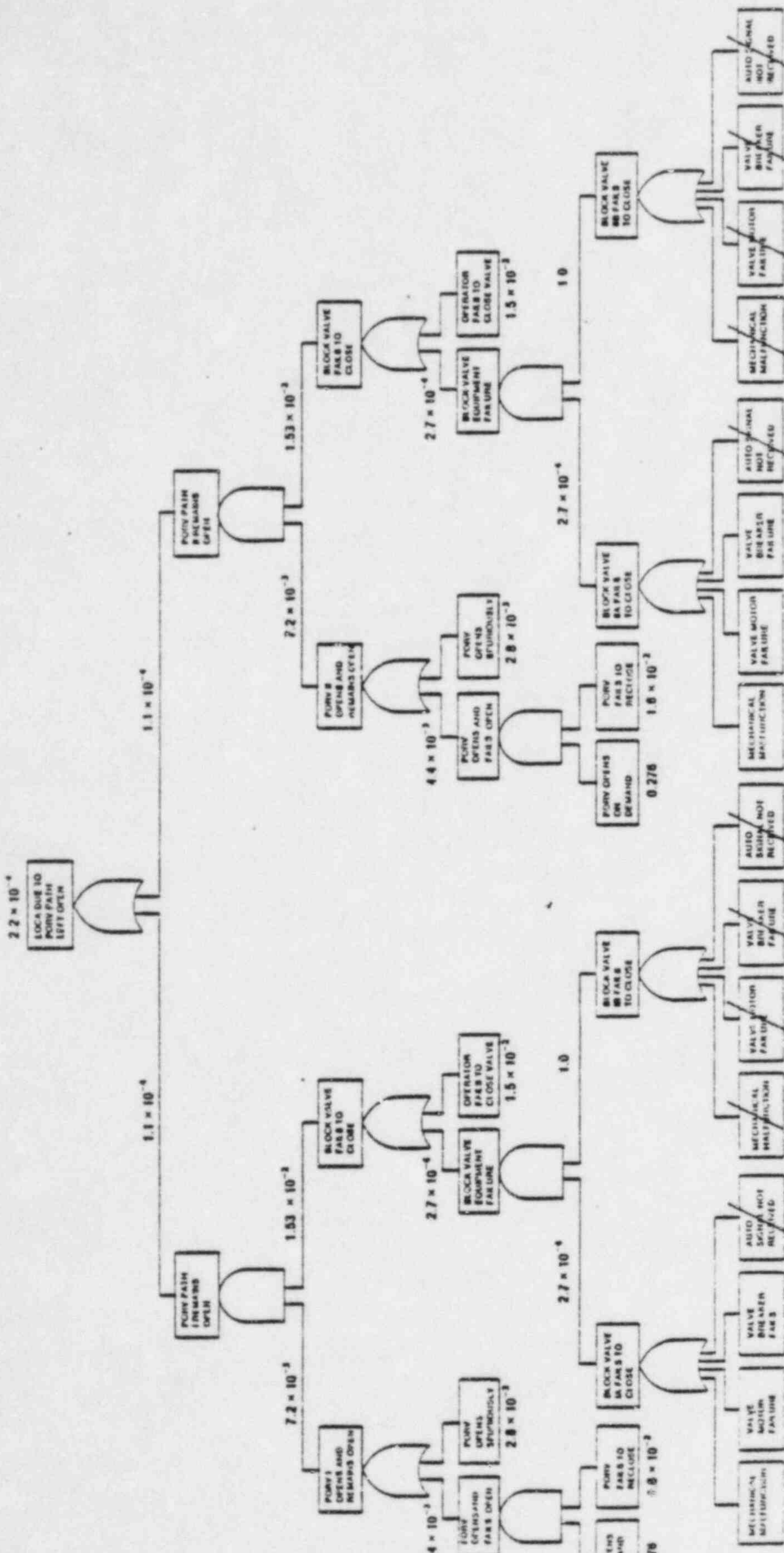

Figure 2. Combustion Engineering Basic Fault Tree, Revised

occurs (complete loss of load) and no credit is taken either for the reactor trip which occurs coincident with the turbine trip or for PORV opening. The analysis of Section 14.5 shows that in this scenario, primary system pressure peaks at approximately 2540 psia (Figure 14.5-4 of the FSAR). Since the primary safety valves are set at 2500 and 2565 psia, at least one of the safety valves will be challenged. These data indicate that there is a substantial likelihood that any initiator event which challenges the PORV will challenge a safety valve, if neither PORV opens.

In Reference 4, CE states that the Palisades plant has operated with both PORV block valves shut since 1972. In addition, the ANO-2 plant does not have PORVs installed. Consequently, for these two plants, there is a quite high likelihood that an SRV will be challenged in an overpressure condition. With regard to the remaining CE-designed plants, CE stated that licensees have operated from time to time with one or both PORV block valves shut due to PORV leakage. Making the conservative assumption that PORV leakage is sufficient to cause the block valve to be shut 50% of the time, there is a 25% probability that both block valves will be shut at any given time at these units (0.5 x 0.5). Therefore, it appears that an evaluation of SRV challenges should be performed for two cases: one where there is no contribution whatsoever from the PORVs (Palisades and ANO-2) and the other where the PORV block valves are shut 25% of the time (the remaining CE units).

With regard to probable SRV failures, CE stated that because of the lack of challenges to the SRVs, a direct quantitative estimate of their reliability based on experience could not be made. CE further stated that, assuming main steam safety valve (MSSV) data are to some degree applicable to primary safety valves, a failure rate of 1.24×10^{-3} could be estimated. This rate was based on seven reported MSSV failures in 5650 estimated demands.

