OCT 1 2 1983

DCS MS-016
Docket File

Docket Nos. 50-317
and 50-318

Mr. A. E. Lundvall, Jr.
Vice President - Supply
Baltimore Gas & Electric Company
P. O. Box 1475
Baltimore, Maryland 21203

Dear Mr. Lundvall:

SUBJECT: NUREG-0737 ITEMS II.K.3.1 - AUTOMATIC PORV ISOLATION AND II.K.3.2 - REPORT ON PORVs FOR CALVERT CLIFFS UNITS 1 AND 2

Item II.K.3.2 of NUREG-0737 required licensees of pressurized water reactors to submit a report to the NRC staff 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 these actions constitute sufficient improvements in reactor safety. Safety valve failure rates based on past history of the operating plants designed by the specific nuclear steam supply system (NSSS) vendor were to be included in the report. Licensees had the option of submitting either a plant specific report or a generic report. Where a generic report was submitted, each licensee was required to document the applicability of the generic report to its plant.

Based upon the results of the report submitted in response to item II.K.3.2, licensees were to assess whether an automatic PORV isolation system was required. If required, licensees were to submit a system design that uses the PORV block valve to automatically protect against a small break LOCA caused by a stuck open PORV. Documentation was to include piping, instrumentation diagrams, electrical schematics and be in conformance with IEEE 279-1971 requirements.

In response to Item II.K.3.2 the Combustion Engineering (CE) Owners Group submitted a generic report to the NRC titled "PORV Failure Reduction Methods," December 1980 (CEN-145).

Your response to the subject NUREG-0737 items dated February 20, 1981 and August 11, 1981 adopted the conclusions reached in the CE Report as applicable for your facility, namely that the concept of an automatic PORV block valve closure system, which closes the PORV isolation valves when lower pressure is sensed subsequent to a PORV failing to close, cannot be warranted on the basis of providing additional protection against a PORV LOCA."

On this basis you proposed no modifications to provide automatic isolation of the PORVs in response to Item II.K.3.1.

OFFICE ▶							
SURNAME ▶	8310310180 831012 PDR ADOCK 05000317 P PDR						
DATE ▶							

We have completed our review of your responses to the subject NUREG-0737 items including the CE Owners Group Report. Our findings are contained in the enclosed Safety Evaluation (SE) with our contractor's, Franklin Research Center's, Technical Evaluation Report (TER) attached evaluating the data contained in the CE Report Based upon our review, we find that the requirements of NUREG-0737 Item II.K.3.2 are met with the existing PORV safety valve and reactor high-pressure trip setpoints and that an automatic PORV isolation system is not required for Calvert Cliffs. This completes the staff's review of the subject NUREG-0737 items for Calvert Cliffs Units 1 and 2.

Sincerely,

Original signed by:

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosure:
Safety Evaluation with attached
Technical Evaluation Report

Docket File
NRC PDR
Local PDR
ORB#3 Rdg
DEisenhut
OELD
EJordan
DJaffe
PMKreutzer
RFerguson
NSIC
JTaylor
ACRS (10)
FRC*
JRMiller

OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL	ORB#3:DL			
SURNAME	PKreutzer	DJaffe:dd	IColburn	JRMiller			
DATE	10/7/83	10/7/83	10/11/83	10/11/83			

Baltimore Gas and Electric Company

cc:

James A. Biddison, Jr.
General Counsel
Baltimore Gas and Electric Company
P. O. Box 1475
Baltimore, MD 21203

George F. Trowbridge, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. R. C. L. Olson, Principal Engineer
Nuclear Licensing Analysis Unit
Baltimore Gas and Electric Company
Room 922 - G&E Building
P. O. Box 1475
Baltimore, MD 21203

Mr. Leon B. Russell
Plant Superintendent
Calvert Cliffs Nuclear Power Plant
Maryland Routes 2 & 4
Lusby, MD 20657

Bechtel Power Corporation
Attn: Mr. J. C. Ventura
Calvert Cliffs Project Engineer
15740 Shady Grove Road
Gaithersburg, MD 20760

