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August 13, 1982

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RE: In the Matter of Consumers Power Company (Big Rock Point Nuclear Power Plant), Docket No. 50-155-OLA (Spent Fuel Pool Modification)

Gentlemen:

The Atomic Safety and Licensing Board requested, during the recent hearings (Tr. 1325), a copy of the portion of Consumers Power Company's Probabilistic Risk Assessment ("PRA") for the Big Rock Point Plant concerning containment integrity and its failure probabilities. Pursuant to this request, I am enclosing Chapter 5.0 of the PRA and the related material in Appendix IV of the PRA.

I am also enclosing a letter dated July 20, 1982 from Mr. Vincent of Consumers Power Company to Mr. Crutchfield of the NRC Staff. Revisions to some of the above-cited PRA material are enclosed with Mr. Vincent's letter. Specifically, the probability of containment isolation failure was re-analyzed with the result being a reduction in the probability of such a failure from .25/demand to .06/demand. This matter is discussed in more detail on the second page of Mr. Vincent's letter.

Sincerely,

Joseph Gallo

Joseph Gallo

Encl.: As stated.

cc: Service List

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5.0 RADIONUCLIDE RELEASE ANALYSIS

5.1 INTRODUCTION

In the evaluation of risk from the Big Rock Point Plant, two ingredients are necessary: The probability of events which can lead to degradation of the core, and the consequences of these events. The previous chapter presented the development and probabilistic quantification of the various accident sequences with potential to contribute to public health risk. The result of this analysis is a listing of key accident sequences developed by considering their probability of occurrence and, in a qualitative manner, their potential for producing serious health consequences. The purpose of this chapter is to present the approach taken in quantifying the releases of radionuclides from the fuel and ultimately to the environment outside of containment. The next chapter will deal with the quantification of the effect of the various radionuclide releases on the health of the population surrounding the Big Rock Point site.

The Big Rock Point release analysis is presented in detail in Appendix V and a brief methodology overview was presented earlier in Chapter 3.0. This chapter will concentrate on the selection of sequences for analysis and on a summary of the results of this analysis.

The calculation of radionuclide releases to the environment associated with accidents involving serious core damage is an exercise in evaluating the integrity of the various barriers designed to prevent release of this material. The principal barriers of interest are the fuel and cladding, the primary system, and the containment. Each of these barriers must be violated to produce a significant release of radionuclides to the environment. Since the initial step in producing a release of radioactive material to the environment involves release from the fuel and cladding, analysis of the potential for the occurrence of this release was first required. This analysis is discussed in Section 5.2.

A significant factor in determining the severity of a radionuclide release to the environment is the state of the containment during and after the core degradation process. For this reason, an evaluation of containment failure modes and the conditions necessary to produce those containment failure modes provided the basis for selecting accident sequences to be analyzed. This selection process is discussed in Section 5.3. Finally, the results of the radionuclide release analysis and the categorization of releases is presented in Section 5.4, the association of radionuclide releases with accident sequences is discussed in Section 5.5, and a discussion of various accidents not involving the core is summarized in Section 5.6.

5.2 CORE MELT EVALUATION

As discussed in Appendix V, the release of radionuclides from the fuel is predicted, consistent with the analysis in WASH-1400, to occur in several components, including:

- Gap release
- Melt release
- Steam explosion release
- Vaporization release

Because the melt release is by far the most significant component of those listed above, and because melting of the core is required prior to the occurrences of both the steam explosion (ex-vessel) and the vaporization releases, it was decided to carefully review the potential for melting of the core. The need to perform this careful evaluation was further highlighted by the fact that the Big Rock Point core is both significantly smaller in diameter and significantly shorter than the core in current BWRs (6 feet vs 12 feet).

The analysis performed to assess the potential for core melting at Big Rock Point, which is reported in more detail in Appendix IV, evaluated the following conditions:

(a) By one of a variety of means the core was assumed to be devoid of all water. RDS actuation for a variety of sequences could produce this state;

(b) The decay power was assumed to be 1% of full power (this condition will not exist until approximately six (6) hours after shutdown); and

(c) No active cooling in the form of core spray was available.

Under these conditions, the potential for decay heat removal was evaluated considering the following mechanisms:

(a) Decay heat was radiated to the wall of the reactor vessel; and

(b) Decay heat was removed from the core by natural convection using high-pressure steam as the fluid and the surface of the steam drum as the heat sink.