There are sufficient similarities between the SRVs and the MSSVs to justify using MSSV failure data to determine the SRV failure rate. However, because of the nature and design of a nuclear plant, the MSSVs are exercised more frequently than are the SRVs; also, the MSSVs relieve a high quality steam, while the SRV may occasionally serve as a water relief. In view of these two considerations, a conservative approach would entail the use of a

somewhat higher failure rate for the SRV. For the purpose of this analysis, the SRVs have been assumed to fail at a rate 10 time larger than that of MSSVs (1.24×10^{-2} per demand). This failure rate is slightly lower than the PORV failure rate (1.6×10^{-2}), which is consistent with the fact that the PORV is a more complicated valve with more possible failure mechanism.

Incorporating the above data into the event trees of Figures 3 and 4, the probability of a small-break LOCA from a stuck-open SRV is estimated as follows:

All Plants Except Palisades and ANO-2 (Figure 3)	8.6×10^{-4} per reactor-year
Palisades and ANO-3	3.4×10^{-3} per reactor-year

The parameters used in these event trees are as follows:

Node	Value	Reference/Rationale
Transient Initiator Event	0.276 per year	Section 3.2.1 of this report
PORVs Not Blocked (Figure 3)	Yes: 0.75 No: 0.25	Section 3.2.4 of this report
PORV Opens on Demand	Yes: 0.9999 No: 1.0×10^{-4}	Reference 8
PORVs Not Blocked (Figure 4)	Yes: 0 No: 1.0	Section 3.2.4 of this report
PORV Recloses on Demand	No: 1.6×10^{-2}	Section 3.2.2 of this report
PORV Blocked Closed After PORV Failure	No: 1.53×10^{-2}	Section 3.2.2 of this report
SRV Opens on Demand		
No PORV Opening	Yes: 1.0	Conservative assumption
With PORV Opening	Yes: 1×10^{-3}	If a PORV opens, no SRV set-point will be reached. A probability of 1×10^{-3} is assigned to conservatively account for a possible premature opening under elevated RPS pressure conditions.
SRV Recloses on Demand	No: 1.24×10^{-2}	Section 3.2.4 of this report

Regarding the results of Figure 3 and 4, the following observations should be made:

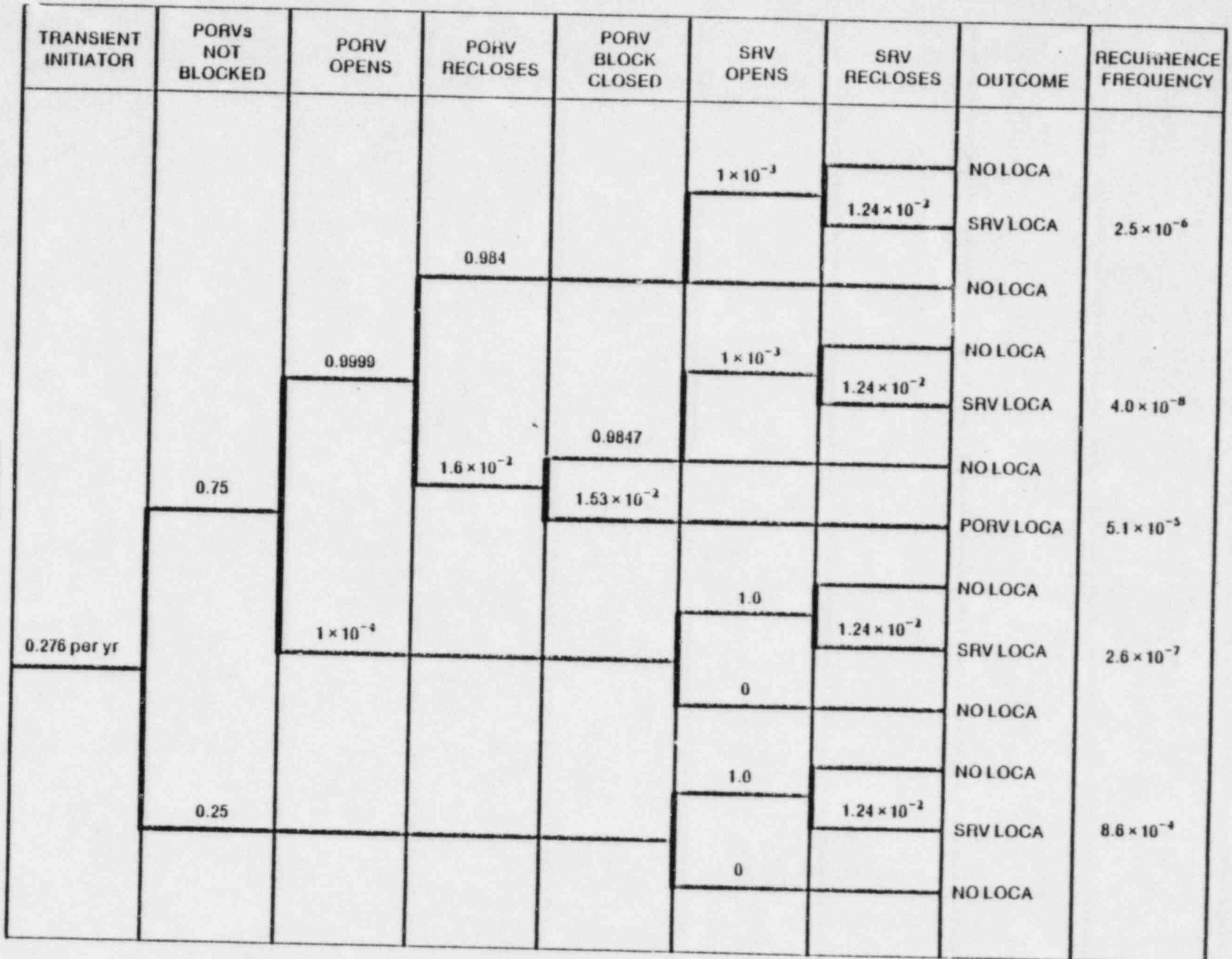

Figure 3. Small-Break LOCA from Stuck-Open SRV (All CE Plants Except Palisades and ANO-2)