Combustion Engineering, Inc.
Attn: Mr. R. R. Mills, Manager
Engineering Services
P. O. Box 500
Windsor, CT 06095

Mr. R. M. Douglass, Manager
Quality Assurance Department
Baltimore Gas & Electric Company
Fort Smallwood Road Complex
P. O. Box 1475
Baltimore, MD 21203

Mr. S. M. Davis, General Supervisor
Operations Quality Assurance
Calvert Cliffs Nuclear Power Plant
Maryland Routes 2 & 4
Lusby, MD 20657

Ms. Mary Harrison, President
Calvert County Board of County
Prince Frederick, MD 20768

U. S. Environmental Protection
Region III Office
Attn: Regional Radiation Repres
Curtis Building (Sixth Floor)
Sixth and Walnut Streets
Philadelphia, PA 19106

Mr. Ralph E. Architzel
Resident Reactor Inspector
NRC Inspection and Enforcement
P. O. Bos 437
Lusby, MD 20657

Mr. Charles B. Brinkman
Manager - Washington Nuclear Ope
Combustion Engineering, Inc.
7910 Woodmont Avenue
Bethesda, MD 20814

Mr. J. A. Tiernan, Manager
Nuclear Power Department
Calvert Cliffs Nuclear Power Pl
Maryland Routes 2 & 4
Lusby, MD 20657

Mr. W. J. Lippold, Supervisor
Nuclear Fuel Management
Baltimore Gas and Electric Compa
Calvert Cliffs Nuclear Power Pla
P. O. Box 1475
Baltimore, Maryland 21203

Mr. R. E. Denton, General Super
Training & Technical Services
Calvert Cliffs Nuclear Power Pla
Maryland Routes 2 & 4
Lusby, MD 20657

Administrator, Power Plant Sit
Energy and Coastal Zone Admini
Department of Natural Resources
Tawes State Office Building
Annapolis, MD 21204

Regional Administrator
Nuclear Regulatory Commission,
Office of Executive Director f
631 Park Avenue
King of Prussia, Pennsylvania

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

SAFETY EVALUATION OF
COMBUSTION ENGINEERING LICENSEES' RESPONSES
TO NUREG-0737 ITEM II.K.3.2

INTRODUCTION

According to NUREG-0737 Item II.K.3.2, the licensees were required to perform the following actions:

(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 (SBLOCA) caused by a stuck-open power-operated relief valve (PORV) and show how those actions constitute sufficient improvements in reactor safety.

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

The requirements of NUREG-0737 allowed each licensee the option of preparing and submitting either a plant-specific or a generic report. If a generic report were submitted, each licensee was to have documented the applicability of the generic report to his plant. All CE licensees referenced a CE report (CEN-145) prepared by the CE Owners Group to address the staff's concerns. Licensees asserted that CEN-145 was applicable to their plants but did not provide any supporting documentation. The CE report claims that the requirements of NUREG-0737 Item II.K.3.2 are met with the existing PORV, SV and high-pressure reactor trip setpoints, and that no automatic PORV isolation system is required for CE plants. Therefore, our review, which was mainly based on the technical evaluation performed by our contractor, Franklin Research Center (FRC), was concentrated in two areas, namely, the adequacy of the CE report, and its applicability to any CE plant. Our review included the effects of plant-specific data reflecting the post-TMI improvements. The data was obtained through the project managers, who obtained the information from the licensees. Our contractor's review is contained in the attached Technical Evaluation Report (TER).

REVIEW

A. CONTENTS OF CEN-145

The CE report considered a spectrum of initiating events that may lead to PORV/SV opening. The fault tree methodology was utilized to estimate the SBLOCA frequency due to a stuck-open PORV (SBLOCA-PORV frequency).

The initiating event frequencies were based on the operating experience of CE plants, and the estimate of SBLOCA-PORV/SV frequency was obtained from the frequencies of the initiating events, the failure probabilities of PORVs and block valves, and the probability of operator failure to close the block valves.

