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Docket Nos. 50-269,50-270
and 50-287

Mr. Hal B. Tucker
Vice President - Nuclear Production
Duke Power Company
P. O. Box 33189
422 South Church Street
Charlotte, North Carolina 28242

Dear Mr. Tucker:

SUBJECT: NUREG-0737 ITEMS II.K.3.1 - AUTOMATIC PORV ISOLATION AND II.K.3.2 - REPORT ON PORVs FOR OCONEE NUCLEAR STATION

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 Babcock and Wilcox Owners Group submitted a generic report to the NRC staff titled "Report on Power Operated Relief Valve Opening Probability and Justification for Present System and Setpoints," December 1980, Babcock and Wilcox 12-1122779.

Your response to the subject NUREG-0737 items dated January 2, 1981 adopted the conclusions reached in the B&W Report as applicable for your facilities 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."

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

We have completed our review of your responses to the subject NUREG-0737 items including the B&W 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 B&W 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 Oconee Nuclear Station. This completes the staff's review of the subject NUREG-0737 items for your facilities.

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosure:
Safety Evaluation with attached
Technical Evaluation Report

cc w/enclosure:
See next page

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Duke Power Company

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION OF B&W 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 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.

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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. However, all B&W licensees referenced a B&W report prepared by the B&W Owners Group to address the staff's concerns. Licensees asserted that the report was applicable to their plants but did not provide any supporting documentation. The B&W 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 B&W plants. Therefore, the staff's 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 B&W report, and its applicability to any B&W plant. The staff review included the effects of plant-specific data reflecting the post-TMI improvements. This 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 B&W REPORT

The B&W report considered the following five categories of PORV opening events in estimating the PORV challenge frequency:

1. PORV opening on an overpressure transient.
2. PORV opening on a transient with delayed auxiliary feedwater.

3. PORV opening due to operator action under abnormal transient operating guidelines (ATOG).
4. PORV opening due to instrumentation and control faults.
5. PORV opening on recovery from overcooling transients.

The PORV challenge frequency due to an overpressure transient was first estimated using a Monte-Carlo approach based on the variation of pressure overshoot in an overpressure transient, and on the variations of setpoints for opening the PORV and for the reactor high-pressure trip. The PORV challenge frequencies due to the other four categories were estimated mainly from generic data. The B&W analysis concluded that the post-TMI PORV challenge frequency from all causes is about 2.4×10^{-2}/reactor-year. This value for the PORV challenge frequency is the value after the post-TMI plant modifications, which included modification to the high reactor pressure trip point and the pressurizer PORV opening setpoint, were made; the PORV challenge frequency was much higher before these modifications were made. The staff estimate of the PORV challenge frequency, given in section C, was based on post-TMI data.

To estimate the PORV failure probability, B&W also relied on the operating data from B&W and concluded that the PORV failure probability was about 2×10^{-2}/demand. This failure probability was estimated from five PORV failures to reseat in 250 estimated demands; B&W estimated the number of demands from 148 documented demands on reactor trips. The staff believes that the use of 250 demands instead of 148 demands in the calculation of the PORV failure probability appears reasonable because there may be multiple PORV openings in the reactor trip events, and there may also exist other undocumented PORV openings in overpressure transients that did not lead to reactor trips. Combining the PORV challenge frequency from all causes with the PORV failure probability, B&W estimated that the SBLOCA frequency due to a stuck-open PORV (SBLOCA-PORV frequency) was about 4.8×10^{-4}/reactor-year.

B. ADEQUACY OF B&W REPORT

Based on our review, the staff believes that there has been a substantial reduction in the PORV challenge frequency; this is primarily due to the changing of the setpoints for the opening of the PORV and for the reactor high-pressure trip. Prior to the TMI accident, the PORV opened before the reactor high-pressure trip during an overpressure transient. With the new setpoints, the reactor high-pressure trip occurs at 150 psig below the PORV setpoint, thereby making the lifting of the PORV much less likely.

In general, the staff finds that the probablistic data in the B&W report appear reasonable; however, it is recognized that there are inherent uncertainties in the B&W analysis. Moreover, the staff believes that the estimates for SBLOCA-PORV/SV frequencies may be higher as a consequence of the considerations such as the following that are not included in the B&W analysis:

(1) Manual Actuation of PORV

The B&W analysis considers manual actuation of the PORV only for the recovery from steam generator tube rupture events. However, there are other 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 noncondensible 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. Manual cycling of a PORV during accident conditions might be necessary if an operator wants to reduce primary system pressure in order to initiate safety injection.

The staff notes 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.

(2) PORV Block Valve Availability

A number of B&W plants have operated with the PORV block valve shut to minimize valve leakage. If a plant operates with a PORV 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.

C. APPLICABILITY OF B&W REPORT

To ascertain that the generic B&W report applies to a specific B&W plant, we need plant-specific information such as the PORV/SV challenge frequencies and the fraction of time the PORV block valve is closed. 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 useful for 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 the 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 B&W plants have supplied the PORV/SV operational data for the period from April 1, 1980 to March 31, 1983. The staff has utilized this more recent operational data together with the previous data given in the B&W report to estimate SBLOCA-PORV/SV frequencies.

(1) Estimate of SBLOCA-PORV Frequency

According to the data given, there were no PORV challenges in the 3-year period (April 1, 1980 to March 31, 1983) for all of the B&W plants listed in Table 1, except for ANO-1, where there was one PORV challenge during the 3-year period. If one PORV challenge in 3 years is used, the upper 95% confidence limit on the PORV challenge frequency is about 1.6/reactor-year. Moreover, assuming that (i) the PORV is 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 at 5×10^{-2}/demand, the staff estimates that the SBLOCA-PORV frequency is about 1.6×10^{-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). The staff believes that its estimate of SBLOCA-PORV frequency is conservative because the operational data from ANO-1 is bounding for the other 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, the B&W report indicates that there were 3 SV challenges during the operation of the B&W plants. None of these events resulted in failure of a SV to reseat. Two of these SV lifting events were related to failures of non-nuclear instrumentation

TABLE 1

PORV/SV CHALLENGES IN B&W PLANTS FROM
APRIL 1, 1980 TO MARCH 31, 1983

PLANT	Number of PORV Challenge [1]
ANO-1	1[2]
Crystal River	0
Davis Besse	0
Oconee 1/2/3	0
Rancho Seco	0
TMI-1[3]	0

Notes: [1]No SV Challenges were known for the plants during the time period.

[2]A loss-of-offsite power event occurred at ANO-1 on April 7, 1980, and the operator opened the PORV.

[3]TMI-1 has not operated in the time period.

(NNI) power supplies; the lifting of SVs on failure of NNI power supplies is less likely now because of changes to NNI power supplies. The third event was lifting of a SV on a loss of load test. However, because of the changes made after the Three Mile Island accident, the reactor high-pressure trip occurs at about 2300 psi while the SVs still lift at about 2500 psi. The probability of challenging a SV on a high-pressure reactor trip is only about 9×10^{-6}, with the new setpoints, according to FRC analysis. The staff has therefore determined that the pre-TMI experience on SV liftings is not applicable. Accordingly, the staff has used only the experience for the 3-year period from April 1, 1980 to March 31, 1983. There were no SV challenges in the 3-year period (April 1, 1980 to March 31, 1983) for all of the B&W plants listed in Table 1.

The staff estimates the SBLOCA-SV frequency as follows:

(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 no SV challenges in about 20 reactor-years of operation of B&W plants.

The 95% upper confidence limit on the SBLOCA-SV frequency is then estimated to be about 1.5×10^{-3}/reactor-year, which falls with the range of SBLOCA frequencies given in WASH-1400 (10^{-2} to 10^{-4} per reactor-year). The FRC estimate of SBLOCA-SV frequency was 4.7x10-3/reactor-year. The FRC estimate did not include consideration of post-Crystal River-3 event modifications.

D. PORV Leakage Problem

The staff's review indicates that many B&W plants operate with a blocked-off PORV. The intentional blocking of a PORV 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 B&W 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 a PORV blocked off may depend on SVs to relieve pressure. Considering that

(1) there is no block valve to terminate a SV release, and

(2) the SV capacity is usually much larger than the PORV capacity,

the consequences of a stuck-open SV may be more severe that those of a stuck-open PORV. In addition, if a PORV is not blocked off, it will 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, the staff concurs, for the licensees given in Table 1, with the licensees' conclusions that the requirements of NUREG-0737 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

Dated: November 22, 1983

The following NRC staff personnel have contributed to this Safety Evaluation: E. Chow.

TECHNICAL EVALUATION REPORT

OPERATING REACTOR PORV REPORTS (F-37)

TMI ACTION PLAN REQUIREMENTS

GENERIC REPORT -- BABCOCK & WILCOX DESIGNED UNITS

NRC DOCKET NO. Various

NRC CONTRACT NO. NRC-03-81-130

FRC PROJECT C5506

FRC ASSIGNMENT 7

FRC TASK 410

Prepared by

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FRC Group Leader: G. J. Overbeck

Prepared for

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Lead NRC Engineer: E. Chow

July 20, 1983

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TECHNICAL EVALUATION REPORT

OPERATING REACTOR PORV REPORTS (F-37)

TMI ACTION PLAN REQUIREMENTS

GENERIC REPORT -- BABCOCK & WILCOX DESIGNED UNITS

NRC DOCKET NO. Various

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FRC PROJECT C5506

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FRC Group Leader: G. J. Overbeck

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July 20, 1983

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Principal Author	Project Manager	Department Director
Date: 7/20/83	Date: 7/20/83	Date: 7-20-83

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CONTENTS

CONTENTS (Cont.)

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 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 a generic report of Babcock & Wilcox (B&W) designed units prepared 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," and Item II.K.3.7, "Evaluation of Power Operated Relief Valve Opening Probability During Overpressure Transient." This evaluation was performed with the following objectives:

- o to ensure that the B&W Report is complete and properly documents the information required by NUREG-0737, Items II.K.3.2 and II.K.3.7
- o to ensure that the estimated probabilities of the B&W Report satisfy the review criteria.

1.2 GENERIC BACKGROUND

In NUREG-0565 [2], "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants," the Nuclear Regulatory Commission's (NRC) Bulletins and Orders Task Force recommended the following:

- o "Provide a system which will assure that the block valve protects against a stuck-open PORV. This system will cause the block valve to close when RCS pressure has decreased to some value below the pressure at which the PORV should have reseated. This system should incorporate an override feature. Each licensee should perform a confirmatory test of the automatic block valve closure system.
- o In order to minimize the opening of the PORV, most overpressure transients should not result in the PORV opening. Licensees of B&W plants should document that the PORV will only open in less than five percent of all anticipated overpressure transients using the revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle.
- o All failures of PORVs to reclose should be reported promptly to the NRC. All challenges should be reported in annual reports.

- Licensees should submit a report to the NRC which discusses the safety valve failure rate experienced in B&W operating plants.
- All failures of safety valves to reclose should be reported promptly to the NRC. All challenges should be reported in annual reports."

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 remaining recommendations were included in Items II.K.3.2 and II.K.3.7. 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."

With regard to Item II.K.3.7, NUREG-0737 stated:

"Based on its review of best-estimate calculations performed by Babcock and Wilcox (B&W), the NRC staff believes that the frequency of PORV challenges has been reduced using the revised PORV and high-pressure reactor trip setpoints and assuming that the anticipatory reactor trips function as designed. At this time, however, the staff is unable to make a quantitative judgment of the expected frequency. Therefore, licensees with B&W-designed plants should perform additional analyses of anticipated transients which indicate the sensitivity of PORV challenges to (1) the variation in core physics parameters which may occur in the plant cycle; (2) single failures in mitigating systems; and (3) transients which do not actuate the anticipatory reactor trips. Analytical assumptions should include those specified in the plant final safety analysis reports (FSARs). The results of these more-detailed and extensive analyses should be used to determine the expected frequency of PORV openings for overpressure transients. This frequency should be less than 5% of the total number of overpressure transients, thereby confirming the findings of the staff's review.

The results of this study should be documented and submitted for staff review by the scheduled date."

1.3 PLANT-SPECIFIC BACKGROUND

In response to NUREG-0737, Items II.K.3.1, II.K.3.2, and II.K.3.7, licensees of B&W-designed plants endorsed and submitted to the NRC a report entitled "Report on Power Operated Relief Valve Opening Probability" [4]. A preliminary review of the report resulted in the NRC's sending a request for additional information (RAI) to one of the licensees on December 16, 1981 [5]. The Licensee responded to the RAI in a letter to the NRC dated October 22, 1982 [6]. This TER evaluates the information provided in References 4 and 6, 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 B&W response to NUREG-0737, Items II.K.3.2 and II.K.3.7, was evaluated against the acceptance criteria provided by the NRC in a letter dated July 21, 1981 [7], which outlined Tentative Work Assignment F. Specifically, the response to NUREG-0737, Item II.K.3.2 was 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."

On July 26, 1982 [8] and September 17, 1982 [9], the NRC clarified that the probability of a small-break LOCA due to a stuck-open PORV or safety valve did not necessarily have to be less than 10^{-3} per reactor-year. Instead, a comparison of pre-TMI and post-TMI data should demonstrate that plant modifications have reduced the probability of a small-break LOCA due to a stuck-open PORV or safety valve and that this reduction should be sufficient to approach the WASH-1400 [10] median probability of a small-break LOCA S_2 with a break diameter between 0.5 and 2.0 in.

The response to NUREG-0737, Item II.K.3.7 was to contain the following information:

"1. Licensees with B&W-designed plants shall do an analysis of anticipated overpressure transients which give the sensitivity of PORV challenges to the following:

   a. variation of core physics parameters during plant cycle;

   b. single failures in mitigating systems;

   c. transients which do not actuate the anticipatory reactor trips.

2. The expected frequency of PORV openings for overpressure transients shall be less than 5% of the total number of expected overpressure transients.