The analysis reported in Appendix IV concluded that neither of the above heat removal mechanisms was sufficient to prevent melting of the core. The analysis also concluded that, given the presence of water in the core, core melting could be prevented. This analysis was corroborated by the results obtained with the

BOIL code, in which the core was predicted to melt under the conditions defined above.

5.3 SELECTION OF ACCIDENT SEQUENCES FOR RELEASE ANALYSIS

5.3.1 CHALLENGES TO CONTAINMENT INTEGRITY

In defining the range of severity of radionuclide releases from the Big Rock Point Plant, the state of the containment during and after the occurrence of core damage is the dominant factor. For this reason, the process of selecting accident sequences for calculation of in-plant consequences leading to radionuclide releases to the environment began with the development of a logic model. This model depicted the processes contributing to the inability of the containment to retain radionuclides. This logic model is shown as Figure 5.1.

Figure 5.1 is a logic tree in which the top event is "Enclosure Fails To Contain Radionuclide Inventory." This model has been developed under the condition that an event has occurred in which radioactive material has been released from the fuel and the containment must prevent the release of this material to the environment. The remainder of this section will be devoted to summarizing the accident sequences for which radionuclide releases have been calculated (see Table 5.1) in the light of the requirements for analysis which are depicted in the logic tree in Figure 5.1. The format for this discussion will be to consider all of the notes shown in Figure 5.1. A more detailed description of the analysis summarized in this section is presented in Appendices IV and V. The accident sequences in Table 5.1 are described in Appendix V.

<u>NOTE 1</u>: One possible way in which containment can fail to completely contain its radionuclide inventory is by normal leakage at a rate of 0.5 percent volume per day. Three typical leakage sequences were analyzed to represent this inventory loss. These are Numbers 1, 3 and 6 in Table 5.1.

<u>NOTE 2</u>: Several possible containment system failures can be classified as failures to isolate. Among these, the most significant are: The failure of the vent valve to close, leakage through the vent valves or other leakage path and leakage from the primary system to regions outside the containment. An estimate of the probability of the failure of the vent valve to close (together with other holes of effective diameter greater than 1/2 inch) has been developed in Appendix IV as 0.10 per demand. Sequence Number 2 in Table 5.1 has been analyzed as a representative case of the releases from containment associated with a core melt sequence with failure of the vent valve to close. It should be noted that a modification of the vent valve currently being considered will lower the best estimate of the probability of failure to close to 0.025 per demand.

