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A PRESSURIZED THERMAL SHOCK EVALUATION OF THE H. B. ROBINSON UNIT 2 NUCLEAR POWER PLANT

Appendix B: Pressurized Thermal Shock Initiating Event Frequency and Branch Probability Screening Estimates for the H. B. Robinson Unit 2 Nuclear Power Station

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B.0. PRESSURIZED THERMAL SHOCK INITIATING EVENT FREQUENCY AND BRANCH PROBABILITY SCREENING ESTIMATES H. B. ROBINSON UNIT 2 NUCLEAR POWER STATION

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PRESSURIZED THERMAL SHOCK INITIATING EVENT FREQUENCY AND BRANCH PROBABILITY SCREENING ESTIMATES H.B. ROBINSON UNIT 2 NUCLEAR POWER STATION

B.1. Introduction

Initiating event frequency and event tree branch probability estimates have been developed for use in quantifying event sequences in the Robinson 2 pressurized thermal shock evaluation. These estimates have been developed for initiators and system/component failures specified by ORNL.

The complete LER data base for Robinson 2 was reviewed for initiating event occurrences and system failures, as well as for a general overview of the performance of plant systems of interest. However, in lieu of relying solely on Robinson information, Westinghouse-specific and PWR-specific operational information was employed when available and when it was considered that Robinson operational experience did not provide an adequate data base. Additional information was obtained from the NREP Generic Data Base (Ref. 1), the Nuclear Power Plant Operating Experience Summaries (Ref. 2, 3, 4 and 5), as well as other sources. With the constraints imposed by programmatic needs and the availability of operational data, only simplified approaches to frequency and probability estimation were permitted. The estimates are, however, considered acceptable for use as screening estimates. Table 1 includes the estimates developed, the rationale used, relevant information, and information sources.

As stated above, a number of the estimates included in Table 1 have been developed from generic sources. This is necessary, since many of the failures of interest are sufficiently infrequent that they will only be seen (if at all) over a large operating period. The estimates may not be representative of Robinson failure probabilities if Robinson systems and components differ significantly from systems and components used throughout the industry; although potential differences have been considered in developing the estimates in Table 1.

A number of initiating transients have been found to be of significance from previous pressurized thermal shock analyses. In general these include three initiator classes: (1) reactor trip; (2) steam line break (SLB); and, (3) loss of coolant accident (LOCA), including steam generator tube ruptures. Several LOCA and SLB situations are of interest - whether a break is small or large; whether it is isolable or nonisolable; and whether the plant is at full power or at hot standby. Although separate event trees may be appropriate to describe all these situations, many of the plant responses of interest are expected to be common among the trees. Also, considering the amount of data available, the frequency estimate developed for one of the initiating events is sometimes an appropriate estimate for others.

B.2. Initiating Event Frequency Estimates

Initiating event frequencies have been developed based on the number of observed events within selected periods of operation. The calculational method is consistent with that developed in Ref. 9, and utilized the χ^2 distribution to estimate a conservative lower bound on mean time between failures, and hence a conservative upper bound on frequency. This frequency estimate is $\chi^2_{1-\alpha}(2r+2)/2T$, where $1-\alpha$ is the confidence level, r the number of observed failures, and T the total observation time. A 50% confidence level was employed.

For some initiators, it may be necessary to estimate the frequency of events in a particular operating mode. The 1980, 1981 and 1982 operating experience of Robinson 2 identified in References 4 and 5 was reviewed to estimate the fraction of time the units were at power and in hot shutdown and cold shutdown modes. The fractions of time in these three states were: power operation, 61.3%; hot shutdown, 1.2%; and cold shutdown, 37.5%.

For medium and large steam line breaks and medium break LOCAs, no data exists as to the relative incidence of these initiators at power and hot shutdown; and fractions based on the amount of time in hot shutdown and at power were used. These fractions are: power operation, 98.1%; hot shutdown, 1.9%.

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Historic small steam line breaks and small LOCAs identified in reference 6 and 7 were reviewed to estimate the fraction of these initiators occurring at hot shutdown and at power. For small steam line breaks, 25% occurred at hot shutdown and 75% occurred at power. For small break LOCAs, 9% occurred at hot shutdown and 91% occurred at power.

B.3. Branch Failure Probability Estimates

Branch failure probability estimates on a per-demand basis were developed using the effective number of failures observed within a period of time and estimating the number of demands expected within that same period.** If no failures on demand were observed, and no other information was available with which to estimate a failure-on-demand probability, then a Poisson approximation of a Binomial process (the number of demands was always large) was assumed applicable and the probability estimated by assuming there was a 50% probability of observing the zero failures actually observed. In such a case,

$P(r = 0) = e^{-m} (m)^0/0!$, where r is the number of failures and m the expected number of failures.

The expected number of failures, m, is equal to the probability of failure (p) multiplied by the number of demands (D). If the probability of zero observations is 0.5, then $P(r = 0) = 0.5 = e^{-m} = e^{-pD}$. If an estimate of D is available, $p \approx .7/D$. (It is interesting to note that the initiating event frequency estimate reduces to ~ .7/T for zero observed events.)

**Failure-per-demand probability estimates developed primarily from test demands may overestimate the actual failure probability up to a factor of two if the actual failures are time dependent and the test demands are spaced at regular intervals. However, based on events documented in the ASP program (Ref. 6 and 7), there appears to exist in many cases a greater likelihood of failure on demand following an actual initiating event than that determined based on testing. These two effects tend to offset one another; the per-demand estimates developed herein are considered acceptable for screening purposes.

In estimating the likelihood of multiple valve failures, conditional probabilities of subsequent component failure given failure of the first component were developed based on the multiple failure rates identified in Reference 10. These estimates are:

- o For air-operated valves in a system of three valves; 0.094 for a specific second valve failing given the first is failed, and 0.081 for both remaining valves failing given the first is failed.
- o For motor-operated valves in a system of three valves, 0.020 for a specific second valve failing given the first is failed, and 0.012 for both remaining valves failing given the first is failed.

As with all event trees, the probability associated with a particular branch is conditional on the prior branches in the sequence. Although event tree development was not in the scope of this phase of the work, certain conditionalities were accounted for when appropriate. Questions of conditionality and potential system interaction effects (which are being considered separately) must be carefully considered prior to the use of Table 1 estimates with a particular event tree. In addition, quantification of human error was not in the scope of the study, and many of the estimates included in Table 1 do not consider plant-specific potential operator recovery actions.