3. Analytical assumptions shall include those specified in the plant FSAR."

3. TECHNICAL EVALUATION

3.1 REVIEW OF THE B&W REPORT FOR COMPLETENESS

In Reference 4, B&W approached the question of PORV reliability by first determining the probability of the PORV opening and combining it with the probability of the valve failing to close. The PORV opening probability was determined by two methods, an analytical approach and an operating data approach. To determine the frequency by which a PORV will fail to close on demand, a combination of operating data and analysis was employed. B&W determined that the probability of a small-break LOCA from a stuck-open PORV was 5.04×10^{-4} per reactor-year, by the analytical method, and 4.7×10^{-4} per reactor-year, based on operating data.

B&W concluded the report of Reference 4 by saying:

> "Both the analytical prediction and the estimate based on historical data result in values for a stuck-open PORV from all causes which meet the requirements given in II.K.3.2. Note that no credit had been assigned for the operator closing the block valve given an open PORV. Analytical predictions (given proper auxiliary feedwater response) result in a value less than .01% of PORV openings for overpressure transients (taking into account the most limiting non-anticipatory trips) and historical data shows the frequency to be less than 1.6% which satisfies the criterion (less than 5%) specified in II.K.3.7.
>
> Since the requirements of II.K.3.7 and II.K.3.2 are met with the current PORV configurations and set point, it is not necessary to address the requirement for an automatic block valve closure system per II.K.3.1."

With regard to the pressurizer safety valves, it was stated in Reference 4 that no estimate was made of failure rates because there were so few data points. In Reference 6, however, the frequency of a small-break LOCA due to a stuck-open safety valve was estimated to be 2.6×10^{-5} per year.

3.1.1 Small-Break LOCA Probability Calculations

The first method used in Reference 4 to determine a recurrence frequency of a small-break LOCA from a stuck-open PORV involved determining the PORV opening probability, based upon analysis, and then multiplying by the probability that the PORV fails to close.

3.1.1.1 PORV Opening Frequency

In order to analytically determine a PORV opening frequency, B&W first determined a PORV opening frequency during a loss-of-feedwater (LOFW) or turbine trip event to be 3.9×10^{-6} per reactor-year, with a PORV setpoint of 2450 psig. This frequency was based on the assumption that the high pressure trip setpoint was 2300 psig with a standard deviation of 1.4 psi and that the actual setpoint at which reactor trip occurred was a random variable which is normally distributed. The small standard deviation was based on the fact that the PORV and reactor protection system (RPS) actuation points are not completely independent in that they share a common source, i.e., sensor and instrument string. Thus, parts of the string errors were perfectly correlated and cancelled one another in the analysis. Other parts of the relevant string error were not correlated, and it was upon these that the 1.4 psi standard deviations were based.

In Reference 6, further justification was provided for using such a small standard deviation:

> "The difference between the setpoints for the high pressure trip and the PORV actuation is of interest and one contribution to this difference is due to electronic module accuracies. Accuracy of individual modules were obtained from the manufacturer (BMCo) and are .1% of range. The range of interest is approximately 1000 psi resulting in a value of .001 x 1000 or 1 psi. The standard deviation is derived as $\sum_{i=1}^{n} [(\text{Accuracy}) \times (\text{Range})]^2$. Both the pressure trip and the PORV share common modules that need not be included in this assessment as errors will cancel out (e.g., if module error is high then both the trip and the PORV are high but the difference is not affected). There are four non-common modules in these two strings, a bistable in the RPS channel and a buffer amp (from either RC3A-PT1 or RC3B-PT1), an inverter (RC 3 PIC) and a H/L monitor (RC 3-P58) in the PORV string. The SD is therefore $\sqrt{1 \text{ psi} + 1 \text{ psi} + 1 \text{ psi} + 1 \text{ psi}}$ or 2 psi. The reference incorrectly used 1.4 as the standard deviation. Note however that the standard deviation of the overall calculation $\sqrt{(2)^2 + (2)^2 + (27.52)^2}$ is dominated by the third term which is associated with the rollover data. In fact the module accuracies can be as large as 10 psi without impacting the standard deviation."

With regard to the trip setpoint of the PORV, a similar analysis was undertaken, which also assumed this trip to be a normally distributed random variable. The assumption of normality for the actuation of either the high pressure trip or the PORV was just an assumption; no data are available to justify or deny the validity. The RCS pressure rise above the RPS high pressure trip setpoint (referred to as "pressure rollover") during a LOFW or turbine trip was determined by a combination of plant data and engineering analysis. Pressure rollover data from the operating plants were compiled from available data. However, these data points represented situations in which the PORV could open, thus decreasing the amount of pressure overshoot. Therefore, it was necessary to correct for the PORV opening, since the situation of concern was when the PORV remained closed. This was accomplished by benchmarking the CADD code to a transient in which the PORV was isolated. After satisfactory duplication of this transient, the code was rerun, modeling proper functioning of the PORV. The resulting pressure correction to the rollover data was 17.4 psi. The rollover data were tested and were statistically acceptable as normally distributed, with a mean of 9.2 and a standard deviation of 27.52 psi.

Using the above data and assumptions, a Monte Carlo simulation of the relation

$$\text{PORV} - \text{RPS} - \text{EXCESS} - \text{BIAS} = \text{SAMPLE}$$

was conducted. The terms in the above relation are defined as follows:

PORV = PORV setpoint, a normally distributed random variable

RPS = High pressure trip setpoint, also a normally distributed random variable

EXCESS = Pressure rollover, a randomly distributed normal variable

BIAS = A constant (17.4 psi) defined by analysis which compensates the rollover data for the fact that the PORV will remain closed.

Six thousand sample values of the above alogrithm expression were calculated using the SAMPLE code. A negative value of the above expression implied that the PORV opened. In the computer trials, no negative values in 6000 instances were observed.

It was then assumed that the random variables described above were independent in the probabilistic sense, so an analytic approach was applied. The sum or difference of several independent normal distributions is also a normal distribution with mean equal to the algebraic sum of the means and with standard deviation equal to the square root of the sum of variances. In this case, the mean was 2450 - 2300 - 9.23 - 17.4 = 123.37 and the standard deviation was 27.59. The frequency by which the PORV will open during overpressure transients, under those conditions, was calculated to be 3.9×10^{-6} per reactor-year. In Reference 6, it was noted that this frequency was incorrectly stated as 3.9×10^{-6} events per reactor-year rather than 3.9×10^{-6} events per overpressure transient. Estimating 10 transients per year, it was determined in Reference 6 that the frequency of PORV openings during overpressure transients was 3.9×10^{-5} events per reactor-year.

In order to determine the frequency by which the PORV will open due to any cause, the above calculated frequency of opening during overpressure transients had to be added to other possible initiating events. B&W grouped PORV opening events into five categories with annual probabilities as follows:

		Per Reactor-year
1.	PORV opening on overpressure transient	3.9×10^{-6}
2.	PORV opening on transient with delayed auxiliary feed	1.4×10^{-3}
3.	PORV opening on operator action under ATOG guidelines	1.58×10^{-2}
4.	PORV opening due to instrumentation control faults	5×10^{-3}
5.	PORV opening from additional consideration from II.K.3.7	1.8×10^{-3}
	Total	2.40×10^{-2}

In Reference 4, brief discussions were provided regarding the derivation of each of the above probabilities. In Reference 6, the additional information was submitted in specific areas in response to NRC questions. A brief summary of these derivations follows:

Transient with Delayed Auxiliary Feed

The value of 1.4×10^{-3} per reactor-year was determined by combining EPRI data for loss of main feedwater and loss of offsite power (LOOP) with B&W auxiliary feedwater unavailability. It was assumed that if auxiliary feedwater was unavailable, the PORV would open (i.e., probability of 1.0).

In Reference 6, a revised value of 7.6×10^{-4} per reactor-year was determined by updating the analysis with data obtained since Reference 4 was completed.

Operation Action Under ATOG Guidelines

The demand on the PORV, given a steam generator tube rupture, depends upon the availability of offsite power. If power is available, only one PORV opening is required. If offsite power is not available, as many as 23 PORV openings are required. However, since there is no causal connection between steam generator tube rupture and LOOP, the WASH-1400 value of LOOP (1×10^{-3}) was used.

Since there have been no tube ruptures in B&W experience, a chi-square 50% confidence value with zero failures of 1.54×10^{-2} per reactor-year was used. The initatior event frequency was calculated as follows:

$$
\begin{array}{ll}
1.54 \times 10^{-2}/\text{rx-yr} \times 1 \text{ demand} & = 1.54 \times 10^{-2}/\text{rx-yr} \\
1.54 \times 10^{-2}/\text{rx-yr} \times 10^{-3} \times 23 \text{ demands} & = 3.54 \times 10^{-4}/\text{rx-yr} \\
\hline
 & 1.58 \times 10^{-2}/\text{rx-yr}
\end{array}
$$

Instrumentation and Control Faults

Reference 4 estimated PORV openings due to instrumentation and control failures at 5×10^{-3} per reactor-year. Reference 6 justified this value by taking the following failure rates that will cause the PORV to open from IEEE Std 500-1977:

Pressure transmitter fails high	0.25×10^{-6}/hr
Bistable functions without signal	0.206×10^{-6}/hr
I/E converter fails high	0.31×10^{-6}/hr
Summer module fails high	0.31×10^{-6}/hr
Short across contacts	0.01×10^{-6}/hr

The sum of these failure rates (approximately 0.8×10^{-6} per hr) was multiplied by the number of hours per year of plant operation (assuming an average annual plant availability of 80%) to arrive at the initiator frequency of 5×10^{-3} per reactor-year.

However, Reference 6 stated that the value of 5×10^{-3} per reactor-year had been incorrectly summed with the other categories of Reference 4. In this case, since the PORV opens due to instrument failure, it must be assumed that the failure will keep the PORV open. Therefore, these failures must be treated differently than the other initiator frequencies which assume a calculated closing probability.

Overcooling Transients That Initiate High Pressure Injection (HPI)

In Reference 4, it was estimated that the PORV opening frequency from overcooling transients that initiate HPI (with operator failure to throttle or terminate flow before the PORV setpoint) is 1.8×10^{-3} per reactor-year.

In Reference 6, however, the frequency was revised to be 8.4×10^{-4} per reactor-year, stating:

> "The prediction value of 1.8×10^{-3}/rx-yr for overcooling events that initiate HPI and result in subsequent PORV actuation was determined from operating experience and operator failure probabilities. The calculation consisted of 8 overcooling events in 392 reactor trips x expected number of reactor trips per year x probability of operator failing to throttle HPI given the overcooling event. There have been 8 HPI initiations due to overcooling events (exclusive of PORV initiated events): Oconee-1, 02/14/78, Davis Besse-1, 10/23/77, Rancho Seco, 01/05/79, TMI-2, 03/29/78, TMI-2, 04/23/78, TMI-2, 11/07/78, TMI-2, 12/02/78. In addition there have been two events that could have started HPI (according to pressure trace of transient) but did not. Conservatively including these two events (Oconee-1 05/05/73 and Davis Besse-1 11/29/77) results in 10 events in 392 Rx-trips. Seven of these were due to auxiliary feedwater cooling. As pointed out previously, ANO-1 is upgrading the EFW system which will preclude auxiliary feedwater overcooling. The expected frequency of overcooling events then is 3/392 per reactor trip. Reactor

trip frequency in 1981 for ANO-1 was 6. The capacity factor was .658 for 1981. Since other calculations in this report assumed a future capacity factor of .80 the trip frequency used here is 7.3 (i.e., 8/.658x6). The operator failure probability is 1.5×10^{-2} demand (see attached event tree). The overall probability is therefore $3/392 \times 7.3 \times 1.5 \times 10^{-2}$ = 8.4×10^{-4}/rx-yr."

In summary, the determination of a PORV opening frequency, from all causes, is as follows:

Event	Reference 4 Per Reactor-year	Reference 6 Per Reactor-year
Overpressure transients	3.9×10^{-6}	3.9×10^{-5}*
Transients with delayed auxiliary feedwater	1.4×10^{-3}	7.6×10^{-4}
Operator action under ATOG guidelines	1.58×10^{-2}	1.58×10^{-2}
Instrumentation and control faults	5.0×10^{-3}	1.7×10^{-3}**
Overcooling transients that initiate HPI	1.8×10^{-3}	8.4×10^{-4}
Total	2.4×10^{-2}	1.9×10^{-2}

3.1.1.2 Small-Break LOCA Calculations

In order to determine the probability of a small-break LOCA from a stuck-open PORV, B&W multiplied the total PORV initiator event frequency of Reference 4 (2.4×10^{-2} per reactor-year) by the probability of the PORV failing open on demand. B&W estimated the failure rate to be 2.1×10^{-2} failures per demand, based upon operational data of 5 (non-installation) failures in 250 recorded demands, combined with an analysis of other mechanical failures which

*Assumes 10 overpressure transients per year.

**Includes only the component attributed to failure (high) of the pressure transmitter. The remaining instrument failures (approximately 5.6×10^{-3} per reactor-year) are treated separately because these initiator events must also be presumed to result in failure of the PORV to close.

would prevent valve closure. The probability of a small-break LOCA from a stuck-open PORV was established as: (2.4×10^{-2} events per reactor-year) x (2.1×10^{-2} failures per demand) = 5.04×10^{-4} per reactor-year. This estimate took no credit for operator action to terminate the LOCA by closing the block valve.

In Reference 6, however, it was stated that the probability of an open PORV flow path was the product of a stuck-open PORV and a failure of the block valve to close. The formula used in Reference 6 was:

[(Sum of PORV Initiator Events) x (Failure of PORV to Close)
+ (PORV Openings due to Instrument Fault)]
x [Failure of Block Valve to Close] = Open PORV Flow Path

The following values were applied to the equation:

Sum of PORV initiator events:	1.9×10^{-2} per reactor-year - from Section 3.1.1.1 above
Failure of PORV to close:	1.7×10^{-2} per demand - This figure was revised from the 2.1×10^{-2} per demand of Reference 4 by including data from C-E plant experience and EPRI testing.
PORV opening due to instrument faults:	5.6×10^{-3} per reactor-year ($0.8 \times 10^{-6} \times 0.8 \times 8760$) from Section 3.1.1.1 above, excluding the component attributed to the pressurizer pressure transmitter failing high.
Failure of the block valve to close:	2.42×10^{-2} per demand - 2.22×10^{-2} per demand for valve-related failure + 1.95×10^{-3} per demand for loss of motive power.