Figure 5.1. LOGIC TREE DEPICTING CONTAINMENT FAILURE PROCESSES

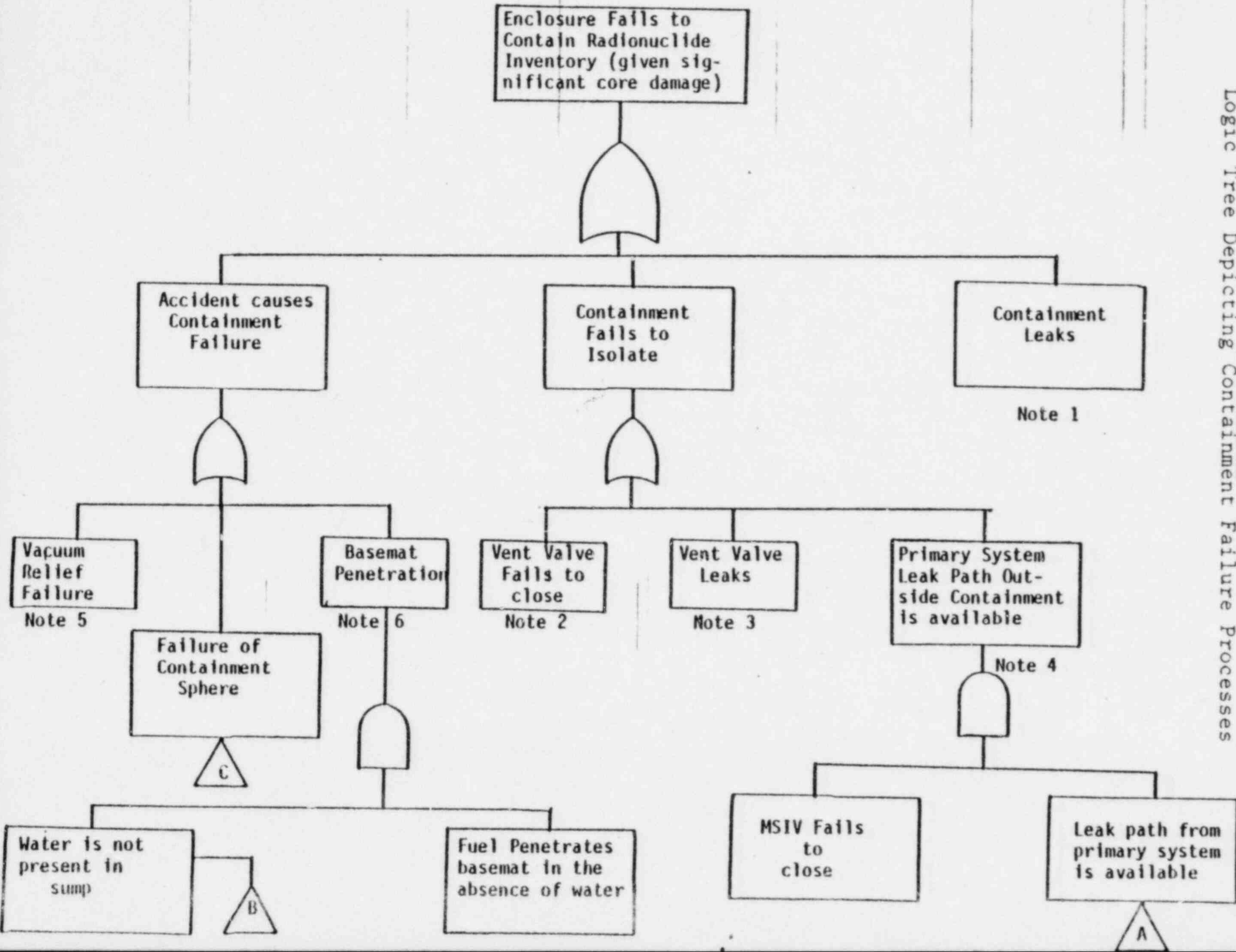

FIGURE 5.1
Logic Tree Depicting Containment Failure Processes

Figure 5.1. LOGIC TREE DEPICTING CONTAINMENT FAILURE PROCESSES, (Cont.)