It should also be noted that, for traceability, numerical values included in Table 1 have been developed to two significant figures. This is not to imply a lack of error bands on the estimates. The error bands associated with many of the estimates are expected to be large - at least an order of magnitude in either direction considering the generic nature of much of the data base and the small amount of information on particular initiators and multiple component failures available.

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Table 1. Event Tree Frequencies and Branch Probabilities for Screening Purposes

Function	Discussion	Screening Estimate
Initiators		
1) Reactor Trip	During 1980, 1981, and 1982 Robinson 2 experienced 26 manual and auto scrams from power (Ref. 4, Ref. 5). This results in a reactor trip estimate of 26/3 years = 8.7/yr.	8.7/ry
2) Steam Line Break		
a) Large Break	Two early events of potential importance to steam line break frequency have been recorded in the LER data: 1) Turkey Point Safety Valve Header Failure 2) Robinson Safety Valve Header Failure. Both of these events occurred before criticality and in view of the fact that no large breaks have been observed in the 577 combined BWR and PWR years of post-critical operation, an alternate estimate of main steam line break frequency has been developed. Using the χ^2 distribution and zero observations with a 50% confidence level, this estimate is 1.2×10^{-3}/year. This estimate applies for breaks greater in area than typical valve-dominated small break and for both isolable and nonisolable breaks.	1.2×10^{-3}/ry
b) Small Break	One event at Robinson 2 was observed involving failure of steam side relief valves to close (NSIC 76461). In addition, 4 small SLB occurrences were observed in the Accident Sequence Precursor (ASP) program (Ref. 6, Ref. 7) over a 288 year observation period. This screening estimate has been developed using the χ^2 distribution with 5 occurrences in 288 reactor years. This estimate, 2×10^{-2}/yr, does not include the	2.0×10^{-2}/ry

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Table 1. (Continued)

Function	Discussion	Screening Estimate
	potential for recovery. Small steam line break occurrences observed in the ASP program were considered 40% non-recoverable. However, the steam-side PORV's at Robinson 2 are not isolable using a series isolation valve; and therefore small SLB's associated with these PORV's will be less recoverable than in the industry in general, unless the valve failures are dominated by valve operator failures and the valve operators are accessible.	
3) Loss of Coolant Accident		
a) Due to Failed Open Safety Valve	No major events involving stuck open safety valves have been observed. However, a safety valve apparently did open below set point pressure at St. Lucie 1 and depressurized the RCS from 2410 to 1670 psig in late 1981. Because of a lack of detailed information concerning this event, it has not been used in developing a frequency estimate. Using the χ^2 distribution with zero observations and an observation period from 1969 through July 1983 (406 PWR reactor years), a value of 1.7×10^{-3} is estimated.	1.7×10^{-3}/ry
b) Due to an Open PORV	NUREG-0611 (Ref. 8) reports 50 applicable PORV lifts at Westinghouse plants. Assuming NUREG-0611 covered the period up to September 1979, which includes 164 Westinghouse reactor years of operation, the Westinghouse PORV lift rate is: 50/(164 Westinghouse RY) = 0.30/ry. A value of 0.027 for PORV failure to close, once open, is developed from Reference 13 (4 failures to close in approximately 150 actuations). Utilizing these two values results in an estimate for LOCA due to an open PORV of: 0.30/ry * 2 valves * 0.027, or	1.6×10^{-2}/ry

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Table 1. (Continued)

Function	Discussion	Screening Estimate
	1.6×10^{-2}/ry. Consideration of operator response to close the block valve associated with a stuck open PORV will reduce this estimate.	
c) LOCA Due to Isolable Breaks Other Than PORV Occurrences	One minor event involving a leak from the letdown piping, followed by SI, occurred at Robinson 2 (NSIC 164149). Because of the minor nature of this event and in light of no other substantial data, the χ^2 distribution with zero occurrences in the total number of PWR reactor years (406 yrs) has been used to estimate the recommended value of 1.7×10^{-3}/year.	1.7×10^{-3}/ry
d) LOCA Due to Non-Isolable Breaks	Reactor coolant system leakage is considered a LOCA if it is large enough to initiate safety injection. One such event was observed at Robinson 2: a reactor coolant pump seal failure (NSIC 103077). A seal failure at Arkansas Nuclear One, Unit 1 also initiated SI. Two events involving tube ruptures followed by SI occurred at Ginna and Prairie Island. (Robinson 2 experienced a significantly greater than average number of SG tube leak LERs, but has plans to replace the SG bundles in the near future.) The estimates provided here are based on the above events and are as follows:	
	1) SG tube ruptures - 2 events in 406 PWR reactor years (1969-July 1983).	6.6×10^{-3}/ry
	2) Other LOCA's - 2 events in 406 PWR reactor years.	6.6×10^{-3}/ry
	These values are considered consistent with the NREP screening values of 10^{-2}/year.	
	No medium break LOCAs have been observed. A screening value of 1×10^{-3}/ry is recommended, based on reference (1).	1×10^{-3}/ry

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Table 1. (Continued)

Function	Discussion	Screening Estimate
e) Overall Small Break LOCA Not Immediately Isolated	An overall estimate for a small break LOCA not immediately isolated (~10 minute time frame) was obtained using the frequency estimates developed in (a), (b), and item (2) of (d), above and assuming a probability of 0.05 of the operator not isolating a blowing down PORV and a probability of 0.9 that an initially blowing down safety valve will not reseat. This results in an overall estimate of 8.9×10^{-3}/ry.	8.9×10^{-3}/ry
f) Overall Small Break LOCA Isolated in the Long Term	Assuming that ~40% of all PORV LOCAs not isolated in the short term are eventually isolated and other small break LOCAs cannot be isolated results in a probability estimate for late isolation of $(0.05 * 1.6 \times 10^{-2}) / 8.9 \times 10^{-3} = 0.039$.	0.039

It should be noted that several of the operational events used to develop the above estimates were associated with Robinson 2. While each of these events was considered a random event across the entire reactor population for the purposes of developing these screening estimates, the number which occurred at Robinson 2 may indicate an actual frequency for such events at this plant higher than that for the industry as a whole.

