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RESPONSE TO NRC QUESTIONS

ON

PRESSURIZER RELIEF AND SAFETY VALVES

PROPRIETARY INFORMATION

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INTRODUCTION

This report is being submitted in order to satisfy the Nuclear Regulatory Commission's Request for Additional Information (RFI) on Arkansas Power and Light Company's (AP&L) response to NUREG-0737, Items II.K.3.2, "Report on Overall Safety Effect of Power-Operated Relief Valve (PORV) Isolation System," and II.K.3.7, "Evaluation of PORV Opening Probability During Overpressure Transient." Specifically, NUREG-0737 requested the following information/justifications:

1. II.K.3.2
   - Compile operational data regarding pressurizer safety valves to determine safety valve failure rates
   - Perform a probability analysis 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 ($<10^{-3}$ per reactor year), or whether the automatic PORV isolation system specified in Task Item II.K.3.1 is necessary.
2. II.K.3.7
   - Perform an analysis to assure that the frequency of PORV openings is less than 5% of the total number of overpressure transients.

In December 1980, a report (Ref. 10) was issued on behalf of all B&W operating plants which addressed the aforementioned concerns of NUREG-0737. Franklin Research Center was subcontracted by the NRC to review the B&W generic response. During the course of their review, Franklin has accumulated a list of items that require clarification before a final evaluation can be accomplished. The intent of this report is to provide clarification to Franklin's concerns and update the former response in light of more relevant information. Arkansas plant specific data was incorporated wherever possible.

The format of this report presents a listing of each of the questions and its associated response. A final section is included which contains the overall results and comments on the impact of any updates in this response.

Response to Item II.K.3.2

Question

1. A detailed description of the various actions taken to decrease the probability of a small-break LOCA caused by a stuck-open PORV, other than the revised high pressure reactor trip and PORV opening setpoints.

Response

1. 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 or PORV discharge.

Question

2. A calculation of safety valve failure rates based on past history of the operating plants designed by the NSSS vendors.

Response

2. Failure rates for the pressurizer safety valves (PSVs) can be ascertained by examining the failure rates of the main steam safety valves (MSSVs). This is possible because both operate on the same principle; i.e., they both work against the closing force of a spring, and they both require an additional sudden opening force when they reach their trip setpoints.

Differences between the PSV and MSSV must also be pointed out:

- The fluid passing through a PSV should contain fewer suspended particulates than that passing through an MSSV.
- The PSV is stainless steel whereas the MSSV is predominantly carbon steel. Rusting of the carbon steel will introduce additional foreign matter into the fluid.
- The PSV is an ASME Class I component, while the MSSV is an ASME Class II valve.
- The PSV must operate with a variable backpressure, while the MSSV operates with a fairly constant backpressure. As a result, the PSV design is more sophisticated and has more components that may fail.

The first three differences suggest that the PSV may have a lower failure rate than the MSSV, while the last point suggests the opposite.

Failure of the PSV was considered to be any instance where blowdown exceeded 35%. This corresponds to the low pressure ESFAS setpoint.

Cumulative B&W operating experience indicates that there have been approximately 2950 MSSV demands. In this MSSV history there has been one case where the blowdown exceeded 35%; however, the valve closed with less than 50% blowdown. Using this data the calculated failure rate for steam relief was found to be 3.39×10^{-4} per demand. The failure rate for water relief was estimated to be 100 times larger than for steam relief, i.e., 3.39×10^{-2} per demand.

The safety valve failure rate was determined using a Bayesian updating procedure. The prior distribution was assumed to be a lognormal with a mean of 3.39×10^{-2} per demand. This lognormal distribution was then combined with the evidence of five safety valve demands with no failures, to determine the probability of failure. The recent EPRI safety valve testing program* accounted for four of the demands. The Dresser safety valve model 31739A performed successfully for three water tests and one steam-to-water transition test. This valve also operated properly during a two phase/water relief at Crystal River on February 26, 1980. Incorporating these instances results in a PSV water discharge failure rate of 3.12×10^{-2} per demand.

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*T. Auble and J. Hosler, "EPRI PWR Safety and Relief Valve Test Program - Safety and Relief Valve Test Report," Research Project V102, Electric Power Research Institute, Palo Alto, California, April 1982.

Question

3. Analysis of the probability of a small break LOCA caused by a stuck-open safety valve.

Response

3. A small break LOCA due to a failed-open safety valve may occur along either of two pathways. The dominant pathways identified include overcooling with subsequent repressurization and overheating transients. However, no attempt was made to quantify the contribution due to overheating transients. This method was chosen because the existing auxiliary feedwater design is very reliable and, in the event of a total loss of feedwater, HPI feed along with some form of pressurizer bleed would be used to cool the core.