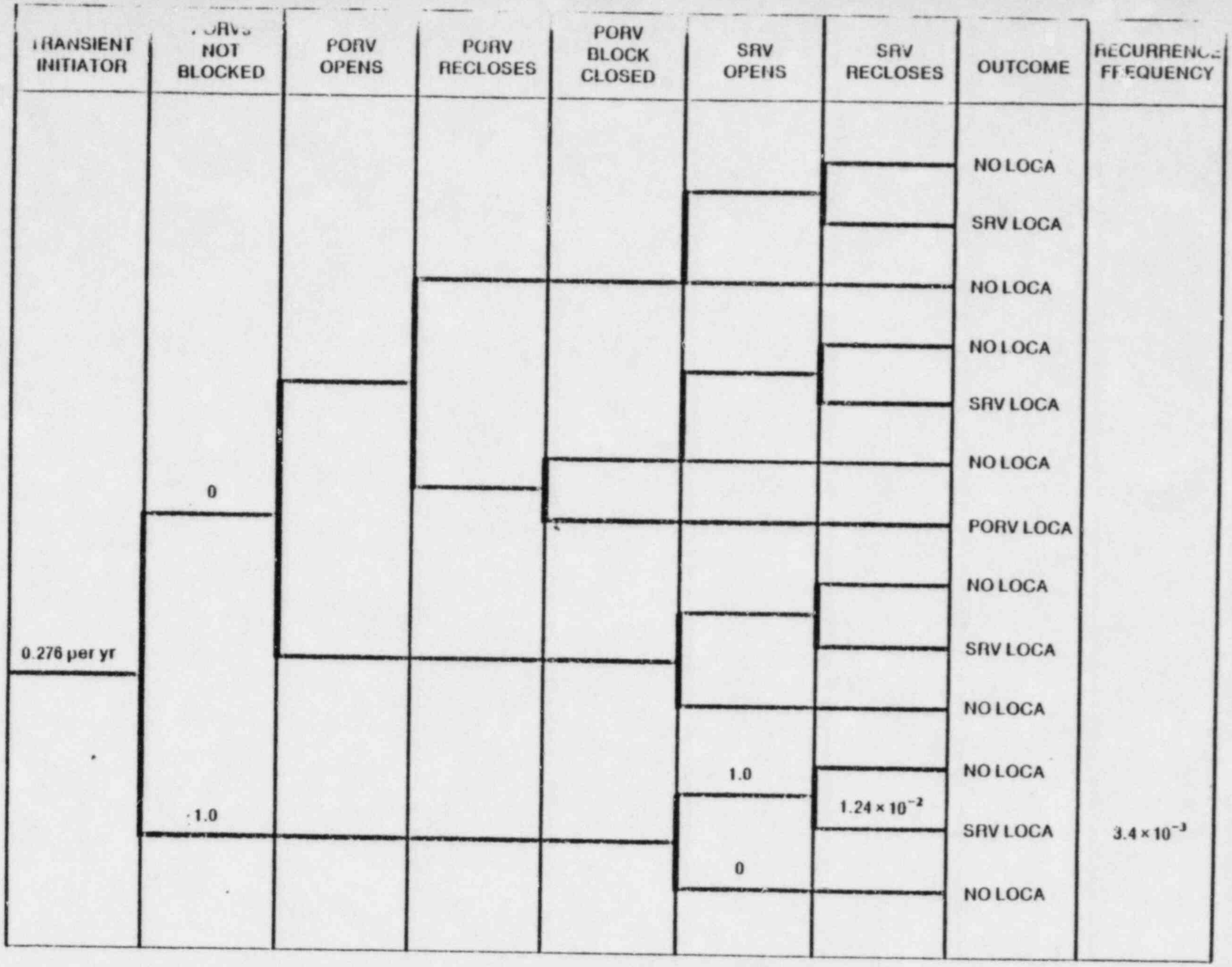

Figure 4. Small-Break LOCA from Stuck-Open SRV
(Palisades and ANO-2)

1. The results must be considered extremely conservative because:

   a. No credit has been taken for the effect of the reactor trip which will occur before the SRV setpoint is reached.

   b. The SRV failure rate was conservatively set as 10 times the calculated MSSV failure rate.

   c. The assumption that PORV leakage causes its block valve to be shut 50% of the time is a conservative assumption.

2. The calculations of Figures 3 and 4 yield a small-break LOCA probability on a per-valve basis. CE plants have two to three SRVs per unit (some set at identical setpoints and others set at different setpoints). Nevertheless, in view of the fact that no credit has been taken from the pressure reduction associated with the reactor trip which would preceed the SRV opening, it is logical to assume that the first SRV to open would terminate the pressure increase and that only one SRV per initiator event opens. For the same reason, multiple SRV cycles per initiator event need not be considered.

