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Abstract

The potential for degraded core conditions resulting from failure of the vapor suppression system in BWR plants equipped with Mark I and Mark II containments during anticipated transients has been examined. It is postulated that this failure is due to the rupture, within the wetwell airspace, of a safety/relief valve discharge line coupled with failure of the valve to close following its actuation in response to the transient. In the absence of any mitigating action, it is found that the resulting inflow of steam leads to rapid pressurization of the containment to levels which exceed containment rupture pressure. The very high probability that core coolant inventory cannot be maintained in this circumstance implies eventual core meltdown. In the present study, it is found that timely actuation of the containment spray system can mitigate the accident; that is, prevent breech of containment. In general, however, such actuation must be accomplished manually implying a relatively low probability of success because of the short times available. This circumstance, when coupled with the high frequency of anticipated transients, leads to an overall accident sequence frequency which is comparable to many of those now considered to be significant contributors to risk. It is concluded, therefore, that the accident sequence identified in this study should be included in any ongoing or future PRA's relating to Mark I and Mark II BWR plants.

ACKNOWLEDGEMENTS

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1.0 Introduction

BWR plants are equipped with safety/relief valves (SRV's) to control primary system pressure transients. These SRV's are the major component of the reactor overpressure protection system whose function is to prevent failure of the reactor coolant pressure boundary (RCPB). They accomplish this by opening at appropriate pressure levels and allowing steam to escape from the reactor vessel. The SRV's are mounted on the main steam lines inside the drywell but the discharged steam is routed away from the drywell into the suppression pool via a system of pipes which are installed for this purpose. Figures 1 and 2 show schematically how this routing is executed for a typical Mark I and Mark II plant. These pipes, or SRV discharge lines (SRVDL's) are subjected to severe thermal and mechanical loading whenever an SRV actuation occurs. Such actuations are expected to occur many times during the life of a BWR and the SRVDL's are designed accordingly.

Because of the function they perform, the SRVDL's can be considered part of the vapor suppression system (VSS) in the same sense that the downcomers in a Mark I or Mark II plant are. That is, their common function is to rout any steam that escapes the RCPB to the suppression pool so that it will be condensed. Failure to fulfill this function results in a severe challenge to containment integrity because of the high pressures generated when the escaping steam is not condensed.

In the Reactor Safety Study,[1] failure of the VSS was considered in connection with LOCA type accident initiators. The criterion used to define this failure was steam bypass (i.e. direct steam inflow to the wetwell airspace) sufficient to prevent clearing of the water column in the downcomers. For a large LOCA it was determined that this criteria was fulfilled by the rupture of one downcomer or one vent or two vacuum breakers (See Figure 1). This type of failure was also found to lead to containment rupture by overpressure within 30 seconds. No indication is given that accident mitigation with any of the available safety systems was considered, presumably because of the very short time available to take any action.

Despite its severity, this accident sequence did not represent a dominant contributor to risk because the low probability of pipe failure was coupled to a low frequency initiating event (LOCA) leading to overall sequence frequencies which were low compared to that of accident sequences with transient initiators.* Accident sequences involving VSS failure with transient initiators were not considered in this earlier work. The objective of the present study is to analyze such an accident sequence with the postulated rupture of an SRVDL replacing the ruptured downcomer as the cause of steam bypass.** Also, the LOCA break is replaced by an open SRV. In particular, the SRV must be postulated to remain open (stuck) after it has served its normal pressure limiting function. This is necessary because normal valve firing times are too short to significantly pressurize the containment because of the steam bypass (see Section 3.1). A more complete and detailed description of the entire accident sequence is provided in Section 3. The scope and method of approach employed in this study is detailed in the next section.

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*Current estimates are that transient initiators occur at a rate ranging from one thousand to one million times greater than LOCA initiators.

**The Mark III containment design and SRVDL routing arrangement precludes steam bypass in the event of an SRVDL rupture. In these plants the SRVDL enters the wetwell below the suppression pool surface (Figure 3) so that a path for direct steam entry to the airspace does not exist. Accordingly, the present study is applicable only for Mark I and Mark II BWR plants.

2.0 Scope and Method of Approach

The approach used to achieve the objectives of this study was essentially the PRA methodology employed in the Reactor Safety Study[1] but with a much more limited scope. This limitation refers primarily to determination of the consequences of the accident sequence but reduction in scope relative to a complete PRA was also imposed in terms of accident sequence development and frequency evaluation.

With respect to consequences, no actual consequence calculations were performed. The severity of the accident sequences was only characterized in terms of a so-called "release category". This could be done with reasonable certainty making use of certain key features of the accident sequence as discussed in Section 3.3.2. Additional limitations included restriction of the study to only the most dominant sequences that could be associated with the SRV pipe failure. Thus, for an initiating event only the most frequent anticipated transient was used. Also, additional system failures were minimized and only one accident mitigating system was considered. The final limitation was to restrict our evaluations to BWR plants for which PRA's were already available. Then the frequency of the accident sequences developed in this study could be used to compare with those previously evaluated to establish how significantly they contributed to overall risk.

In the next section, the major findings of our study are presented. The details of how these findings were obtained are described in the several Appendices which are provided. Our conclusions and recommendations based on these findings are presented in Section 4.0.

3.0 Description of Key Findings

This study required the completion of three distinct tasks. These were to analyze the pressurization in the containment (Task 1), analyze SRV pipe capability (Task 2) and assess the significance of the postulated accident (Task 3). The key findings in each of these areas are enumerated below by task.

3.1 Containment Pressure Response to Steam Bypass (Task 1)

The containment pressure response to the steam bypass resulting from the SRVDL rupture was obtained using the methods described in Appendix A. Some typical results are shown in Figure 4. Bypass fraction refers to the fraction of total SRV steam flow which actually enters the wetwell airspace. The rupture pressure which is shown for the Mark I containment is derived in WASH 1400* for the Peach Bottom plant. The rupture pressure cited for a Mark II plant corresponds to that for the Limerick Generating Station (LGS) and is reported in their PRA.[2] The design pressure for these plants, which was obtained from the same sources, is indistinguishable on the scale of this graph. As can be seen, with 100% bypass design pressure is exceeded within four minutes for either plant while pressures exceeding rupture levels occur in about ten minutes. In Figure 5, the effect of actuating the Containment Spray System (CSS) on a typical Mark II transient is depicted. It is apparent that this action provides a very effective means of mitigating the accident provided it is accomplished in a timely fashion.

Based on these results it is concluded that a stuck open SRV (SORV) combined with a ruptured SRVDL inevitably will lead to containment rupture by overpressure unless the CSS is actuated within about 10 minutes of the transient initiator. Also, significant design pressure exceedance will occur unless the CSS is actuated within four minutes.

*In the ensuing discussion the Reactor Safety Study[1] will be referred to as WASH 1400.

3.2 SRVDL Failure Rates (Task 2)

A key element of this study was to obtain estimates of SRVDL failure rates. These are not necessarily identical with those used for LOCA pipe ruptures because the latter are process piping that experience high but essentially steady loads, whereas the SRVDL's are standby pipes which experience a cyclic type of loading. Estimates of failure rates were obtained by two different methods. These included an evaluation of operating experience similar to that used in WASH-1400 to estimate the LOCA frequency as well as by use of so-called Probabilistic Mechanical Design methods (PMD). The latter methods were also used to develop estimates of the effect of load cycles and corrosion. The details of the development are given in Appendix B and C. The key findings of these studies are summarized in Table 3.1. As can be seen, a wide range of failure rates can be expected depending primarily on plant life.* At the present time, the most likely failure rate that would be expected based on PMD would be about 10^{-7} compared with what we consider to be a bounding value of 7.4 E-5 based on operating experience. Note, however, that if the pipe experiences a sufficient number of load cycles, the failure rates predicted by PMD approach this bounding value. Determination of the appropriate number of load cycles that will be experienced by a SRVDL during its lifetime is normally done by the plant A/E. Our "guesstimate" of over 7000 (but less than 14000) for a 40 year plant life is based on one such study.[3]

Based on these results, it is concluded that the most appropriate value to be used for SRVDL pipe failure rate is 7.4 E-5 per demand with the recognition that it probably represents an upper bound rather than a best estimate or mean value.

3.3 Estimated Contribution to Risk of the Accident (Task 3)

The determination of the contribution to risk of any particular accident requires the determination of its frequency of occurrence, together with an estimate of the resulting consequences. Our findings relative to these two requirements are detailed separately below.

*But see also Appendix B for a discussion of the uncertainty in failure rates deduced by PMD methods due to uncertainties in material properties.

3.3.1 Accident Frequency

3.3.1.1 Event Tree for Accident Sequences

Accident frequency is determined with the aid of an event tree which defines the sequence of events that make up the accident. These events will include an initiating event, as well as the performance of various safety systems whose successful functioning can mitigate the accident or whose failure allows the accident to proceed. The event tree developed and evaluated for the present study is shown in Figure 6. The event tree is structured in a conventional manner. Following the initiating event (in this case an anticipated transient), the tree shows in columnar fashion the functions that need to be successfully performed to prevent an accident (degraded core condition)* from occurring. It also shows the system (or systems) that can be exploited to perform these functions. The ordering of the functions is in total agreement with event trees previously developed with the single exception of the inclusion of the "limit containment pressure" function in a transient event tree (see discussion in Section 1.0). As indicated in Figure 6, this represents the point of departure for the present study. This function has been placed before the coolant and heat removal functions because its failure implies very early rupture of the containment (Section 3.1) which leads, with high probability, to failure to maintain coolant inventory and therefore, to eventual core meltdown.

In terms of the notation defined in Figure 6 this study is concerned only with the TPD sequences or branches of the event tree. A complete risk analysis would consider all such sequences. However, consistent with the limited scope of the present study (Section 2.0), only some of the more dominant sequences are considered. In Figure 6 the sequences which are not examined are indicated by the dashed branches which are not continued (e.g. TPDQ). Another simplification

*It has been determined[4] that the highest risk to the public from operation of a nuclear power plant is presented by postulated accidents involving the reactor core. Other sources of radioactivity at a nuclear plant make a negligible contribution to the public risk.

implicit in the event tree is that only the CSS is shown as capable of limiting containment pressure given failure of the VSS (via SRVDL rupture). Note that the symbol Z chosen to represent failure of the CSS is not a conventional notation. The representation is unique to this report.