In addition, the CE report considered various methods for reducing the SBLOCA-PORV frequency. Among them, elimination of the turbine runback feature was considered to be the most effective in reducing PORV challenges without adverse impact on plant operation. In addition, CE assessed the impact of other countermeasures to reduce the SBLOCA-PORV frequency. These countermeasures included improving operator capability and automatic closure of the PORV.

B. ADEQUACY OF CEN-145

Based on our review, we find that the fault tree methodology used in the CE report is a valid approach to estimating the SBLOCA-PORV frequency. In general, we find that the probabilistic data in the CE fault tree appear reasonable; however, we recognize that there are inherent uncertainties in the CE analysis. The result of the CE analysis indicates that the SBLOCA-PORV frequency, with credit for operator action, is about 1.8×10^{-3}/reactor year, after being reduced by a factor of about 15 due to the elimination of the turbine runback feature and the provision of direct indication of PORV position. As discussed in the attached TER, FRC has performed calculations and verified these estimates, given the validity of the CE data. However, considerations such as the following were not included in the CE analysis:

(1) Manual Actuation of PORV

The CE analysis does not consider manual actuation of PORV. However, there are instances in which manual actuation of PORV may be needed as discussed below:

(i) Venting of Noncondensible Gases

An operator may use a PORV to vent the noncondensable gases in the pressurizer. For example, an operator may open a PORV to vent the noble gases that have leaked from the fuel into the primary coolant.

(ii) Depressurizing the Primary System

To depressurize the primary system, an emergency procedure may require an operator to cycle a PORV several times. For example, during recovery of a steam generator tube rupture event, an operator may use a PORV to depressurize the primary system to minimize leakage to the secondary system.

We note that the operator error for failing to close a block valve, given a stuck-open PORV, is less likely when the PORV has been opened manually than when it is opened automatically.

The approach of the staff implicitly took these considerations into account, since it was based on operating data.

(2) PORV Block Valve Availability

The CE plants have operated with the PORV block valves shut to minimize valve leakage. If a plant operates with PORVs blocked off, its SBLOCA-PORV frequency would be greatly reduced but at the expense of having a higher SBLOCA-SV frequency which may lead to more adverse consequences.

(3) Overcooling Transients

The CE analysis does not consider the challenges to the PORV/SVs due to the actuation of the high-head safety injection system during recovery from overcooling transients such as the overfeeding of a steam generator. As discussed in the attached TER, FRC has estimated the SBLOCA-PORV/SV frequency due to overcooling transients.

C. APPLICABILITY OF CEN-145

To ascertain that the generic CE report applies to a specific CE plant, we need the plant-specific information such as the PORV/SV challenge frequencies, the fraction of the time the PORV block valves are closed, and the various post-TMI modifications that may have reduced the PORV/SV challenge frequencies. Because the various post-TMI modifications may have reduced the PORV/SV challenge frequencies, the operational data on PORV/SV challenge frequencies in the time interval before the post-TMI modifications were imposed is not directly applicable to the prediction of future challenge frequencies. The PORV/SV operational data is available because NUREG-0737 Item II.K.3.3, "Reporting SV and RV Failures and Challenges", requires that all PWR licensees promptly notify NRC of the PORV/SV failures and periodically report the PORV/SV challenges in annual or monthly reports beginning April 1, 1980. This requirement to report PORV/SV operational data was imposed because, prior to the TMI accident, there was insufficient data to portray accurately the operational PORV/SV failures and challenges.

The project managers for the various CE plants have supplied us with the PORV/SV operational data for the period from April 1, 1980 to March 31, 1983. We have utilized this more recent operational data, together with the operational data given in the CE report, to estimate SBLOCA-PORV/SV frequencies.