3. The small-break LOCA from a stuck-open SRV probabilities of Figures 3 and 4 remain well within the 10^{-2} to 10^{-4} range of WASH-1400 for small-break LOCAs. Even in the extremely unlikely case of two SRVs opening in the same event, the probability of a small-break LOCA remains within the WASH-1400 range for all CE units.

4. Figure 3 shows a small-break LOCA probability from a stuck-open PORV to be 5.1×10^{-5}. This value differs from that of Figure 2 because Figure 2 accounts for two PORVs opening simultaneously and also because Figure 2 does not consider the block valves to be shut 25% of the time. By doubling the Figure 3 value and dividing by 0.75 (percentage of time the block valves are open), the valve of Figure 2 is obtained (1.4×10^{-4}).

In addition to the above analysis performed by the authors of this TER, Reference 7 presented another analysis of the probability of a small-break LOCA from a stuck-open SRV, performed by one licensee of a CE-designed plant. This analysis determined the recurrence frequency of a small-break LOCA from a stuck-open SRV to be 1.1×10^{-4} per reactor-year, with the predominant failure path being through inadvertent opening of the SRV. Using data from a recently completed IREP report, values of 2×10^{-2} per event and 3×10^{-3} per demand were used for the probabilities of premature opening per valve and failure to close once open, respectively. This recurrence frequency is

comparable to the value of Figure 3, which is the figure applicable to the licensee submitting Reference 7, although Figure 3 is somewhat higher due to its conservative approach.

In summary, it is concluded that the small-break LOCA frequency range of WASH-1400 satisfactorily bounds the probability of a stuck-open SRV for all CE-designed units.

3.3 ADDITIONAL CONSIDERATIONS RELEVANT TO SMALL-BREAK LOCA FROM STUCK-OPEN PORV OR SAFETY VALVE

Although not addressed in the CE submittals, three other items should be considered relative to small-break LOCA from a stuck-open PORV or safety valve. These items are (1) events which require the operator to open the PORV, (2) overcooling events which challenge the PORV or safety valves through operation of the safety injection systems, and (3) low-temperature, overpressure events. These items are discussed in the following subsections.

3.3.1 Events Which Require the Operator Action to Open the PORV

Certain situations make administrative use of the PORV to depressurize the reactor coolant system. The more significant cases are:

1. use of the PORV in the plant recovery from a steam generator tube rupture event
2. use of the PORVs in "feed and bleed" operations in response to inadeqate core cooling (ICC) scenarios
3. use of the PORV to vent the reactor coolant system to remove air or non-condensable gases.

In any situation in which the operator wishes to depressurize the reactor coolant system, the operator can use the PORV to accomplish reactor coolant system depressurization. By cycling the PORV open and shut, the operator is generally able to control the reactor coolant system pressure. It is also noted that relatively rapid repetitive cycling of the PORV has the potential to increase the failure rate of the PORV to close when demanded.

Although not specifically addressed by CE in Reference 5, it is concluded that this problem is not a significant contributor to the expected frequency of a small-break LOCA from a stuck-open PORV. The main reason for this conclusion is that in any situation in which the operator is manually cycling the PORV open and shut, the operator would be particularly aware of the position of the PORV and the increase or decrease of primary system pressure. If the PORV should fail to close on demand (as indicated by decreasing pressure), the operator would immediately shut the PORV block valve, terminating the small-break LOCA. The limiting component of this scenario will most likely be the motor-operated block valve. The failure rate of a motor-operated valve to operate on demand from NUREG/CR-1363 [14], "Data Summaries of Licensee Event Reports of Valves at Commercial Nuclear Power Plants," is 4×10^{-3} per demand. Therefore, even assuming that the failure rate of the PORV is conservatively estimated at 1.5×10^{-2} per demand and that the frequency of these events is 0.1 per reactor-year (10 times the recorded frequency of a steam generator tube rupture), the contribution of these administratively required operator-induced PORV cyclings and subsequent failures is only 6.0×10^{-6} per reactor-year, which is not a significant contributor to the overall expected frequency of a small-break LOCA from a stuck-open PORV.

3.3.2 Overcooling Events

The contribution of overcooling events to the total expected frequency of a small-break LOCA from stuck-open PORV or safety valve is discussed and quantitatively evaluated in Appendix A of this report. The generic plant evaluated in Appendix A is assumed to have a high-head safety injection system capable of developing sufficient pressure in the reactor coolant system to challenge (demand open) the PORV(s) and/or safety valves. A generic plant of this design with a high-head safety injection would therefore be the limiting or bounding case to be evaluated for its contribution to the expected frequency of a small-break LOCA from all sources. From the calculations shown in Appendix A, it can be concluded that overcooling events which initiate the

safety injection system are not a significant contributor to the expected frequency of a small-break LOCA from a stuck-open PORV or safety valve.

3.3.3 Consideration of Low-Temperature, Overpressure Events

In August 1976, the matter of low-temperature, overpressure protection was raised, and licensees initiated procedures and proposed systems to mitigate postulated overpressure events while at reduced temperatures. The main concern was with the low-temperature modes of cooldown and heatup, during which overpressurization could cause brittle fracture of the reactor vessel. In most cases, licensees proposed a manually enabled low-pressure setpoint on the existing PORVs, supplemented by procedures and technical specifications, as the means of presenting overpressurization while at low temperatures.