The sequences considered to be the dominant contributors to risk and which were therefore selected for quantification in this study are the TPD, TPDW and TPDZ sequences as indicated by an entry in the last column. A detailed discussion of each of these sequences will be given below. The frequency with which these accident sequences occur is indicated in functional form by the products entered in the last column. These frequencies, when quantified, will be compared to the frequency of some other dominant contributors to risk (e.g. TW, TPW) which have been evaluated by others.[1,2] A decision can than be made regarding the significance of the TPD sequences of accidents in terms of overall reactor safety.

3.3.1.2 Detailed Discussion of Dominant Accident Sequences

3.3.1.2.1 The TPD Sequence

The initiating event for this sequence is considered to be a turbine trip (with bypass) which is generally considered to be the transient with the highest frequency of occurrence[5,6]. Following this transient it is postulated that the reactor is successfully scrammed and that the SRV's actuate in a normal fashion to limit reactor pressures. When the latter has been reduced to levels below the SRV setpoints, it is then postulated that one of them fails to reseat (the event P). It is further assumed that the SRVDL associated with this valve has ruptured and is allowing steam to bypass to the wetwell air space (the event D). For this sequence it is then assumed that the CSS is actuated in a timely fashion (ie. before containment rupture by overpressure) and that the coolant and heat removal functions operate successfully to bring the reactor to a cold shutdown. This scenario or sequence is not an accident in the sense defined earlier (i.e., does not result in a degraded core condition) but is considered to lead to some release of radioactivity because it is assumed that the design pressure was exceeded for a period of time. That is, it is assumed that the CSS was actuated before containment rupture, but after design pressure exceedance.

3.3.1.2.2 The TPDW Sequence

This sequence proceeds exactly as the TPD sequence described above but the additional failure to remove the residual heat from the containment is postulated (the event W). This ultimately leads to containment pressures which are sufficiently high to rupture the containment. This is considered to lead, with probability near unity, to a core meltdown because of failure to maintain coolant inventory.

3.3.1.2.3 The TPDZ Sequence

In this sequence, it is postulated that the CSS is not actuated before the containment ruptures (the event Z). Here again, this is assumed to lead, with high probability to degraded core conditions.

3.3.1.3 Frequency of The Sequence Events

To quantify the frequency of the dominant accident sequences discussed above the frequency of the initiating event T, as well as the events P,D,Z and W are required. Estimates of these frequencies are developed in the ensuing subsections.

3.3.1.3.1 Frequency of The Initiating Event T

In WASH 1400 no distinction between different types of transients is made. Based on operating experience an estimate of about 10 transients of all kinds per reactor year is employed. Since that study more detailed examination of operating experience has been made[5,6]. These indicate that the total number of transients is about 7 per reactor year and that, of these, about 4 correspond to the transient we have discussed above - the turbine trip with bypass. In the quantification presented in Section 3.3.1.4 we utilize T=4 per reactor year when comparing sequence frequencies with Limerick sequence frequencies since they employ the more recent data. When comparing with WASH-1400 frequencies, we will adjust them to account for their use of T = 10 per reactor year.

3.3.1.3.2 Frequency of the Event P

In a review of Licensee Event Reports (LER's) performed by EG&G for the NRC[7] it was found that there were 17 instances of SRV's failing to reseat following actuation caused by 482 transients which required reactor scram. The total number of reactor years represented by the study was 62. This implies a failure rate of $17/62 \approx 0.3$ SORV's per reactor year. Note however, that this does not represent the failure rate P as defined in the event tree (Figure 6) but the product T*P. Alternately, one can get a P failure rate by utilizing the known number of transients; thus P = 17/482 = 3.5 E-2 per transient. This would be an appropriate value to use except that it does not make any distinction between the type of transient which caused the SRV's to actuate. Note that the EG&G estimates indicate that the frequency with which transients of all types occur is $482/62 \approx 7$ per reactor year consistent with the rate indicated in Section 3.3.1.3.1.

The EG&G study also develops a failure rate based on demand (i.e., per SRV actuation) by taking into account SRV population in each plant. On this basis a demand failure rate of 4.5 E-3 is obtained with an error range of 1.5. Note that by use of this form of failure rate, plants with more numerous valves would be penalized more severely (ie, would have higher probability of failure) than plants with fewer valves.

In the present study, the demand failure rate deduced in the EG&G report is employed to develop the failure rate for the event P. Specifically, for quantification of the dominant accident sequences we take

$$P = 1.5 * N * 4.5E-3$$

where N represents the number of valves actuated by the selected transient and the 1.5 multiplier, corresponding to the error range, is employed for conservatism.

3.3.1.3.3 Frequency of the Event D

In the present context, failure of the VSS is considered to have occurred if the SRVDL associated with the SORV has ruptured. Therefore, the failure rate for this situation derives from combining the probability that an SRVDL has ruptured due to valve actuation with the probability that the ruptured line is connected to the SORV. The latter probability, of course, is 1/N, where N has been defined in Section 3.3.1.3.2. Based on these considerations the failure rate for the event D is taken to be

$$D = 7.4E-5/N$$

where we have utilized the bounding value of pipe failure rate as discussed in Section 3.2.

3.3.1.3.4 Frequency of the Event Z

Since actuation of the CSS represents simply one of the modes in which the RHR system is designed to operate, its unavailability would not be expected to differ greatly from that of the latter under ordinary circumstances. For the Limerick plant, the unavailability of the RHR is quoted as being 3.5 E-8 in the absence of the P event increasing by a factor of about 30 when challenged by two SORV's. The WASH-1400 value corresponding to the first of these unavailabilities is 1.0E-6 per challenge. If probabilities of this order of magnitude were employed for the failure rate Z, it would clearly imply accident frequencies too low to represent a significant contribution to risk. Of course, the circumstances for which we need the failure rate Z are not "ordinary"; specifically they correspond to a sequence of events in which the P and D failures have already occurred. Thus, as with the 30-fold increase in unavailability due to the P failure, the effect of the D failure will also influence the unavailability of the CSS.

The influence of the D failure on the CSS unavailability is primarily a function of the short time interval available. In the present context, we consider that the Z failure has occurred if the CSS has not been actuated within 10 minutes of the occurrence of the transient. This is a very short time in terms of operator reliability particularly when it is recognized that the prevailing conditions are "stressful". (Containment pressure exceeding design within 4 minutes after the transient is judged to be very stressful.) Under such conditions, the WASH-1400 estimate that the operator would fail to act correctly (here, actuate the CSS) lies somewhere between a 10 to 90% probability. There is also some indication that in some plants actuation of the CSS under the condition of high drywell pressure is precluded for a period of ten minutes or more due to the so-called "lockout" feature.[8] This logic is introduced to assure that all LPCI pumps remain dedicated to maintaining coolant inventory.

Based on all of these considerations, we judge that manual actuation of the CSS is dominated by operator reliability and its successful actuation is as likely to occur as not. Accordingly, the failure rate for the event Z is assigned the value 5.0 E-1 per demand.

3.3.1.3.5 Frequency of the Event W

The inability to remove residual heat from a BWR containment given the events P and D and successful actuation of the CSS has not been quantified to date. Thus, since the limited scope of the present study precludes fault tree analysis of the relevent systems, only a qualitative bounding estimate for this failure rate is employed. For this purpose, we identify certain key features which arise in the TPDW sequence; namely, the fact that the containment (and any equipment contained therein) are exposed to design pressure and temperature exceedance for about 15 to 20 minutes during this scenario (Figure 5), as well as the potential for exposure to flooding due to actuation of the CSS. With respect to the first of these effects, we judge that they are only slightly more severe than the conditions to which the equipment is exposed during a LOCA. In this case, although design parameters are not exceeded, the exposure times extend for many hours, if not days. With respect to flooding induced degradation of system capability, GE spokesmen[9] have indicated that no such effect is to be expected. Based on these considerations, we judge that the unavailability of the heat removal systems in the TPDW sequence is not likely to exceed the unavailability during a LOCA by more than an order of magnitude. The WASH 1400 value for this event was 1.25 E-4 while for the Limerick plant it is taken as 1.6 E-4. Accordingly, for the present study, we take

W = 1.5 E-3.

3.3.1.4 Quantification of Accident Frequencies

Using the values developed in Section 3.3.1.3, the frequency of the dominant accident sequences defined in Sections 3.3.1.1 and 3.3.1.2 become

$$f(TPD) = T*P*D*(1-Z)$$

$$= (4.0)*(1.5*N*4.5\text{ E-}3) * \left(\frac{7.4\text{E-}5}{N} \right) * (5.0\text{E-}1)$$

$$f(TPD) = 1.0\text{ E-}6$$

$$f(TPDW) = T*P*D*(1-Z)*W$$

$$= (1.0\ E\text{-}6)*(1.5\ E\text{-}3) = 1.5\ E\text{-}9$$

$$f(TPDZ) = T*P*D*Z = 1.0\ E\text{-}6$$

3.3.1.5 Comparison of Accident Frequencies With Earlier Sequence Frequencies

In Table 3.2, the frequencies of the accident sequences developed in this study are compared with those of selected sequences taken from WASH-1400 and the Limerick PRA. All of the latter are considered to be significant contributors to risk. The frequencies for sequences involving transient initiators are all adjusted to correspond to the turbine trip as the initiator. Thus, for the sequences taken from WASH-1400, the frequencies reported therein were reduced by a factor 4/10 (see Section 3.3.1.3.1).

The comparison clearly indicates that the frequencies developed in this study are comparable to those analyzed previously. If the consequences of these sequences are also comparable, it would imply that the TPD sequences are also significant contributors to risk. The consequences of these three sequences are considered in the next section.

3.3.2 Accident Consequences

3.3.2.1 The TPD Sequence

As indicated in Section 3.3.1.2.1, this sequence does not lead to a core melt. It is postulated, however, that some leakage out of the containment occurs due to design pressure exceedance. This situation closely resembles the A accident sequence of WASH-1400 which corresponds to a large LOCA but with all safety systems functioning successfully. Thus, the consequences of this accident are relatively minor and are characterized by being placed in Release Category 5. We judge that the TPD sequence is most likely in this category as well.

Since its frequency is two orders of magnitude less than the A sequence (Table 3.2), we conclude that this sequence will not be a significant contributor to risk.