The probability of a small break LOCA due to an overcooling transient with subsequent repressurization is simply the product of three terms: the frequency of applicable overcooling transients, the probability of the operator failing to throttle HPI, and the probability of the safety valve failing to reseat. The operator failure probability is estimated to be 1.49×10^{-2} per demand. For the sake of conservatism the larger water discharge failure rate of 3.12×10^{-2} per demand will be implemented. In order to determine the frequency of overcooling transients a review of all B&W transients leading to reactor trip was conducted. From this a list was accumulated of all occurrences of low pressure ESFAS initiations. Some of these events can no longer occur due to plant modifications. These events were not considered. Out of the 392 reactor trips reviewed, three were currently applicable.

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. Incorporating these factors the resulting small break LOCA probability due to a stuck open safety valve is:

$$\left(\frac{3 \text{ events}}{392 \text{ rx-trips}}\right)\left(\frac{7.3 \text{ rx-trips}}{\text{yr}}\right)(1.49 \times 10^{-2})\ (3.12 \times 10^{-2}) = \frac{2.6\text{x}10^{-5}}{\text{yr}}$$

Question

4. An analysis of the effect of operating with the PORV blocking valve shut, except as required for depressurization under operating guidelines (e.g., steam generator tube rupture). In this analysis, examine the increased potential for causing a stuck open safety valve and the overall effect on safety (e.g., effect on other accidents).

Response

4. Plant operation with the PORV blocking valve closed is not a normal mode of operation. The blocking valve is normally only closed if there is a slight leakage through the PORV. Operation of the plant with the PORV blocking valve closed involves a trade-off between a decreased probability of a stuck-open PORV and an increased probability of a stuck-open safety valve. With the revised PORV setpoint (2450 psig), the majority of transients will cause the PORV to open will also cause the safety valves to open. Therefore, operation with the PORV blocking valve closed does not significantly impact the probability of a stuck-open safety valve.

Question

5. Further clarification of the references and the method used to determine the initiator frequency of 1.4×10^{-3} per reactor year for a PORV opening on a transient with delayed AFW.

Response

5. The initiator frequency of 1.4×10^{-3}/rx-yr for a PORV opening on a transient with delayed AFW was calculated with data from Ref. 1, 2, 6, and 7. However, more relevant data is available now, and these values will be redetermined.

The format used in the following calculation will simply be the unavailability of the aux feed system multiplied by the frequency of its corresponding initiating event. Aux feed system unavailabilities were obtained from Ref. 5 and are broken down into the standard three cases outlined in NUREG 0611. Obtaining the initiating event frequencies was fairly routine, except for the case of total loss of AC (LOAC). This value is calculated to be the probability of LOOP x probability both diesels fail. Both diesels failing was calculated to be failure of one diesel times a coupling factor. The revised PORV opening frequency due to delayed AFW will be obtained as the sum of the three cases. Table 5.1 references the probabilities that were used, while Table 5.2 outlines the calculational procedure which results in an initiator frequency of 7.6×10^{-4}/rx-yr.

TABLE 5.1

Event	Value	Reference
LMFW	2.0	6
LOOP	.15	4
Diesel	1.3×10^{-2}	7
Coupling factor	.1	8
$\overline{AFW}$/LMFW	3.4×10^{-4}	5
$\overline{AFW}$/LOOP	5.2×10^{-4}	5
$\overline{AFW}$/LOAC	1.4×10^{-2}	5

TABLE 5.2

Case	Initiating Event Frequency	AFW Unavailability	Contribution
LMFW	2.0	3.4×10^{-4}	6.8×10^{-4}
LOOP	.15	5.2×10^{-4}	7.8×10^{-5}
LOAC	$(.15)(1.3 \times 10^{-2})(.1)$	1.4×10^{-2}	2.7×10^{-6}

Σ 7.6×10^{-4}/rx-yr

Question

6. A justification of the estimated initiator frequency of 5×10^{-3} per reactor-year for a PORV opening due to instrumentation control faults.

Response

6. Six potential instrumentation related faults were considered that could produce an open PORV condition. These include faults in (1) power supplies, and in the signal processing equipment such as (2) pressure transmitter (3) Bistable (4) I/E converter (5) summer module in addition to (6) the control circuitry for the PORV itself. The first category, power supply faults was determined to be negligible. The failure mode of interest is failure of the power source such that an open signal is generated. Faults with both the ±24 VDC sources do not produce on open signal as the power sources are tripped. Faults with the pressure transmitter power supply are insignificant due to the fact that the sensor is a current mode generator with tight voltage specifications on power sources. The failure mode of no output, while significant, is not a mode of interest. The pressure transmitter could fail, but since the operator has the opportunity to switch to the other pressure transmitter channel, this fault is included elsewhere. The remaining modules that produce an "open" PORV signal if they fail in the high direction are listed below. Failure rates from Ref. 9 were used for this assessment.