With the reduced pressure setpoint in effect, transients or plant conditions normally associated with the shutdown, cooldown plant can cause PORV actuation (and hence possible small-break LOCA), such as inadvertent operation of the pressurizer heaters or excessive charging. Although not addressed by CE in Reference 5, it is considered that the low-temperature, overpressure situation need not be considered with the other transients which can result in a small-break LOCA from a stuck-open PORV. The reasons for this conclusion are:

- When reduced pressure setpoints are in effect, the plant will generally be in a long-term cooling mode using the RHR system. RHR can maintain system water inventory in spite of an open PORV.
- When reduced pressure setpoints are in effect, the operator has less equipment running and can readily diagnose abnormal conditions. The operator is in a less stressful condition and can be expected to react in a positive manner.
- When reduced pressure setpoints are in effect, the plant has been shut down for some period of time, and therefore decay heat rates are lower, providing more reaction time before thermal limits are approached.
- The temperature of the coolant released from the PORV under these conditions will normally be such that flashing to steam will not occur. The water will merely be collected in the containment sump.

4. APPLICABILITY

4.1 APPLICABILITY OF THE CE REPORT TO SPECIFIC CE-DESIGNED PLANTS

4.1.1 Initiator Event Frequencies

As discussed in Section 3.2.1 of this report, CE determined a PORV initiator event frequency of 0.276 per reactor-year based upon a total of eight PORV opening events in the 29 years of CE plant operation. The eight opening events were distributed among the plant as follows:

Calvert Cliffs Unit 1	2
Fort Calhoun	1
Palisades*	4
St. Lucie Unit 1	1
	8

With an average lifetime of the CE-designed plants of 4 to 5 reactor-years, it is apparent that each CE plant has a plant-specific initiator event frequency equal to or less than 0.276 per reactor-year, with the exception of Calvert Cliffs Unit 1 and the Palisades plant. While it is not statistically valid to draw inferences from such a small data base, it can nevertheless be stated that it appears that Calvert Cliffs Unit 1 exceeds the average event rate by approximately a factor of 2, while the Palisades plant exceeds the average rate by approximately a factor of 4.

Even if a specific plant had an initiator event frequency four times the average rate, the probability of a small-break LOCA from a stuck-open PORV (as calculated by CE) would be increased by a factor of 4 to 7.2×10^{-3} per reactor-year, which remains within the WASH-1400 range of 1×10^{-2} to 1×10^{-4} per reactor-year for a small-break LOCA. Furthermore, increasing the initiator frequency by a factor of 4 in the revised calculation of Figure 2, a more realistic result of 6.3×10^{-4} per reactor-year is obtained, which is below the WASH-1400 median frequency of 1×10^{-3} per reactor-year.

*The Palisades plant reports it has operated with PORV block valves shut since 1972.

4.1.2 PORV Failure Rates

All CE-designed plants, except for Arkansas Nuclear One Unit 2, (ANO Unit 2), which does not have PORVs installed, are equipped with Dresser electromatic solenoid pilot-operated PORVs. For this reason, CE chose to use failure data from B&W-designed plants which have the same type of valve installed with a more substantial data base (150 B&W operational openings versus 38 CE valve openings*). Since all CE plants have the Dresser valve, except ANO Unit 2, these failure data are applicable to all CE-designed plants except ANO Unit 2.

4.1.3 SRV Data

As discussed in Section 3.2.4, estimates of small-break LOCAs from stuck-open SRVs were made by extrapolating information from MSSV failure data and by considering two different conditions (one where the PORV block valves are normally open and the other where the block valves are always shut or PORVs are otherwise not available). By determining small-break LOCA probabilities for these two different conditions, LOCA probabilities applicable to each of the CE-designed plants have been provided.

4.2 SUMMARY

In view of the foregoing information, portions of this report related to PORV reliability are applicable to all CE-designed units except ANO Unit 2, which does not have PORVs, and the SRV portions of the report are applicable to all CE-designed units.

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*Note: The reason there have been 38 openings in CE units while the initiator event frequency considers only 8 operational openings is that a substantial number of PORV openings were attributed to the turbine-runback feature which has been eliminated in order to improve PORV reliability. When the openings due to the turbine-runback feature are eliminated from the data base, the number of operational openings is reduced to 8.

5. CONCLUSIONS

The conclusions resulting from evaluation of the CE report against the review criteria of Section 2 are as follows:

- CE's approach to developing an initiator event frequency is considered to be technically valid. An initiator event frequency of 0.276 events per year which challenge the PORV is considered valid, when the turbine runback feature is removed.

- By combining CE and B&W PORV failure data, a failure rate of 1.6×10^{-2} failures per demand can be used rather than the 2.0×10^{-2} failures per demand rate from the B&W data alone.

- The Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Plant Applications (NUREG-CR/1278) yields an expected operator error rate of 1.5×10^{-2} errors per event, which is less than the 0.155 errors per event rate used by CE.

- CE calculated the recurrence frequency of a small-break LOCA from a stuck-open PORV to be 1.8×10^{-3} events per year, which is well within the WASH-1400 frequency of a small-break LOCA of 10^{-2} to 10^{-4} events per year. A revised calculation of this frequency in this report shows a more realistic frequency to be 2.2×10^{-4} events per year, which is below the WASH-1400 median probability of 1×10^{-3}.

- The probability of a small-break LOCA from a stuck-open safety relief valve is 8.6×10^{-4} per reactor-year for all CE plants except Palisades and ANO-2 and 3.4×10^{-3} per reactor-year for Palisades and ANO-2. These frequencies are both within the 10^{-2} to 10^{-4} per reactor-year range of WASH-1400.