With regard to the block-valve failure rate, Reference 6 stated:

"Two dominant contributors were identified which would not allow the block valve to close. These were valve related faults including local power and the absence of 480 VAC motive power. The dominant instance of motive power unavailability will occur as a result of LOOP (LOOPxdiesel fails; 1.95×10^{-3}). The conclusion is conservative since if this condition existed (i.e., LOOP) some of initiator events could not occur. The block valve failure rate was determined using a Bayesian updating procedure. A value of 8.1×10^{-4} for failure to close per demand was calculated from Ref. 11. This failure rate was used to construct a lognormal distribution (mean = 8.1×10^{-4}, range factor = 10), which was then used as the prior in the Bayesian analysis. A review of Reference 12 produced 34 failures in 1433 demands, which was then implemented to update the prior distribution. This resulted in a posterior mean of 2.22×10^{-2} with 5th and 95th percentile values of 1.63×10^{-2} and 2.89×10^{-2} respectively."

Inserting these values into the formula, it was concluded that the probability of having an open PORV flow path (i.e., the probability of a small-break LOCA from a stuck-open PORV) was 1.43×10^{-4} per reactor-year:

$$[(1.9 \times 10^{-2})(1.7 \times 10^{-2}) + 5.6 \times 10^{-3}]\ 2.42 \times 10^{-2} = 1.43 \times 10^{-4}/\text{rx-yr}$$

Reference 6 was concluded by stating:

"The results of this study indicate that the probability of having an open PORV flow path is 1.43×10^{-4}/Rx-yr. This value does not significantly impact the small break LOCA probability for all causes. A sensitivity study was also conducted in order to determine the effect of multiple PORV challenges with certain initiator frequency groups. As mentioned in the response to Question 7, multiple PORV openings could occur with causes 2 and 5. To illustrate the potential impact of these increased PORV demands, causes 2 and 5 were assumed to initiate 10 PORV openings. The results of this investigation demonstrate that the small break LOCA probability would only be perturbed 2.1% in both cases."

3.1.2 Small-Break LOCA Probability Based on Operational Data

Reference 4 stated that there had been 190 reactor trips of B&W units which would have lifted the PORV with the old PORV/reactor trip setpoints, but that only 3 of these events would have reached the new PORV setpoints. In addition, modifications had been made so that PORV action would now be precluded given those three initiating events, as well. Consequently, B&W conservatively estimated that there could be one PORV actuation in the 45-year life of a B&W plant for a resultant probability of 2.22×10^{-2} per reactor-year. Combining this initiation frequency with the PORV failure rate in Reference 4 of 2.1×10^{-2} failures per demand, B&W concluded the probability of a small-break LOCA from a stuck-open PORV to be 4.7×10^{-4} per reactor-year ($2.22 \times 10^{-2} \times 2.1 \times 10^{-2}$).

In Reference 6, further justification of the estimate of one PORV actuation in 45 years was provided:

"From the pressure responses associated with various actual transients, three transients could have actuated the PORV with the revised setpoints (Oconee-3, 04/30/75, Rancho Seco, 03/20/78, Crystal River-3, 02/26/80). However, changes have been made to the plant that would have precluded the initiating events that caused these three transients. Even with the revised setpoints and other changes it was assumed that if one event (not specified) could occur in the 45 years of operation then the probability of occurrence would be 2.22 x 10^{-2}/Rx-yr. Although this assumption was made, a closer estimate of 0 events in 45 reactor years is believed to be a better indicator of future event frequency."

3.1.3 Percentage of PORV Openings During Overpressure Transients

As indicated in Section 3.1.2, B&W stated in Reference 4 that 3 overpressure transients out of 190 would have lifted the PORV with the new setpoint. Further, plant modifications had been installed to preclude even these three events. Nevertheless, even with 3 of 190, less than 1.6% of the overpressure transients would cause a PORV opening. B&W stated that this satisfies the 5% criteria of Item II.K.3.7.

3.1.4 Actions Taken to Reduce PORV Challenges

In Reference 6, the following statements were made relative to reducing the probability of a small-break LOCA from a stuck-open PORV:

"In addition to the elevated PORV setpoint AP&L has taken many steps to reduce the probability of a stuck-open PORV. These actions have been directed in three major areas: reducing the PORV challenge potential, equipment upgrades that rectify past problem areas, and an increased emphasis on operator awareness.

The potential for challenging the PORV has been greatly reduced by incorporating two anticipatory trips and improvements in auxiliary feedwater control. ANO-1 has installed anticipatory reactor trips on loss of feedwater and on turbine trip. They are also upgrading the EFW system which will preclude auxiliary feedwater overcooling.

A review of B&W operating history has identified three transients which could have challenged the PORV at its elevated setpoint: (Oconee 3, 04/30/75, Rancho Seco, 03/20/78, Crystal River-3, 02/26/80). An investigation into the failure mechanisms which caused these pressure excursions has led to a variety of equipment upgrades. As a result, actions have been taken to avoid short circuits that would permit PORV opening, to enable proper response on loss of single power supplies to NNI control circuitry, and to upgrade power supply reliability. Changes have been made in the PORV control system along with power upgrades.

In the event of a small break LOCA, measures have been taken to increase operator awareness to permit valid diagnosis and actions. The presence of an alarmed acoustic monitor at the outlet of the PORV will facilitate the action of the operator closing the block valve. In addition AP&L has implemented the ATOG program. The training the operator receives in this program is very extensive; areas which pertain to this discussion are:

- For overcooling events the operator is instructed to throttle HPI to prevent pressurizer filling in the presence of both subcooled reactor coolant and the return of pressurizer level,
- Recognition of pressurizer steam space breaks,
- Quench tank pressure/level changes are an indicator of PORV discharge."

3.1.5 Small-Break LOCA from Stuck-Open Safety-Relief Valve (SRV)

In Reference 4, B&W stated there have been three cases in which SRVs have lifted on B&W plants. None of these resulted in failure of a valve to reseat. Because of the lack of data, no estimate of SRV failure rate was made.

In Reference 6, however, an SRV failure rate was established of 3.12×10^{-2} per demand for an SRV water discharge. This rate was calculated by using main steam safety valve (MSSV) data (1 partial failure in 2950 demands), estimating a water-relief failure rate to be 100 times larger, and adjusting for recent EPRI test data and an actual lift of an SRV at the Crystal River plant in 1980.

In Reference 6, it was estimated that the predominant events leading to SRV challenges were overcooling and overheating events. Overheating events were not considered due to the reliability of the existing auxiliary feedwater system design. Overcooling events were estimated by reviewing B&W transients leading to reactor trip. Of the 392 trips reviewed, a list was created of those which resulted in low pressure ESFAS initiations. Of these events, many were no longer possible because of plant modifications. Three remained applicable. The probability that the operator would fail to throttle or terminate HPI flow prior to reaching the SRV lift point was estimated to be 1.49×10^{-2} per demand.

The resulting probability of a small-break LOCA from a stuck-open SRV was then calculated to be:

(3 events/392 reactor-trips)(7.3 reactor-trips/year) X
$(1.49 \times 10^{-2})(3.12 \times 10^{-2}) = 2.6 \times 10^{-5}$ per reactor-year.

3.2 EVALUATION OF THE B&W REPORT SUBMITTED IN RESPONSE TO NUREG-0737, ITEMS II.K.3.7 AND II.K.3.7

The following sections evaluate the information provided in References 4 and 6, and summarized in Section 3.1 above, as related to Items II.K.3.2 and II.K.3.7 of NUREG-0737.

3.2.1 Small-Break LOCA Probability Calculations

The evaluation of the analytical approach to determining the probability of a small-break LOCA from a stuck-open PORV will be conducted in two parts: (1) an evaluation of the PORV opening frequency as developed by B&W, and (2) an evaluation of the probability that the PORV will remain open following any given demand opening.

3.2.1.1 PORV Opening Frequency

B&W determined that the PORV will be opened with a frequency of 1.9×10^{-2} openings per reactor-year. The frequency was determined by summing five initiator events. Three of these initiator events (transients with delayed auxiliary feed, instrumentation and control faults, and overcooling transients) are minor contributors to the resultant frequency. Detailed discussion of these events has not been provided below because detailed information has been provided by B&W as to the derivation of the initiator frequencies and also because it is reasonable to assume that these events would not have a major impact on the total opening frequency. The remaining two events (overpressure transients and operator action under ATOG guidelines) are further discussed below because (1) overpressure transients have historically been the major contributor to PORV openings (before the setpoint changes) and (2) operator action is currently the major factor in the determination of a combined initiator frequency.

Overpressure Transients

Prior to the TMI accident, overpressure transients were a major factor in the lifting of the PORV. In Reference 4, B&W stated that there were 190 recorded events on B&W plants that would have lifted the PORV with the pre-TMI setpoints. In Reference 6, however, it was concluded that the probability of lifting the PORV in a single overpressure transient was 3.9×10^{-6} per event. Estimating 10 overpressure events per reactor-year, Reference 6 concluded that the recurrence frequency of PORV openings was 3.9×10^{-5} per reactor-year.

This dramatic decrease in the number of PORV challenges due to overpressure transients is the result of changes in both the high pressure reactor trip and PORV actuation setpoints following the TMI accident. The pre- and post-TMI setpoints are as follows:

	Pre-TMI	Post-TMI
Operating Pressure	2155	2155
PORV Opens	2255	2450
HP Reactor Trip	2355	2300
SRV Opens	2500	2500

As can be seen from the above table, not only has the PORV setpoint been increased by nearly 200 psig, but also the high pressure reactor trip, which previously did not occur until after the PORV had been signaled to open, occurs 150 psig before the PORV setpoint is reached. Both changes significantly affect the probability that a given overpressure transient will be terminated prior to reaching the revised PORV setpoint. It is reasonable to expect, therefore, that a change in setpoints of these two trip functions will greatly reduce the recurrence frequency of a PORV opening during an overpressure transient.

Part of B&W's calculation that the probability of a PORV opening during an overpressure transient is 3.9×10^{-6} per event, however, depends upon a standard deviation of 2 psig for the high pressure reactor trip setpoint and the PORV actuation setpoint. B&W justifies the use of such a small standard deviation by identifying four non-common modules in the two strings (each module having a 1-psig accuracy) and by stating that the errors in the common modules exactly cancel each other in the analysis. A possible problem with this analysis is that the high pressure reactor trip signal is generated by a 2-out-of-4 logic circuit, whereas the PORV is actuated by selecting one of two possible input signals. Consequently, even if the pressure transmitter selected to actuate the PORV is one of the two transmitters that will eventually actuate the high pressure trip logic, a second and completely independent (of the PORV) transmitter must operate in order to cause the high pressure trip. Therefore, it appears that more than just the four modules identified by the B&W must be activated. Furthermore, if the transmitter selected to the PORV is calibrated well below the remaining three pressure transmitters, peak system pressure will be much closer to the PORV setpoint than the combined normal distribution with a mean of 123.37 and standard deviation of 27.59 used by B&W in the calculations.

Taking the above logic one step further, the authors of this TER developed the following normal distributions by taking data from Reference 4 and also from the EPRI PWR Safety Valve and Relief Valve Test Program [11]:

	Mean (psig)	Standard Deviation (psig)	Reference(s)
Peak System Pressure	2327	27.5	4
PORV Opening	2450	21.7	4/11
SRV Opening (both)	2500	29.5	4/11

Using these distributions, the authors of this TER calculated the probability of a PORV opening given a reactor trip on overpressure and also the probability of an SRV opening given a reactor trip on overpressure (with no PORV opening). The calculation essentially involved a determination of the interference of one normal distribution with another (e.g., the interference of the reactor trip distribution with the PORV distribution). The calculations indicated the following opening probabilities:

PORV	2.2×10^{-4} per overpressure event
SRV	9.0×10^{-6} per overpressure event

Assuming 10 overpressure events per year, PORV openings would occur with a frequency of 2.2×10^{-3} per reactor-year and SRV openings with a frequency of 9.0×10^{-5} per reactor-year (assuming the PORV to be blocked or otherwise inoperative). These values must be considered to be extremely conservative, particularly since they assume complete independence between the reactor trip and PORV opening or because the effect of the PORV is not considered in the SRV analysis. Nevertheless, they show that the contribution of overpressure events to the total PORV openings remains small, even in view of the conservative assumptions involved.

Operator Action Under ATOG Guidelines

Since there were no steam generator tube ruptures in B&W experience, the determination of an initiator for operator action under ATOG guidelines in Reference 4 depends upon the determination of a chi-square 50% confidence value with zero failures. Calculating back from the B&W value of 1.54×10^{-2} events per reactor-year, it was determined that B&W assumed a 45-year plant life. This is consistent with average plant life as estimated in other parts of the report and is also consistent with the typical design lifetime of modern nuclear plants. In view of the fact that there have been no tube ruptures in the history of B&W plants, this approach appears to be a reasonable method of establishing an initiator frequency.

B&W has assumed that, in a steam generator tube-rupture event, there will be one demand on the PORV if offsite power is available (e.g., reactor coolant pumps are available to transfer heat to the steam generators) and that there will be 23 PORV demands without offsite power (e.g., natural circulation conditions). With regard to this portion of the analysis, several observations should be made:

1. Although it is possible to complete a post-tube-rupture cooldown and depressurization with only one PORV cycle (offsite power available), it is more likely, and experience at PWRs other than B&W units indicates, that multiple PORV cycles may be necessary. It does not

appear unreasonable to assume that as many as five cycles may be required.

2. In a situation where as many as 23 PORV cycles may be required in a fairly short period of time, a failure rate somewhat larger than the predicted failure rate for the PORV should be used. Reference 6 indicates that a sensitivity study by B&W shows that the failure rate is increased by 2.1% after 10 cycles of the PORV. It is not logical to assume, however, that this value can be linearly extrapolated to a value such as 23 cycles. At some point, the increasing number of cycles in a short time period will greatly affect the probable failure rate. It appears that an assumption of a failure rate increased by 50% would not be unreasonable.