3.3.2.2 The TPDW Sequence

This sequence has the following features in common with the TW, TPW and TQW accident sequences; all scram successfully and all have containment failure by overpressure prior to core melt (the W failure). On this basis, we would expect the consequences for all to be essentially the same. One difference, however, can be cited which would indicate that the TPDW sequences may have more severe consequences. This relates to the fact that, due to the steam bypass, some percentage of the fission product release during core melt will not experience the scrubbing action provided by passage through the suppression pool. Since the consequences could be somewhat more severe and since the frequency of this sequence is comparable to that of the TQW sequence (Table 3.2), we conclude that the TPDW sequence may represent a signficant contributor to risk. A more detailed examination of the effects of design pressure (and temperature) exceedance and flooding effects on the heat removal systems unavailability would need to be performed to arrive at a more definite conclusion.

3.3.2.3 The TPDZ Sequence

This sequence has a relatively high frequency, particularly when compared to the LGS-PRA results (Table 3.2). Also, because containment rupture occurs only ten minutes after reactor scram, we would expect the consequences to be more severe than any of the sequences involving the W event, since, in these cases, containment rupture does not occur until 20-25 hours after scram. The severity is further compounded by fission product bypass of the suppression pool. In our judgement, we would expect this sequence to contribute significantly to risk in all of the more severe release categories defined in WASH-1400.

4.0 Conclusions and Recommendations

As a result of this study, it is concluded that transient initiated accident sequences involving the failure of an SRV to reseat combined with the rupture of the associated SRVDL can occur at frequencies comparable to or greater than that of many accident sequences now considered to be significant contributors to risk. It is also found qualitatively that the consequences of these sequences are potentially severe.

At the present time, such sequences have not been included in any risk assessment that we are aware of. There is no plausible reason for this to be so. We recommend that such sequences be included in any on-going or future PRA's for Mark I and Mark II plants.

There are three major areas of uncertainty associated with quantification of the accident sequences examined in this study. These are the pipe and SRV failure rates, operator reliability, and the effect of design pressure and temperature exceedance and flooding on safety systems. Considerable additional work in these areas is needed and would be appropriate.

It is our judgement that the first of these can fruitfully be approached generically by the NRC staff. For example, Licensee Event Report summaries could be restructured to provide a continuous updating of the number of challenges experienced by SRV's and SRVDL's in BWR plants. Such updating is particularly crucial in the case of SRV's since these are sometimes replaced with improved equipment for the explicit purpose of reducing failure rates.[10] Also, a serious effort to establish the statistical material properties of the pipes used for SRVDL's should be mounted under the auspices of the NRC MEB.

Finally, insofar as the other areas of uncertainties are concerned, additional work is best left, in our judgement, to the individual plant when performing the PRA's. Operator reliability and the unavailability of a manual system such as the CSS involve too many plant unique features to be usefully treated generically.