Pressure Transmitter Fails High	$.250 \times 10^{-6}$
Bistable Functioned Without Signal	$.206 \times 10^{-6}$/hr
I/E Converted Fails High	$.310 \times 10^{-6}$/hr
Summer Module Fails High	$.310 \times 10^{-6}$/hr
Short Across Either of 2 N.O. Contacts	1×10^{-8}/hr

These faults may cause an open PORV anytime the plant is up, therefore the sum of these failure rates ($\sim .8 \times 10^{-6}$/hr) times the hours the plant is up per year ($\sim 8760 \times .8$) = 5.6×10^{-3}/Rx-yr. The value of 5×10^{-3} per reactor year was incorrectly summed with the other categories in the reference document, it should be treated separately as described in the discussion of overall results.

Question

7. A justified estimate of how many PORV openings (multiple openings per transient) could be expected with each initiator frequency group.

Response

7. This analysis generally assumed one opening per transient which results in a compact fault tree. For example, category 2 and 5 assumed one opening which is accurate for the most probable scenarios; however, less probable scenarios would have multiple lifts. In category 1 the highest frequency transient is turbine trip which produces one PORV demand. Category 3 uses different number of PORV demands based on whether offsite power is available or not. For category 4 it is assumed that the PORV cannot close therefore the number of demands is irrelevant. For the transients above that show 1 PORV lift, the value is based on operating experience. For transients where multiple lifts might occur (categories 2 and 5) the exact number of lifts cannot be accurately estimated, but a sensitivity estimate shows that even if the number of lifts were increased by a factor of 10 the probability of small break LOCA through the PORV is increased by only a factor of 2.1 percent.

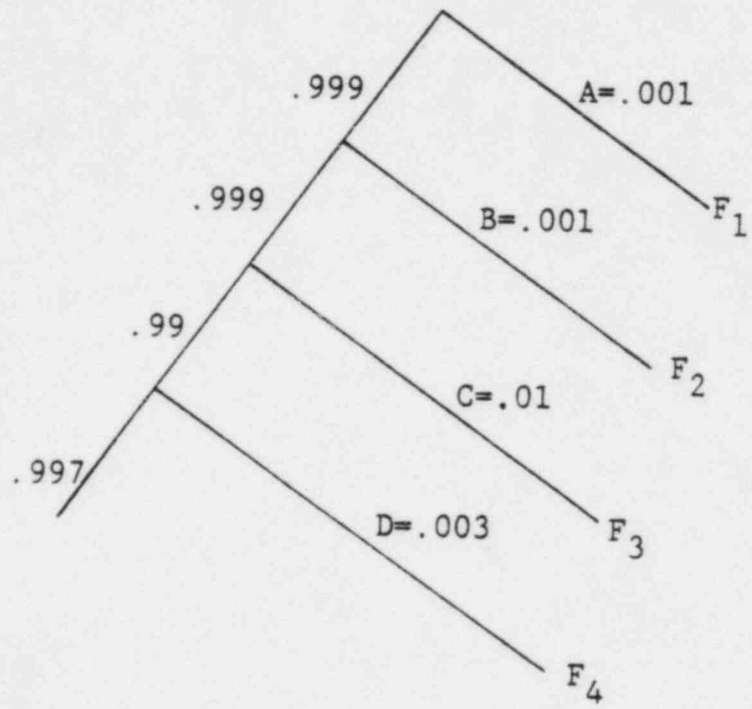
Question

8. A justification of the numbers used to determine the initiator frequency of 1.8×10^{-3} per reactor-year from PORV opening of overcooling transients that initiate high pressure injection (HPI) and result in an overpressure condition when the operator fails to throttle or terminate HPI.

Response

8. 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 overcooling. 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 $3/392 \times 7.3 \times 1.5 \times 10^{-2} = 8.4 \times 10^{-4}$/rx-yr.

HPITHROC - Operator fails to throttle HPI



$P(F) = F_1 + F_2 + F_3 + F_4$

$P(F) = 1.49 \times 10^{-2}$

"A" = Operator fails to realize ESFAS initiates HPI pumps (Table 20-3).*

"B" = Fails to resume attention to legend light (Table 20-3).

"C" = Fails to recognize the return of pressurizer level on ATOG scope (Table 20-5).

"D" = Fails to throttle HPI and realign normal make-up (Table 20-13).

*Note: Tables identified are from NUREG/CR 1278.

Question

9. A justification of the estimate of 250 demands used to determine the mechanical contributor to the rate of failure of the PORV to close on demand.