(1) Estimate of SBLOCA-PORV Frequency

According to the data given us, there were no PORV/SV challenges in the 3-year period (April 1, 1980 to March 31, 1983) for many of the CE plants listed in Table 1. The maximum number of PORV challenges to any of the plants was 4 in the 3-year period. We make the conservative assumption that the plants with the high numbers of PORV challenges have such a high number because of plant-specific difference, and not because of random statistical fluctuation in the frequency of challenges. If we use 4 PORV challenges in 3 years, then the upper 95% confidence limit on the PORV challenge frequency is about 3.1/reactor-year. Moreover, assuming (i) that the PORVs are not isolated, (ii) the PORV failure probability is 2×10^{-2}/demand, and (iii) the operator error probability in not isolating a stuck-open PORV is conservatively estimated

TABLE 1

PORV/SV CHALLENGES IN CE PLANTS FROM
APRIL 1, 1980 TO MARCH 31, 1983

PLANT	Number of PORV Challenges	Number of SV Challenges
ANO-2[1]	0	0
Calvert Cliffs-1	0	0
Calvert Cliffs-2	3	0
Fort Calhoun	0	0
Maine Yankee	1	0
Millstone-2	4	0
Palisades	0	0
St. Lucie	4	2

Notes: [1]ANO-2 has no PORV.

at $5x10^{-2}$/demand, we estimate that the SBLOCA-PORV frequency is about $3.1x10^{-3}$/reactor-year which still remains within the range of the SBLOCA frequencies given in WASH-1400[3] (10^{-2} to 10^{-4} per reactor-year). We believe that our estimate of SBLOCA-PORV frequency is conservative because the use of 4 PORV challenges in 3 years is the maximum for any of the plants, and because the 95% confidence limit is used for estimating the PORV challenge frequency. Moreover, depending on the fraction of time that PORVs are actually blocked off due to leakage, the PORV challenge frequency would be somewhat less.

(2) Estimate of SBLOCA-SV Frequency

Based on its survey, CEN-145 indicates that there were no SV challenges during the period of operation of the CE plants covered in the survey. Since the publication of CEN-145, St. Lucie experienced an overpressure event on December 19, 1981 (Licensee Event Report 355-81-56) due to an inexplicable closure of MSIVs at near full power causing PORVs to lift, and a SV to lift once below its setpoint and another time near the operating pressure of the pressurizer. We estimate the SBLOCA-SV frequency based on the following:

(i) The SV failure probability is 10^{-2}/demand according to WASH-1400 and the recent IREP study on ANO-1.[4]

(ii) There were 2 SV challenges during the 3-year period among 8 CE plants.

The SBLOCA-SV frequency based on the 95% upper confidence limit on the SV challenge frequency is then estimated to be about 2.5×10^{-3}/reactor-year, which falls within the range of the SBLOCA frequencies given in WASH-1400 (10^{-2} to 10^{-4} per reactor-year).

D. PORV Leakage Problem

Our review indicates that many CE plants operate with PORVs blocked off a substantial fraction of the time. The intentional blocking of PORVs is done to eliminate PORV leakage and to ensure that the reactor coolant system (RCS) leakage does not exceed the technical specification limit. Since there are many CE plants which have blocked off PORVs, it may imply either that PORVs need to be modified to correct the leakage problem or that there should be some maintenance or repair work on PORVs on a periodic basis. A plant that operates with PORVs blocked off may depend on SVs to relieve pressure. Considering the fact that the SV capacity is much larger than the PORV capacity, and there is no block valve to terminate a SV release, the consequences of a stuck-open SV may be more severe than those of a stuck-open PORV. In addition, if PORVs are not blocked off, they supply additional pressure relieving capacity in an ATWS (anticipated transient without scram) event. The NRC staff is considering the need for imposing a technical specification limit on the amount of time a plant can operate with PORVs blocked. The need for upgrading the reliability of PORVs is a proposed generic issue (see the memorandum from D. DiIanni on the subject, "Proposed Generic Issue - PORV and Block Valve Reliability"[5]).

CONCLUSION

Based on the review of the licensees' responses, we concur, for the licensees given in Table 1, with the licensees' conclusions that the requirements of NUREG-0737 Item II.K.3.2 are met with the existing PORV, SV and high-pressure reactor trip setpoints, and that the automatic PORV isolation system is not required.