- Additional events which can challenge the PORV or safety valves, such as operator cycling of the PORV and overcooling events and low-temperature, overpressure event, have been considered and do not significantly influence the frequency of possible small-break LOCAs.

- CE's report sufficiently addressed actions taken to reduce the frequency of a small-break LOCA from a stuck-open PORV.

- The CE report is applicable to all CE-designed plants except for the estimate of PORV reliability which is not applicable to ANO Unit 2 which does not have PORVs.

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APPENDIX A

APPENDIX A

EVALUATION OF THE CONTRIBUTION FROM OVERCOOLING EVENTS TO THE TOTAL PROBABILITY OF A SMALL-BREAK LOSS-OF-COOLANT ACCIDENT FROM A STUCK-OPEN POWER OPERATED RELIEF VALVE OR SAFETY VALVE

Purpose

To review the available literature and operational historical data to ascertain whether or not Combustion Engineering and Westinghouse-designed nuclear steam supply system plants need to consider the contribution from overcooling events to the total probability of a small-break LOCA from a stuck-open PORV or safety valve.

Background

Overcooling events can cause a rapid depressurization of the primary system and subsequent initiation of the high pressure safety injection system. To plant operators, a rapid depressurization appears to be very similar to a small-break LOCA. As a consequence of the TMI-2 accident, operator guidelines were instituted to require the PORV blocking valve(s) to be shut, thus terminating a depressurization, if it was caused by a stuck-open PORV. Regardless of the cause of the depressurization, operator action is required to terminate high pressure safety injection upon subsequent repressurization to prevent challanges to safety valves (or PORV if unblocked). The following is a technical evaluation of whether such events can significantly contribute to the number of challenges experienced by the PORV and/or safety valve.

Evaluation

Secondary side overcooling transients usually occur because of overfeeding of a steam generator, demanding too much steam from the steam generators, or introducing excessive amounts of relatively cold auxiliary feedwater into the steam generators. NUREG-0667 [1], "Transient Response of Babcock & Wilcox-Designed Reactors," describes the sensitivity of the once-through steam generator (OSTG) in B&W designs to such overcooling transients. Specifically, it was concluded that:

"Because the heat removed is proportional to the transfer area, the amount of heat removed by an OTSG is essentially directly proportional to the height of liquid on the secondary side. As such, any change in secondary coolant level directly affects the amount of heat capable of being removed. This, coupled with the relatively smaller secondary side liquid inventory, results in a fairly rapid primary system response to secondary coolant system perturbations."

Reference 1 also describes the sensitivity of a U-tube steam generator, such as the kind presently used in Westinghouse and Combustion Engineering plants. It was concluded that:

"Since the heat removal rate is proportional to the product of the heat transfer coefficient, heat transfer area, and temperature difference, and because the product of the heat transfer area and heat transfer coefficient is usually high, only small changes in primary to secondary temperature difference are needed to accommodate rather large changes in heat removal rate. Because of this and because the volume of water on the secondary side surrounding the U-tubes is large, perturbations on the secondary side of the inverted U-tube steam generator, such as feedwater from changes or system pressure changes, do not readily affect the behavior of the primary coolant system."

Based upon both of these descriptions, it can be concluded that Babcock & Wilcox designed reactors are more susceptible to depressurizations caused by overcooling transients than reactors designed by Westinghouse or Combustion Engineering. This conclusion is supported by historical operational data. A Babcock & Wilcox generic report [2], "Report on Power-Operated Relief Valve Opening Probability and Justification for Present System and Setpoints," states that 8 overcooling transients have initiated high pressure safety injection system flow in 392 reactor trips, and that the current frequency of reactor trips is six trips per reactor-year per plant. Thus, for Babcock & Wilcox-designed reactors, the frequency of overcooling events with subsequent high pressure safety injection system flow equals 0.122 events per reactor-year. For plants designed either by Westinghouse or Combustion Engineering, very little pre-TMI information is readily available concerning plant response to events that overcooled the primary system in excess of the normal cooling expected following a reactor trip. Reference 1 states, "Since TMI-2, three events that depressurized the primary system to the HPI actuation setpoint have occurred in plants with reactors designed by Westinghouse and Combustion Engineering." Two of these events involved stuck-open turbine

bypass valves, and one was the result of a steam generator tube rupture. Since the steam generator tube rupture is a separate initiating event, it can be excluded from this study. During the 2 years between the TMI-2 accident and the completion of Reference 1, 41.7 reactor operating years were recorded by Westinghouse and Combustion Engineering plants. Therefore, the frequency of overcooling events with subsequent high pressure safety injection system flow equals 4.8×10^{-2} events per reactor-year for Westinghouse and Combustion Engineering plants.

To quantify the probability that an overcooling event will lead to a small-break LOCA from a stuck-open PORV or safety valve, an event tree was constructed. This event tree is shown in Figure A-1. The following paragraphs describe the branch nodes which are used in the construction of the event tree. Paths branching upward at these nodes represent a "yes" response to the question, while those paths branching downward represent a "no" response. When quantifying the event tree, the probabilities shown in Table A-1 the probabilities represent the probability that the answer to the question is yes or no, rather than the availability and unavailability of a system.

Node A

Operator stops HPI prior to PORV setpoint pressure	Upward paths at this node indicate that the operator has throttled or secured the high pressure safety injection system prior to the reactor coolant system pressure reaching the PORV opening setpoint pressure. The recommended PORV opening setpoint pressure is 2350 psia on Westinghouse-designed plants. Downward paths at this node indicate that the operator has failed to throttle or secure the high pressure safety injection prior to the reactor coolant system pressure reaching the PORV opening setpoint pressure.