Branch Probabilities

1) Turbine Fails to Trip on Demand	PWR LERs were reviewed for turbine trip, turbine stop valve, etc., failures. While there have been several failures of individual stop valves (single steam line) to close, only one event (NSIC 92449 at Turkey Point 3, 4) identified a total failure of turbine stop valves. Assuming ~12 shutdowns/plant year (Ref. 3) and ~406 PWR years applicable to this review, the number of turbine stop valve demands is ~4900. One failure in this number of demands results in a failure estimate of ~2×10^{-4}. This estimate	4×10^{-5}

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Table 1. (Continued)

Function	Discussion	Screening Estimate
	does not consider use of the turbine control valves in isolating the turbine if the turbine stop valves fail to close. Consideration of the turbine control valves would reduce this estimate somewhat, perhaps by a factor of up to 10. Assuming a value of 0.2 for the conditional probability of control valve failure given stop valve failure results in a screening estimate of $2x10^{-4} * 0.2$ or $4x10^{-5}$. (Considering zero observed failures of the turbine to isolate on demand over the ~406 year period results in an estimate of $\sim 1.4x10^{-4}$ based on observation alone.)	
2) Steam Side PORV's Fail to Close on Demand	Based on a review of the Robinson 2 FSAR, these valves appear to open on power runbacks of greater than 70%. Two failure-to-close occurrences at Robinson 2 were reported in the same LER (NSIC 76461). Since the failures did not occur simultaneously, no deductions were made concerning common mode coupling and the events were assumed to be independent. The estimates provided here were based on the 2 failures and a demand estimate. The number of demands was estimated based on the number of reactor trips at Robinson 2 from greater than 70% power in 1978, 1979 and 1980 (15), the number of PORV's (3), and 12.7 reactor years of operation at Robinson from criticality through May 1983: $\frac{\text{15 demands/valve}}{\text{3 years}} * 12.7 \text{ reactor years} * 3 \text{ valves} = 190 \text{ demands}$ Recognizing that the valves open only on trips above 70%, the following effective demand probability estimates for a general reactor trip result:	

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Table 1. (Continued)

Function	Discussion	Screening Estimate
	a) For a single valve failure ((2/190 = $1.05x10^{-2}$) * (5/8.7)).	$6.3x10^{-3}$
	b) For failure of any one of 3 valves (3 * $1.05x10^{-2}$ * (5/8.7)).	$1.8x10^{-2}$
	c) For failure of any two valves (3 * $1.05x10^{-2}$ * 0.094 * (5/8.7)).	$1.7x10^{-3}$
	d) For failure of all three valves ($1.05x10^{-2}$ * 0.081 * (5/8.7)).	$4.9x10^{-4}$
3) Steam Dump Valves (SDVs) Fail to Close on Demand	No failures-to-close of SDV's at Robinson 2 were reported in the LERs. Based on zero observed failures, the failure on demand probability can be estimated using the Poisson approximation to the binomial discussed earlier and an estimate of the number of demands. All 5 SDVs open when demanded (Ref. 12) and assuming they are demanded on startup as well as on shutdown, the number of demands at Robinson 2 is: 5 valves * 12.7 Robinson 2 reactor years * (17.3 startups/yr + 17.3 shutdowns/yr (Ref. 4 and 5)) or ~2200 demands. The following probability estimates result:	
	a) For a single valve failure (0.7/2200 = $3.18x10^{-4}$).	$3.2x10^{-4}$
	b) For failure of any one of 5 valves (${}_5C_1$ * $3.18x10^{-4}$).	$1.6x10^{-3}$
	c) For failure of any two of 5 valves (${}_5C_2$ * $3.18x10^{-4}$ * 0.094**).	$3.0x10^{-4}$
	d) For failures of any three of 5 valves (${}_5C_3$ * $3.18x10^{-4}$ * 0.081**).	$2.6x10^{-4}$
	e) For failure of any four of 5 valves (${}_5C_4$ * $3.18x10^{-4}$ * 0.081**).	$1.3x10^{-4}$
	f) For failure of five valves ($3.18x10^{-4}$ * 0.081**).	$2.6x10^{-5}$
	g) For failure of three or more valves ((d) + (e) + (f)).	$4.2x10^{-4}$

**Reference 10 indicates no substantial differences in failure rates for air-operated valves in systems of greater than three components than those in three component systems.

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Table 1. (Continued)

Function	Discussion	Screening Estimate
4) Main Feedwater System Fails to Correctly Run Back Following Non-Specific Reactor Trip (not applicable for other initiators)	Potential main feedwater end states following a non-specific reactor trip are developed in Attachment A. Based on the frequency estimates developed in Attachment A, the following end-state conditional probability estimates are developed:	
	a) Probability of utilizing main feedwater system and bypass valves following non-specific reactor trip $((\lambda_{RT} - \lambda_{LOFW}\ [1/ry])/\lambda_{RT})$ = ((8.7 - 1)/8.7) = 0.89.	0.89
	b) Probability of requiring auxiliary feedwater following RT $((\lambda_{RT} - \lambda_{LOFW}) * P_{op.error} + \lambda_{LOFW})/\ _{RT}$ = ((8.7 - 1) * 0.03 + 1)/8.7 = 0.14.	0.14
	Development of steam generator overfill probabilities requires development of failure probabilities for main feedwater control valve closure and main feed pump trip:	
	Main Feedwater Control Valves Fail to Close on Demand. The feedwater control valves on Robinson 2 are run back on each shutdown. Failure to runback is usually not reportable, but has occurred at some PWR's. Because of the lack of observational information concerning these valves, the failure probability for an air operated valve included in reference 1 ($1x10^{-3}$) has been utilized in conjunction with conditional probabilities of multiple valve failures developed from reference 10. This results in the following estimates:	
	o For a single valve failure, $1x10^{-3}$.	
	o For failure of any one of 3 valves, ${}_3C_1 * 1.0x10^{-3} = 3x10^{-3}$.	

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Table 1. (Continued)

Function	Discussion	Screening Estimate
	o For failure of any two of three valves, $_3C_2$ * $1x10^{-3}$ * 0.094) = $2.8x10^{-4}$. o For failure of three valves, $1x10^{-3}$ * 0.081 = $8.1x10^{-5}$.	
	Main Feedwater Pumps Fail to Trip on Demand. The main feedwater pumps are tripped on high SG level (2/3 signals in any SG) and by a safety injection signal. Given the existence of one of these signals, the likelihood of main feedwater trip, is considered high. A value of 10^{-3} per pump is recommended.	
	Using the failure logic combinations developed in Attachment A, and recognizing that the probability of failing to generate a main feed pump trip signal on high steam generator level is small compared to the likelihood of failing to trip a main feed pump given the trip signal has been generated, the following probabilities of steam generator overfeed given a non-specific reactor trip are estimated:	
	c) Probability of one steam generator overfeed following RT ((λ_{RT} - λ_{LOFW}) * P(any one feed control valve fails to close) * P(either main feed pump fails to trip/SG high level signal or failure to generate SG high level signal)/ λ_{RT} = (8.7 - 1) * ($3x10^{-3}$) * ($2x10^{-3}$)/ 8.7 = $5.3x10^{-6}$.	$5.3x10^{-6}$
	d) Probability of two steam generators overfeed following RT ((λ_{RT} - λ_{LOFW}) * P(two of three feed control valves fail to close) * P(either main feed pump fails to trip/SG high level signal or failure to generate SG high level	$5.0x10^{-7}$