3. Finally, the probability of the operator's potential failure to close the block valve in a situation in which the operator has opened the PORV is not as great as the condition when the PORV has automatically lifted. Since the operator is controlling the PORV, he is much more likely to be aware of the effect of the PORV on plant parameters, and therefore he should immediately be aware of a malfunction of the PORV. The expectation that an operator who is controlling a PORV which sticks open will immediately shut the block valve was recently demonstrated during a steam generator tube rupture at the Ginna Power Plant. Consequently, when considering the probability of operator action to shut the block valve in response to a stuck-open PORV under steam generator tube rupture conditions, only the probability of mechanical or electrical valve failure need be considered.

In view of the above discussion, the number of PORV openings per reactor-year due to steam generator tube ruptures is calculated as the frequency of a tube rupture event times the probability of retaining or losing offsite power:

$$1.54 \times 10^{-2} \times 0.999 \times 5 = 7.69 \times 10^{-2}$$
$$1.54 \times 10^{-2} \times 0.001 \times 23 = 3.54 \times 10^{-4}$$
$$\overline{7.73 \times 10^{-2}}$$

In view of the above discussions, a comparison of the B&W-calculated PORV opening recurrence frequency and that recalculated in this TER is provided below:

Event	B&W $(\text{Rx-yr})^{-1}$	TER $(\text{Rx-yr})^{-1}$
Overpressure transients	3.9×10^{-5}	2.2×10^{-3}
Transients with delayed auxiliary feedwater	7.6×10^{-4}	7.6×10^{-4}

Event	B&W $(Rx\text{-}yr)^{-1}$	TER $(Rx\text{-}yr)^{-1}$
Operator action under ATOG guidelines	1.58×10^{-2}	7.73×10^{-2}
Instrumentation and control faults	1.7×10^{-3}	1.7×10^{-3}
Overcooling transients that initiate HPI	8.4×10^{-4}	8.4×10^{-4}
Total	1.9×10^{-2}	8.28×10^{-2}

3.2.1.2 Small-Break LOCA Calculations

Having determined a PORV opening frequency, B&W calculated the probability of a small-break LOCA from a stuck-open PORV by multiplying the opening frequency by the probability of the PORV sticking open and the probability of the block valve failing to close.

B&W calculated the failure rate of the PORV to be 2.1×10^{-2} per demand using actual valve operating history. In Reference 6, this value was revised to 1.7×10^{-2} per demand by including additional operating history from Combustion Engineering-designed plants and EPRI test data. These failure rates can be justified because they incorporate actual valve operating experience and because they are in keeping with similar rates established in other studies and reports relative to similar relief valves. Use of the failure rate of 1.7×10^{-2} is considered both justified and appropriate.

In the case of potential failure of the block valve, however, B&W does not consider the possibility that the operator could fail to recognize the PORV as having failed open or, even if the PORV failure is correctly diagnosed, that the operator could fail to take appropriate action. The potential failure of the block valve to close on demand must be combined with the potential of the operator to fail to initiate the demand, for all cases except those for which the operator is intentionally operating the PORV.

In Reference 6, a failure rate of 2.4×10^{-2} per block valve demand was determined by B&W based upon two dominant contributors to failure: valve-related failures and absence of 480 VAC motive power. The contribution of valve-related failures was determined using a Bayesian updating procedure. The prior in the Bayesian analysis (8.1×10^{-4} mean failure rate, taken from Reference 12) was updated with data of 34 failures in 1433 demands [13]. Although B&W did not thoroughly justify the use of the 8.1×10^{-4} mean failure rate prior, the update data of 34 failures in 1433 demands totally dominate the Bayesian prediction, and a classical statistical estimate predicts a failure range similar to the B&W analysis. Consequently, the B&W anaysis is considered to provide a reasonable assessment of block valve reliability with respect to valve-related failures.

With regard to the absence of motive power, B&W makes the statement that dominant instance of motive power unavailability occurs as result of a loss of offsite power (LOOP x diesel failure). B&W states that this is a conservative conclusion because under LOOP conditions, some of the initiator events could not occur. This statement is interpreted to mean that under LOOP conditions, some of the valve-related failures (associated with electrical faults) could not occur. There are undoubtedly certain block valve failure modes which require the unavailability of electrical power, and the combination of motive power failure and valve-related faults in the B&W analysis is considered to be conservative. In summary, the block valve failure rate of 2.4×10^{-2} per demand determined by B&W appears to be a reasonable assessment of failure probability. However, when combined with the probability of the operator failing to close the valve (1.5×10^{-2} derived from NUREG/CR-1278 [14]), a combined probability of the block valve remaining open per event of 3.9×10^{-2} is obtained.

As shown in Figure 1, B&W determined the recurrence frequency of a small-break LOCA from a stuck-open PORV by multiplying the probability of the PORV remaining open with the probability of the block valve remaining open. The probability of the PORV remaining open is the sum of openings due to instrumentation faults and openings due to other demands multiplied by the probability of the PORV to fail to close. Figure 1 shows that B&W determined the probability of a small-break LOCA from a stuck-open PORV to be 1.4×10^{-4} per reactor-year.

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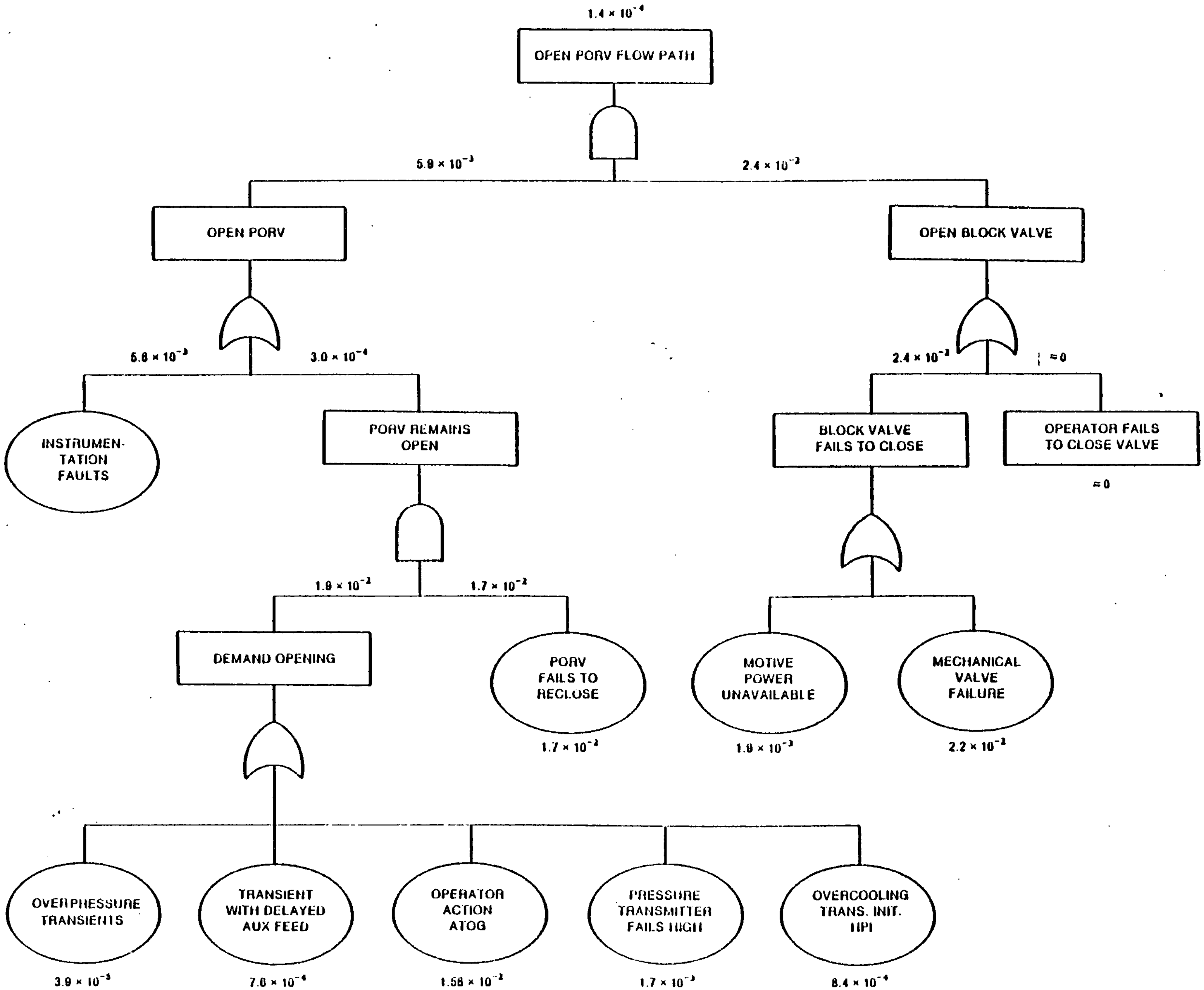

Figure 1. B&W Calculation

Figure 2 shows the same calculation revised in accordance with the comments and discussion developed in this section of the report. It includes changes made to the initiator frequency for overpressure transients (2.2×10^{-3} rather than 3.9×10^{-5}), the initiator frequency for operator action under ATOG guidelines (7.7×10^{-2} rather than 1.58×10^{-2}), and the probability of the operator failing to shut the block valve when required (1.5×10^{-2} per demand rather than 0). As result of this recalculation, the probability of a small-break LOCA from a stuck-open PORV is 2.53×10^{-4} per reactor- year. While this value is somewhat larger than the value determined by B&W, it is quite comparable and is also well below the median WASH-1400 frequency for a small-break LOCA of 1×10^{-3}.

3.2.1.3 Multiple PORV Cycles

With regard to the question of more than one PORV opening per initiator event, the following statement was made in Reference 6:

> "As mentioned in the response to question 7, multiple PORV openings could occur with causes 2 and 5. To illustrate the potential impact of these increased PORV demands, causes 2 and 5 were assumed to initiate 10 PORV openings. The results of this investigation demonstrate that the small break LOCA probability would only be perturbed 2.1% in both cases."

Although not expressly stated, B&W has apparently discounted the possibility of multiple PORV cycles under overpressure transient conditions because of the new high pressure trip/PORV setpoints. Similarly, multiple cycles of the PORV are not a consideration under conditions of failure of the pressure transmitter. In the case of operator action (ATOG), the matter of multiple valve cycles has already been considered (e.g., 23 PORV openings if offsite power is not available). This leaves cause 2 (transients with delayed auxiliary feed) and cause 5 (overcooling transients which initiate HPI) as initiator events with the possibility of more than one PORV opening.

B&W's statement that the investigation of these two causes with 10 multiple cycles each yielded a 2.1% increase in both cases considered each cause separately. Logically, to determine the overall probability of a