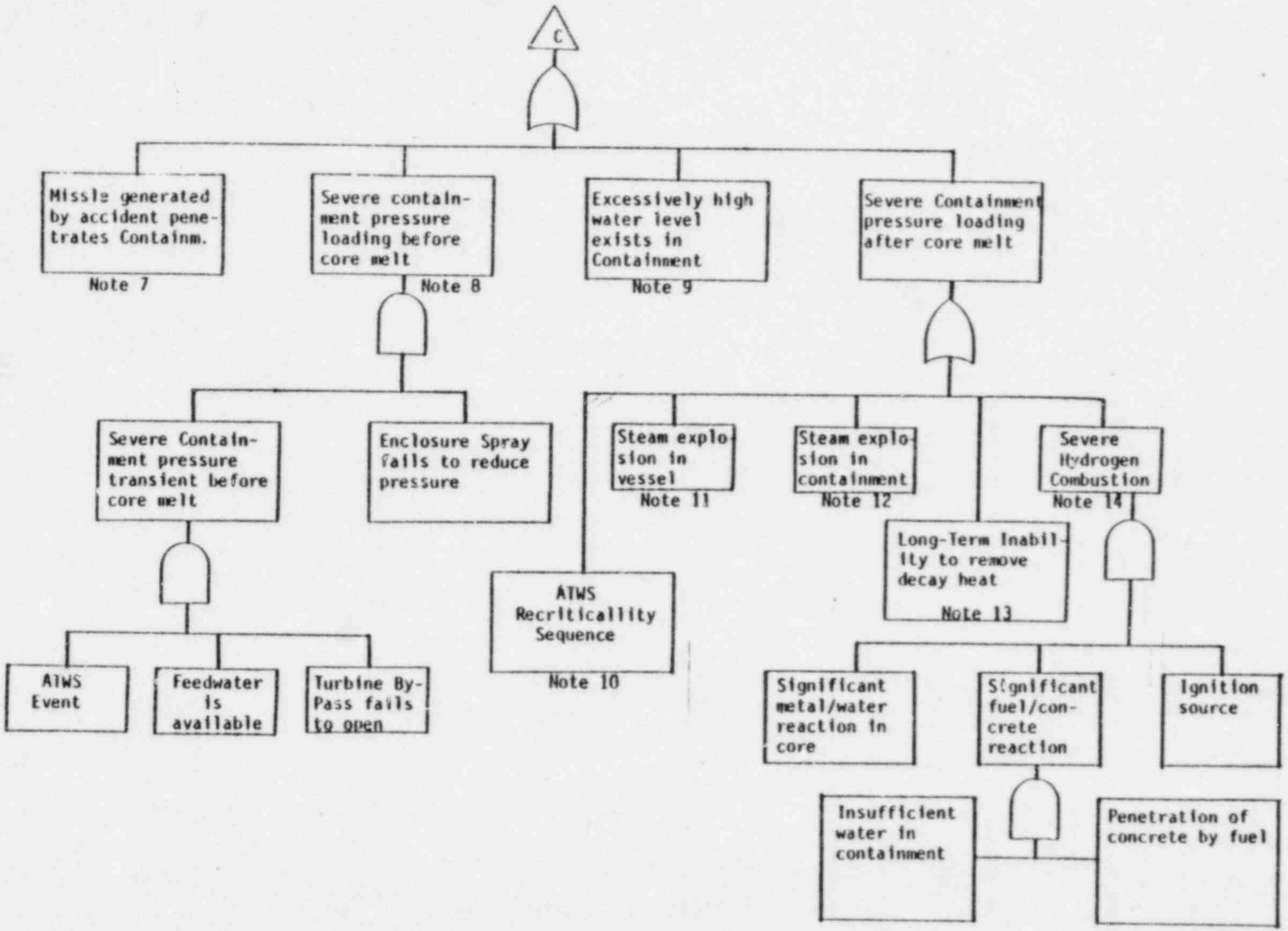

FIGURE 5.1

Logic Tree Depicting Containment Failure Processes ma0381-0902a-72-..

FIGURE 5.1
Logic Tree Depicting Containment Failure Processes

Figure 5.1. LOGIC TREE DEPICTING CONTAINMENT FAILURE PROCESSES (Cont)