Attachment: FRC Technical
Evaluation Report

Principal Contributer

Ed Chow, DST

REFERENCE

1. CEN-145, "PORV Failure Reduction Methods," December 1980.

2. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3. WASH-1400, "Reactor Safety Study," October 1975.

4. NUREG/CR-2787, "Interim Reliability Evaluation Program: Analysis of the Arkansas Nuclear One-Unit 1 Nuclear Power Plant," June 1982.

5. Memorandum dated June 6, 1983 from D. DiIanni for W. Minners through R. Clark, "Proposed Generic Issue - PORV and Block Valve Reliability."

TECHNICAL EVALUATION REPORT

OPERATING REACTOR PORV REPORTS (F-37)

TMI ACTION PLAN REQUIREMENTS

COMBUSTION ENGINEERING OWNERS GROUP (CEN-145)

NRC DOCKET NO. Various

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 7

FRC TASK 409

Prepared by

Franklin Research Center
20th and Race Streets
Philadelphia, PA 19103

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Author: G. J. Overbeck
S. M. Jenkins
T. J. DelGaizo

FRC Group Leader: G. J. Overbeck

Lead NRC Engineer: E. Chow

July 18, 1983

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Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

CONTENTS

TECHNICAL EVALUATION REPORT

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Prepared by

Franklin Research Center
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Philadelphia, PA 19103

Author: G. J. Overbeck
S. M. Jenkins
T. J. DelGaizo

FRC Group Leader: G. J. Overbeck

Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: E. Chow

July 18, 1983

Prepared by:	Reviewed by:	Approved by:
T. J. DelGaizo	G. J. Overbeck	D. P. Carfagno
Principal Author	Group Leader	Department Director
Date: 7-15-83	Date: 7-15-83	Date: 7-18-83

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

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 Units," 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 ...n Developed as a Result of the TMI-2 Accident." The first recommendation

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.

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

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,'

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:

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

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

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

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.