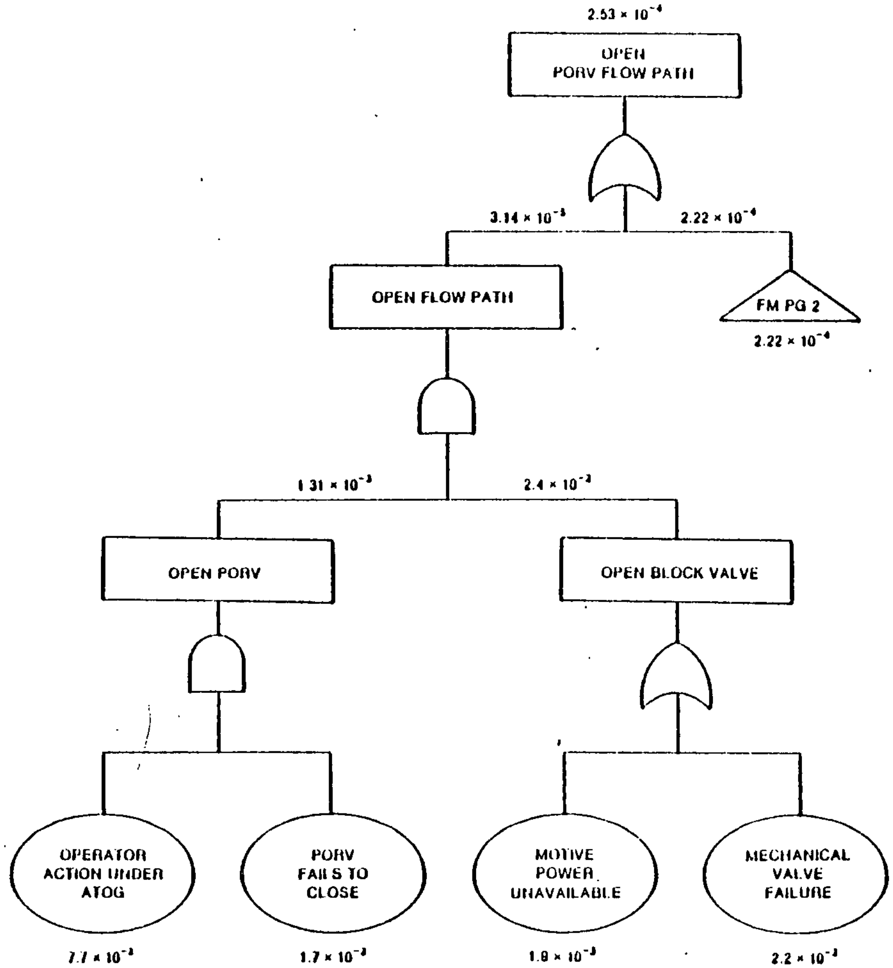

Figure 2. (Page 1) Revised Calculation

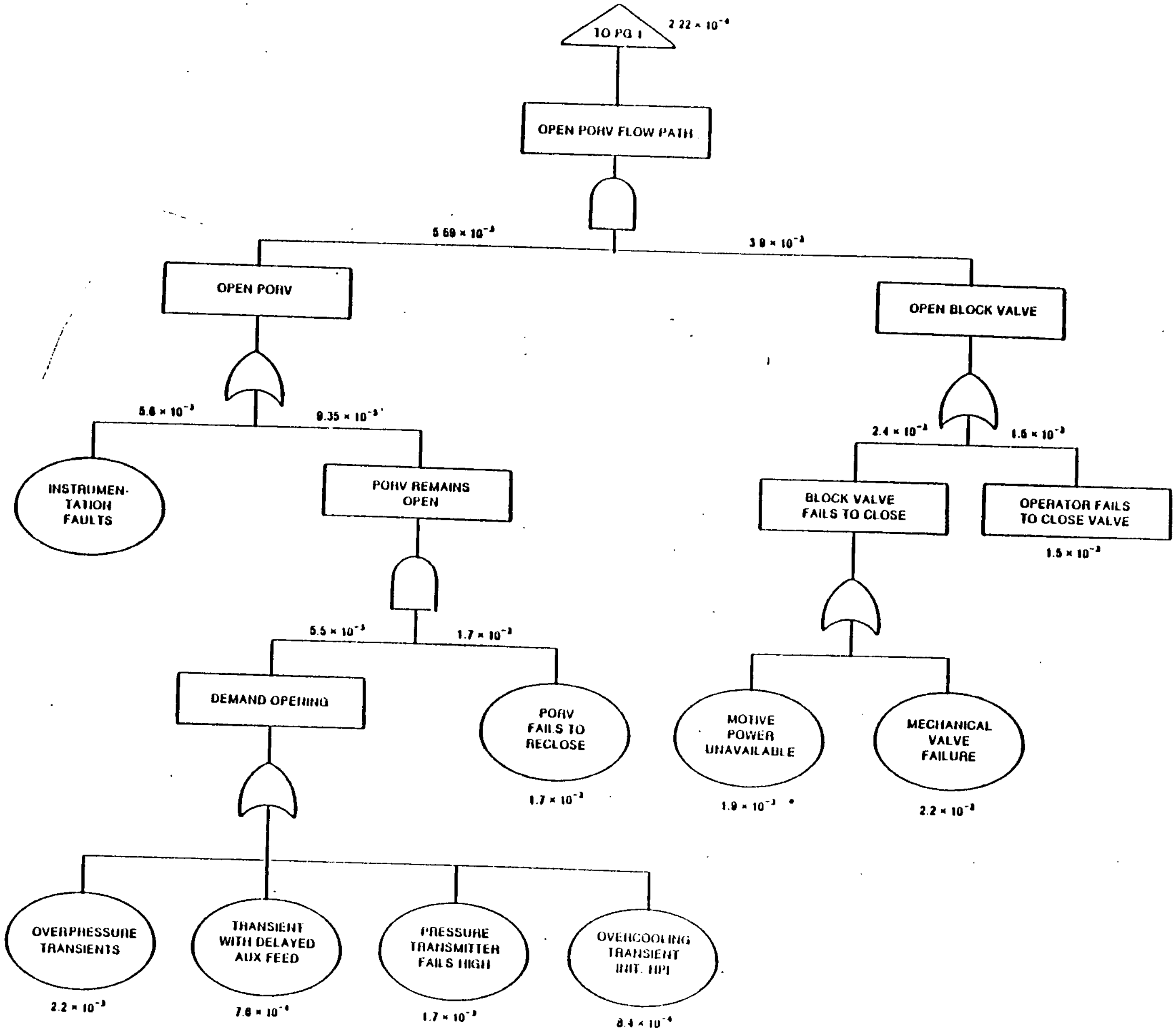

Figure 2. (Page 2) Revised Calculation

small-break LOCA from a stuck-open PORV (considering possible multiple cycles), multiple cycles during these events should be analyzed concurrently.

The revised calculation of Figure 2 (discussed in Section 3.2.1.2) has been updated to include the effect of multiple PORV cycles. The results of this updated analysis are given in Figure 3. Figure 3 shows a small-break LOCA probability of 2.61×10^{-4} per reactor-year, a 3.3% increase over the value calculated in Figure 2. These calculations show that multiple PORV cycles is not a primary concern for B&W-designed units.

The results of these PORV small-break LOCA calculations are summarized below:

	Probability of Small-break LOCA from Stuck-open PORV (per reactor-year)
Figure 1 (B&W Calculation)	1.4×10^{-4}
Figure 2 (Revised calculation described in Section 3.2.1.2)	2.53×10^{-4}
Figure 3 (Revised calculation of Figure 2 with additional consideration of multiple cycles)	2.61×10^{-4}

3.2.2 Small-break LOCA Probability Based on Operational Data

In Reference 4, B&W determined the recurrence frequency of a small-break LOCA from a stuck-open PORV to be 4.7×10^{-4}, based upon operational data. B&W used a theory that of the 190 reactor trips which had lifted the PORV in the history of B&W plants, only 3 of these events would lift the PORV with the new setpoints. Further, B&W reasoned that subsequent modifications had been made to B&W plants since these three events such that none of them would cause a PORV lift today. Consequently, B&W estimated that if one PORV lift occurred in the 45 years of life of a B&W unit, the probability of PORV openings would be 2.22×10^{-2} per reactor-year. Combining this frequency with the previously determined PORV failure rate (2.1×10^{-2} per demand), the small-break LOCA frequency was calculated. It should be noted that this

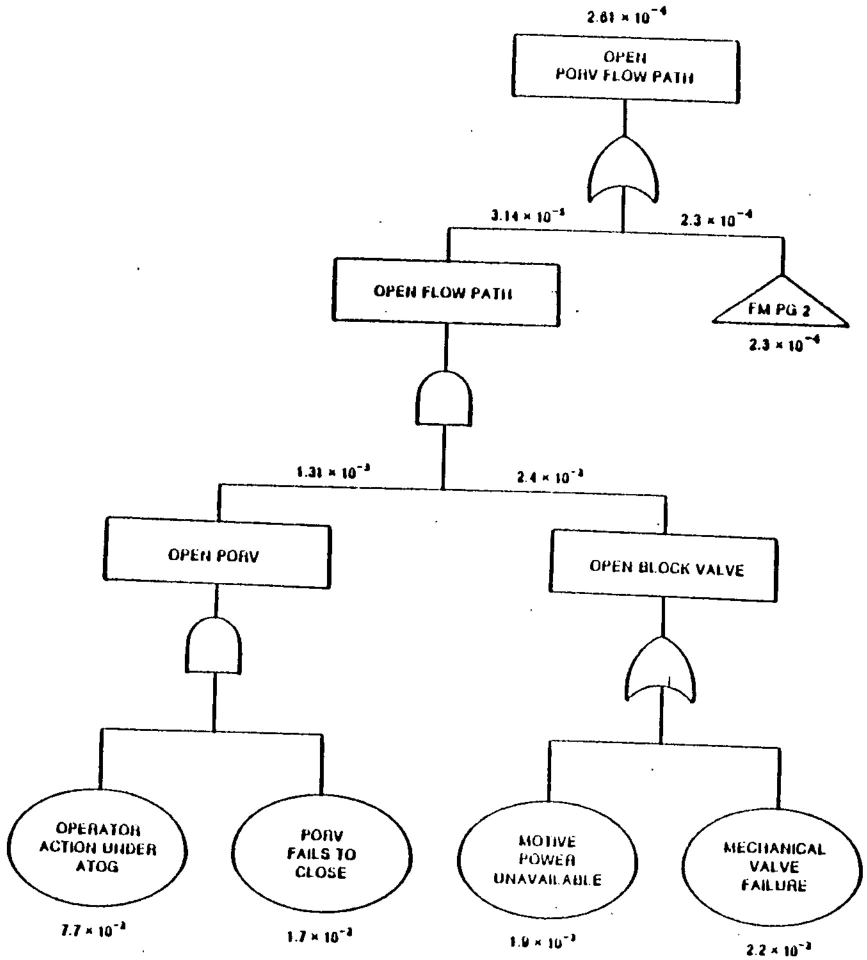

Figure 3. (Page 1) Revised Calculation, Considering Multiple PORV Cycles

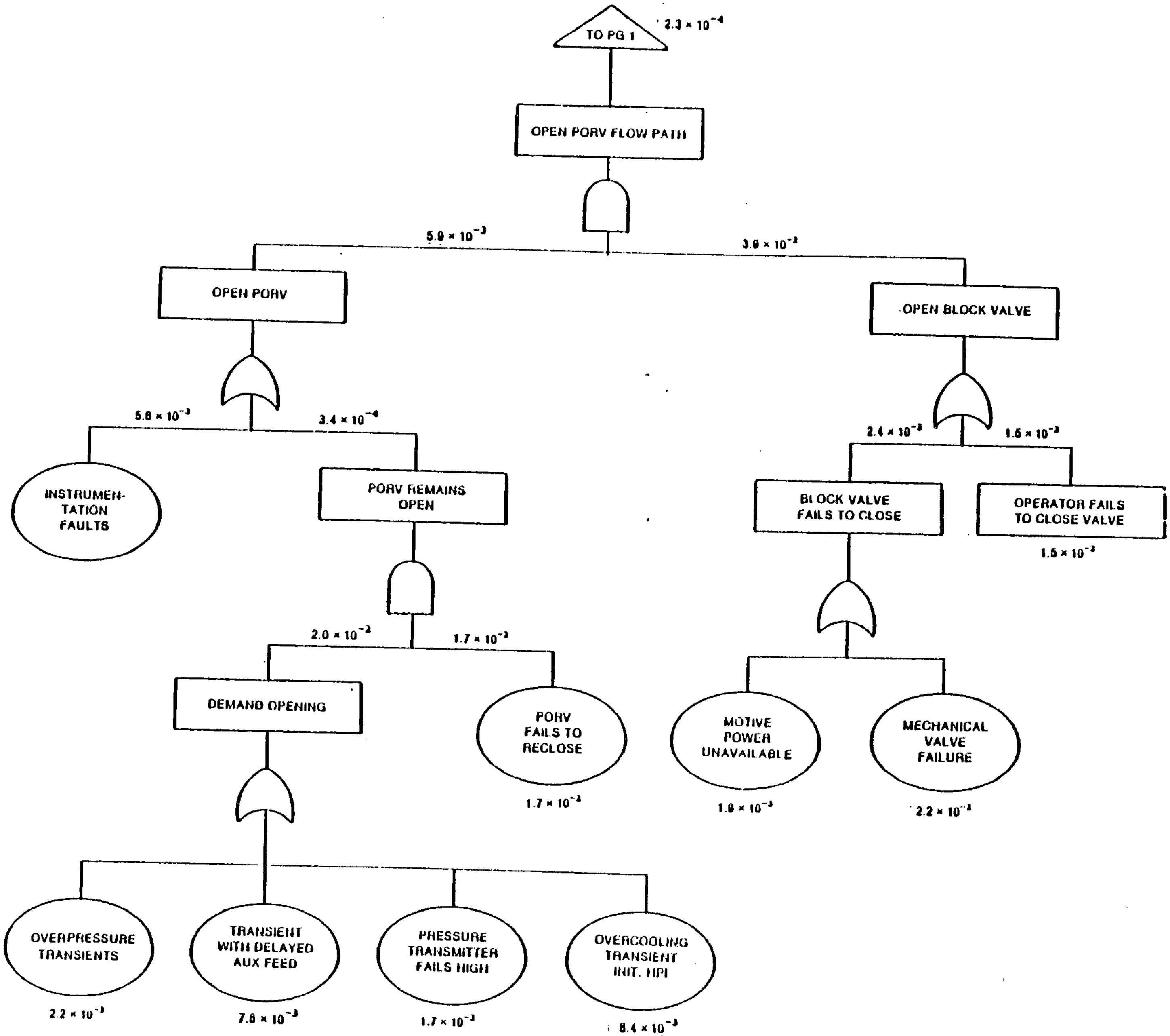

Figure 3. (Page 2) Revised Calculation, Considering Multiple PORV Cycles

calculation does not take credit for operator action to terminate the small-break LOCA by shutting the associated block valve.

While this analysis is theoretically based on operational data, it should be noted that the actual data (148 reactor trips with PORV actuation prior to the TMI-2 accident and 3 reactor trips with PORV actuation after the setpoint changes) are not used in the calculation. Rather, B&W estimates that, of the 190 reactor trips which would have lifted the old setpoints, only 3 would have lifted the PORV with the new setpoints and even these would not cause PORV actuation today because of additional plant changes. Consequently, B&W is using an analysis to determine that, under present plant conditions, no actual PORV lifts would have occurred; hence, the estimate of one PORV lift in a 45-year plant life is used.

In view of the dependence of B&W's analysis on theoretical information, the authors of this TER determined that an independent confirmatory analysis, based on operational data to the maximum practical extent, was desirable. In order to do this, it was necessary to consider only the operational data recorded after the TMI setpoint changes. Although there are only limited operational data since the TMI setpoint changes were initiated, there was one significant event (at Crystal River Unit 3 in February 1980), during which both the PORV and an SRV were opened.

It is not the intent of this TER to provide a detailed review of the Crystal River event. However, the following significant facts are relevant to this report:

1. The loss of a non-nuclear instrumentation (NNI) bus caused the PORV to open and remain open.
2. The erroneous control input signals (due to loss of NNI) caused a high pressure transient which resulted in a reactor trip (2300 psig).
3. Decreasing RCS pressure caused initiation of HPI (1500 psig).
4. The operator isolated the open PORV by shutting the block valve.
5. The HPI system filled the RCS to a solid water condition, lifting code safety valve RCV-8 (SRV).

6. After establishing emergency feedwater to one steam generator, HPI flow was throttled to establish an RCS pressure of 2300 psig, reseating the SRV.

It must be noted that certain plant modifications have been made to preclude NNI power supply failures since the above incident at Crystal River Unit 3. These modifications were made pursuant to IE Bulletin 79-27, published in response to the Crystal River event. Nevertheless, failures of NNI power supplies subsequent to these modifications, resulting in at least partial loss of non-nuclear instruments (e.g., partial loss of NNI at the Davis-Besse plant on June 24, 1981), indicate that PORV/SRV challenges as a result of NNI failures are still credible events. Consequently, consideration of the Crystal River event in an analysis of post-TMI PORV/SRV operating data remains valid, although the results of the analysis will be fairly conservative in view of the above-mentioned plant modifications.

Since the Crystal River was not a classical overpressure transient but was caused by failure of instrument power supplies, the event tree used to analyze these data has been specifically tailored to an overcooling situation (i.e., where HPI operation challenges the PORV/SRV). In this scenario, operator control of the block valve and of HPI discharge pressure are key considerations in challenging the PORV/SRV. (Note: Even though the PORV opens on loss of NNI power supply, a probability is assigned to the PORV reseating which presumes that NNI power is regained within a short period of time.)