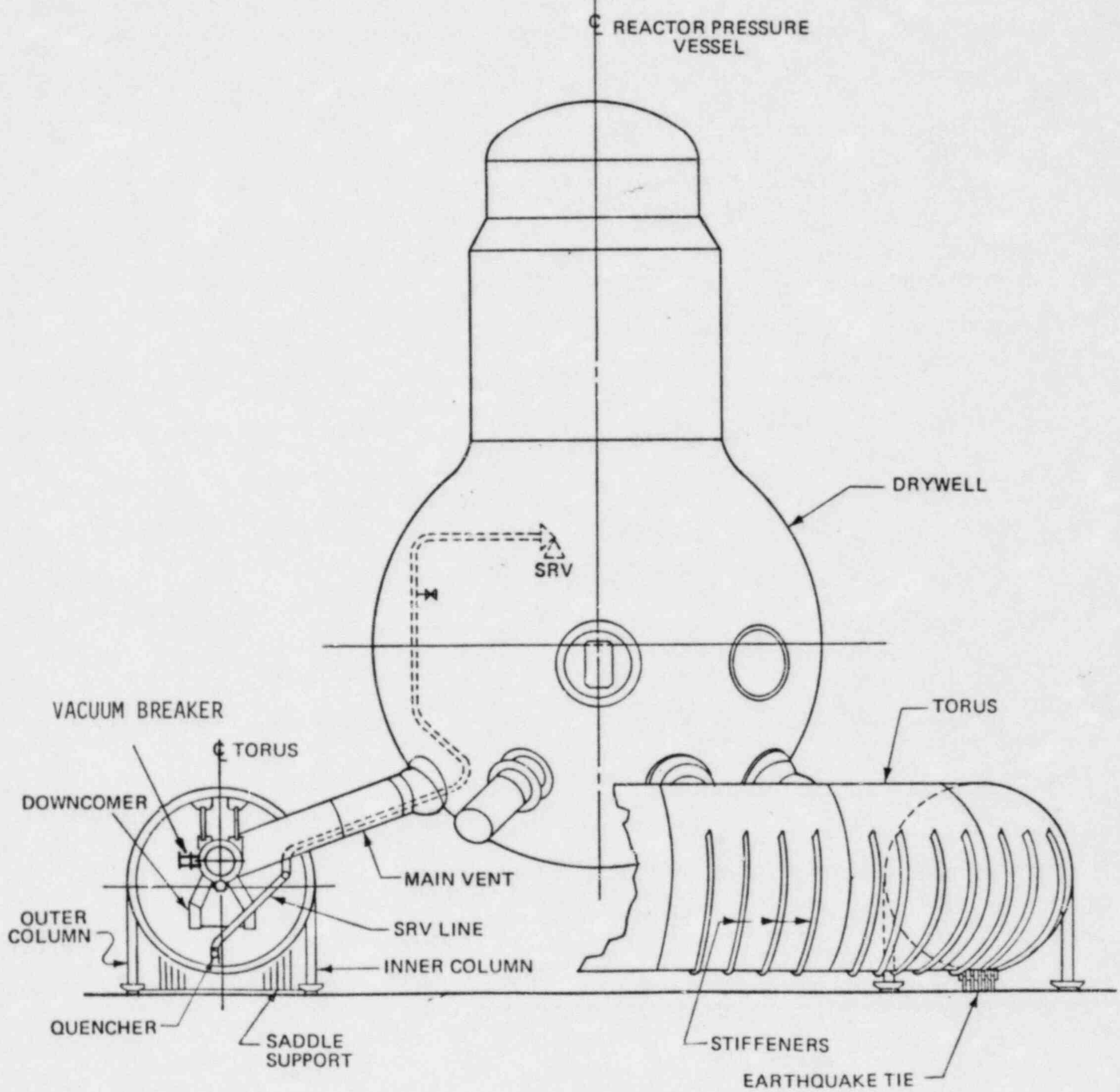


Figure 1 - Mark I Pressure Suppression Containment

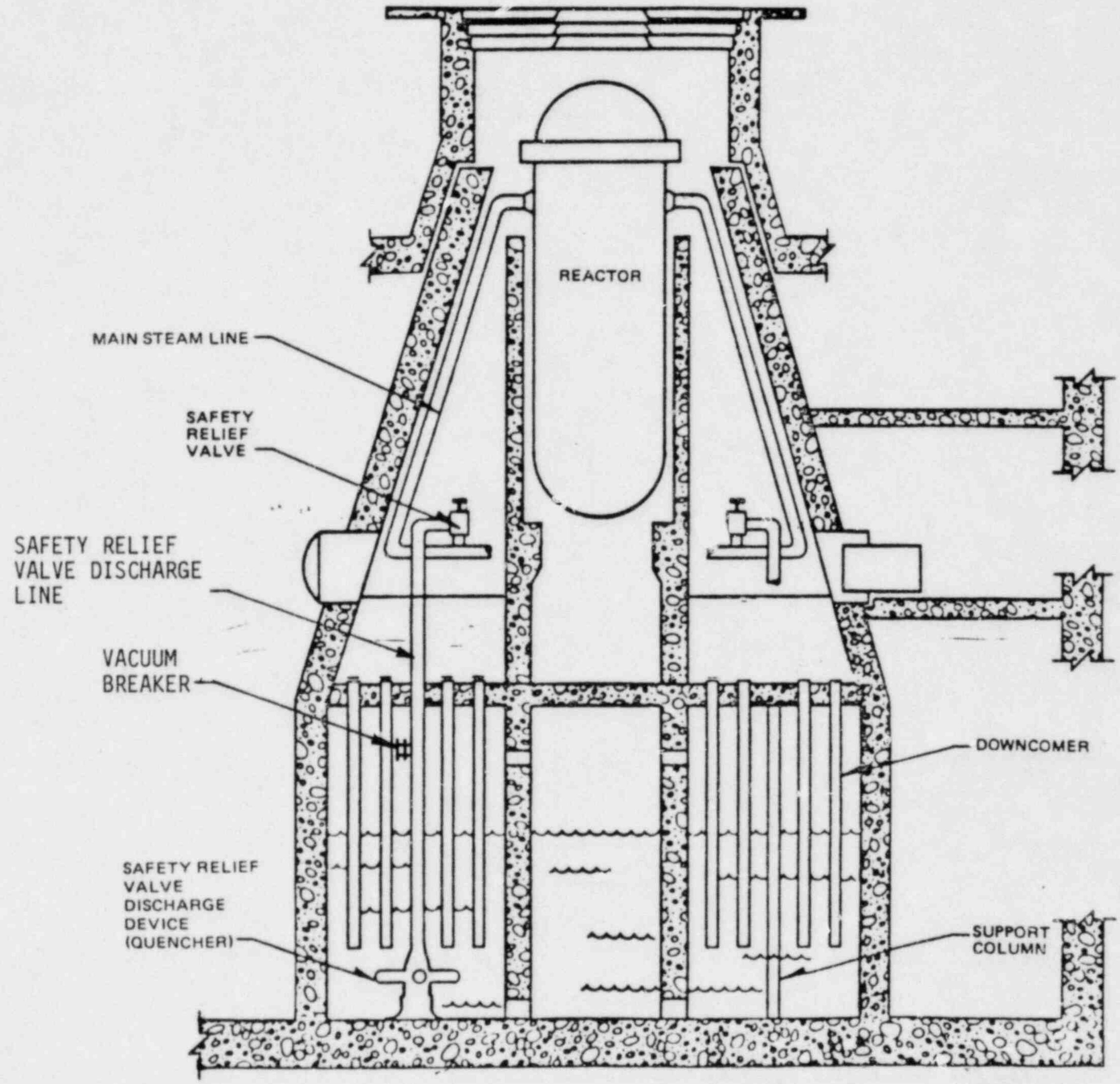


Figure 2 - Mark II Pressure Suppression Containment

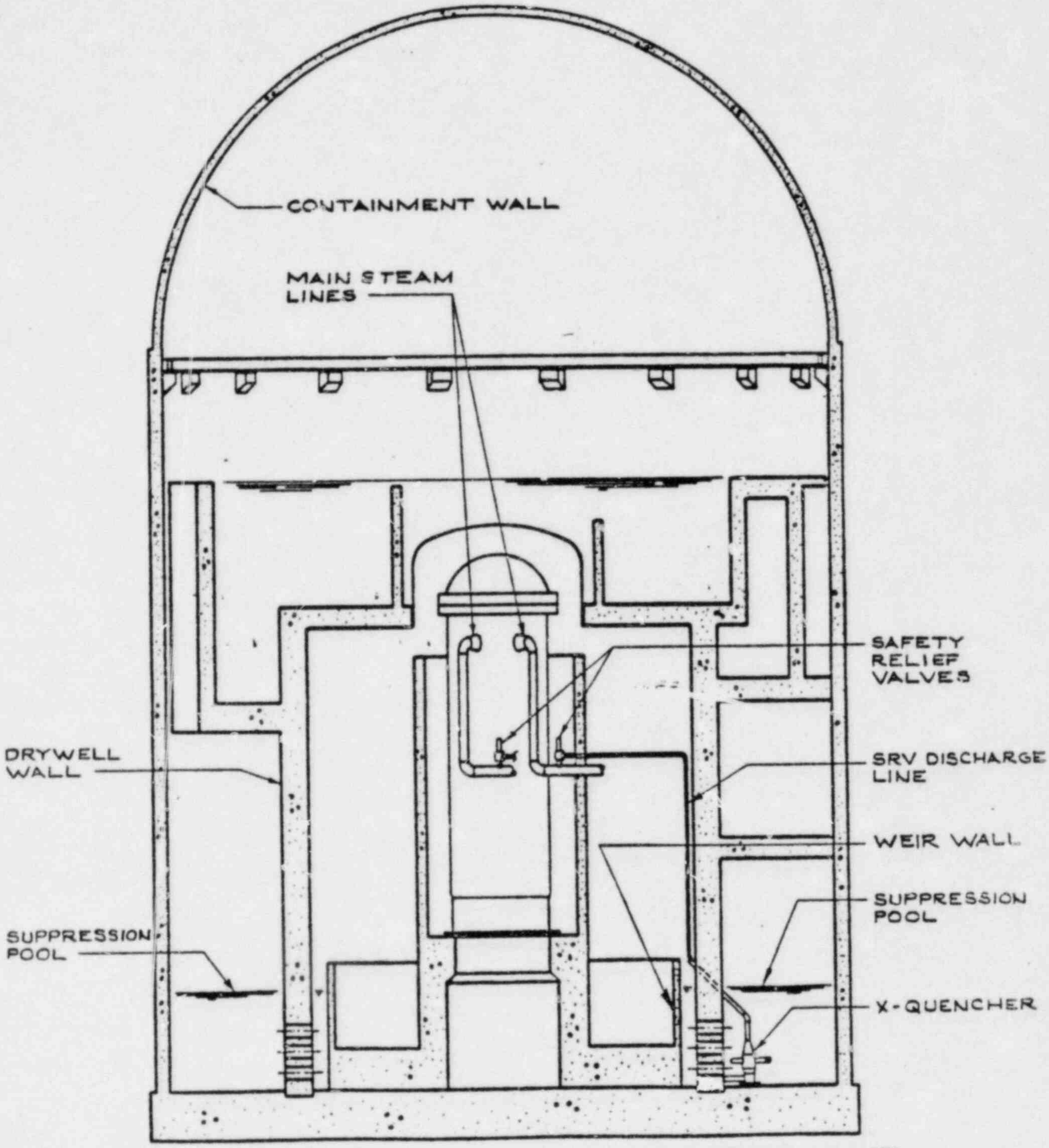


Figure 3 - Mark III Pressure Suppression Containment

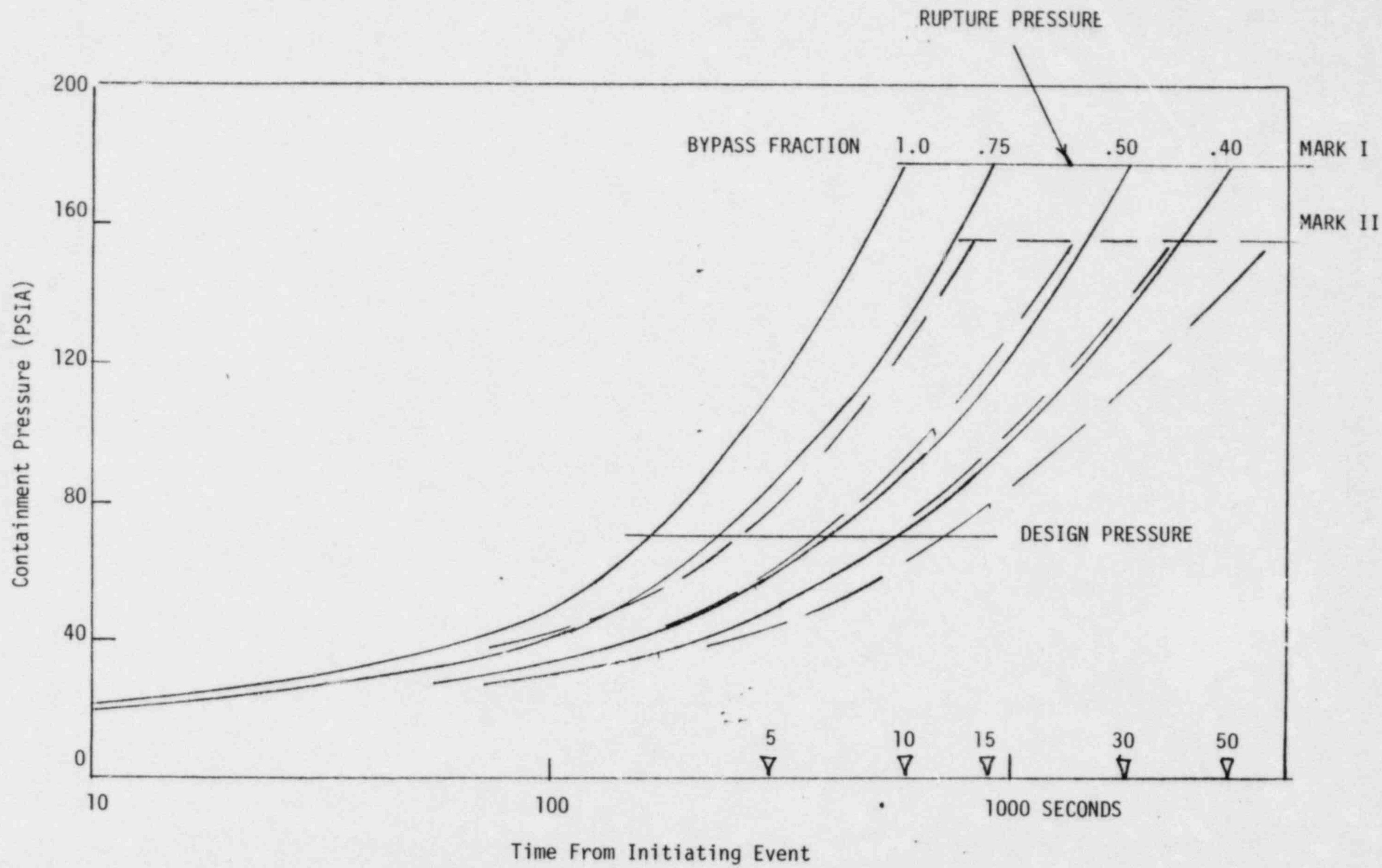


Figure 4 - Containment Response to SORV + SRVDL Rupture

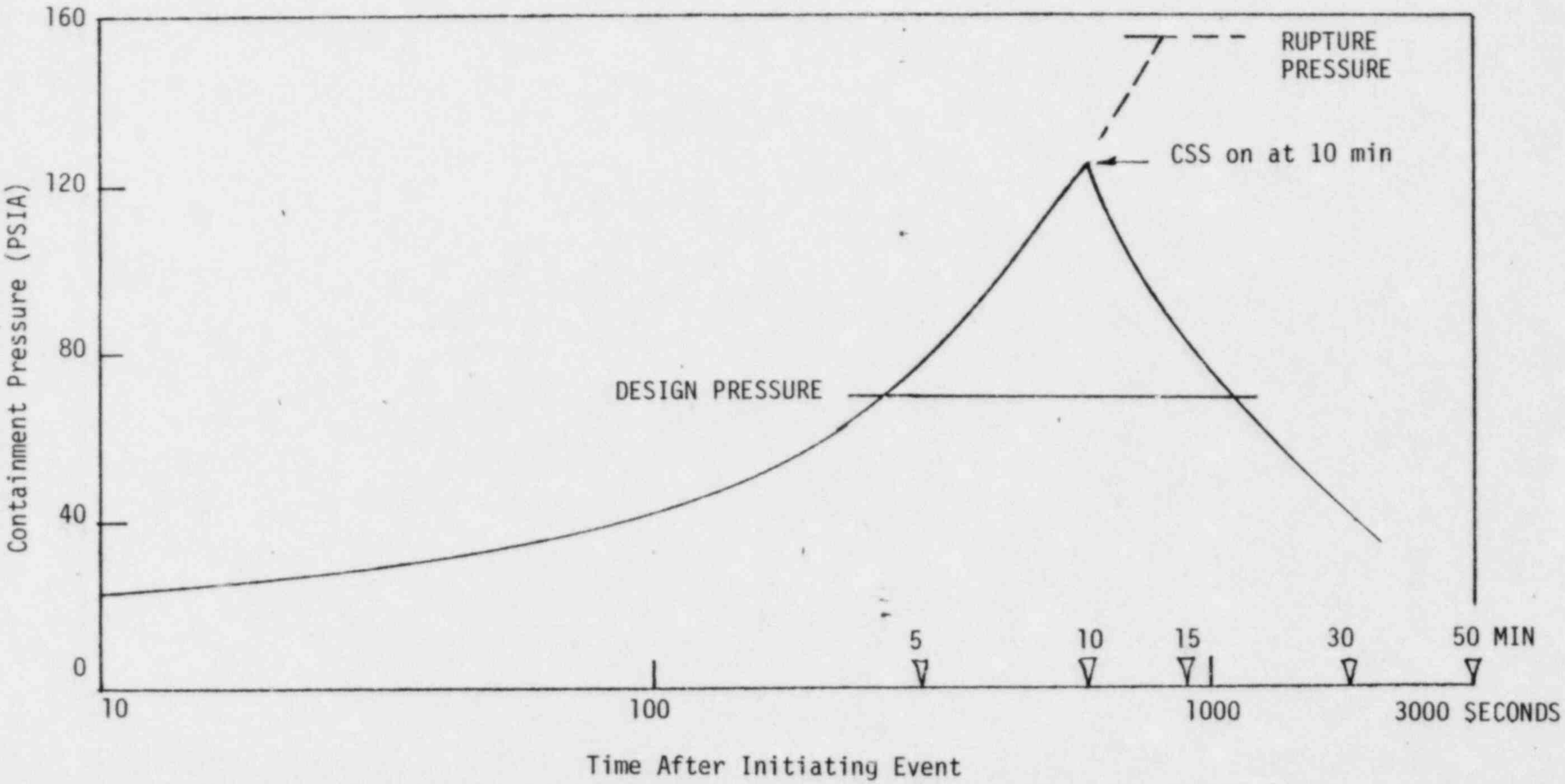
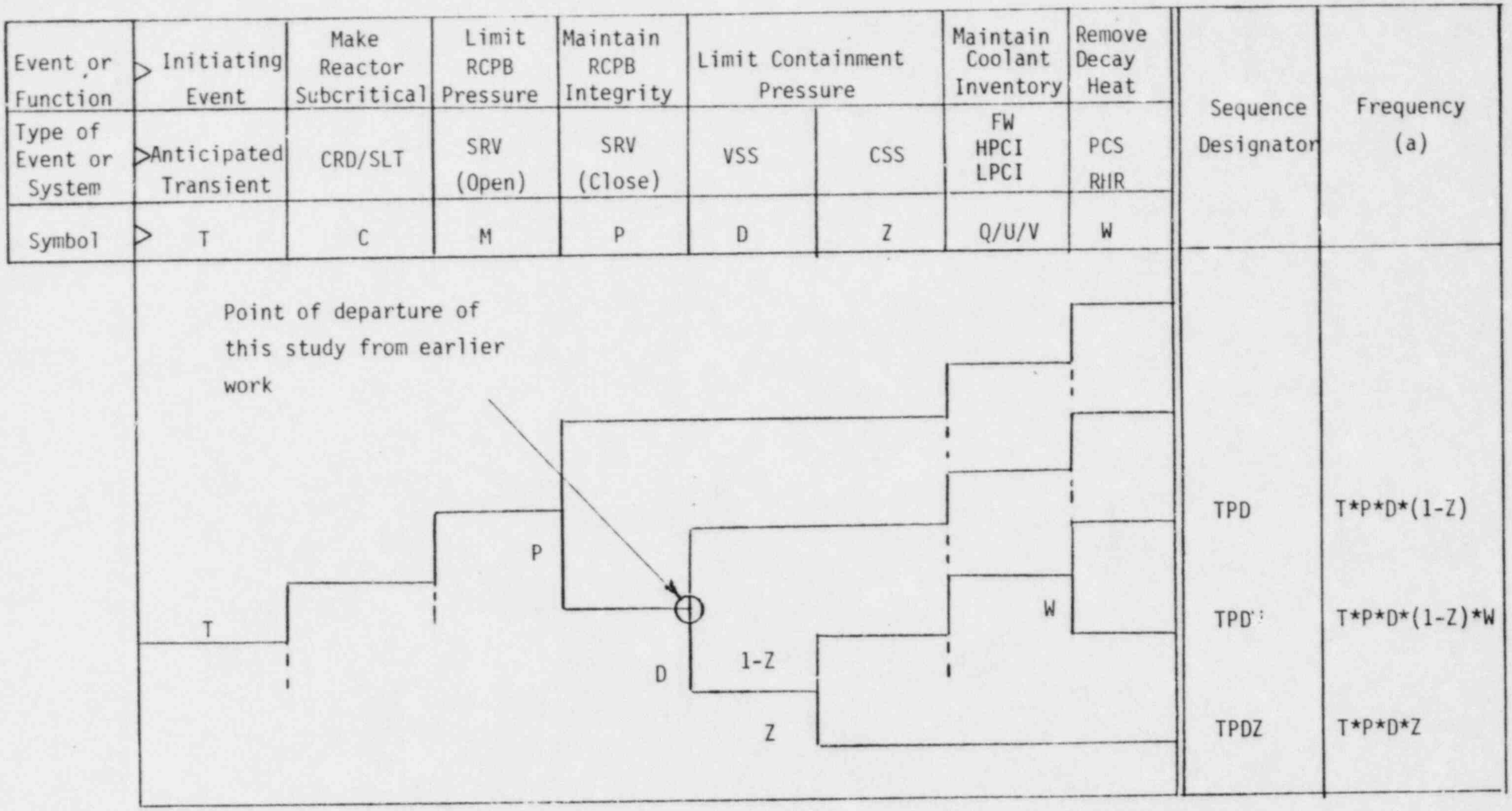


Figure 5 - Mark II Containment Response to SORV + SRVDL
Rupture With 100% Bypass and Actuation of CSS



(a)The symbols T,P, etc. are used to designate both the event and its frequency.

Figure 6 - Simplified Transient Event Tree Used in Present Study

Table 3.1 Pipe Failure Rates

Failure Rate (per demand)	Load Cycles	Corrosion	Method of Estimating	Remarks
7.4E-5	---	--	Operating Experience (Appendix C)	Upper Bound Value
1.0E-7	0	No	PMD (Appendix B)	"As Built" Best Estimate
3.0E-6	7001	No	"	Strength Reduced 10%
1.0E-4	14001	No	"	Strength Reduced 20%
7.5E-7	0	Yes	"	40 Plant-Years
2.6E-5	7001	Yes	"	40 Plant-Years

Table 3.2 Comparison of Dominant Accident Sequence Frequencies

This Study	WASH-1400 (Ref. 1)	LGS-PRA (Ref. 2)
TPD - 1.0E-6	A - 1.0E-4	TPW - 4.0E-7
TPDZ - 1.0E-6	TW - 8.0E-6	TQUV - 3.0E-8
TPDW - 1.5E-9	TQUV - 2.0E-7	AJ - 6.4E-8
	AJ - 1.3E-8	TQW - 3.0E-9

5.0 References

1. NUREG 75/104 (WASH 1400), "Reactor Safety Study. An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," October 1975.

2. Philadelphia Electric Co. "Limerick Generating Station Probabilistic Risk Assessment," Rev. 3, April 1982.

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Appendix A - Containment Pressure Response

This appendix describes the calculations performed to estimate the containment pressure and temperature levels associated with the accident sequences developed in Section 3.3 of the report. The task basically consisted of computer runs to generate containment pressure histories resulting from an SRV pipe failure in a BWR wetwell airspace, thereby bypassing the vapor suppression system. Both the Mark I and Mark II containment systems were considered in the study. The systems modeled consisted of two compartments, wetwell and drywell, connected by vacuum breakers with a range of break sizes being considered.

The computer code utilized for the calculations was the CONTEMPT-LT/028 code as described in Reference 1. CONTEMPT-LT is a computer code capable of treating the long-term response of water-cooled nuclear reactor containment systems due to a postulated loss-of-coolant accident. The program calculates the time variation of compartment pressures and temperatures, mass and energy inventories, heat conducting structure temperature distributions and energy exchange between compartments. In addition, the code also possesses various models to handle compartment mass and energy transfers and describe various engineered BWR safety systems. However, in order to accommodate the steam bypass associated with the SRV pipe failure and subsequent mass and energy addition to the wetwell airspace, modifications and additional coding were required. The program changes were concerned with the following areas:

a. Wetwell mass and energy addition - a table input was added to provide the time history of steam mass and energy being discharged from a primary system safety-relief valve (SRV) corresponding to the reactor transient chosen for consideration. During a normal transient involving SRV actuation, the steam flow described by this table would be discharged to the suppression pool and thereby condensed. The transient used in the present study is shown in Figure A.1. It corresponds to the steam flux expected to issue through a single SORV following a turbine trip with about 30% bypass of the main turbine. The steam flux decreases monotomically, of course, in response to the decay in RPV pressure as indicated in the figure.

In the accident under consideration in the present study, it is postulated that all or part of the steam flux defined in Figure A2 will be discharged into the wetwell airspace thereby bypassing the vapor suppression system. In order to consider the effects of varying break size, a factor was used in conjunction with this table to indicate what fraction of the discharged mass flow enters the vapor space. For those cases with only partial bypass of the vapor suppression system, the fraction of mass not entering the airspace is added to the suppression pool to conserve the mass and energy deposition thus enabling a valid comparison between the various cases considered. The appropriate coding was included to update the mass and energy inventories for the wetwell.

b. Mass and energy transfer between wetwell and vent system - as the pressure increases in the wetwell airspace due to the introduction of steam, the suppression pool water will be forced up into the vent system. In the present scenario, the water slug will be accelerated up into the downcomers until the pressure differential across the slug is alleviated by actuation of the vacuum breakers. The level or hydrostatic head achieved by the slug will asymptotically be equal to the pressure difference at which the vacuum breakers between the wetwell and drywell open. Coding changes were made to account for this phenomenon.

In order to verify that the modifications were correctly coded, numerous qualification checks were performed. These included mass and energy checks at various times during the transient as well as reruns of test cases to confirm that the changes did not affect the original code. In addition to the above, a time step study was performed to assure that an adequately small timestep was used for the study. This study was necessitated by the accuracy required to predict the motion of the water slug in the downcomers.

Because the suppression pool is being bypassed by the SRV steam discharge, the passive heat sinks and containment spray system (CSS) are of prime importance for control of the accident sequence. The passive heat sinks utilized in the calculations for both the Mark I and Mark II containments are indicative of the structures available for storage of heat and for condensation of the steam in the wetwell and drywell airspaces. The structures modeled included the containment walls, both steel liner and concrete where applicable, and the vent system components. The Uchida condensing steam heat transfer coefficient was used in the current study. This heat transfer coefficient option is dependent on the compartment mass ratio of air to steam and is capable of handling both sensible heat and mass transfer for either saturated or superheated conditions. A more detailed discussion of the boundary condition option can be found in Reference 1.