Response

9. The Dresser PORV is used at ANO-1. The cumulative experience of the B&W operating plants using the valve reveals 127 recorded reactor trips. Due to the setpoint values prior to the TMI accident, each "normal" trip produced one demand. Unusual trips produce multiple demands. Additionally there have been plant upsets in which the PORV functioned as designed even though a reactor trip was precluded. Because the total number of PORV demands is not recorded, B&W operating personnel estimated the number of demands. To be conservative the lower bound of these estimates, 250, was used. There have also been 38 demands (0 failures) from the CE operating plant experience and 27 demands (0 failures) in the ERPI valve test program on this Dresser PORV which if combined with the B&W operating plants' known experience would produce 192 (127 + 38 + 27) recorded demands. We believe that the expected total number of demands is about 400; 250 is conservatively used here. The failure probability to reclose is 4/250 or 1.6×10^{-2}/demand.

Question

10. An explanation of the analysis performed to arrive at the non-mechanical contributor to PORV failure rate of 1×10^{-3} per demand.

Response

10. The non-mechanical contribution to PORV failure consists of control circuitry and solenoid related faults. Four potential fault conditions were identified in this area that could lead to a stuck open PORV. One fault was determined to have negligible probability: failure of the pressure transmitter (and/or sensor) to change with a change in the process variable. This would cause an open PORV if this failure mode occurred in the short time after the PORV had opened. However, if the PORV were to have opened, the transmitter (sensor) would have been operating correctly up to that point and a failure in the short time is highly unlikely. This is a random failure with no identified causal relationships involved. The probability therefore of a random failure in the time interval from opening to demand for closure is insignificant.

The other three fault conditions are: (1) Bistable fails to operate when signalled, (2) short across a normally closed contact, (3) solenoid fails to deenergize on demand. The unavailability due to the first two faults was calculated by failure rate x time between tests. The third value was derived on a per demand basis. The failure rate method gives the more conservative results because the PORV circuitray is tested during every startup. The average number of startups per year at ANO-1 has averaged about 7/yr since commercial operation. During the last three years it has been 6, and in the most recent year three startups have occurred. Using the most conservative value here (3) the average time between operability verification is (8760)/(2)(3). A capacity factor of 0.8 is applied to obtain a conservative but realistic time between tests. Failure rates for the bistable and solenoid were obtained from Ref. 9, while the probability of a short was taken from Ref. 8. The values are summarized below:

Bistable $\frac{(.822\times10^{-6})(8760)(.8)}{(2)(3)} = 9.6 \times 10^{-4}$

NC Contact $\frac{(1\times10^{-8})(8760)(.8)}{(2)(3)} = 1.2 \times 10^{-5}$

Solenoid $= \frac{4\times10^{-6}/d}{\sum 9.76\times10^{-4} \approx 1\times10^{-3}/d}$

Question

11. A detailed analysis and justification of the estimate that one PORV demand opening event could occur in 45 years of B&W plant operation. Include a detailed description of the specific plant reconfigurations that have upgraded the AFW system, the control circuitry of the PORV, the NNI power sources, and the AC power sources that have contributed to the aforementioned initiating frequency estimate.

Response

11. 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 2.22×10^{-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. Refer to response for question #1 for a brief discussion of plant changes.

Question

12. A discussion of the applicability fo the B&W generic report to ANO-1 design.

Response

12. The report is applicable. Each B&W plant has a slightly different trip profile; i.e. causes for trips are somewhat different. For ANO-1 the profile shows that a loss-of-offsite power is a larger component than at other sites, although the average number of plant trips for any reason is not significantly different. The loss-of-offsite power challenges the auxiliary feedwater system, but the upgraded control system will limit PORV lifts due to overcooling induced transients.

Response to Item II.K.3.7

Question

1. A more detailed and extensive analysis which demonstrates 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 trips. The analysis provided should document that the PORV will open in less than 5% of all anticipated transients using revised setpoints and anticipatory trips for the range of plant conditions which might occur during a fuel cycle. The analysis provided should identify the FSAR analytical assumptions used.

Response

1. The intent of this question, namely that less than 5% of all anticipated transients lift the PORV, has been adequately addressed throughout this report. Use of FSAR assumptions are not generally incorporated into probabilistic analyses of this nature. For example, one assumption used in FSAR analyses, to increase the decay heat to 120% of the ANS curve is obviously not pertinent because the probability of operating at this level is infinitesimal. Another assumption would be to use BOL parameters; for this case the probability analysis could reflect the probability that the plant would be operating for the fraction of time represented by BOL. The FSAR analysis uses other such conservative assumptions, and usually those assumptions are compounded. The probability of compounding several conservative assumptions in combination with the probability of initiating events is too small to justify analytical predictions.

   The question "single failures in mitigating systems" is not clear. The report addresses failures of components, single and otherwise that affect the PORV challenge rate. Further clarification would be needed on this concern as to exactly what failures in which systems are contemplated. Note that failures in systems such as auxiliary feedwater have already been included in the report.