As shown in Figure 4, the small-break LOCA probabilities resulting from this analysis are:

SRV (per rx-yr)	PORV (per rx-yr)
3.17×10^{-3}	7.12×10^{-5}
1.55×10^{-3}	6.38×10^{-5}
4.72×10^{-3}	1.35×10^{-4}

The results indicate that the probability of a small-break LOCA from a stuck-open PORV or SRV remains within the WASH-1400 range of 10^{-2} to 10^{-4}

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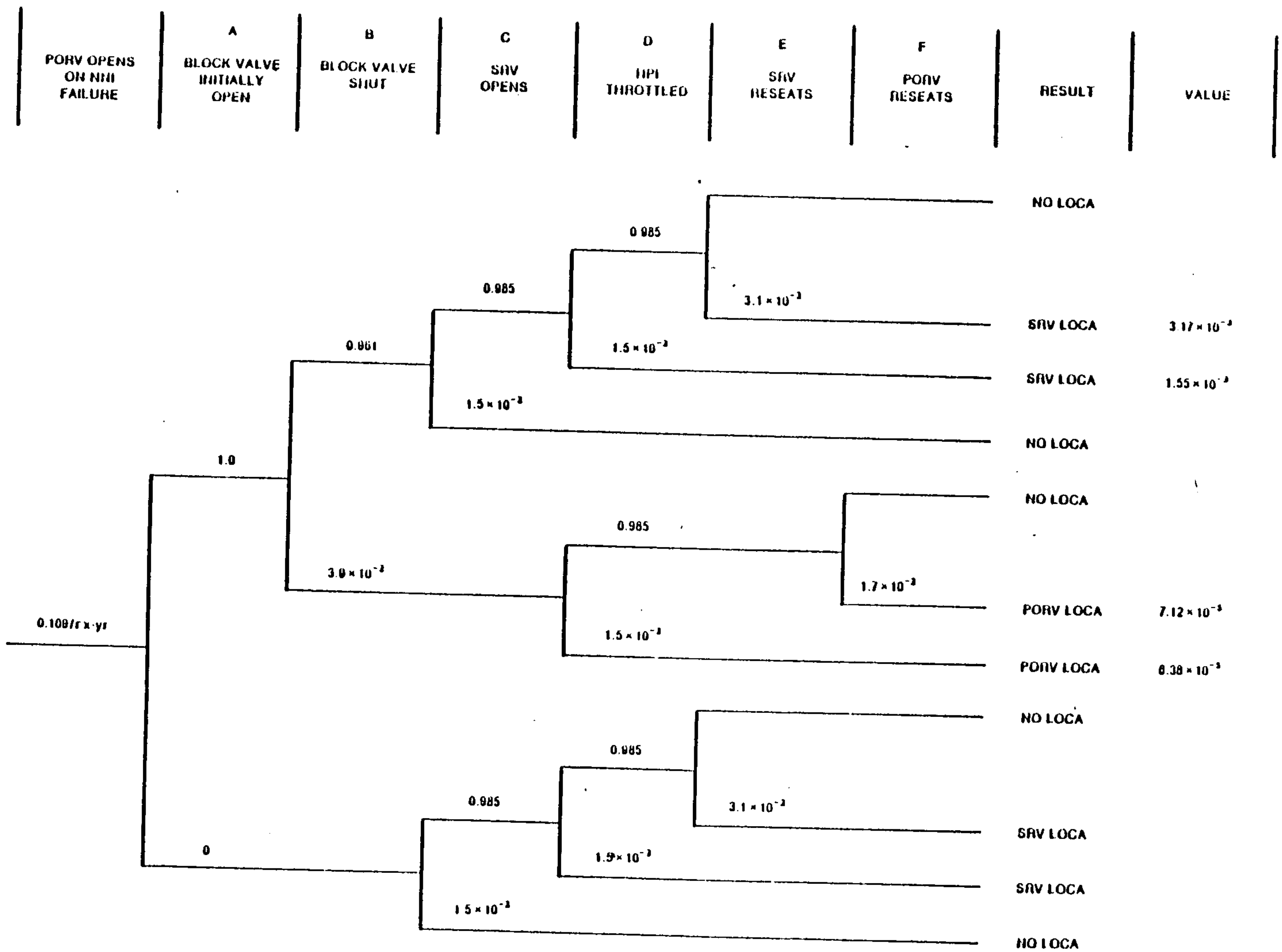

Figure 4. Operational Data Event-Tree

NOTES TO FIGURE 4

1. PORV Opens on NNI Failure:

   No. of events: 1 (Crystal River Unit 3, NUREG-0667)

   No. of reactor-years of B&W unit operation from April 1980 to June 1983: 9.2 (assumes a 60% availability of 7 units throughout the period)

   No. of openings per reactor year: 1/9.2 = 0.109 events/rx-yr

2. Block Valve Initially Open: Yes (Conservative assumption)

   Note: The effect of assuming that the block valves are open less than 100% of the time is to decrease the calculated probability of a small-break LOCA from a stuck-open PORV while the calculated probability of a small-break LOCA from a stuck-open SRV remains essentially constant as follows:

Initial Block Valve Opening	PORV (per rx-yr)	SRV (per rx-yr)
100%	1.35×10^{-4}	4.72×10^{-3}
50%	6.75×10^{-5}	4.80×10^{-3}
0%	0	4.89×10^{-3}

3. Block Valve Shut: 0.961 (Figure 2, page 2 of this report)

4. SRV Opens: 0.985 (For all plants except Davis-Besse)

   With the PORV isolated, the SRV setpoint is exceeded by HPI system pressure unless the operator throttles HPI sufficiently prior to reaching the setpoint. The same probability is assigned for operator failure to throttle as for operator failure to shut the block valve (1.5×10^{-2} per demand; see page 2 of Figure 2 on page 28 of this report).

   1×10^{-3} (for Davis-Besse)

   HPI system pressure at shutoff head does not exceed the SRV setpoint at the Davis-Besse plant. A probability of 1×10^{-3} is assigned to account for possible premature opening at elevated RPS system pressures. The effect of this reduces the probability of an SRV LOCA at the Davis-Besse plant to 4.79×10^{-6} per reactor-year.

5. HPI Throttled After SRV Opening: No: 1.5×10^{-2}

   The operator failure probability following SRV opening is assigned the same value as the probability of failure to throttle HPI prior to SRV opening.

6. SRV Fails to Close on Demand: 3.1×10^{-2} (Reference 6).

7. PORV fails to Close on Demand: 1.7×10^{-2} (Section 3.1.1.2 of this report).

for small-break LOCA. Furthermore, when considering that plant modifications have been made which will substantially lower the probability of the PORV opening and remaining open on loss of NNI power supplies and that less instrumentation will be lost on loss of NNI power supplies, the small-break LOCA probabilities will be lowered to the bottom of or below the WASH-1400 range. Since the modifications to the NNI power supplies affect the event initiating frequency, these modifications will affect the small-break LOCA probabilities of both the PORV and SRV.

In summary, both the B&W analysis based upon operating data and the conservative independent analysis of this TER using only post-TMI setpoint change data indicate that the probability of a small-break LOCA from a stuck-open PORV or SRV remains within the WASH-1400 range for small-break LOCA.

3.2.3 Percentage of PORV Openings During Overpressure Transients

In Reference 4, B&W stated that with new setpoints, the PORV would have opened in 3 of 190 overpressure transients. B&W further stated that plant modifications had been installed which would preclude even these PORV openings. Nevertheless, even if the three openings were not precluded, the percentage of PORV openings per overpressure transient would still be 1.6%, which met the 5% criteria of Item II.K.3.7.

It was noted in Section 3.2.2 of this report that since the TMI accident, there have been 59 reactor trips of which one PORV actuation occurred. This information indicates that 1.7% of overpressure transients cause PORV openings. In addition, in Section 3.2.1 of this report, a value of 2.2×10^{-3} PORV openings per reactor-year due to overpressure transients was determined out of a total of 8.28×10^{-2} PORV openings per reactor-year from all sources (other than certain instrument failures). These values lead to a determination that approximately 2.7% of all openings will be associated with overpressure conditions.

The compilation of this information indicates that the criteria of Item II.K.3.7 have been met.

3.2.4 Actions Taken to Reduce PORV Challenges

As has been developed previously in this report, prior to the TMI accident, approximately 148 reactor trips resulted in PORV actuations. Since the TMI accident, there have been 59 reactor trips, 42 of which would have lifted the PORV with the old setpoints but only one of which actually caused the PORV with new setpoints to lift. Subsequent to that singular event, additional modifications were made to preclude a repeat of that lifting. The primary reason for the substantial reduction of PORV actuations at B&W plants is the resetting of the high pressure reactor trip setpoint and the PORV actuation setpoint. Prior to the TMI accident, during an overpressure situation, the PORV lifted before a high pressure reactor trip would occur. The PORV was used to reduce the number of high pressure trips. Since the setpoint revision, the roles of these two functions have been reversed. The reactor trip now occurs 150 psig prior to reaching the PORV setpoint. The PORV now performs only its primary function of limiting safety-relief valve operation under overpressure conditions.

Other measures taken at B&W plants to reduce recurrence of PORV openings, such as increased operator awareness, installation of certain additional anticipatory trips, improvements in auxiliary feedwater controls, and other circuitry changes, have undoubtedly contributed to the observed reduction in PORV openings. There is no need to attempt to quantify these contributions to the overall reduction, however, since the overall reduction has been so substantial. At the same time, it is considered that these additional measures have had only a minor effect in comparison to that of the setpoint changes. The setpoint changes are considered the primary contributor to ensuring that the probability of a small-break LOCA from a stuck-open PORV remains below the WASH-1400 median frequency of a small-break LOCA.*

*Note: Reference 6 stated that ANO-1 had installed two anticipatory reactor trips as measures to reduce PORV challenges. It is not known if these trips have been installed at all B&W units; however, this particular information is not considered relevant because of the substantial reduction in the PORV challenge rate resulting from the setpoint changes.

3.2.5 Small-Break LOCA from Stuck-Open Safety-Relief Valve (SRV)

In Reference 6, the probability of a small-break LOCA from a stuck-open SRV was determined to be 2.6×10^{-5} per reactor-year per SRV. The calculation used a failure rate taken from main steam safety valve data, correlated to water relief valves. The calculation also relied upon data from overcooling events that resulted in high pressure engineered-safety-feature system activations which could challenge the SRV, if operator actions to limit the pressure increase were not taken. The other initiator event which could lead to initiation of high pressure safety injection, overheating, was not considered in this calculation because the improved reliability of the auxiliary feedwater system essentially precludes overheating events (see Section 3.1.5 of this report).

In addition to this calculation by B&W, a calculation based upon only operational data since the change in setpoints following the TMI event was provided in Section 3.2.2 of this report. That analysis indicated an extremely conservative probability of a small-break LOCA from a stuck-open SRV was 4.72×10^{-3} per reactor-year. The primary focus of that analysis was on overcooling events in which the SRV is lifted by pressure head from the HPI system because this is the only actual operational event to cause an SRV opening since the TMI accident. In order to investigate the probability of an SRV opening during a possible overpressure transient, the authors of this TER conducted the theoretical analysis described below.

The probability of a small-break LOCA from a stuck-open SRV initiated by an RCS system overpressure transient was determined by considering the interaction of three normal distributions; the RPS trip setpoint (with pressure overshoot), the PORV setpoint, and the SRV setpoint. The mean values and variances for each setpoint were previously described in Section 3.2.1.1. The values are:

	Mean (psig)	Variance (psig)
RPS trip (with overshoot)	2327	27.5
PORV setpoint	2450	21.7
SRV setpoint	2500	29.5

By integrating throughout the entire normal distribution of the RPS trip, the following probabilities were developed regarding PORV and SRV openings:

PORV opening per overpressure RPS trip	2.2×10^{-4}
SRV opening per overpressure RPS trip (no PORV opening)	9.0×10^{-6}
Both PORV and SRV opening per overpressure RPS trip	8.8×10^{-7}
SRV opening given that the PORV has opened	4.0×10^{-3}

Applying these probabilities to the event trees of Figures 5 and 6, the following small-break LOCA probabilities were determined:

	Block Valves 100% Open (Figure 5)	Block Valves 100% Closed (Figure 6)
Stuck-open PORV per rx-yr	1.5×10^{-6}	Not applicable
Stuck-open SRV per rx-yr	3.1×10^{-6}	2.8×10^{-6}

The results of this analysis must be tempered by the following considerations:

1. The assumption of normality is at its worst in the tails of the distributions, the areas of primary focus of this analysis.
2. The dependence of the PORV and the SRV is not accounted for, i.e., if the PORV opens, the RCS peak pressure will probably be less than if it does not open.
3. The variances (standard deviations) of the PORV and SRV setpoint distributions are based upon a limited amount of test data and are not based upon actual in-plant operating conditions [11].

In addition, the resulting probability of a small-break LOCA from a stuck-open PORV is based upon only the overpressure transient, which is not the controlling initiating event for this accident. Consequently, there is no direct correlation between the above results and those discussed in Sections 3.2.1 and 3.2.2 of this report. Nevertheless, in considering the probability of a small-break LOCA from a stuck-open SRV due to an overpressure condition in the RCS system, this analysis shows that the probability is quite small and that it is amply bounded by the predictions of WASH-1400.

The possibility of two SRVs opening during a given initiator event (overpressure or overcooling) has not been considered (for plants with two SRVs) because the opening of either SRV will terminate the pressure increase. The possibility of both SRVs opening simultaneously is considered to be extremely remote.