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Table 1. (Continued)

Function	Discussion	Screening Estimate
	signal)/λ_{RT} = ((8.7 - 1) * $(2.8x10^{-4})$ * $(2x10^{-3})$/8.7 = $5.0x10^{-7}$.	
	e) Probability of three steam generator overfeed following RT ((λ_{RT} - λ_{LOFW}) * P(all three feed control valves fail to close) * P(either main feed pump fails to trip/SG high level signal or failure to generate SG high level signal)/λ_{RT} = ((8.7 - 1) * $(8.1x10^{-5})$ * $(2x10^{-3})$/ 8.7 = $1.4x10^{-7}$.	$1.4x10^{-7}$
5) Failure of Engineered Safety Features System to Actuate	Failure to actuate ESFS will result in unavailability of trip signals for main steam line isolation, main feedwater isolation, and SI initiation, necessitating manual trip of the affected components and manual initiation of AFW and SI. A general multichannel instrumentation failure probability of $3x10^{-5}$ is recommended for screening purposes.	$3x10^{-5}$
6) Failure to Isolate Main Feedwater Following Initiators Other Than Non-Specific Reactor Trip (not applicable for non-specific reactor trip)	Boolean expressions for the probability of continued main feed flow to one or more steam generators are developed in Attachment B. Development of numeric estimates requires estimation of main feed isolation valve failure to close probabilities: MFIVs Fail to Close. A preliminary estimate developed in the IPRDS program (Ref. 11) for the failure of a motor operated valve to close, based on review of maintenance records at a small number of PWR plants is $6.4x10^{-3}$/demand. No failures of these valves have been observed at Robinson 2. Since these valves are tested quarterly, a failure estimate of 0.7/(12.7 ry * 3 valves/test * 4 tests/ry) = $4.6x10^{-3}$ can be developed, which is consistent with the IPRDS value.	

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Table 1. (Continued)

Function	Discussion	Screening Estimate
	The failure probability estimates for combinations of these valves are: o Any one of three valves fail to close (${}_3C_1$ * $4.6x10^{-3}$ = $1.4x10^{-2}$). o Any two of three valves fail to close (${}_3C_2$ * $4.6x10^{-3}$ * 0.020 = $2.8x10^{-4}$). o All three valves fail to close ($4.6x10^{-3}$ * 0.012 = $5.5x10^{-5}$).	
	Based on this development, other component failure probabilities developed previously, the Attachment B Boolean expressions, and assuming limited failure coupling between main feed control valves and main feed isolation valves (a conditional probability of 0.01 was assumed, to recognize some maintenance coupling), the following estimates can be made:	
	a) Continued feed flow to any one generator = 6 * (probability of one MFW pump failing to trip) * (probability of a single MFIV failing to close) * (probability of the associated MFCV failing to close) = 6 * $1x10^{-3}$ * $4.6x10^{-3}$ * 0.01 = $2.8x10^{-7}$.	$2.8x10^{-7}$
	b) Continued feed flow to any two steam generators = 6 * (probability of one MFW pump failing to trip) * (probability of two MFIVs failing to close) * (probability of both associated MFCVs failing to close) = 6 * $1x10^{-3}$ * $2.8x10^{-4}$ * (0.01 * 0.094) = $1.5x10^{-9}$.	$1.5x10^{-9}$
	c) Continued feed flow to all three steam generators = 2 * (probability of one MFW pump failing to trip) * (probability of all three MFIVs failing to close) * (probability of	$1x10^{-10}$

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Table 1. (Continued)

Function	Discussion	Screening Estimate
	all three MFCVs failing to close) = 2 * $1x10^{-3}$ * $5.5x10^{-5}$ * (0.01 * 0.081) = $1.0x10^{-10}$.	
	The attachment B expressions for failure of main feedwater isolation while in hot shutdown require estimation of the probability of the main feed bypass valves failing to close on demand. These valves are normally closed during power operation but are opened for decay heat removal using the main feed system while in hot shutdown. For this development, failure probability and common-mode coupling values equivalent to those used for the MFCVs have been assumed. Based on these values, the probability estimates for continued main feedwater flow given SI and while in hot shutdown are:	
	a) Continued feed flow to any one steam generator = 3 * (probability of MFW pump failing to trip) * (probability of single MFBV failing to close) = $3x10^{-3}$ * $3x10^{-3}$ = $9x10^{-6}$.	$9x10^{-6}$
	b) Continued feed flow to any two steam generators = 3 * (probability of MFW pump failing to trip) * (probability of two MFBVs failing to close) = $3x10^{-3}$ * $2.8x10^{-4}$ = $8.4x10^{-7}$.	$8.4x10^{-7}$
	c) Continued feed flow to all three steam generators = (probability of MFW pump failing to trip) * (probability of three MFBVs failing to close) = 10^{-3} * $8.1x10^{-5}$ = $8.1x10^{-8}$.	$8.1x10^{-8}$
	It should be noted that the above failure estimates have been developed without considering potential common-cause failure effects, which, for the multiple steam generator overfeed	

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Table 1. (Continued)

Function	Discussion	Screening Estimate
	situations, would be expected to dominate.	
7) Failure to Isolate Main Feedwater on SI Signal Given Failure to Runback Main Feedwater (Non-Specific Reactor Trip Only)	Given a failure to runback main feedwater to any steam generator, closure of the associated MFIV is required for isolation. Based on the conditional probabilities utilized in 6) above, for failure of these valves to close given failure of the associated MFCV to close, results in the following probability estimates:	
	If one line fails to runback:	
	a) One line fails to isolate.	$1x10^{-2}$
	If two lines fail to runback:	
	a) One line fails to isolate.	$2x10^{-2}$
	b) Both lines fail to isolate.	$9.4x10^{-4}$
	If three lines fail to runback:	
	a) One line fails to isolate.	$3x10^{-2}$
	b) Two lines fail to isolate.	$2.8x10^{-3}$
	c) Three lines fail to isolate.	$8.1x10^{-4}$
8) Multiple Steam Generators Blow Down Following Steam Line Break	Boolean expressions for the probability of multiple steam generator blowdown given a steam line break are developed in Attachment C. To develop numeric estimates from these expressions, the probability of multiple MSIV failure must first be estimated.	
	Main Steam Isolation Valves Fail to Close on Demand. One instance potentially involving failure of an MSIV to fully close was reported at Robinson 2 (NSIC 146521). Considering this one occurrence, a demand failure probability is estimated for the 12.7 years of operation at Robinson 2 as (1)/ (12.7 yr * 12 test demands/yr/valve	