Transients which do not actuate the anticipatory trips are the ones that have been addressed in the report. Those transients that activate anticipatory trips will not reach the PORV setpoint.

In summary, less than 5% of all transients reach the PORV setpoint with any set of reasonable assumptions concerning plant status.

Questions

2a. The basis for including only LOFW and turbine trip anticipated transients instead of all possible overpressure initiators.

2g. The method used to determine that the probability of the PORV opening during an overpressure transient is 3.9×10^{-6} per reactor year. Specifically, identify the number of overpressure transients per reactor-year used in the analysis.

Responses

2a. &g. The basis for using the LOFW transients is due to its pressure response. The pressure response resulting from LOFW envelopes the pressure response of other anticipated transients. Anticipated transients are defined consistent with the PC-2 category of ANS 51.1 which includes loss of external electrical load, loss of condenser vacuum, inadvertent closure of main steam isolation valves, inadvertent boron dilution. There is one anticipated transient - inadvertent control assembly group withdrawal - which is less probable than LOFW; its pressure response is comparable to the LOFW pressure response if expected operating conditions are assumed. The reasoning behind using LOFW was to calculate the probability of the most severe pressure response reaching the PORV setpoint as the less severe pressure responses would necessarily have a lower probability of reaching the same setpoint.

It has been calculated that the probability of opening the PORV by LOFW without anticipatory trips is approximately 3.9×10^{-6} per transient; therefore other overpressure transients have an even smaller probability. Note: The report incorrectly states 3.9×10^{-6}/reactor year. Note that with the incorporation of anticipatory trips the LOFW probability is less than the value given above. Even making the conservative assumptions that all overpressure transients have a probability of 3.9×10^{-6}/transient and that there are 10 transients per year results in a value of approximately 4×10^{-5}/reactor year. In summary, LOFW was selected as a bounding transient.

Question

2b. A justification of the use of a standard deviation of 1.4 psi for the high pressure reactor trip setpoint of 2300 psig and the PORV opening setpoint of 2450 psig.

Response

2b. 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 $\sqrt{\sum_{i=1}^{N} [(\text{Accuracy})(\text{Range})]^2}$. 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 inverted (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 $\sqrt{(\partial)^2 + (\partial)^2 + (27.52)^2}$ 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.

Question

2c. A justification of the use of a constant 17.4 psi pressure correction to the rollover data and a description of the method by which the 17.4 constant was calculated.

Response

2c. A pressure correction to the rollover data (identified by bias in the reference report) was needed to adjust the operating plant data because the setpoints are now reversed. The data obtained with pre-TMI setpoints will shows a "faster" rollover because the PORV opens before reactor trip. Post-TMI setpoints will show a "slower" rollover because the PORV does not open. Since no post-TMI operating plant data was available, an adjustment had to be made. The data source was a single plant transient with the PORV block valve closed with a pressure trace available. The CADD computer code was benchmarked to this pressure transient. This resulted in a correction factor of 17.4 psi which is subtracted from the difference between PORV and RPS trip setpoints (i.e. reduces the range of difference making it more likely to actuate PORV on any given RC trip).

The mean of the rollover data has been calculated to be 9.2 psi with a standard deviation of 27.5 psi. The supporting calculations and W test* follow. The data given in Table 2.1-1 is also plotted in the attached figure. The parameters of the data were calculated with the W value of .97756 supporting the assumption of normality.