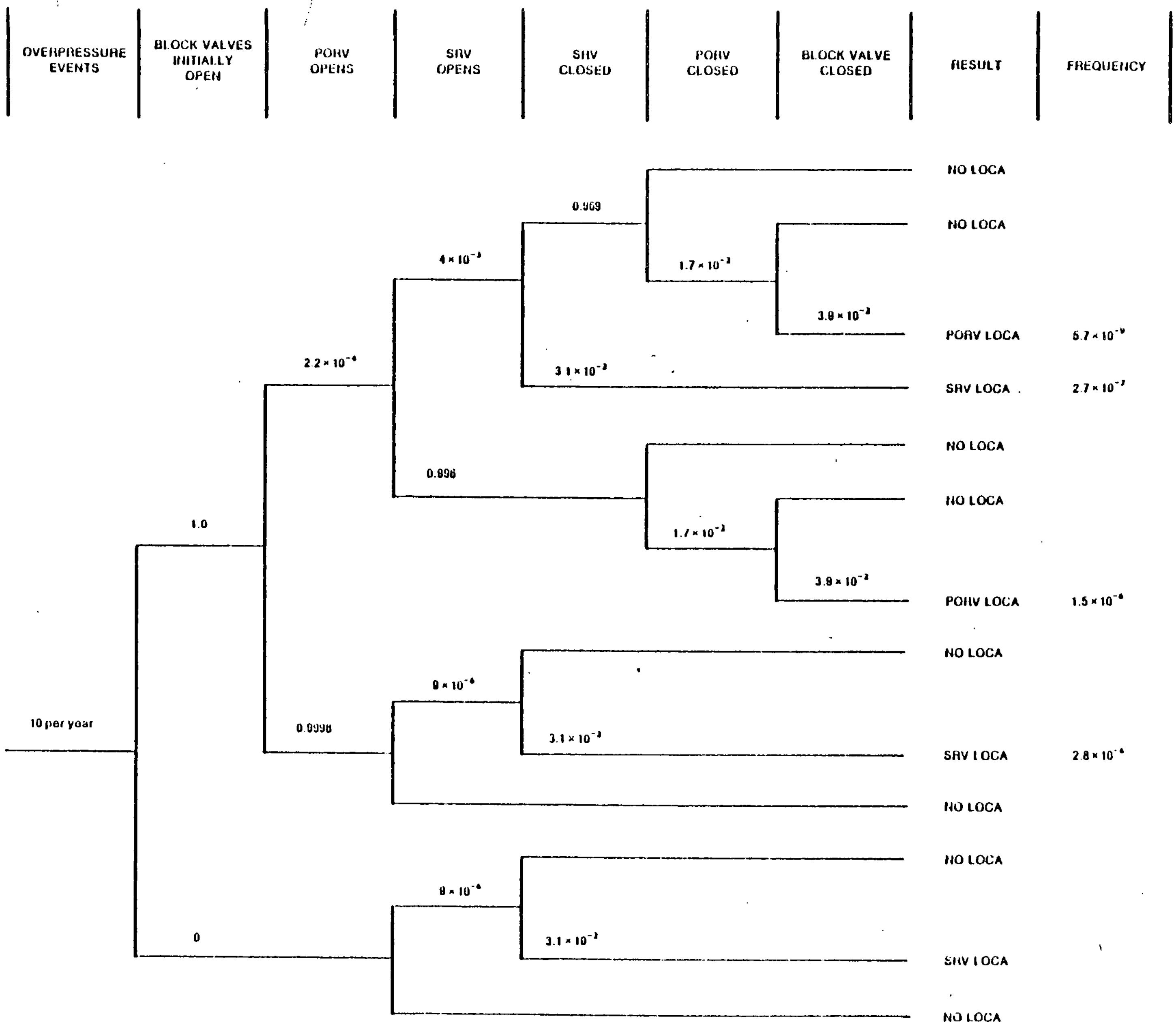

Figure 5. Overpressure Events - Theoretical Calculation Based on Normal Distribution of Setpoints (Block Valves Open)

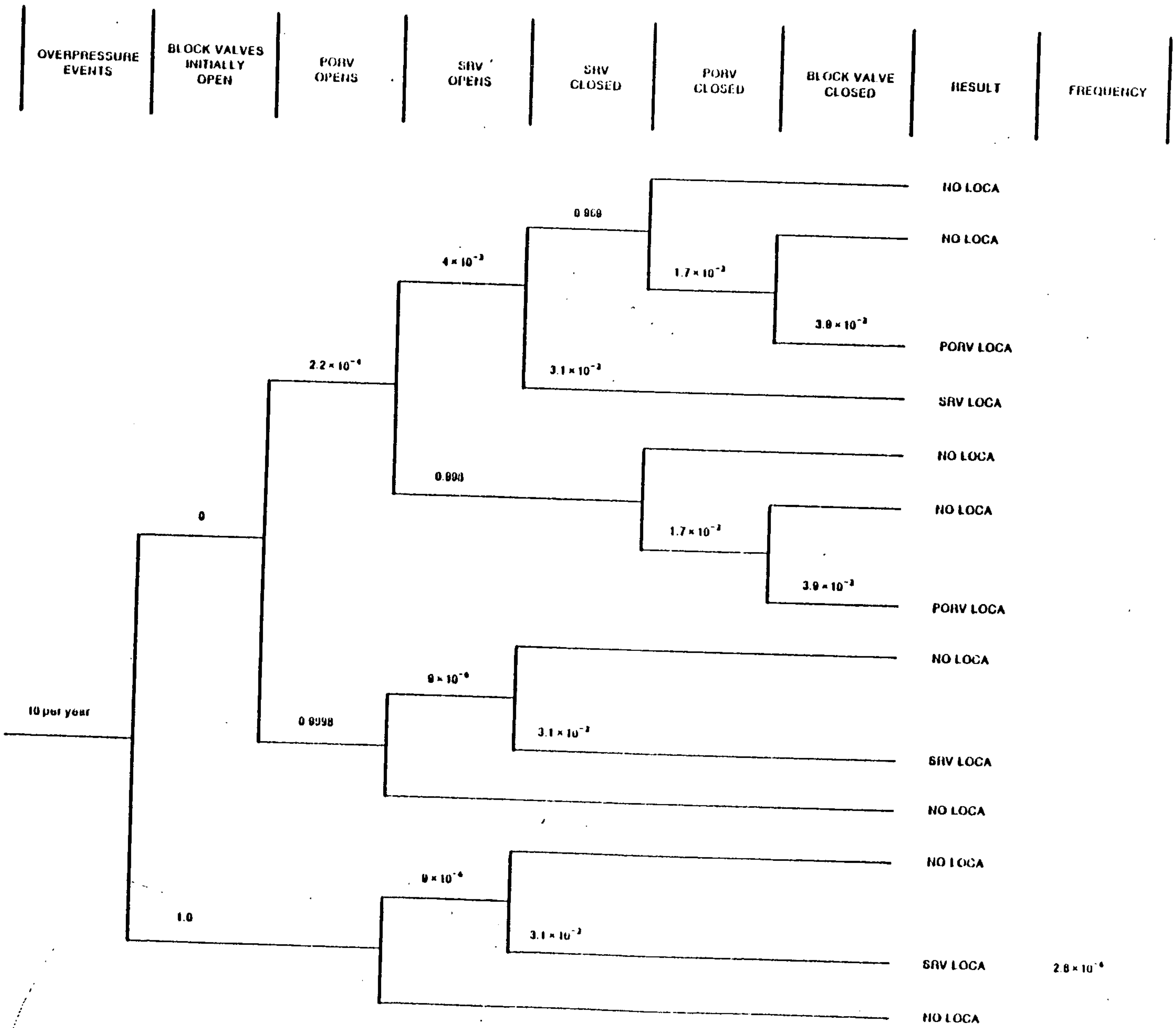

Figure 6. Overpressure Events – Theoret[illegible] [illegible]ulat[illegible]
Normal Distribution of Setpoints (B[illegible] V[illegible]ves C[illegible]d)

NOTES TO FIGURES 5 AND 6

Overpressure Events: 10 per year - Conservative B&W estimate (Reference 6)

Block Valves Initially Open: Yes (Figure 5)
No (Figure 6)

PORV Opens: Yes: 2.2×10^{-4} per demand

Determined by integrating (from negative to positive infinity), the expression of the normal distribution of the RPS trip pressure (P_{RPS}) times the probability that, for any RPS trip pressure, the PORV setpoint (P_{PORV}) is less than P_{RPS}. The probability that P_{PORV} is less than P_{RPS} for any given reactor trip was expressed as:

$$pr\left\{P_{PORV} < P_{RPS} \mid P_{RPS}\right\} = 1 - \frac{1}{\sqrt{2\pi}} \int_{\frac{P_{RPS} - \mu_{PORV}}{\sigma_{PORV}}}^{\infty} e^{-z^2/2}\, dz$$

where $z = \dfrac{P_{PORV} - \mu_{PORV}}{\sigma_{PORV}}$

SRV Opens (PORV Open): Yes: 4×10^{-3} per demand

Determined by integrating (from negative to positive infinity), the expression for the normal distribution of P_{RPS} times the probability that for any P_{RPS}, both P_{PORV} and P_{SRV} (the SRV setpoint) are less than P_{RPS}, divided by the probability of P_{PORV} being less than P_{RPS}, i.e.,:

$$\frac{pr\left[\left\{P_{PORV} < P_{RPS} \mid P_{RPS}\right\} \text{ and } \left\{P_{SRV} < P_{RPS} \mid P_{RPS}\right\}\right]}{pr\left\{P_{PORV} < P_{RPS} \mid P_{RPS}\right\}}$$

SRV Opens (No PORV Opening): Yes: 9×10^{-6} per demand

Determined similarly to the calculation of PORV opening (above), using the normal distribution for the SRV rather than the PORV.

SRV Closed: No: 3.1×10^{-2} per demand (Reference 6)

PORV Closed: No: 1.7×10^{-2} per demand (Reference 6)

Block Valve Closed: No: 3.9×10^{-2} per demand (Figure 2, page 2 of this report)

4. APPLICABILITY

4.1 APPLICABILITY OF THE B&W REPORT

In considering the applicability of the B&W report, two main factors need to be considered: (1) the type of PORV installed in the various B&W units and (2) the value of the revised setpoints at the different units. In the first instance, all B&W units have Dresser PORVs installed except for the Davis-Besse plant, which has a Crosby PORV installed. With regard to the revised setpoints, all B&W units have reset the PORV to 2450 psig with exception of the Davis-Besse plant, which is set at 2400 psig. Each of these factors is considered separately below.

4.1.1 Type of PORV

In Reference 4, B&W made the following statement regarding the type of PORVs installed at the different units:

> "Note that all plants except Davis-Besse (Crosby PORV) have Dresser valves; however, the entire B&W operating plant experience was used to arrive at a generic PORV sticking open probability as follows: There have been ten stuck open PORV events, five of which could be classified as mechanical failure of the PORV (the other five were basically installation errors). Using all of these five failures in determination of future frequency is considered conservative since two of the failures OC-3, 6/13/75 and CR-3, 11/75) were rectified by design changes, another (TMI-2, 3/28/79) cause is unknown. OC-2, 11/6/73 could be considered as a burn-in failure and the DB-1, 10/13/77 event is a Crosby valve. Using five failures in 250 demands results in a value of 2×10^{-2} to fail to reclose on demand. This value is considered conservative not only due to the inclusion of all five failures but also the number of demands is probably much higher than 250. There have been 148 documented PORV openings on reactor trips; however, there is not a listing of PORV demands when the reactor did not trip (e.g., ICS runback) nor is consideration given to transients that could have actuated the PORV numerous times during an event. The value of 250 demands is conservatively used here. An analysis was also performed to include values for other than mechanical failure that keep the PORV open. The results of this analysis is summed with the mechanical contributor (2×10^{-2}/d) to arrive at the value for failure to reclose on demand (2.1×10^{-2}/d)."

The above statement by B&W does not appear to address the rationale for developing a single failure rate for both the Dresser and Crosby valves. The implication is that because a conservative failure rate has been determined, it is representative of both of the valves concerned. This implication would be correct if the failure mechanisms and the failure characteristics of each of the valves are similar enough that the combined data adequately represent each valve individually. However, no information has been presented relative to such a conclusion.

The 148 documented PORV openings on reactor trips discussed by B&W are identified in NUREG-0667 [15]. A review of Reference 15 shows that 29 PORV openings on reactor trip occurred at the Davis-Besse plant prior to the TMI accident, one of which resulted in a PORV failure (October 13, 1977). In order to compare these data to the B&W-determined PORV failure rate of 2.1×10^{-2} per demand, however, the number of PORV openings must be increased by the factor 1.68 (250/148), which accounts for PORV openings that did not result in reactor trips. Consequently, 49 pre-TMI PORV openings at the Davis-Besse plant (29×1.68) must be considered. One failure in 49 openings yields a failure rate of 2.04×10^{-2} per demand. This value coincides closely with the B&W value determined using the combined Dresser and Crosby data. In view of the fact that B&W considers the 250 openings to be a conservative value, it is concluded that combining the Dresser and Crosby data and determining a single PORV failure rate is technically justified and that the B&W report is applicable to both the Dresser valves and the Crosby Valve at the Davis-Besse plant.

4.1.2 PORV Setpoints

In Reference 4, B&W determined that the probability of the PORV lifting during a loss of feedwater is approximatey 3.9×10^{-6} per overpressure event for PORVs with a setpoint of 2450 psig. For the Davis-Besse plant [16], this probability was determined to be approximately 3.9×10^{-3} per overpressure event, since the Davis-Besse plant has a PORV set at 2400 psig. Assuming 10 overpressure events per year (as assumed in Reference 6), the probability of the PORV lifting during a loss of feedwater is approximately 3.9×10^{-2} per reactor-year. Substituting this value into the determination of the PORV opening frequency shown on page 13 of this report, the PORV opening frequency for the Davis-Besse plant becomes:

Event	Per Reactor-Year
Overpressure transients	3.9×10^{-2}
Transients with delayed auxiliary feedwater	7.6×10^{-4}
Operator action under ATOG guidelines	1.58×10^{-2}
Instrumentation and control faults	1.7×10^{-3}
Overcooling transients that initiate HPI	8.4×10^{-4}*
Total	5.8×10^{-2}

Using the Davis-Besse PORV opening rate of 5.8×10^{-2} per reactor-year, the calculation of Figure 1 of this report was repeated. The resulting probability of a small-break LOCA from a stuck-open PORV was as follows:

All B&W units except Davis-Besse Figure 1	Davis-Besse Figure 7
1.4×10^{-4}/rx-yr	1.7×10^{-4}/rx-yr

Next, the calculation shown in Figure 2 of this report was repeated for a unit with a PORV setpoint of 2400 psig (e.g., Davis-Besse). In this case, a reactor trip setpoint of 2300 psig with an overshoot of 27 psig and a standard deviation of 27.52 psig was compared with a PORV with a 2400 psig setpoint and a 21.7 psig standard deviation. The analysis showed that the PORV would open with a frequency of 1.9×10^{-2} per event, given an overpressure transient causing a reactor trip. This value was then inserted into Figure 2 and the resulting small-break LOCA probability was determined as follows:

*Although not discussed by B&W, the Davis-Besse safety injection system is a low-head system and cannot challenge the PORV. Nevertheless, this transient is never a significant contributor to the total opening frequency.