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Table 1. (Continued)

Function	Discussion	Screening Estimate
	(partial stroke testing) * 3 valves) = 2.2×10^{-3}. Based on a review of all MSIV LER's, the number of failures to close for single valves is on the same order as for multiple valves failing to close. Thus it can be concluded that the potential for common mode coupling among these valves is large. Consistent with these observations, 0.3 was chosen for the conditional probability of a specific second valve failing, given that one has failed; and 0.8 was chosen for the conditional probability of the third valve failing given that the other two have failed. This results in the following estimates:	
	o For failure of a particular MSIV to close: 2.2×10^{-3}.	2.2×10^{-3}
	o For failure of a particular set of two MSIV's to close: $2.2 \times 10^{-3} * 0.3 = 6.6 \times 10^{-4}$.	6.6×10^{-4}
	o For failure of three MSIV's to close: $2.2 \times 10^{-3} * 0.3 * 0.8 = 5.3 \times 10^{-4}$.	5.3×10^{-4}
	Based on the probability expressions developed in Attachment C and the above valve failure probabilities, the following probabilities of steam generator blowdown following an arbitrary large steam line break can be estimated:	
	a) No steam generator blows down.	0.5
	b) One steam generator blows down.	0.5
	c) Two steam generators blow down ($1.5 * 6.6 \times 10^{-4}$).	9.9×10^{-4}
	d) Three steam generators blow down ($0.5 * 3.3 \times 10^{-4}$).	1.7×10^{-4}
	The above estimates are also applicable to small steam line breaks which result in MSIV closure. For small breaks which	

Table 1. (Continued)

Function	Discussion	Screening Estimate
	do not result in MSIV closure, the following estimates, developed in Attachment C, are applicable:	
	a) No steam generator blows down.	0.0
	b) One steam generator blows down.	0.5
	c) Two steam generators blow down.	0.0
	d) Three steam generators blow down.	0.5
	Operator action to close the MSIVs would reduce the estimate for three steam generator blowdown by the operator action probability.	
9) High Pressure Safety Injection Fails to Occur on Demand	At Robinson 2, four failures of SI pumps were reported in LERs in 1979 and subsequent years. Assuming half of the number of Robinson years of operation, since all the failures occurred after 1976, yields a demand estimate of 12 demands/yr * 6.4 yrs * 3 pumps = 228 demands; and a demand failure probability estimate of 1.7×10^{-2} per pump. With conditional probabilities of 0.1 and 0.3 applied for subsequent failures of a second pump and then the third, respectively, a failure probability for the system of 5.2×10^{-4} is estimated. This is consistent with the estimate available from the ASP data base for Westinghouse plants of 4.8×10^{-4}. (The estimate with potential recovery considered is lower by a factor of 0.34.) The recommended estimate is 4.8×10^{-4}, from the larger data base.	6.1×10^{-4}
	Given HPI actuation success, successful HPI injection is dependent on the primary side pressure dropping low enough and check valves in the injection paths opening. There are 3 check valves in each of three injection paths that are inside containment and are not typically tested during monthly HPI testing. Using the Ref. 1 value of	

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Table 1. (Continued)

Function	Discussion	Screening Estimate
	10^{-4}/demand for a check valve failing to open, the probability of any one of 3 valves failing in a given path is 3.0×10^{-4}. Using coupling factors developed from reference 10 for check valves failing to open of 0.5 for failure of a second specific valve and 0.42 for both remaining valves in a set of 3, and assuming that dependent coupling is much more likely for equivalent valves, results in an estimate for the probability of not delivering HPI through any path given actuation success of 1.3×10^{-4}. Combining these values for failure to actuate and failure to deliver flow through the injection paths results in an overall estimate of 6.1×10^{-4}.	
10) AFW Fails to Actuate on Demand	An estimated AFW system failure probability, suitable for screening purposes, has been developed based on the average PWR operational experience from 1969 through 1981 as evaluated in the ASP program (Ref. 6, Ref. 7). This value is 1×10^{-3} without considering potential recovery. Considering potential short term recovery results in an estimate of 3×10^{-4}. Since these values are based on averaged experience and do not consider potential learning (except as evidenced in the averages), they may not be representative of expected future experience at Robinson. However, they are considered consistent with the AFW component failure experience observed to date at Robinson 2: 7 motor pump failures, 2 steam turbine pump failures, and 13 failures of pump discharge valves to open. No total AFW system failures have been observed at Robinson 2.	See discussion

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Attachment A

Development of Main Feedwater End State Probability Estimates for Non-Specific Reactor Trip Initiator

Following a general reactor trip, the main feedwater system is run back and an attempt is made (typically successfully) to utilize the main feed pumps and the bypass valves for decay heat removal. Depending on power level and the extent of SG level shrink, AFW can be initiated on RT. Isolation of AFW following such initiation is a normal part of RT recovery, along with manual opening of the bypass valves.

Following a non-specific RT, the following steam generator feed situations are possible:

- o Main feedwater using bypass valves (normal situation).
- o Auxiliary feedwater (main feedwater not recovered).
- o One steam generator overfeed.
- o Two steam generator overfeed.
- o Three steam generator overfeed.

For the overfeed situations, auxiliary feedwater would be expected to be provided to the isolated steam generators once low level in one steam generator is reached. Responses which include partially or totally faulted auxiliary feedwater are possible, but have not been included because they are considered less conservative than the above situations with respect to PTS.

The above states require the following responses:

- o Main feedwater on, bypass valves used for flow control - operator actions to open bypass valves following control valve closure and secure AFW if initiated due to shrink.
- o Auxiliary feedwater - operator action to open bypass valves and use main feedwater not effective or RT due to LOFW.
- o One steam generator overfeed - failure of feed control valve in one train to run back and failure of either main feed pump to trip on high SG level.

- o Two steam generator overfeed - failure of feed control valves in two of three trains to run back and failure of either main feed pump to trip on high SG level.
- o Three steam generator overfeed - failure of feed control valves in all three trains to run back and failure of either main feed pump to trip on high SG level.