The containment spray system was the only active BWR system investigated during the SRV pipe break calculations. When employed, it was assumed to be operator actuated at 10 minutes into the transient to mitigate the accident. The system was specified to deliver 5500 GPM of water at a temperature of 65 °F with 96% being directed to the drywell airspace. No heat exchangers were considered in the study and it was assumed that the spray system was capable of maintaining the same outlet temperature. This portion of the calculation was purposely kept simple since the primary reason for using the CSS model was to determine if the accident would be mitigated. Therefore, the actual transient which would have been obtained with a detailed calculation of the actual system was of little importance to the study.

The results of the calculations for both the Mark I and Mark II containments are summarized in Figure A-2. Presented in this figure are the containment responses without containment spray as a function of time for varying bypass fractions. It is obvious from this figure that the containment pressure transients are significant for the accident scenario chosen. In fact, the design pressures for the case of 100% bypass are exceeded in about 3 minutes for a Mark I containment and 4 minutes for a Mark II. (The design pressure for the two containments are not identical; they appear so only because of the scale chosen for the graph.) The calculations also indicate that the rupture pressure will be exceeded in about 10 minutes for the Mark I and 14 minutes for the Mark II for the case of 100% bypass. The values of the rupture pressures used in the calculations were obtained from References 2 and 3 for the Mark I and Mark II containments, respectively. An important point illustrated by this figure is that the severity of the accident is shown to diminish with reduced vapor suppression bypass fractions and thus, the time available for corrective action is greatly enhanced. In order to investigate what effect the CSS would have on the accident, the case of a Mark II containment with 100% bypass was considered. An indication of the effectiveness of the containment spray system is shown in Figure A3. The figure illustrates that the CSS is a very effective means for mitigating the accident at even the most severe accident conditions. Similar mitigating effects of the CSS were also obtained for the Mark I containment.