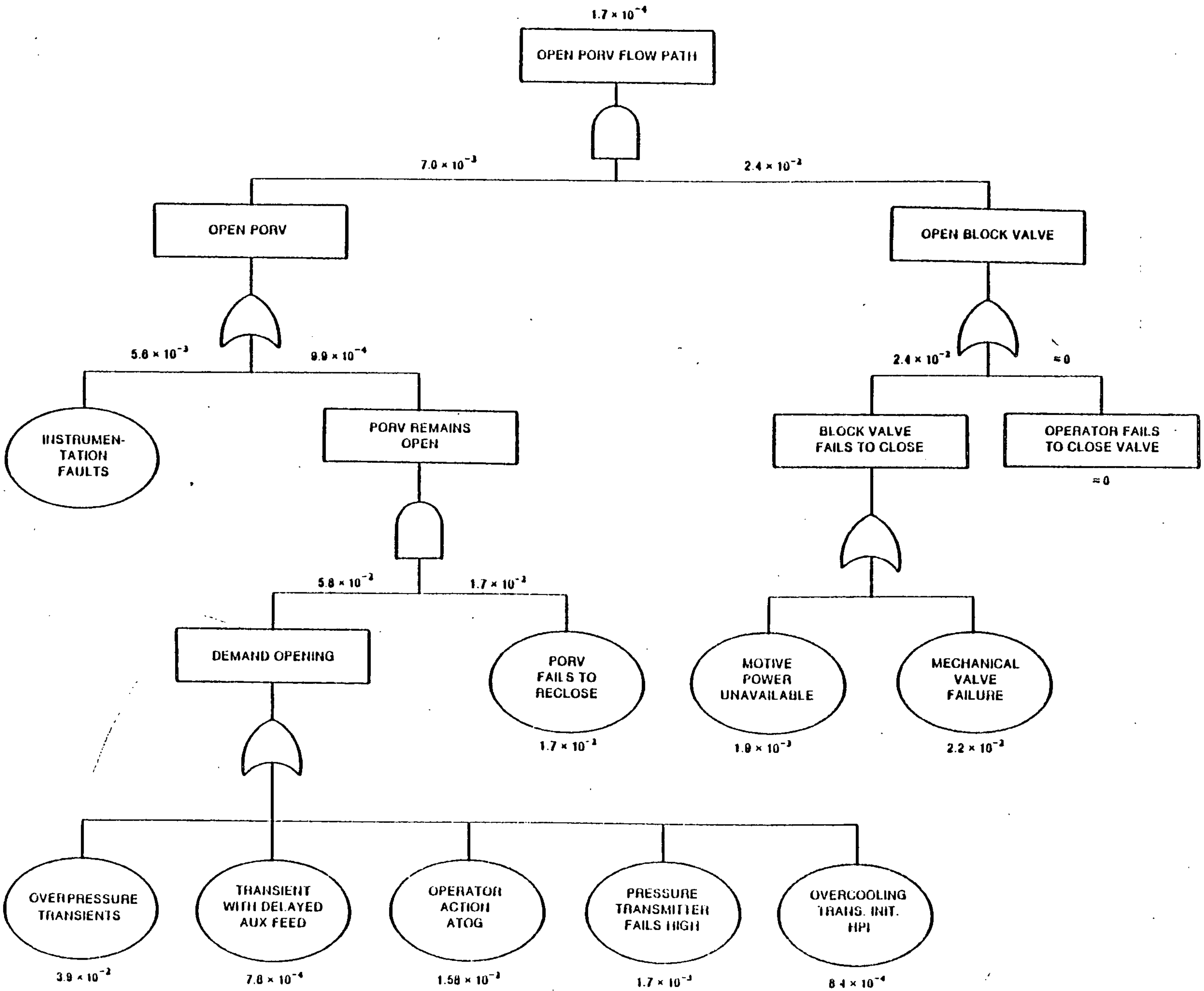

Figure 7. B&W Calculation - Davis-Besse

All B&W units except Davis-Besse Figure 2	Davis-Besse Figure 8
2.5×10^{-4}/rx-yr	3.7×10^{-4}/rx-yr

As a result of these calculations, it can be concluded that, although the probability of a small-break LOCA from a stuck-open PORV at the Davis-Besse plant is greater than the same probability at the other B&W units, the difference is not significant and is probably less than a factor of 2 larger. Furthermore, since all values determined are less than the median WASH-1400 value for small-break LOCA from all causes of 1×10^{-3} per reactor-year, the B&W report can be considered applicable to all B&W units, whether the PORV setpoint is 2450 psig or 2400 psig.

Finally, with regard to the safety-relief valves, the probability of a small-break LOCA from a stuck-open SRV at the Davis-Besse plant will be lower than the probability of the same event at the other B&W units. This is because at the Davis-Besse plant, the PORV opens sooner than at the other units, making it less likely that the SRV setpoint will be exceeded. Consequently, any determinations or conclusions relative to small-break LOCA from a stuck-open SRV are equally applicable to the Davis-Besse plant.

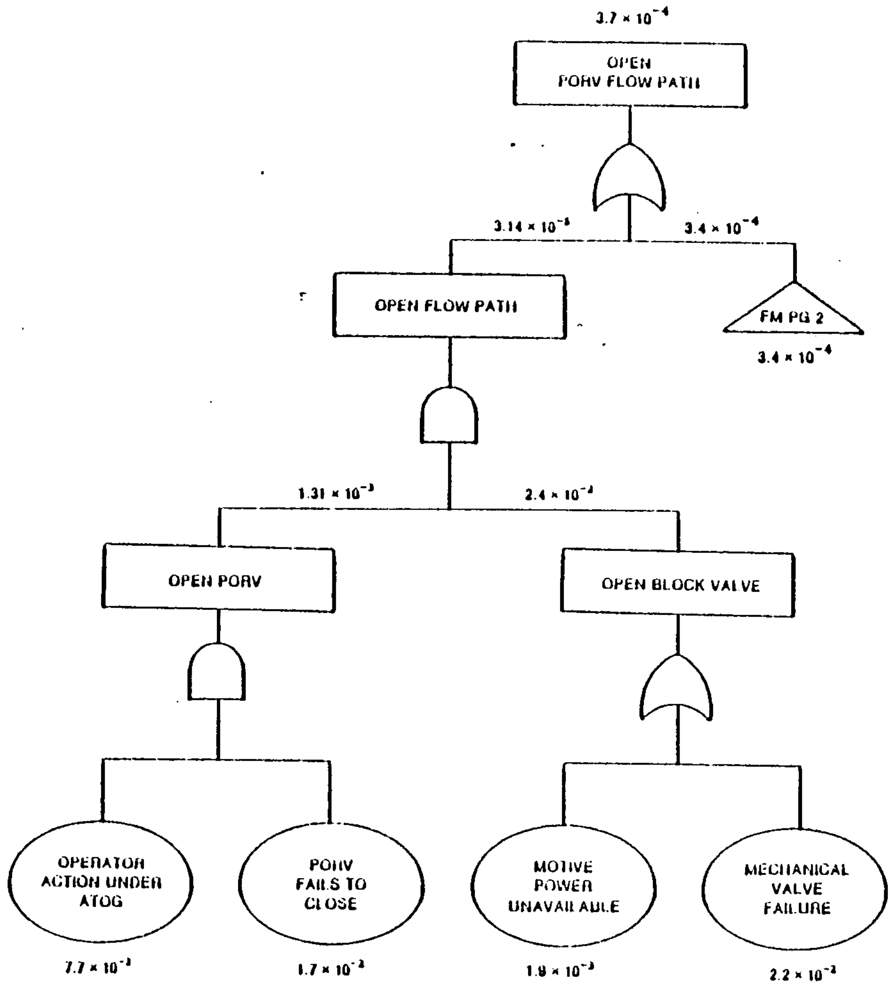

Figure 8. (Page 1) Revised Calculation - Davis-Besse

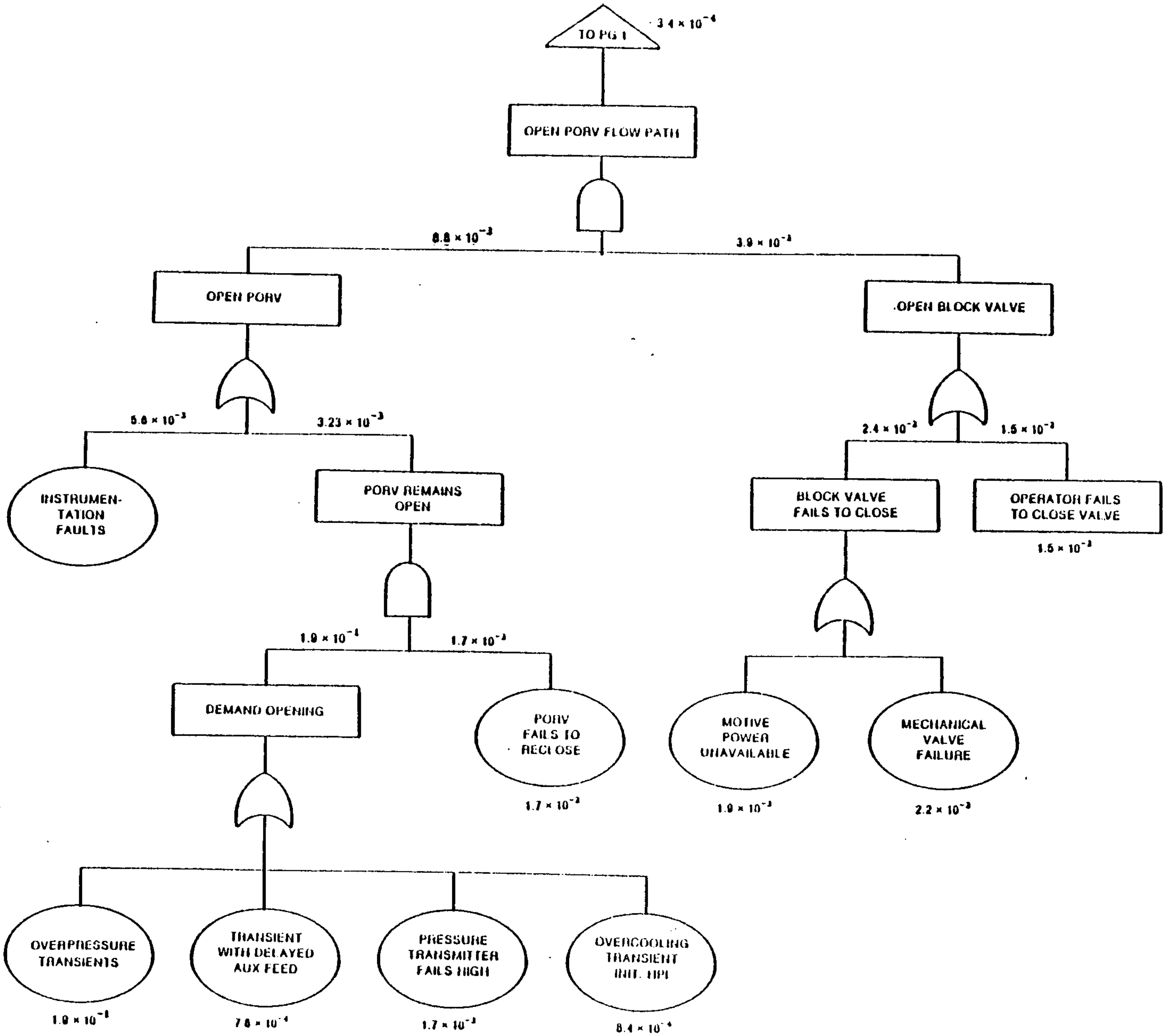

Figure 8. (Page 2) Revised Calculation - Davis-Besse

5. CONCLUSIONS

The conclusions that result from evaluation of the B&W report against the criteria of Section 2 are as follows:

- The B&W report documents the post-TMI modifications instituted at B&W-designed NSSS plants which have made a significant reduction in the expected frequency of a small-break LOCA from a stuck-open PORV or safety valve.

- The methods and results of the B&W report have been reviewed, and the expected frequency of a small-break LOCA from a stuck-open PORV or safety valve is less than the median WASH-1400 frequency for a small-break LOCA 1×10^{-3} per reactor-year. The following is a tabulation of the results of this evaluation:

Event	Recurrence Frequency (per reactor-year) B&W	Recurrence Frequency (per reactor-year) Independent Verification	Report Section
Small-break LOCA from stuck-open PORV (analytical calculation)	1.4×10^{-4}*	2.5×10^{-4}*	3.2.1
Small-break LOCA from stuck-open PORV (operating data)	4.7×10^{-4}**	1.4×10^{-4}*	3.2.2
Small-break LOCA from stuck-open SRV	2.6×10^{-5}	4.7×10^{-3}	3.2.2
		3.1×10^{-6}***	3.2.5

*Includes credit for operator action to shut block valve.
**No credit for operator action to shut block valve.
***Considers overpressure events only.

- Multiple PORV cycles on a single initiator event has been considered and is not significant for B&W units. The effect of 10 cycles per each applicable initiator event raises the probability of a small-break LOCA by about 3%.

- The percentage of PORV openings during overpressure transients is less than 2% of the total number of overpressure transients.

- Information used to verify the assumptions and calculations submitted by B&W has been taken from generic sources applicable to B&W-designed NSSS units. This report is considered applicable to all B&W plants, including Davis-Besse, which has a different PORV than other units and a lower PORV setpoint.

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