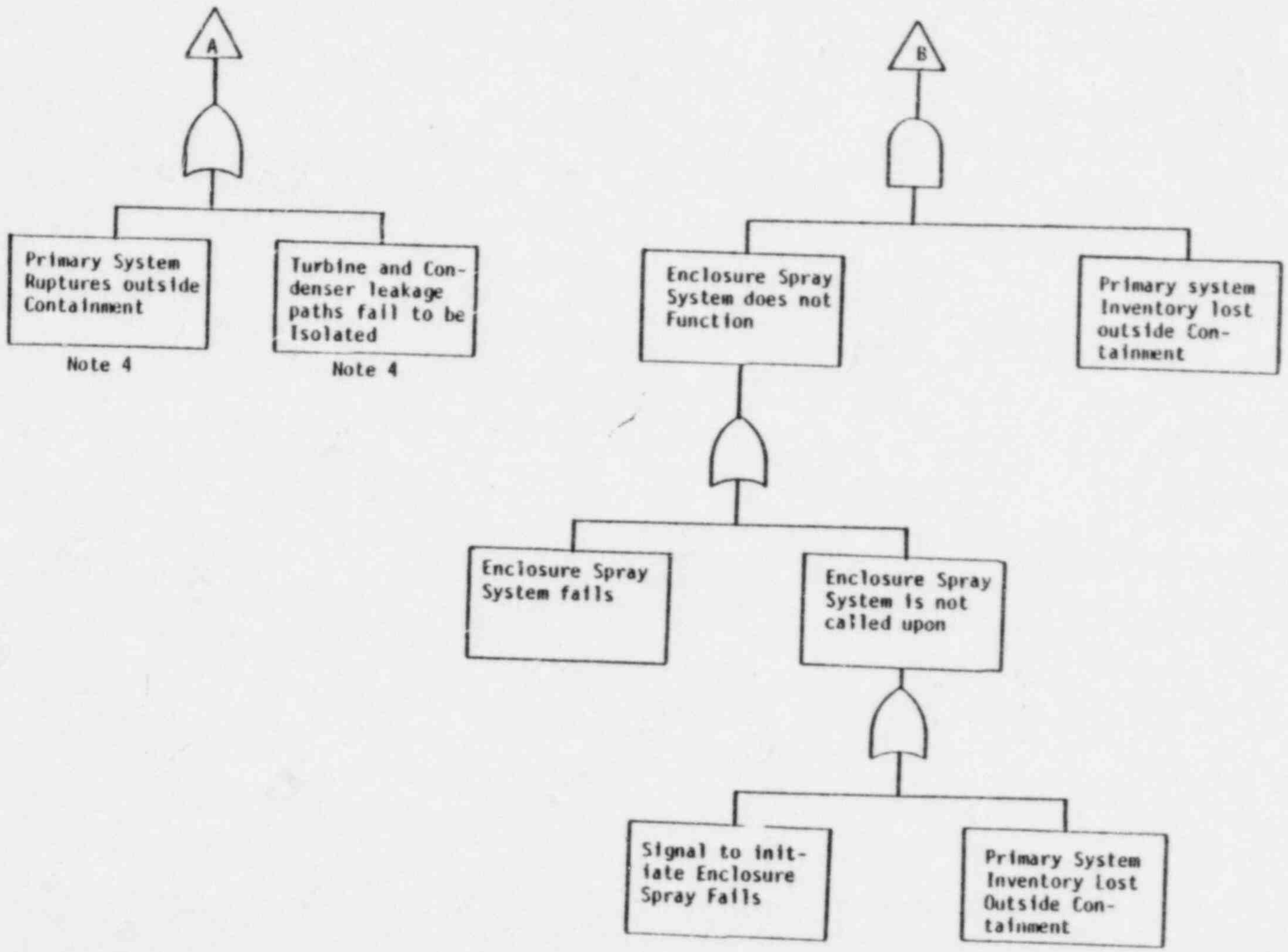

TABLE 5.1
Summary of Accident Sequences for Which Radionuclide Releases Have Been Analyzed for Big Rock Point

Sequence Description	Sequence Number	Important Failures	Containment State
Small Steam Line Break Inside Containment			
Sequence S_3E_mC-1	1	EM-Cond, Core Spray	Isolated
Sequence S_3E_mC-2	2	EM-Cond, Core Spray	Open (Early), No Enclosure Spray
Loss of Off-Site Power			
Sequence PEF_sC-1	3	EM-Cond, Main Cond, RDS	Isolated, No Enclosure Spray
Sequence PEF_sC-2	4	EM-Cond, Main Cond, RDS	Open (at Vessel Melt-Through), No Enclosure Spray
Large Steam Line Break Outside Containment			
Sequence S_8ZC	5	MSIV Closure, Core Spray	Open (Early)
Large LOCA			
Sequence S_7C	6	Core Spray	Isolated
Large Primary Leakage and Into Turbine Building			
Sequence PIF_sYC	7	Primary System Isolation, Main Condenser, Core Spray	Open to Turbine Building (Early)
Long-Term Containment Failure			
Sequence S_3E_mL-1	8	EM-Cond, Core Spray	Isolated Initially (Leakage)
Sequence S_3E_mL-2	9	EM-Cond, Core Spray	Ultimately Open (Late)

TABLE 5.1
Summary of Accident Sequences for Which Radionuclide Releases Have Been Analyzed for Big Rock Point

Sequence Description	Sequence Number	Important Failures	Containment State
Early Release Sequence	10	Nonmechanistic Sequence	Open (Early), Released Defined Nonmechanistically
Release of Noble Gases	11	Nonmechanistic Sequence	Leakage Paths to Turbine Building Available Via Tortuous Routes

NOTE 3: Another manner in which the containment can fail to isolate is for leakage paths to be available from inside to outside of the containment. Again, the containment isolation analysis presented in Appendix IV has indicated that this failure probability is approximately 0.13 per demand. Of this value, leakage paths which can allow communication between inside the containment and the outside atmosphere represent approximately 0.03 per demand.

Although the radionuclide releases associated with these leak paths would be expected to be somewhere between those discussed in NOTES 1 and 2, this evaluation has conservatively assumed that the leakage rate for these cases is characterized by the releases described in Sequence Number 2 in Table 5.1 (see NOTE 2).