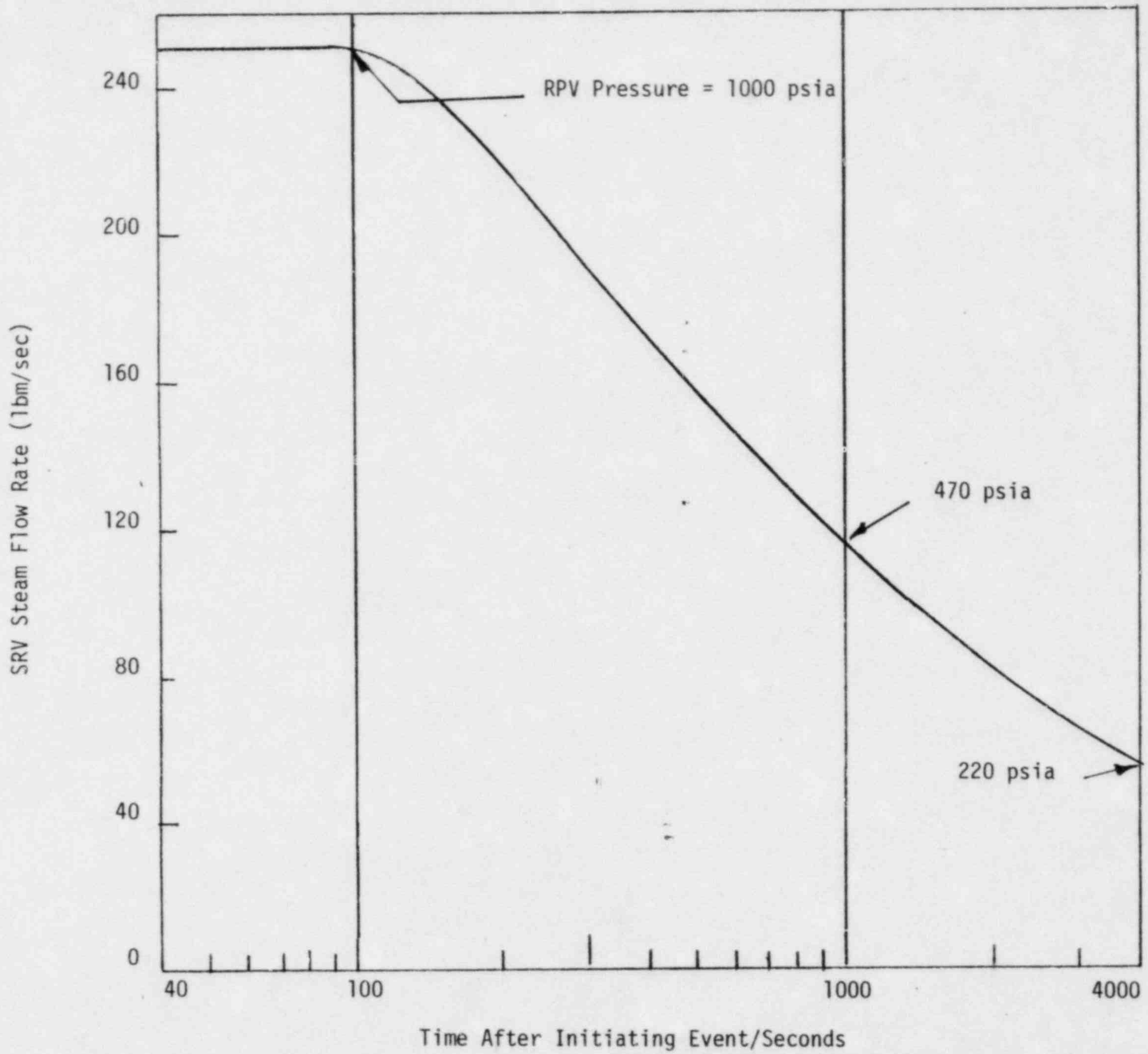

Figure A1 - SRV Steam Flow Rate Transient

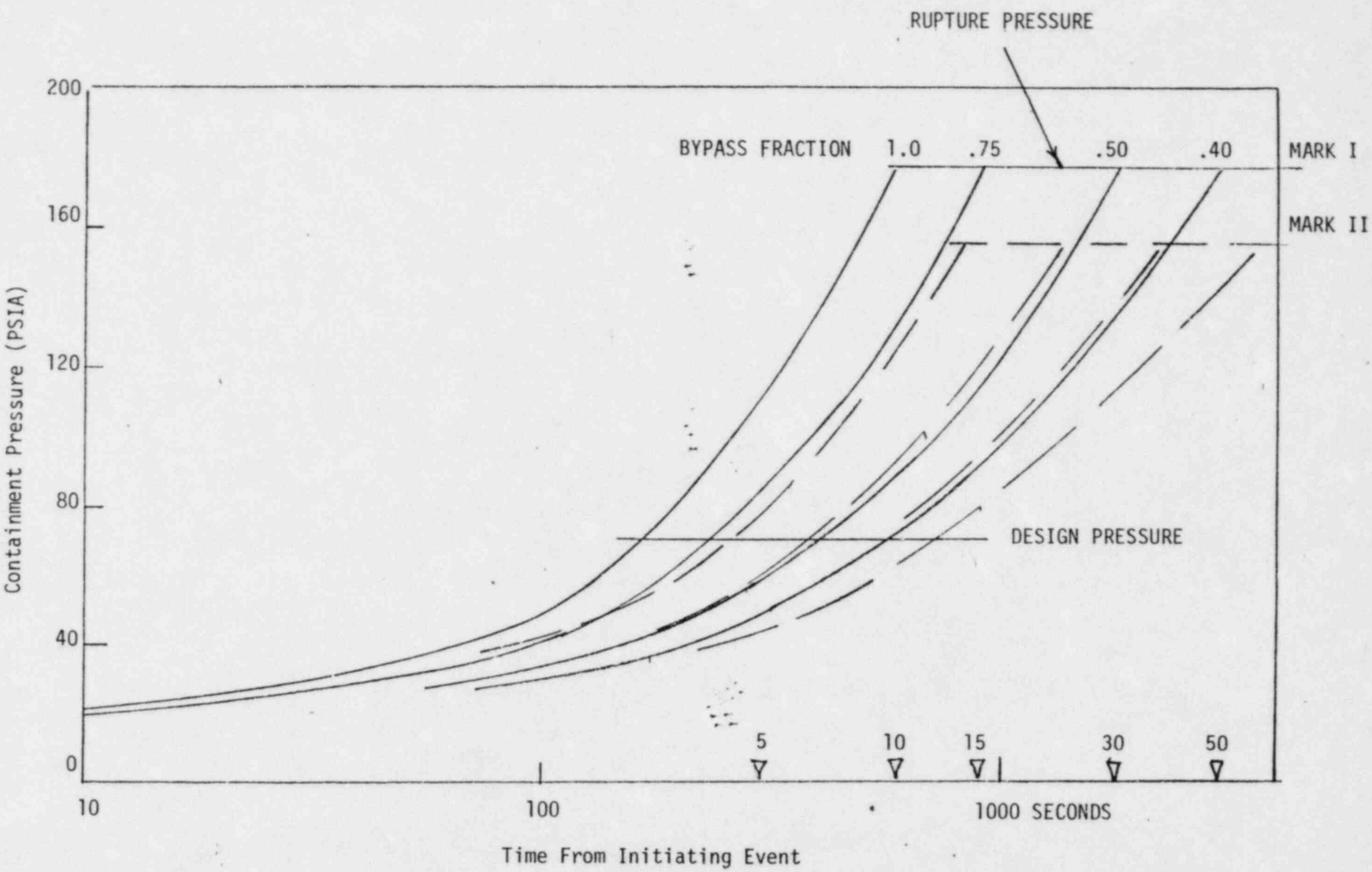

Figure A2 - Containment Response to SORV + SRVDL Rupture

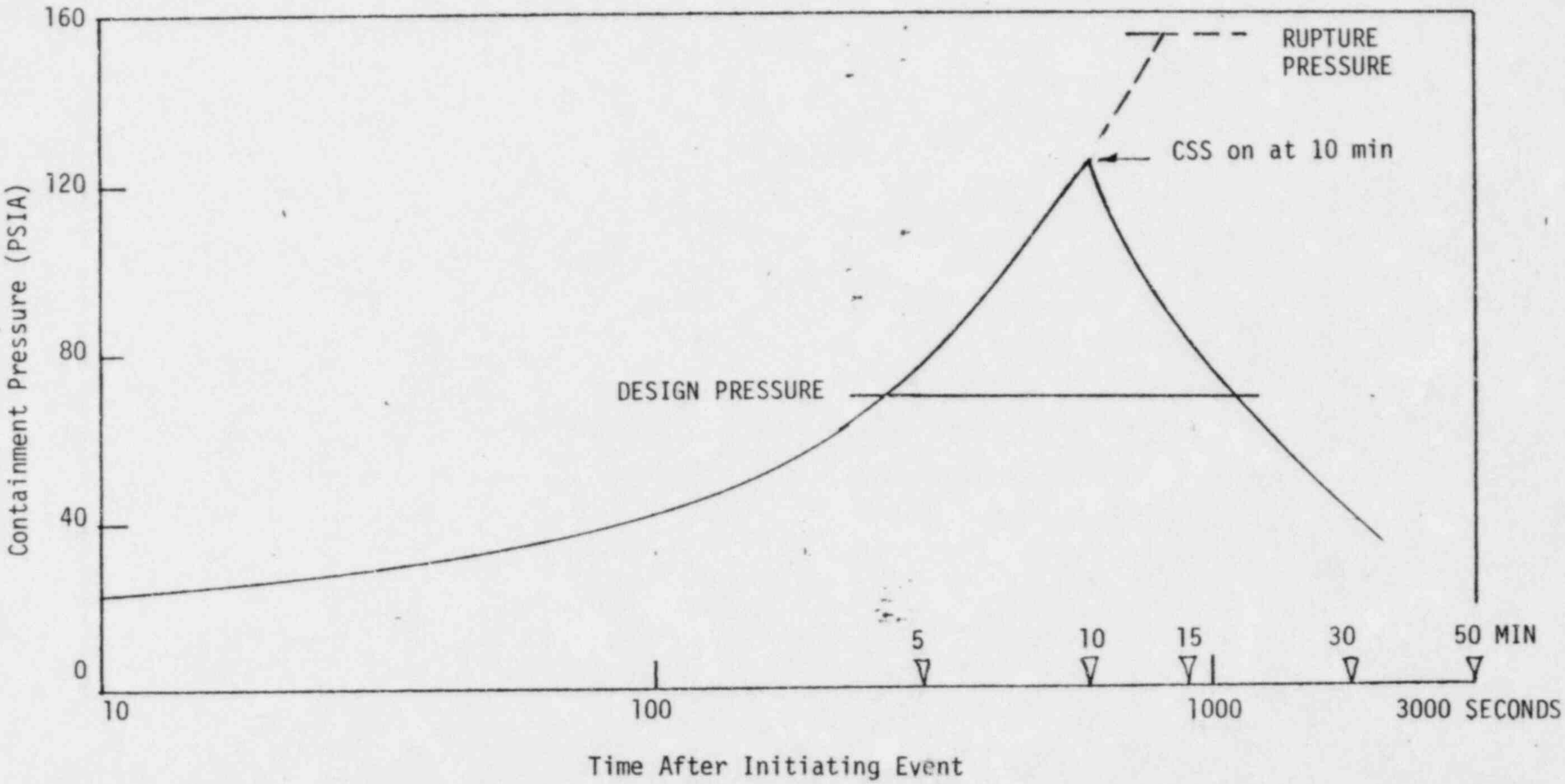

Figure A3 - Mark II Containment Response to SORV + SRVDL
Rupture With 100% Bypass and Actuation of CSS

Appendix A - References

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Appendix B - Evaluation of Pipe Failure Rates via Probabilistic Mechanical Design - The Stress Strength Overlap Method

1. Introduction and Description

The concept of a safety factor in design implies that there is a certain separation of the strength of a particular component from the stresses applied to that component. What must be remembered is that the strength and stress in question are mean values (or conservative estimates of mean values) of a complete strength and stress distribution. For any reasonable distribution, including the normal, an overlap of stress and strength, however small, is unavoidable (Figure B.2-1a). Rather than use a safety factor, the adequacy of a component for which strength and stress distributions are known can be determined from the probability that strength exceeds stress (reliability), or the probability that strength is less than stress (failure). By definition, reliability equals one minus the failure probability. Given the variable nature of the parameters encountered in the physical sciences, the probabilistic approach to design is a natural choice.

The amount of overlap of the strength and stress distribution is a measure of the probability of failure. If the distributions are known, the reliability or, alternately, probability of failure, can be computed. As described in Reference 1, the salient points in the calculation are the following:

Let s, the stress, and S, the strength, be normally distributed random variables with density functions f(s) and f(S). Let δ = S-s. Since f(s) and f(S) are normally distributed, so is $f(\delta)$. Therefore,

$$f(\delta) = \frac{1}{\sigma_\delta\sqrt{2\pi}} \exp\left[- \frac{1}{2} \left(\frac{\delta-\bar{\delta}}{\sigma_\delta}\right)^2\right]$$

where

$$\bar{\delta} = \bar{S}-\bar{s} \text{ and } \sigma_\delta = \sqrt{\sigma_S{}^2 + \sigma_s{}^2}$$

A bar indicates the mean and σ the standard deviation (or standard deviation estimator) of the variables indicated.

Figure B.1-b shows a sketch of $f(\delta)$. The density of δ to the right of zero is the reliability, while the failure density is represented by the portion less than zero. In other words, the reliability R is the probability that $\delta > 0$ and so $R = \int_0^\infty f(\delta)\, d\delta$ and the failure probability P_f is given by 1-R or $P_f = \int_{-\infty}^0 f(\delta)\, d\delta$. To evaluate either of these integrals, one has only to make the transformation which relates δ to the standardized normal variable Z and look up the integral values which are tabulated in standard tables of normal functions. The transformation is given by $Z = \frac{\delta-\bar{\delta}}{\sigma_\delta}$. Since the Z coordinate of interest is the one where $\delta = 0$, $Z = \frac{-\bar{\delta}}{\delta} = \frac{\bar{s}-\bar{S}}{\sqrt{\sigma_S{}^2 + \sigma_s{}^2}}$

or $$Z = \frac{|\bar{S}-\bar{s}|}{\sqrt{\sigma_S{}^2 + \sigma_s{}^2}}$$

Therefore, if the distributions of S and s are known, Z can be computed and the reliability or failure probability found from tables of the standard normal function. (See Reference 1 for a more detailed discussion.)

2. <u>Obtaining Parameter Values</u>

If stress and strength are assumed to be normally distributed, then the mean and standard deviations for each must be found to compute the probability of failure.

The stress data for this study was obtained from measurements made in the SRV lines of the Caorso nuclear plant in Italy during SRV discharge tests. Details of measurement location, strain gage arrangement and data interpretation can be found in Reference 2. Although many measurements were made, Reference 2 cites only 7 data points for which all stresses, i.e. pressure, thermal, etc., are given. Two additional data points can be found in Reference 3. These nine values lead to a mean stress of 23,727 psi, with a standard deviation of 2561 psi. Implicit in the use of this data to determine failure rates for the present purpose is the assumption that the strain gages in Caorso were located on the most highly stressed part of the SRV line.

To obtain the strength parameters, the SRV pipe material properties and their variability must be established. In the wetwell, the Caorso SRV piping is 10-inch, Schedule 80. The material is specified as A106, Grade B, carbon steel. The yield strength for this material is listed as 35 Ksi, while the ultimate strength is given as 60 Ksi. These values were taken to be conservative estimates of the means of the assumed normal yield and ultimate strength distributions. A fairly exhaustive search was made for strength variability data on A106 Grade B but no statistical information for this particular piping material could be found. In order to obtain a reasonable estimate of the standard deviation, data which was available for other carbon steels was used as described in the following: In Appendix 10.A of Reference 4, data on eighteen carbon steels of various kinds are listed. Mean, standard deviation and sample size are given for ultimate tensile strength and tensile yield strength. A weighted average standard deviation as a percentage of tensile or yield strength was found from averaging this data and weighting it by the sample size for each entry. It was found that for the ultimate tensile strength the weighted average standard deviation was 4.6% of the mean, while for the tensile yield strength the weighted average standard deviation was 6.6% of the mean.

A choice must be made regarding the value assigned to the strength at failure, $\bar{S}$, i.e., the plastic collapse load. Choosing the yield strength would be overly conservative, since the carbon steel pipe can be assumed to be quite ductile and therefore can undergo substantial plastic deformation before rupture failure. Construction quality can also be considered high since these pipes are designed and fabricated to certain code requirements. One may be led to conclude then that the ultimate strength would be an appropriate choice for $\bar{S}$. This would be too optimistic an assumption however, since the potential failure at smaller loads due to weldment imperfections or other stress concentrations must be recognized. Therefore, a reasonable value for failure strength $\bar{S}$ is judged to be a stress level halfway between the yield and ultimate strength of A106 Grade B. This choice of $\bar{S}$ agrees with the plastic collapse load chosen by General Electric for a generic evaluation of Mark I SRV discharge line integrity (5). The philosophy behind choosing this average of yield and ultimate stress is the same as that expressed for BWR containment failure criteria in (6).

Therefore, if $\bar{S}$ = (ultimate stress + yield stress)/2, the corresponding standard deviation σ_S is then computed from:

$$\sigma_S = \sqrt{\sigma_{yield}^{2} + \sigma_{ultimate}^{2}}$$

In terms of percent of the mean $\%\ \sigma_s = \sqrt{(4.6\%)^2 + (6.6\%)^2} = 8.0\%$ of $\bar{S}$.

For instance, if the values for tensile and yield strength of A106 Grade B carbon steel cited above are used, $\bar{S} = \frac{35000 + 60000}{2}$ or $\bar{S} = 47{,}500$ psi and σ_s = (.080) 47,500 or $\sigma_s = 3{,}800$ psi.

To get additional insight, standard deviations corresponding to the highest and lowest percent of mean value in Appendix 10.A of Reference 4 were also computed. For σ_S these were calculated to be 12.7% and 3.5% of $\bar{S}$, respectively.

3. Sample Calculation

To illustrate how a particular probability value is computed, consider the case of the Caorso SRV lines again where the stress parameters, as indicated earlier, are given by

$$(\bar{s},\sigma_s) = (23727,\ 2561)$$

and the strength was found to be $(\bar{S},\sigma_S) = (47500,\ 3800)$

Then

$$Z = |\bar{S}-\bar{s}| \Big/ \sqrt{\sigma_S^{\,2} + \sigma_s^{\,2}}$$

or

$$Z = |47500-23727| \Big/ \sqrt{3800^2 + 2561^2} = 5.19$$

From a standard table for $\int_Z^\infty f(Z)dZ$ the probabi[illegible] failure is found to be $P_f = 1.07 \times 10^{-7}$.

4. Fatigue and Corrosion Considerations

As stated in NUREG-0661, SRV piping must be evaluated in accordance with ASME Class 2 Rules. The ASME Class 2 Fatigue requirements provide for a stress range reduction factor which is a function of the number of alternating stress cycles as shown in Table NC-3611.2(e)-1 of the 1977 ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC.