*ANSI N15.15-1974 Assessment of the Assumption of Normality

$$S^2 = \sum_{i=1}^{26} (X_i - \bar{x})^2 = 12934.62$$

$k=13$

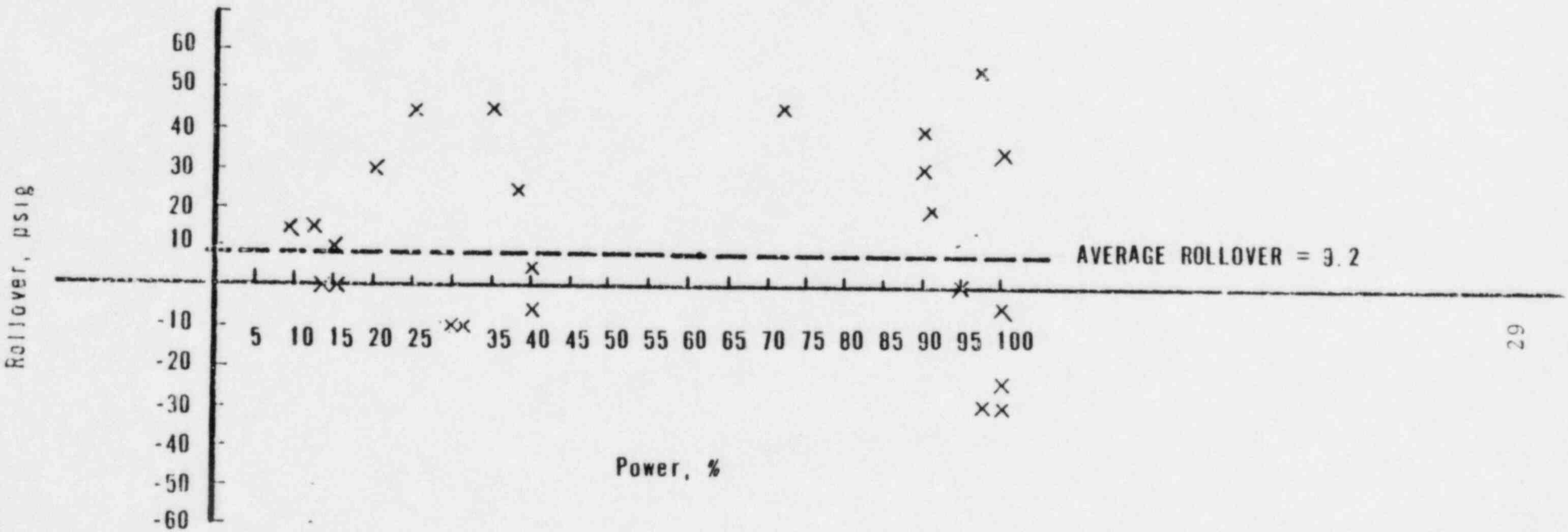
$$b = \sum_{i=1}^{k} A_{n-i+1}(X_{n=i+1} - X_i) = 136.05$$

$$W = \frac{b^2}{S^2} = .97756$$

Coefficients used in the W test for normality with sample size 26.

A_{n-i+1}
.4407
.3043
.2533
.2151
.1836
.1563
.1316
.1089
.0876
.0672
.0476
.0284
.0094

ROLLOVER DATA



Question

2d. The statistical method used to conclude that with a 99% confidence at least 99.99% of all LOFW and turbine trip high pressure transients will not open the PORV set at 2450 psig.

Response

2d. The statement that there is 99% confidence that 99.99% of all LOFW and turbine trip high pressure transients will not open the PORV if set at 2450 is obtained by assuming all distributions used in the simulation are the actual distributions (or constants, as appropriate) and with the stated parameter values for means and variances (standard deviations). Actually, the confidence level is unnecessary since distribution parameters are assumed known, and the tolerance statement is effectively just a simple statement about a distribution. In the report, the basic relation can still be expressed in terms of the variables used in the simulation, i.e.,

Result = PORV - RPS - EXCESS - BIAS

and if Result is less than or equal to zero, the valve will be opened.

With the assumption, the mean of the distribution is 123.37 and the variance is (27.59)*(27.59). Using a standard technique to determine the probability of a random selection from this distribution being less than or equal to 0.0 is determined by computing

$$Z = \frac{0.0-123.37}{27.59} = -4.47 = K.$$

$$\text{Probability } (Z \leq -4.47] \cong 3.9 \times 10^{-6}$$

The write-up implies that if there is one (1) demand due to these transients in a reactor year, then at lest 99.99 of the situations will not result in the PORV opening. This says that the proportion of the population less than -4.47 standard deviations from the mean is 1 - 0.9999 (at most), or at least 99.99 percent of the transients will not open the PORV. The actual percentage is 0.9999961. The above does relate to a single event and on the normality, independence, and known parameter value assumptions.

Question

2e. A discussion of the method employed to determine that three of the past PORV actuations would have lifted the PORV with the revised setpoints.

Response

2e. The primary method used to screen historical data for potential PORV actuations at 2450 was to identify any safety valve lifts since the nominal pressure for pressurizer safety lift is 2500. There have been 2 pressurizer safety valve lifts, one at Crystal River on 26 Feb. 80, the other at Rancho Seco on 20 March 78. The pressurizer safety valve setting at Rancho Seco was low (may have been approximately 2400); however, the exact value is unknown and this event was counted as one that could have lifted the PORV. In addition to those transients that have lifted safety valves all available pressure traces on reactor trip data were analyzed. The Oconee 3 transient of 4/30/75 was the only other identified transient. It indicated that RC pressure may have reached the 2440 range and this event was also counted as a potential PORV actuation.

Question

2f. A detailed description of the modifications incorporated into subsequent plant designs which formed that conclusion that these three PORV openings would have been precluded, given the same initiating events. The response should include a discussion of whether or not these modifications have been implemented at ANO-1.

Response

2f. All three transients designated as potential PORV actuations would have not occurred had present plant modifications such as at ANO-1 had been in place. Refer to question #1 (II.K.3.2).