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

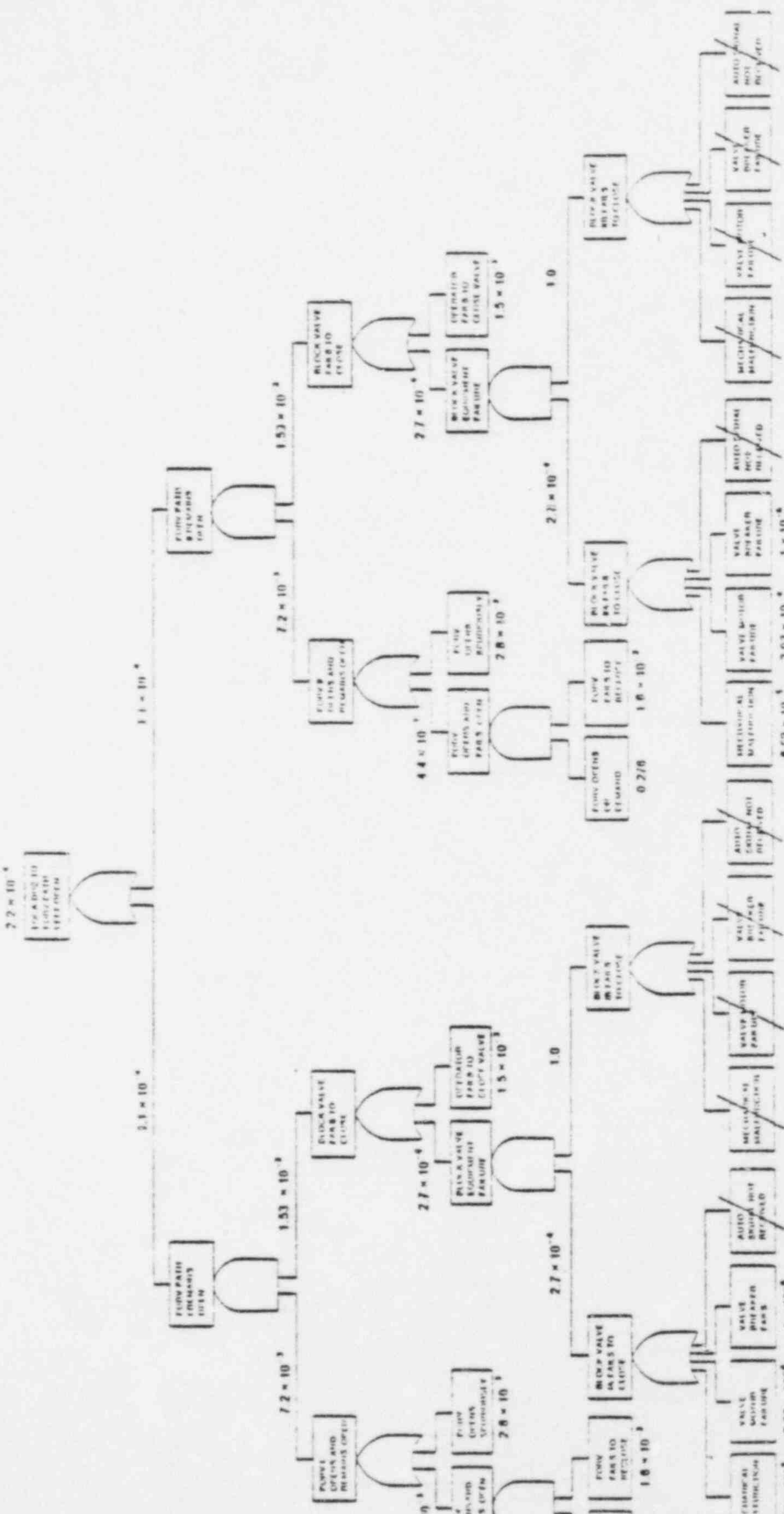

Figure 2. Combustion Engineering Basic Fault Tree, Revised

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 setpoint 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:

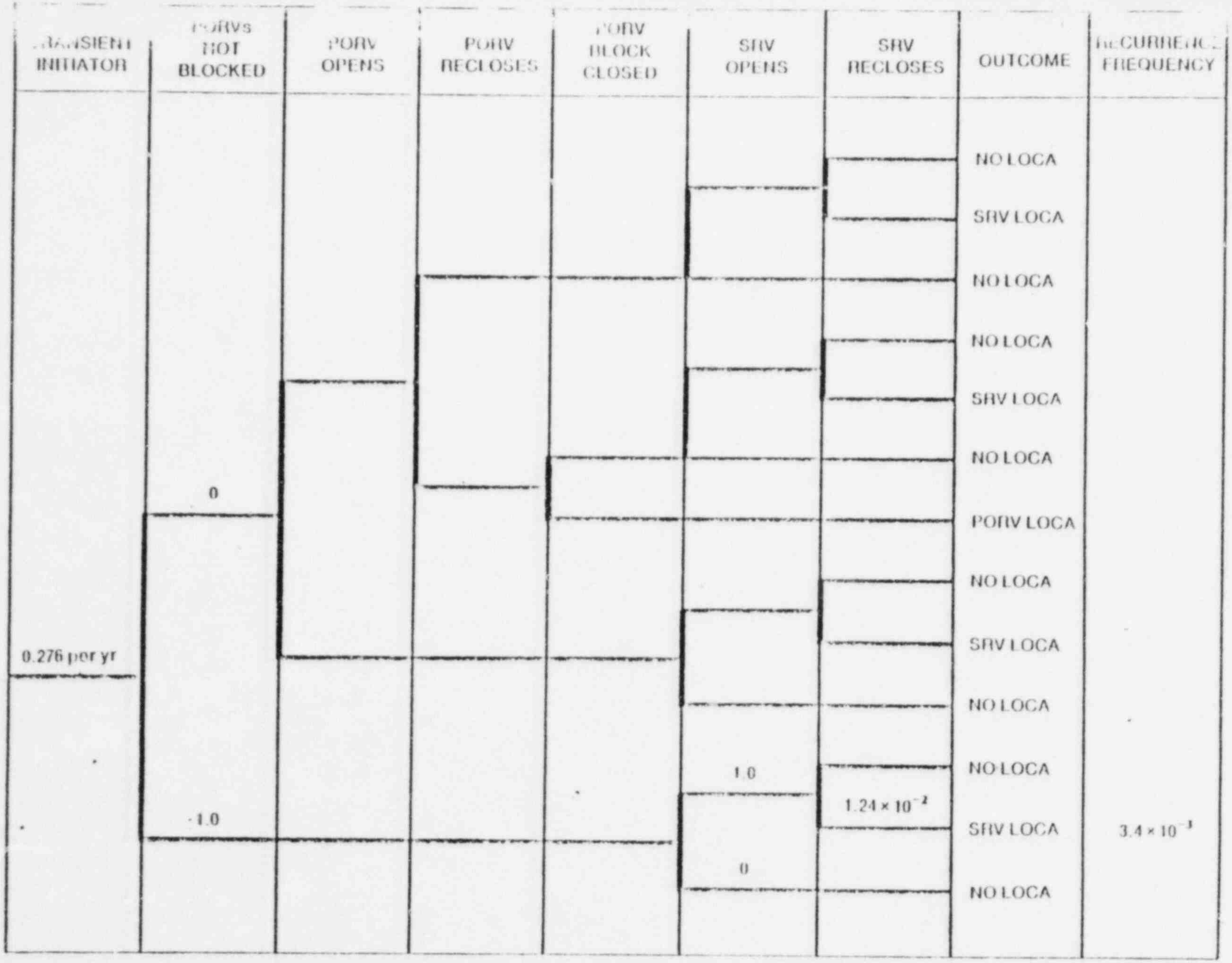

Figure 4. Small-Break LOCA from Stuck-Open SRV
(Palisades and ANO-2)

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.

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.

5.3.3 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.

4.1.2 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.

*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.

6. REFERENCES

1. "Clarification of TMI Action Plan Requirements"
   NRC Office of Nuclear Reactor Regulation, November 1980
   NUREG-0737

2. "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in CE Designed Operating Plants"
   NRC Office of Nuclear Reactor Regulation, January 1980
   NUREG-0635

3. "NRC Action Plan Developed as a Result of the TMI-2 Accident"
   NRC Office of the Executive Director for Operations, May 1980
   NUREG-0660

4. Typical Letter Endorsing CEN-145
   A. E. Lundvall (BG&E)
   Letter to D. G. Eisenhut (NRC)
   Subject: Calvert Cliffs Nuclear Power Plant Units 1 and 2 Response to NUREG-0737
   February 20, 1981

5. "PORV Failure Reduction Methods"
   CE, Inc., Windsor, Connecticut, December 1980
   CEN-145

6. R. A. Clark (NRC)
   Letter to A. E. Lundvall (BG&E)
   Subject: Request for Additional Information in Regard to NUREG-0737 Action Item II.K.3.2 for the Calvert Cliffs Nuclear Power Plant Units Nos. 1 and 2
   January 20, 1982

7. A. E. Lundvall (BG&E)
   Letter D. G. Eisenhut (NRC)
   Subject: Calvert Cliffs Nuclear Power Plant Units 1 and 2 Response to NUREG-0737 Items II.K.3.2 and II.K.3.17
   April 26, 1982

8. A. E. Lundvall (BG&E)
   Letter R. A. Clark (NRC)
   Subject: Calvert Cliffs Nuclear Power Plant Units 1 and 2 Response to NUREG-0737 Items II.K.3.2 and II.K.3.17
   June 7, 1982

9. J. N. Donohew, Jr. (NRC)
   Letter to S. P. Carfagno (FRC)
   Subject: Contract No. NRC-03-81-130, Tentative Work Assignment F
   July 21, 1981

APPENDIX A

Franklin Research Center
A Division of The Franklin Institute
The Benjamin Franklin Parkway, Phila., Pa. 19103 (215) 448-1000

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:

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

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.

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

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-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}