NOTE 4: The final containment isolation failure mechanism considered is that leading to a leak path between the primary system and the region outside the containment. For any such paths to be available, the MSIV must remain open. In addition, either the primary system must rupture outside of containment, or significant leakage paths must exist associated with the turbine or the condenser (eg, steam seals, ruptured disks). Because of the possible differences in severity of releases between large primary system leak paths, which exist directly to the environment, and those which produce leakage flow into the turbine building, two cases were analyzed. These cases are represented by Sequence Numbers 5 and 7 in Table 5.1 for large primary leakage directly to the environment and to the turbine building, respectively.

NOTE 5: Another general category of causes for the enclosure to fail to contain its radionuclide inventory is the situation in which the accident produces the failure of the containment. To provide some insights into the various physical parameters which will influence the ability of the containment to resist accident-related failure, Table 5.2 is presented. This table compares key characteristics of the Big Rock Point containment with those of the Surry containment. As shown, the Big Rock Point power level is approximately 10% that of Surry, the containment volume is half as large, the design pressure is about two-thirds, and the primary inventory is about one-third that of Surry. These factors combine to indicate that the challenge to containment as a result of a blowdown of the primary system is similar for Surry and Big Rock Point. The challenge to containment resulting from physical processes relating to the power level (eg, long-term failure resulting from an inability to remove decay heat from debris) is less than 20% as significant for Big Rock Point as for Surry. Other results in Table 5.2 will be discussed in subsequent notes.

TABLE 5.2
Comparisons of Containment Physical Parameters

	Big Rock Point	Surry
Power Level (MW_t)	240	2,440
Containment Volume (Ft^3)	0.94×10^6	1.8×10^6
Containment Design Pressure (Psig)	27.0	45.0
Energy Releases in Large LOCA Information Required		
· Primary Volume (Ft^3)	3,639	8,387
· Primary Temperature (°F)	566	572
· Primary Pressure (Psia)	1,350	2,295
Pressure in Containment Following a Large LOCA Assuming No Containment Spray (Psig)	20	39.3
Volume of Steam Produced at Atmospheric Pressure by 1% Decay Power (Ft^3/Min) (@100°C)	0.382×10^4	3.88×10^4
Divided by Cont Volume	0.41×10^{-2}	2.2×10^{-2}
Mass of UO_2 (Lb)	27,500	175,600
Mass of Zirconium in Core (Lb)	~11,000	36,300
Percent of H_2 in Containment (Assuming All Zirconium Reacts)	7 to 8	12 to 14
Removal Rate Constant (Per Hour)		
· Iodine (Natural)	1.5	1.4
· Particulate (Natural)	0.9	0.6
· Iodine (Spray)	0.1	3.0*
· Particulate (Spray)	0.6	20.0

*This number applies for boric acid. The removal rate constant is a factor of 10 higher with hydroxide in the spray.

One of the possible causes of accident-related containment failure is underpressure failure following failure of the vacuum relief system. An accident sequence for which this containment failure mode would be relevant, would be characterized by a small steam line break in which the steam flow is insufficient to result in an early automatic vent valve closure. After some time, during which the steam flow would reduce the partial pressure of air in the containment, the vent valves would close either manually or automatically and, following a 15-minute delay (a recent design modification requires the enclosure spray to come on immediately), the enclosure spray would come on. Ultimately, the enclosure spray could condense sufficient steam to produce a demand on the vacuum relief system. Failure of this system to open the vent valves could then lead to an underpressure failure of containment.

A pessimistic evaluation of the radionuclide releases associated with a core damage event in which containment fails, as described above, has been performed. Sequence Number 2 in Table 5.1 is that evaluation. Although this evaluation has been performed, sequences similar to this are not expected to be significant contributors to risk. This is because the product of the probability of sequences involving small steam line breaks inside containment (1.4×10^{-5}) and the probability of failure of the vent valves to relieve vacuum (5.4×10^{-5} per demand) is very small (7.6×10^{-10} per year).

NOTE 6: An alternative way in which to fail containment as a result of an accident is for the fuel and core debris to penetrate the concrete base mat of the containment. Experimental evidence together with analysis presented in Appendix IV indicates that for any core debris penetration of the base mat to occur, there must be an almost total absence of water in the vicinity of the core debris. In addition, even in the absence of water, the debris will not necessarily penetrate completely through the base mat. Indeed, analysis performed for Big Rock Point and reported in Appendix V indicates limited debris penetration in the absence of water. Moreover, should this penetration occur, the public health consequences have been shown not to be of significance. For these reasons, base mat penetration has been judged to be both highly improbable and of little public health consequence. Therefore, radionuclide releases via base mat penetration have not been analyzed in this risk assessment.