Discussion of Overall Results

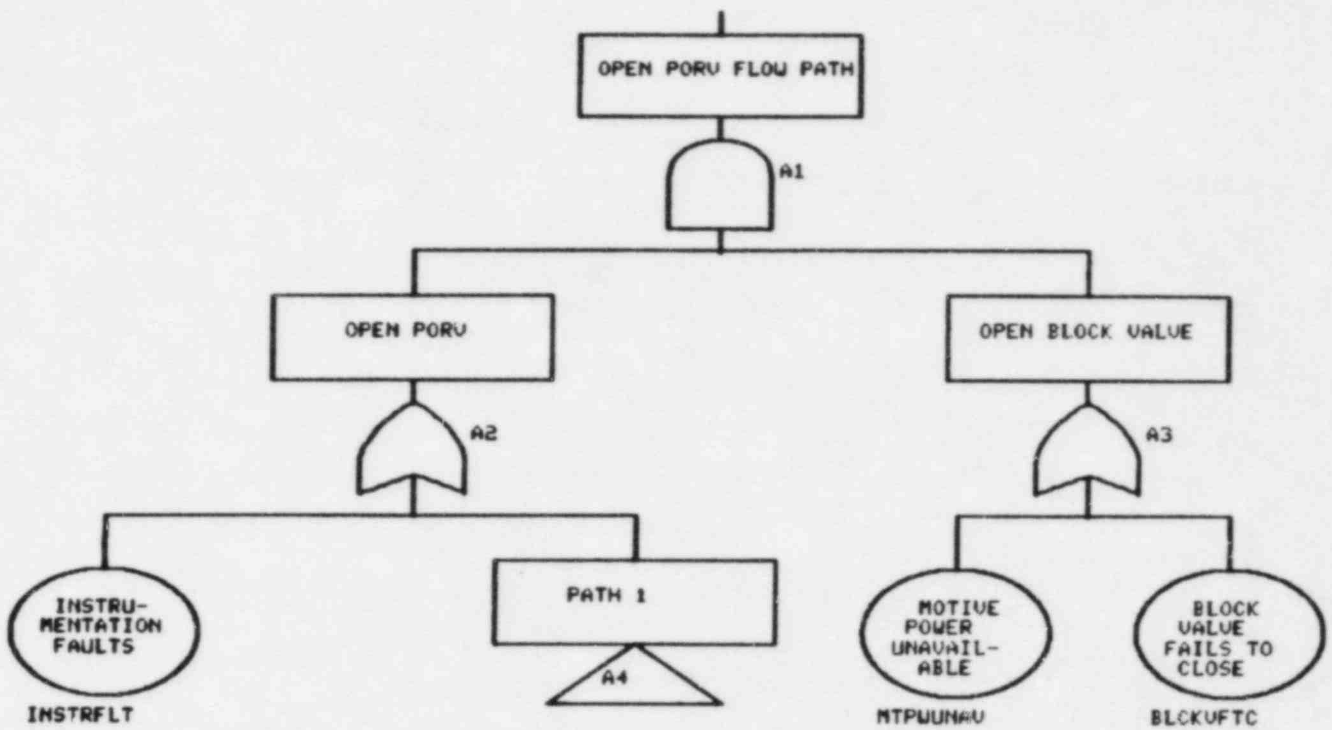
The probability of an open PORV flow path is the product of a stuck open PORV times the probability of the block valve failing to close when required.