The frequencies associated with the above states can then be written as:

- o λ(on bypass valves following RT) $\cong \lambda_{RT}$ * P(operator opens bypass valves following runback) - λ_{LOFW}.
- o λ(on AFW following RT) $\cong (\lambda_{RT} - \lambda_{LOFW})$ * P(operator fails to open bypass valves following RT) + λ_{LOFW}.
- o λ(one SG overfeed) $\cong (\lambda_{RT} - \lambda_{LOFW})$ * P(one feed control valve fails to close) * P(either main feed pump fails to trip/SG high level signal or failure to generate SG high level signal).
- o λ(two SG overfeed) $\cong (\lambda_{RT} - \lambda_{LOFW})$ * P(two feed control valves fail to close) * P(either main feed pump fails to trip/SG high level in either SG or failure to generate SG high level signal in either SG).
- o λ(three SG overfeed) $\cong (\lambda_{RT} - \lambda_{LOFW})$ * P(three feed control valves fail to close) * (either main feed pump fails to trip/SG high level in any steam generator or failure to generate SG high level in any SG).

Main feedwater isolation in the event of overfeed will occur due to closure of the MFIVs on SI if SI occurs. If SI does not occur, or if the applicable MFIV(s) fail to close on demand, then operator action is required to trip the condensate pumps or close the MFIVs (if SI has not been initiated).

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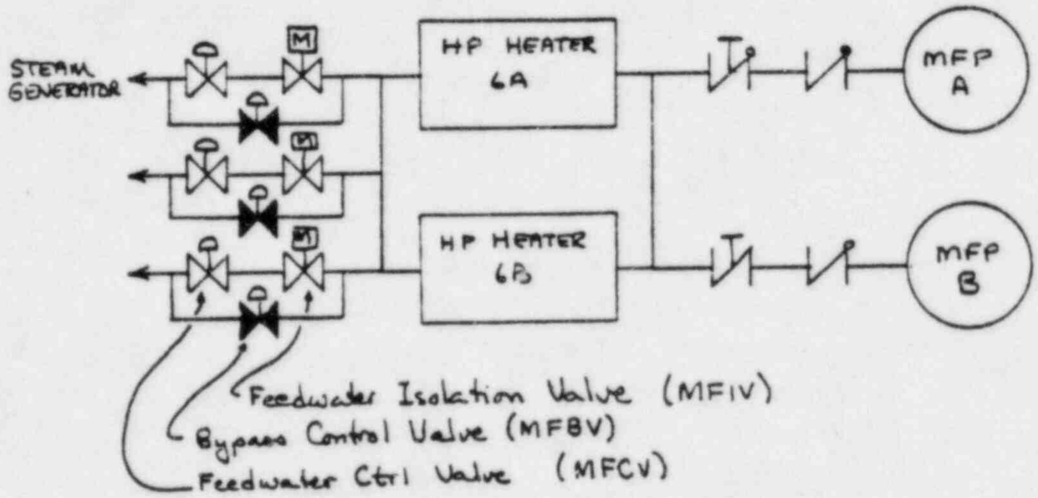
Attachment B

Development of Main Feedwater Isolation Failure Probability Estimates (All Initiators Except Reactor Trip)

1. The H.B. Robinson 2 main feedwater system is arranged as follows:



On reactor trip plus safety injection initiation, the main feed control valves and feedwater isolation valves are commanded shut and the main feed pumps are tripped. Continued feedwater flow to a steam generator will occur if both of the valves in the associated feed line fail to close and either main feed pump fails to trip.

2. Continued feedwater flow to one generator will therefore occur if:

$$(\text{MFP A} \cup \text{MFP B}) \cap (\text{MFIV A} \cap \text{MFCV A} \cap (\overline{\text{MFIV B}} \cup \overline{\text{MFCV B}}) \cap (\overline{\text{MFIVC}} \cup \overline{\text{MFCV C}}) \cup \text{other feed line combinations})$$

3. Continued feedwater flow to two steam generators will occur if two of the three feed lines fail to isolate and either main feed pump fails to trip:

$$(\text{MFP A} \cup \text{MFP B}) \cap (\text{MFIV A} \cap \text{MFCV A} \cap \text{MFIV B} \cap \text{MFCV B} \cap (\overline{\text{MFIV C}} \cup \overline{\text{MFCV C}}) \cup \text{other feed line combinations})$$

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For continued flow to all steam generators, the following is required:

$$(\text{MFP A} \cup \text{MFP B}) \cap (\text{MFIV A} \cap \text{MFCV A} \cap \text{MFIV B} \cap \text{MFCV B} \cap \text{MFIV C} \cap \text{MFCV C})$$

4. To reduce the above equations, it is assumed that system response is symmetric (i.e., the likelihood of pump A failing to trip is equal to the likelihood of pump B failing to trip), and that pump and valve response is loosely coupled.

5. The probability of continued flow to N steam generators is then approximately:

$$P(\text{flow to 1 SG}) \cong 6 * P(\text{MFP}) * P(\text{MFIV}) * P(\text{MFCV}|\text{MFIV})$$

$$P(\text{flow to 2 SGs}) \cong 6 * P(\text{MFP}) * P(\text{MFIV1}) * P(\text{MFIV2}|\text{MFIV1}) * P(\text{MFCV1}|\text{MFIV1, MFIV2}) * P(\text{MFCV2}|\text{MFIV1, MFIV2, MFCV1})$$

$$P(\text{flow to 3 SGs}) \cong 2 * P(\text{MFP}) * P(\text{MFIV1}) * P(\text{MFIV2}|\text{MFIV1}) * P(\text{MFIV3}|\text{MFIV1, MFIV2}) * P(\text{MFCV1}|\text{MFIV1},\ldots) * P(\text{MFCV2}|\text{MFIV1},\ldots) * P(\text{MFCV3}|\text{MFIV1},\ldots)$$

$$P(\text{flow to 0 SGs}) = 1 - P(\text{flow to 1 SG}) - P(\text{flow to 2 SGs}) - P(\text{flow to 3 SGs})$$

6. In hot shutdown the bypass valves (MFBVs) are used to control feedwater flow to the steam generators. Feed flow is typically provided by one pump. The MFBVs are in parallel with the MFIV, MFCV pairs, and hence the closure or MFP trip is necessary for steam generator isolation. In hot shutdown, then, the probability of continued flow to N steam generators is approximately:

$$P(\text{flow to 1 SG}) \cong 3 * P(\text{MFP}) * P(\text{MFBV})$$

$$P(\text{flow to 2 SGs}) \cong 3 * P(\text{MFP}) * P(\text{MFBV1}) * P(\text{MFBV2}|\text{MFBV1})$$