NOTE 7: For missiles to be generated with sufficient energy to cause penetration of containment, a significant localized energy release would be required. Such an energy release could be associated with a steam explosion within the reactor vessel or within the containment. Appendix IV presents an evaluation of the potential for steam explosions and concludes that such events with sufficient energy to produce a missile are extremely unlikely. Although risk analyses have attempted to qualify the probability of these "extremely unlikely" events with estimates ranging from 10^{-2} to 10^{-3} given a core damage event, such a procedure for the Big Rock Point Plant would produce an insignificant perturbation to the estimated risk. For that reason, containment failure resulting from missile generation has been excluded from consideration. The health consequences associated with such events have, however, been estimated. They are the consequences associated with Sequence Numbers 2 and 4 in Table 5.1.

NOTE 8: Another accident-related cause of containment failure is a severe pressure loading produced in advance of the occurrence of significant core damage. Only a limited range of events involving Anticipated Transient Without Scram (ATWS) have the potential to significantly challenge containment integrity in advance of the occurrence of core damage. As discussed in Appendix VII, the plant as presently designed has an insufficient supply of feedwater for ATWS events to provide a severe pressure challenge to containment integrity in advance of the occurrence of core damage. However, some of the modifications being evaluated to reduce the probability of ATWS events at Big Rock Point will assure a longer term supply of feedwater during the event, and thereby increase the severity of the challenge to containment integrity. For this reason, radionuclide release analyses have been performed for sequences representative of this case. These are Sequence Numbers 2 and 4 in Table 5.1.

NOTE 9: In the early phases of a variety of accident sequences, those in which either lines break or safety relief valves lift, there is a need for enclosure spray to assure that the containment temperature does not exceed the value to which key equipment is qualified. Should the accident sequence eventually lead to activation of the reactor depressurization system and the need for core spray, then the principal means of removing decay heat in the long term is by the post-incident system operated in the recirculation mode. For this system to operate, the water level in containment must be above 587 feet and this system is supposed to be activated prior to the level reaching 596 feet. In theory, if the water level were allowed to exceed this value, the containment could fail. To assess the conservatism of this assumption, a containment margin analysis was performed in which the water level required to fail the containment was estimated. This analysis, presented in Appendix IV, indicated that overfill failure would not occur below an enclosure level of 634 feet. Other analysis has concluded that if the containment water level were increased to this height via core spray and enclosure spray, the water in the containment could accommodate all of the decay heat generated over a period of 30 days without exceeding 212°F even without any active heat removal capability. After 30 days, the natural heat removal processes through the enclosure sphere would allow removal of decay heat. For these reasons, containment failure due to high water level is considered to be quite unlikely. Nonetheless, the radionuclide releases associated with an event involving overfill failure of the enclosure can be conservatively estimated by referring to the analysis for Sequence 9 in Table 5.1.

NOTE 10: A variety of causes are possible for severe containment pressure loading subsequent to core melt. This and the next four notes discuss these causes. The discussion of ATWS sequences presented in Appendix I has shown that, for Big Rock Point as presently designed, a significant fraction of such sequences lead ultimately to activation of the RDS. Based on current understanding, it is not possible to demonstrate that following an RDS actuation, recriticality of the core can be prevented as the water level is restored in the vessel by the core spray system. Should this recriticality occur, the containment pressure would rise rapidly and, as analyzed under the assumption in Appendix VII, overpressure failure would result.

During the time when the reactor is critical, its power level has been estimated to be 20% of full power. At this power level with a significant fraction of the length of the core uncovered, the integrity of the fuel cannot be assured. For this sequence, the radionuclide releases from the containment were estimated to be the same as the releases from the fuel in the TMI-2 accident. These releases are reasonable because:

(a) The phenomena in the core region are similar to those which occurred when the core was partially uncovered at TMI;

(b) The radionuclide removal processes which would be in effect during this type of ATWS event at Big Rock Point are significantly less efficient than those which were available in the pressurizer at TMI-2;

(c) Since the containment is expected to fail by overpressure during the event, the radionuclide removal processes within the containment (which normally have time constants on the order of an hour - Table 5.2) would be expected to be less effective in this case.