Node B

PORV block valve(s) open	Upward paths at this node indicate that at least one PORV block valve is open when the challenge to the PORV occurs. This applies both to the case

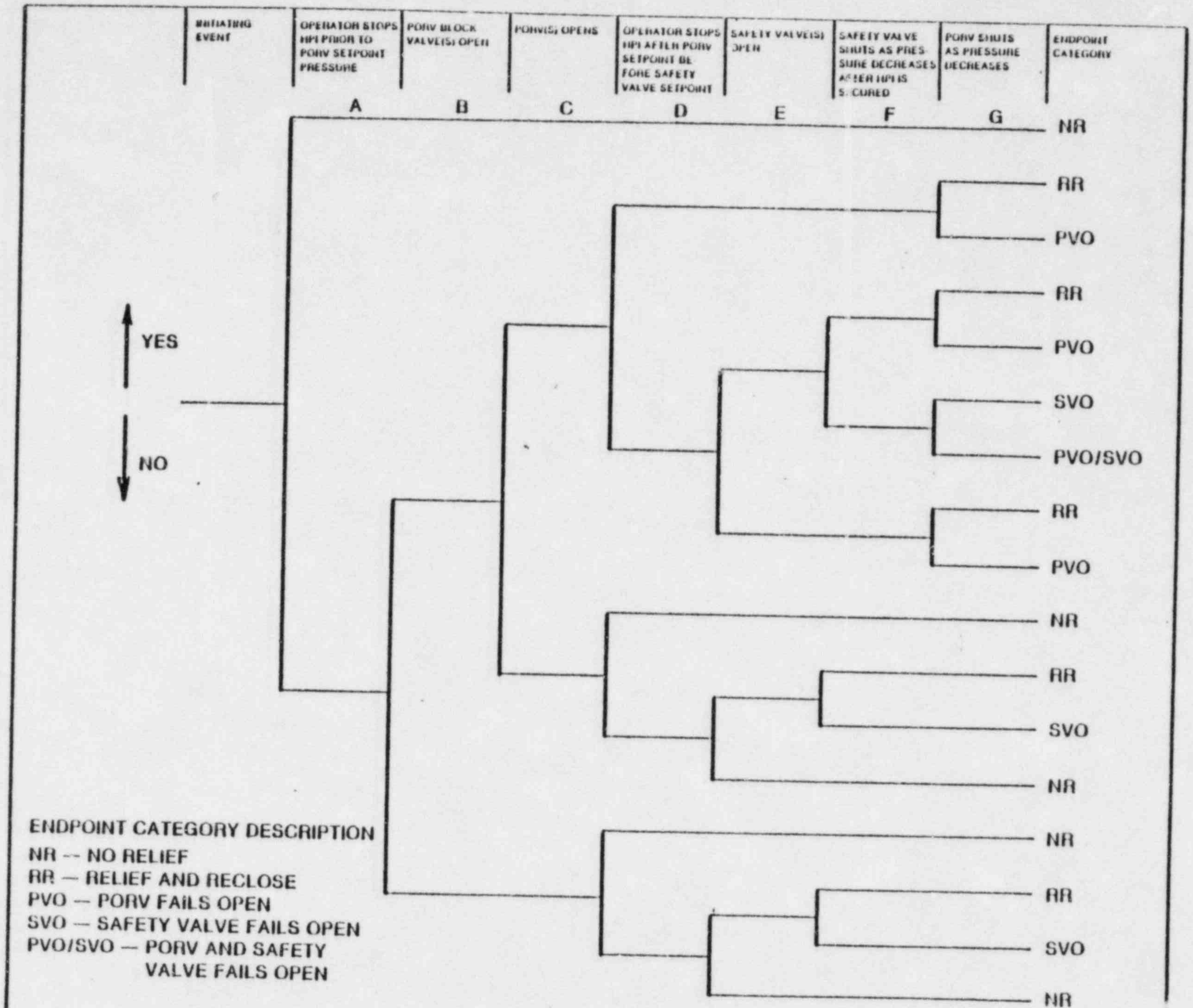

Figure A-1. Overcooling Event Transient Event Tree

Node B (Cont.)

where the PORV block valve is manually positioned, and the case of automatic open/closure systems where the block valve may be automatically moved.

Downward paths at this node represent those events where all the PORV block valves are closed when the PORV opening setpoint pressure is reached.

This node is not considered to be relevant for those events where the PORV opening setpoint pressure is not reached.

Node C

PORV(s) open

Upward paths at this node represent the PORV(s) opening after the PORV opening setpoint pressure is reached.

Downward paths at this node represent the PORV(s) staying closed after the PORV opening setpoint pressure is reached.

Since this node is relevant only for those events where the PORV opening setpoint pressure is reached and the PORV block valves(s) are open, the probability of the PORV staying closed represents the failure of the PORV to open on demand. This probability for the failure of the PORV to open on demand must therefore include such failures as pressure sensors, pressure transmitters, and control channels, as well as those failures associated directly with the PORV.

Node D

Operator stops HPI after PORV setpoint before safety valve setpoint

Upward paths at this node indicate that the operator has throttled or terminated HPI after the PORV opening setpoint pressure has been exceeded but before the safety valve opening setpoint pressure is reached. The recommended opening setpoint for safety valves on Westinghouse-designed plants is 2500 psia.