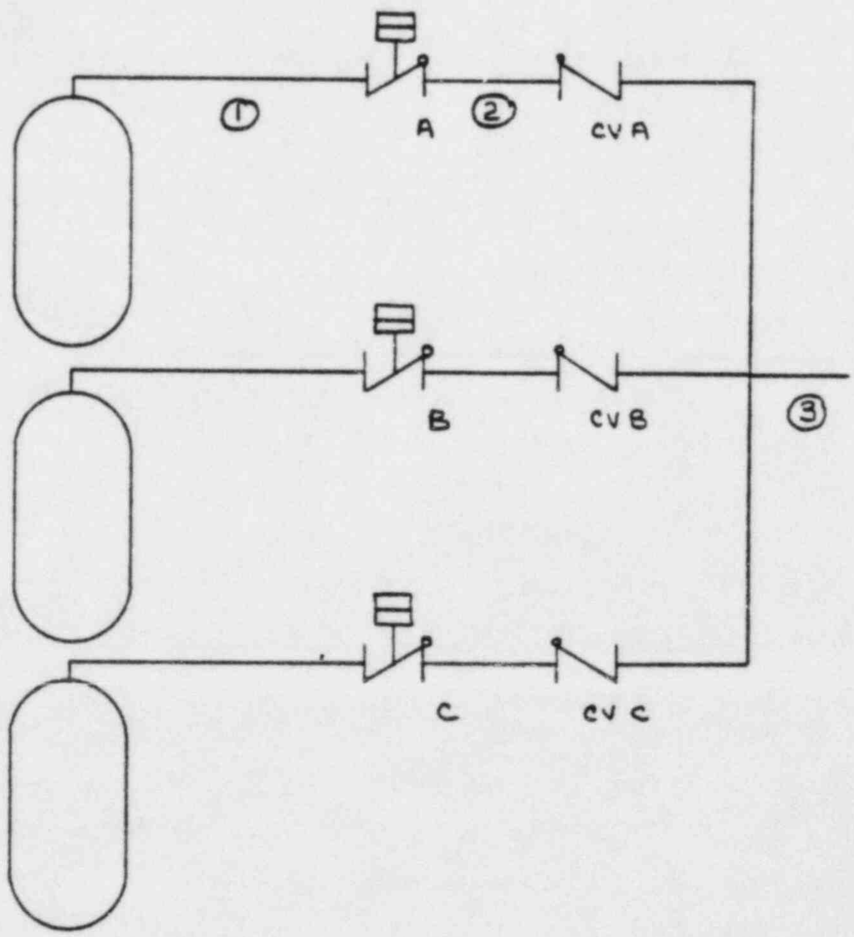
$$P(\text{flow to 3 SGs}) \cong 3 * P(\text{MFP}) * P(\text{MFBV1}) * P(\text{MFBV2}|\text{MFBV1}) * P(\text{MFBV3}|\text{MFBV1, MFBV2})$$

$$P(\text{flow to 0 SGs}) = 1 - P(\text{flow to 1 SG}) - P(\text{flow to 2 SGs}) - P(\text{flow to 3 SGs})$$

Attachment C

Development of Multiple Steam Generator Blowdown
Frequency Estimates

1. The H.B. Robinson 2 steam line arrangement utilizes both MSIVs and check valves for steam generator isolation:



2. Consider potential breaks at location (1) with frequency λ_1, (2) with frequency λ_2 and (3) with frequency λ_3. The break locations and valve states for n steam generator blowdown are:

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End State	Break Location (1)	Break Location (2)	Break Location (3)
0 SGs blowdown	n/a	A closed and (CVA closed or B closed and C closed)	A closed and B closed and C closed
1 SG blowdown	A closed or CVA closed or B closed and C closed	A open and (CVA closed or B closed and C closed) or A closed and CVA open and (B open and C closed or B closed and C open)	A open and B closed and C closed or B open and A closed and C closed or C open and A closed and B closed
2 SG blowdown	A open and CVA open and (B open and C closed or B closed and C open)	A open and CVA open and (B open and C closed or B closed and C open)	A open and B open and C closed or A open and C open and B closed or B open and C open and A closed
3 SG blowdown	A open and CVA open and B open and C open	A open and CVA open and B open and C open	A open and B open and C open

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Considering the frequency of breaks in locations (1), (2), and (3), the frequency of n steam generator blowdown, $\Lambda(n)$, is:

$$\Lambda(0) = \lambda_2 * P[\bar{A} \cap (\overline{CVA} \cup \bar{B} \cap \bar{C})] + \lambda_3 * P[\bar{A} \cap \bar{B} \cap \bar{C}]$$

$$\Lambda(1) = \lambda_1 * P[\bar{A} \cup \overline{CVA} \cup \bar{B} \cap \bar{C}] + \lambda_2 * P[A \cap (\overline{CVA} \cup \bar{B} \cap \bar{C}) \cup \bar{A} \cap CVA \cap (B \cap \bar{C} \cup \bar{B} \cap C)] + \lambda_3 * P[\bar{A} \cap \bar{B} \cap C \cup \bar{A} \cap B \cap \bar{C} \cup A \cap \bar{B} \cap \bar{C}]$$

$$\Lambda(2) = \lambda_1 * P[A \cap CVA \cap (B \cap \bar{C} \cup \bar{B} \cap C)] + \lambda_2 * P[A \cap CVA \cap (\bar{B} \cap C \cup B \cap \bar{C})] + \lambda_3 * P[A \cap B \cap \bar{C} \cup A \cap \bar{B} \cap C \cup \bar{A} \cap B \cap C]$$

$$\Lambda(3) = \lambda_1 * P[A \cap CVA \cap B \cap C] + \lambda_2 * P[A \cap CVA \cap B \cap C] + \lambda_3 * P[A \cap B \cap C]$$

The following assumptions have been made in reducing the above equations for an arbitrary large SLB:

1. Because of the proximity of the MSIVs and check valves, λ_2 is small compared to λ_1 and λ_3.

2. Even if three MSIVs fail to close, the probability of check valve failure is less than 0.1.

3. Break locations (1) and (2) are equally applicable to all steam generators. Furthermore, because of the lack of data, breaks in locations (1) (for all three steam generators taken together) and (3) are equally likely (i.e., breaks upstream of the MSIVs are as likely as breaks downstream of the MSIVs).