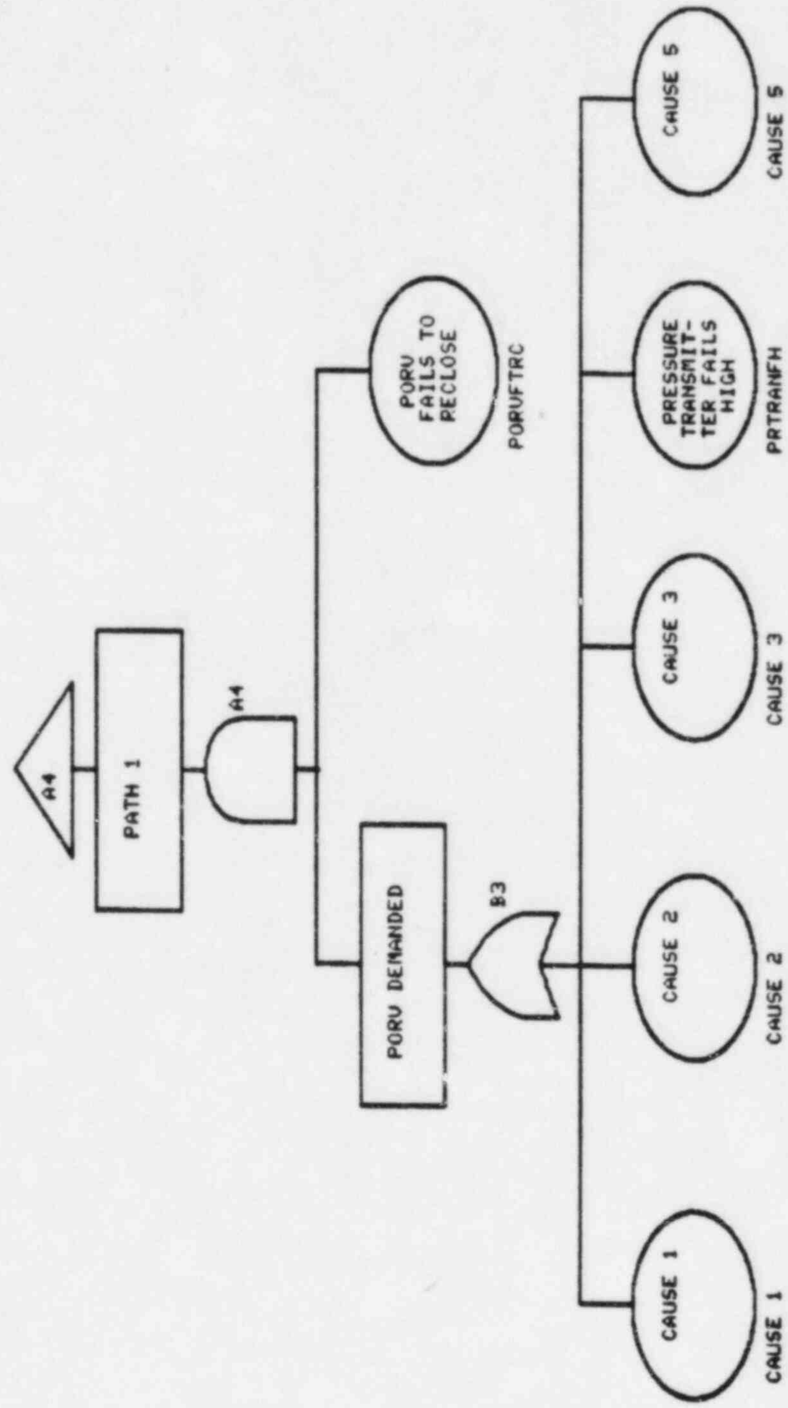
The probability of a stuck open PORV is the sum of two contributor paths. The first path is the product of the sum of PORV demands from causes 1, 2, 3, 5 and the frequency of the pressure transmitter failing high (part of category 4) times the probability of failure to close. The second path consists of the rest of the 4th category, opening due to instrumentation faults. It is assumed here that these faults will keep the valve open.

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 the 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 Ref. 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.

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.

OPEN PORV FLOW PATH




A4
PATH 1
A4
PORV DEMANDED
B3
PORV FAILS TO RECLOSE
PORVFTRC
CAUSE 1
CAUSE 1
CAUSE 2
CAUSE 2
CAUSE 3
CAUSE 3
PRESSURE TRANSMIT-TER FAILS HIGH
PRTRANFH
CAUSE 5
CAUSE 5

FAILURE RATE DATA

CODE	UNAVAILABILITY
INSTRFLT	5.6×10^{-3}
MTPWUNAV	$(.15)(1.3\text{x}10^{-2}) = 1.95 \times 10^{-3}$
BLCKVFTC	2.22×10^{-2}
PORVFTRC	$(4/250) + (1 \times 10^{-3}) = 1.7 \times 10^{-2}$
CAUSE 1	$(10)(3.9 \times 10^{-6}) = 3.9 \times 10^{-5}$*
CAUSE 2	7.6×10^{-4}
CAUSE 3	1.58×10^{-2}
PRTRANFH	$(.25 \times 10^{-6})(.8)(8760) = 1.7 \times 10^{-3}$
CAUSE 5	8.4×10^{-4}

*This assumed 10 overpressure events a year

SENSITIVITY OF MULTIPLE PORV CHALLENGES

CASE	SUM OF IMPLICANTS	% IMPACT
Nominal Value	1.43×10^{-4}	-
10* (CAUSE 2)	$\sim 1.46 \times 10^{-4}$	2.1
10* (CAUSE 5)	$\sim 1.46 \times 10^{-4}$	2.1

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6) B&W Proprietary Data.

7) AP&L Proprietary Data.

8) Reactor Safety Study, NUREG-75/014 (WASH-1400).

9) IEEE Guide to the Collection and Presentation of Electrical, Electronic, and Sensing Component Reliability Data for Nuclear Power Generating Stations), IEEE Std. 500-1977.

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11) Data Summaries of Licensee Events Reports of Valves at U.S. Commercial Nuclear Power Plants, NUREG/CR-1363, W. H. Hubble, et. al., June 1980.

12) Reliability Availability Data Collection and Analysis System, NPGD-TM-597, Babcock & Wilcox, Lynchburg, Virginia, March 1982.