Downward paths at this node indicate that the operator has failed to throttle or terminate the HPI before the safety valve opening setpoint pressure was reached.

Node D (Cont.)

The probability at this node must reflect whether the PORV and the PORV block valve are open, since the probability of reaching the safety valve opening setpoint pressure is significantly reduced if the PORV and PORV block valve are open.

(Note: This analysis assumes that the PORV setpoint is exceeded one time per event, resulting in one PORV opening. Multiple PORV cycles are not considered for the following reasons:

1. The results of this analysis are to be compared to the results of the CE report. The CE report considers one PORV opening per initiator event.

2. The operator who fails to secure HPI prior to the first PORV opening would be alerted to the need to secure HPI prior to the second opening.

Node E

Safety valve(s) open

Upward paths at this node represent the opening of a safety valve at the safety valve opening setpoint pressure.

Downward paths at this node represent the safety valve staying closed after the safety valve opening setpoint pressure is reached.

This node is not considered to be relevant for those events where the safety valve opening setpoint pressure is not reached.

Node F

Safety valves(s) shut as pressure decreases after HPI is secured

Upward paths at this node represent the successful reclosing of the safety valves(s) when the reactor coolant system pressure decreases below the safety valve opening pressure setpoint after the HPI system is secured.

Downward paths at this node represent the failure of the safety valve(s) to reclose when the reactor coolant system pressure decreases below the safety valve opening pressure setpoint after the HPI system is secured.

Intuitively inherent in the probability assigned at this node is the fact that, at some point in the overcooling event, the HPI system will be secured allowing the reactor coolant system pressure to decrease below the safety valve opening setpoint pressure.

Node G

PORV(s) shuts as pressure decreases

Upward paths at this node indicate the successful reclosing of the PORV(s) when the reactor coolant system pressure decreases below the PORV opening pressure setpoint after the HPI system is secured.

Downward paths at this node indicate the failure of the PORV(s) to reclose when the reactor coolant system pressure decreases below the PORV opening pressure setpoint after the HPI system is secured.

As with the probability assigned to Node F, the probability assigned to Node G assumes that at some point in the overcooling event, the HPI system will be secured allowing the reactor coolant system pressure to decrease below the PORV opening setpoint pressure.

Each endpoint path is categorized by a consequence description as defined below:

NR - No PORV or safety valve relief occurs

RR - Relief occurs but the valve(s) recloses on demand

PVO - PORV(s) opens and fails to reclose

SVO - Safety valve(s) opens and fails to reclose

PVO/SVO - PORV(s) and safety valve(s) opens and fails to reclose.

In order to quantify the event tree paths, probability data are needed for each path at each node of the event tree. The probability data represent the answer to the question at that node. The probabilities and the reference source for the probability used for each node are given in Table A-1.

The results of the various endpoint paths are shown on Table A-2. The expected frequencies of a small-break LOCA from a stuck-open PORV or safety

valve from an overcooling initiated transient event are 6.1×10^{-6} per reactor-year and 6.9×10^{-6} per reactor year, respectively. From this, it can be concluded that overcooling events are not a significant contributor to the expected frequency of a small-break LOCA from a stuck-open PORV or safety valve for Westinghouse and Combustion Engineering-designed NSSS plants.

Table A-1. Probabilities Assigned to Overcooling Event Tree Nodes

Node	Node Description	Probability Assigned	Discussion	References
-	Initating transient event frequency	0.048/ reactor-year	Frequency was determined from events reported in Reference 1 and total Westinghouse and Combustion Engineering plant operating time from 4/1/78-4/1/80	1,3,4,5
A	Operator stops HPI prior to PORV set-point pressure	0.985	Probability was determined from Reference 6 for an operator with a moderate to high stress level	6
B	PORV block valves(s) open	0.45	Probability was based on a summary of historical operating data for Westinghouse plants as reported in Reference 7	7
C	PORV(s) open	0.99	Conservative engineering judgment coupled with information from Reference 8 for a single channel non-redundant control system	8
D	Operator stops HPI after PORV set-point before safety valve setpoint	0.999 or 0.1	Note that two probabilities are assigned to this node. The first probability, 0.999, is for the case where the PORV(s) and block valve(s) are open, making it highly unlikely that the safety valve opening setpoint pressure would ever be reached. The second probability, 0.1, is for the case where the PORV(s) or block valve(s) do not or are not open. Both	8,9

Table A-1 (Cont.)

Node	Node Description	Probability Assigned	Discussion	References
D Cont.			probabilities are based on plant and system characteristics from Reference 9 and general human error rate estimates from Reference 8.	
E	Safety valve(s) opens	0.99997	Probability was based on information from Reference 8, Volume V, Page V-38.	8
F	HPI secured, safety valve shuts as pressure descrease	0.981	Probabilty is based on a conservative engineering judgment using information from References 7 and 8 (see Section 3.2.6 of this report for more detailed discussion.)	7,8
G	PORV(s) shuts as pressure decreases	0.981	See discussion of Node F above for information.	7,8

Table A-2. Endpoint Category Description and Frequencies

Endpoint Category	Description	Frequency per Reactor-Year
NR	No PORV or safety valve relief occurs	4.7×10^{-2}
RR	Relief occurs but the valve(s) recloses on demand	6.6×10^{-4}
PVO	PORV(s) opens and fails to reclose	6.1×10^{-6}
SVO	Safety valve(s) opens and fails to reclose	6.9×10^{-6}
PVO/SVO	PORV(s) and safety valve(s) open and fail to reclose	1.2×10^{-10}

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