Conservatively assuming that the entire applied stress can alternate, one can get an estimate of the effect of fatigue on the failure probabilities by applying the stress range reduction factor directly to the mean value of the strength, $\bar{S}$. For example, using values from the Table cited above and the "as built" value for $\bar{S}$ of 47.5 kips for A106 Grade B carbon steel, between 7,000 and 14,000 cycles $\bar{S}$ = 0.9x47.5 kips or 42.75 kips. Between 14,000 and 22,000 cycles $\bar{S}$ = 0.8x47.5 kips or 38.0 kips. Taking the strength standard deviation σ_S as a fixed percentage of $\bar{S}$, e.g. 8.0% as discussed earlier, one can now calculate a failure probability taking into account fatigue by using the modified strength values and the previously used applied stresses. Figure B.2 shows the variation of the demand failure rate with strength $\bar{S}$.

Corrosion can be accounted for by assuming a certain reduction in pipe thickness during the life of the plant. Using a general rule of conventional design practice, one can assume a 1/6 reduction in pipe thickness due to corrosion over 40 plant years. This change in wall thickness will modify the applied stress values obtained from References 2 and 3. For a thin walled cylinder, like the SRV pipe, thermal stresses are relatively unaffected by thickness, while pressure and bending stresses are inversely proportional to the wall thickness. Modifying the stress components from References 2 and 3 appropriately then leads to a mean stress value $\bar{s}$ = 25,126 psi and a σ_s = 2,691 psi. Now a demand failure rate accounting for corrosion can be found by using the modified stress parameters and the original strength parameters.

Obviously, one can evaluate failure rates using both modified stress and modified strength parameters thus accounting for both corrosion as well as fatigue at the same time. The values for several cases are given in Table 3.1 of this report.

5. Comments on Assumptions and Accuracy

Any discussion of the stress-strength overlap method would be incomplete without mentioning its sensitivity to the assumptions regarding distribution shapes and parameter values.

The high sensitivity of the failure probability to variations in the standard deviation is shown in Figure B.2. Three different strength standard deviations taken as 3.5, 8.0 and 12.7% of the mean failure strength give vastly different failure probabilities as shown in the figure. That such small variations in the standard deviation σ lead to such large differences in the failure probabilities is not so surprising since only the extreme tails of the distributions overlap to determine the probabilities and these tails are sensitive to σ. While Figure B.2 show the effect of varying the standard deviation of the strength, a similar effect would be achieved by varying the stress standard deviation.

So far, only a normal distribution has been considered. The sensitivity of the probabilities to distribution shape is obviously also great since it is the extreme tails of the distributions which determine the probabilities. Even if a distribution appears close to normal for moderate distances away from the mean, the shape of the extreme tails may differ greatly, thereby greatly changing the probability from one predicted using a normal distribution assumption.

An interesting and thorough discussion of the failure probability's sensitivity to assumptions of distribution shape and parameters can be found in Reference 7.

The assumptions made in the present report were based on best estimates made from the available data. That this data is much more sparse than is desirable cannot be disputed. It would be very helpful to have many more data points for the stress distribution which a SRV line is subjected to at its critical point. Much more material testing is needed to conclude with confidence that the strength distribution of A106 Grade B is normal and what the value of the standard deviation is.

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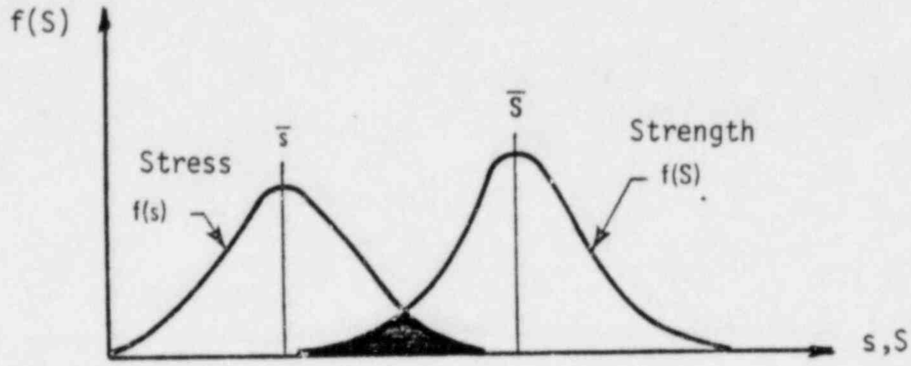
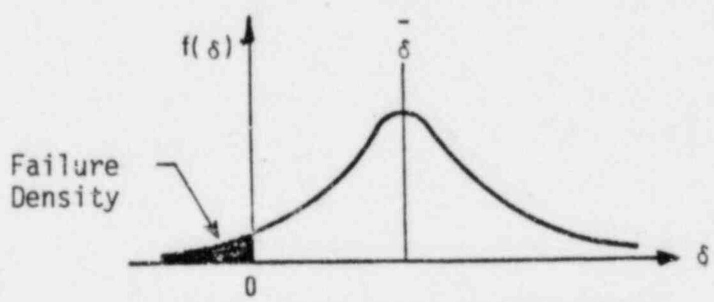
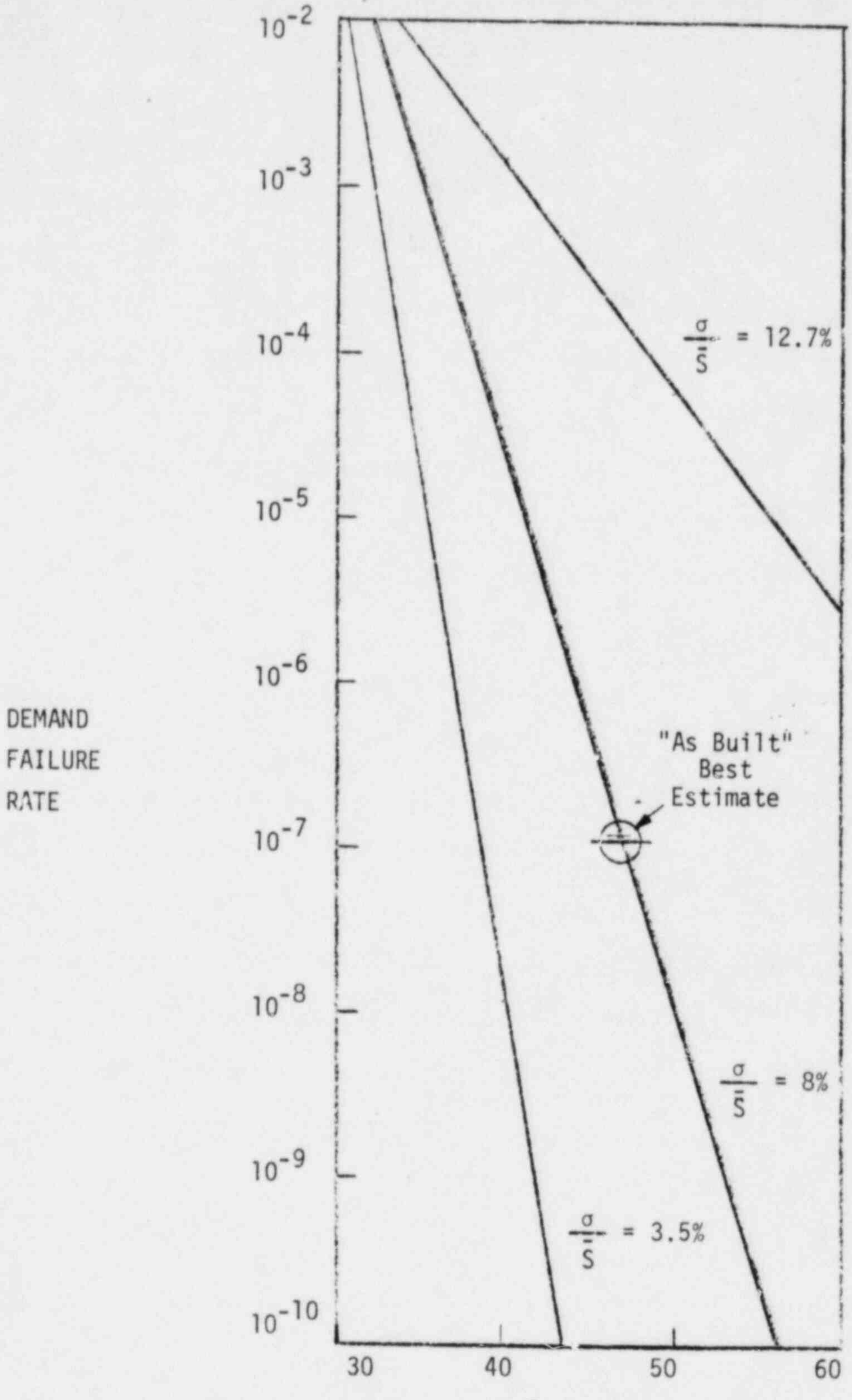


(a) Normal Distributions Showing Overlap



(b) Difference Distribution

Figure B.1



Pipe Failure Rates Estimated by Stress-Strength Overlap Method

Figure B.2

Appendix C - Estimate of Pipe Failure Rates from Operating Experience

The approach used here to obtain a bounding estimate of SRVDL failure rate is similar to that used in WASH 1400 to estimate the probability of a "catastrophic" failure of the RCPB leading to a DBA LOCA. What was done there was to reason that since such an event had not occurred during the total number of plant years in which BWR's were operating, to postulate one such failure during this period would represent a bounding of the frequency of this potential failure. In our case, we have developed this failure rate on a "per demand" basis. Specifically, we have used the results previously cited in Section 3.3.1.3.2[1] to conclude that, on the average, there were a total of $3803/62 \approx 61$ demands made on SRVDL's during a reactor year. We have also estimated from the information supplied in Reference 2 that, through 1981, the total reactor years of BWR operation (all Mark I) is 210. These figures imply that a total of 12,900 challenges to SRVDL's have occurred during BWR experience without a single reported failure.[3] We thus conclude that an estimate of a bounding failure rate can be obtained by postulating one such failure during this period yielding $1/12900 \approx$ 7.8 E-5. The rate shown in Table 3.1 of this report is somewhat smaller than this value because we have also included about 600 demands which have occurred during in-plant and full-scale SRV tests. Brief descriptions of these tests can be found in References 4 and 5.

Appendix C - References

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