The above equations can then be reduced to:

$$\Lambda(0) \cong 3 * \lambda_2 + \lambda_3$$

$$\Lambda(1) \cong 3 * \lambda_1 + 3*\lambda_2 * [P(A) + P(B) * P(CVA|B) + P(C) * P(CVA|C)] + \lambda_3 * [P(A) + P(B) + P(C)]$$

$$\Lambda(2) \cong 3 * \lambda_1 * [P(A) * P(A|B) * P(C|AB) * P(CVA|ABC) + P(A) * P(C|A) * P(B|AC) * P(CVA|ACB)] + 3 * \lambda_2 * [P(A) * P(B|A) * P(C|AB) * P(CVA|ABC) + P(A) * P(C|A) * P(B|AC) * P(CVA|ABC)] + \lambda_3 * [P(A) * P(B|A) * P(C|AB) + P(A) * P(C|A) * P(B|AC) + P(B) * P(C|B) * P(A|BC)]$$

$$\Lambda(3) \cong 3 * \lambda_1 * [P(A) * P(B|A) * P(C|AB) * P(CVA|ABC)] + 3*\lambda_2*[P(A) * P(B|A) * P(C|AB) * P(CVA|ABC)] + \lambda_3 * [P(A) * P(B|A) * P(C|AB)]$$

Assumptions one and two result in the following further simplification:

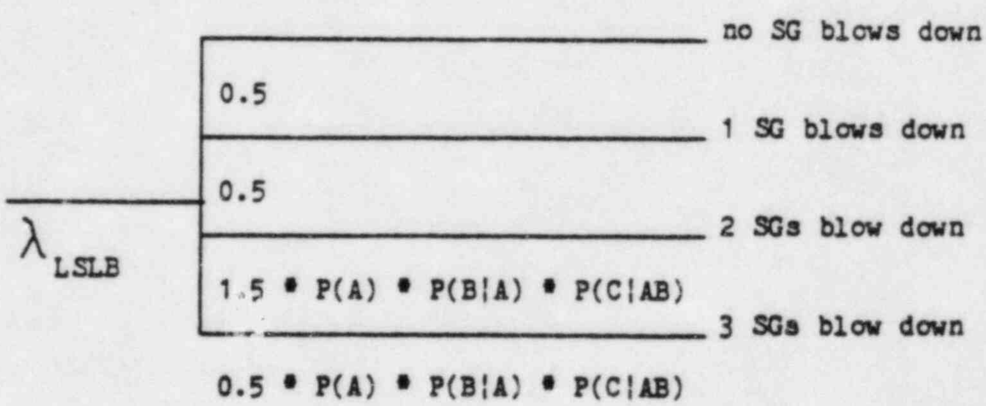
$$\Lambda(0) \cong \lambda_3$$

$$\Lambda(1) \cong 3*\lambda_1 ^{**}$$

$$\Lambda(2) \cong 3 * \lambda_3 * P(A) * P(B|A) * P(C|AB)$$

$$\Lambda(3) \cong \lambda_3 * P(A) * P(B|A) * P(C|AB)$$

Since $3 * \lambda_1 \cong \lambda_3$ and $\lambda_2 << \lambda_3, \lambda_1$, an event tree can be constructed representing potential SG blowdown following an arbitrary large steam line break.



**Since P(A), etc., $\cong 10^{-2}$.

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A review of historic small steam line breaks indicates approximately 50% have been associated with steam line relief valves located upstream of MSIVs, and the remaining have been associated with condenser dump valves, typically located downstream of MSIVs. Because of these ratios, the above development for large steam line breaks is also considered applicable to small steam line breaks, provided the break is sufficiently large to require MSIV closure.

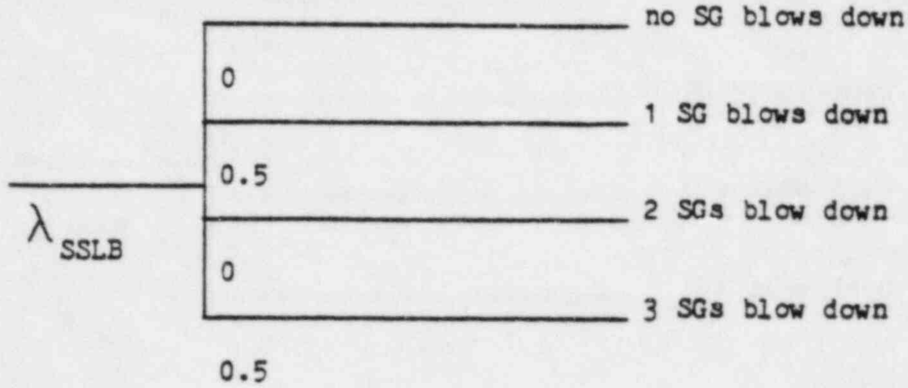
If a small steam line break is large enough to close the appropriate check valve (if it were located upstream of the check valve) but is not large enough to initiate MSIV closure and operator action is not taken to close the MSIVs, then the frequency of multiple steam generator blowdown can be estimated as:

$$\Lambda(0) \cong 0$$
$$\Lambda(1) \cong \lambda_1 \bullet P[CVA] + \lambda_2 \bullet P[CVA]$$
$$\Lambda(2) \cong 0$$
$$\Lambda(3) \cong \lambda_1 \bullet P[CVA] + \lambda_2 \bullet P[CVA] + \lambda_3$$

Small SLBs upstream of the check valves may be associated with any of the steam generators. Since the number of relief valve-related breaks is consistent with the number of condenser dump valve-related breaks and the number of small SLBs associated with region (2) is expected to be small compared with valve-related breaks, the following simplifications of the above equations can be made:

$$\Lambda(0) \cong 0$$
$$\Lambda(1) \cong 3 \bullet \lambda_{1,\ SSLB} \cong 0.5 \bullet \lambda_{SSLB}$$
$$\Lambda(2) \cong 0$$
$$\Lambda(3) \cong \lambda_{3,\ SSLB} \cong 0.5 \bullet \lambda_{SSLB}$$

Potential steam generator blowdown, given a small steam line break which does not initiate MSIV closure, can be represented by the following event tree:


no SG blows down
0
1 SG blows down
0.5
2 SGs blow down
λ_{SSLB}
0
3 SGs blow down
0.5

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November 21, 1984

H. B. Robinson Review Document Number 2

TO: Distribution

SUBJECT: Review H. B. Robinson PTS Report Chapter 3 - Development of Overcooling Sequences

Please find enclosed copies of the H. B. Robinson Unit 2 chapter 3 and appendix B. This material has now been reviewed internally and cleared as draft information for comment. It should be noted that some figures, references, and page numbering are incomplete and will be cleaned up shortly. Please review and return comments by Friday, December 14. If additional time is needed for review, please let me know.

D. L. Selby

D. L. Selby
Engr. Physics and Mathmetics Div.

DLS:nc

Enclosures (2)

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