Radionuclide releases estimated for this sequence have been analyzed to determine their health effects and are represented as Sequence 10 in Table 5.1. In practice, the mechanistically calculated releases associated with Sequences 2, 4 and 7 in Table 5.1 were employed in characterizing releases for ATWS sequences.

NOTE 11: A detailed discussion of in-vessel steam explosions is presented in Appendix IV. The conclusion of that analysis is that the physical conditions for an in-vessel steam explosion of sufficient energy to fail the containment do not exist in the accident sequences for BRP. For this reason, in-vessel steam explosions are not considered in this study as causal events for containment failure.

NOTE 12: Appendix IV presents a discussion of the potential for significant challenge to containment integrity resulting from a steam explosion within the containment. The conclusion of this analysis is that for those sequences in which water is accumulated below the reactor vessel, ex-vessel steam explosions involving a limited amount of core debris can occur. However, these events do not pose any threat to the containment integrity. For this reason, ex-vessel steam explosions are excluded from consideration in this analysis.

NOTE 13: Following an accident sequence resulting in serious core degradation, the potential exists for containment failure caused by inability to remove decay heat in the long term. An analysis performed to assess this potential is discussed in Appendices IV and V. The conclusions of this analysis include:

(a) Decay power level of 0.2% or less can easily be removed from the containment via radiation and convection. This power level will exist after about 32 days.

(b) If the accident sequence being analyzed involves actuation of the RDS, then a sufficient quantity of water would condense on the floor of the containment to allow accommodation of decay heat (via refluxing and heatup of structures) without reaching the containment design pressure for an extrapolated minimum of 10 days after the accident. Best estimate analysis indicates that the containment design pressure of 27 psig will not be reached at any time during this sequence. This conclusion is valid even in the absence of enclosure spray.

(c) If the enclosure spray functions, then sufficient water can be added to the containment to assure that the integral decay heat over a 30-day period can be accommodated. If the spray water is added until the level reaches the 634-foot elevation, then the temperature of the water in the enclosure would never reach 212°F. Despite the fact that the best estimate analysis indicates that long-term inability to remove decay heat via active means would not cause a failure of containment, an analysis was performed to characterize radionuclide releases both in the case of long-term containment integrity and in the case in which overpressure failure is assumed to occur at 10 days after the accident. The sequence numbers in Table 5.1 which are representative of these two cases are Sequences 8 and 9, respectively.

NOTE 14: Certain accident sequences develop in such a way as to produce hydrogen by reaction between the zirconium in the cladding and hot water or steam. An analysis presented in Appendix IV has shown that even if all the zirconium in the cladding were to react with steam, the hydrogen concentration would be below that at which complete combustion and an associated significant pressurization would occur. For this reason, an additional source of hydrogen is required to produce a sufficiently high concentration for combustion to occur and cause containment pressurization. This source could only come from the interaction between fuel debris and concrete. For this interaction to occur, no water can be present in the region where the debris is accumulating. The only sequences for which this situation could exist are sequences in which the primary system water inventory is lost outside containment either as a result of failure to close the enclosure vent valves or as a result of the failure of the MSIV to close. In both of these types of sequences (see, for example, Sequences 2, 5 and 7 in Table 5.1), the enclosure has failed to isolate well in advance of the accumulation of hydrogen. For this reason, the hydrogen accumulation and eventual rapid combustion would have essentially no effect on radionuclide releases for sequences in which these phenomena are possible.

5.3.2 CONTAINMENT INTEGRITY

In the preceding section, reference was made to evaluations of the integrity of the containment in various accident sequences and under various mechanical loadings. The detailed work on this subject is reported in Appendix IV, and a brief summary is presented here.

5.3.2.1 CONTAINMENT ISOLATION

Eliminating leak paths from the containment can be achieved by closing off the containment (closing ventilation valves as well as other possible leak paths) and by isolating the primary system. Isolation of the primary system can be achieved either by closing the MSIV (which is effective even for line breaks outside containment) or by closing a variety of exhaust paths to the outside (including air ejectors and steam seal regulators).

Results of the analysis of the probability of failure to isolate are shown in Table 5.3 for the system as designed in May 1980. The relevant information from that table are the probabilities for all size leak paths through the vent valves and the leaks through the steam line or feedwater piping. Other leak paths are not important either because of the low probability of failure to isolate or because the leak path leads into a confined region outside the containment boundary (such as the radwaste tanks).

For this analysis, the important numbers in Table 5.3 are:

(a) The probability that the vent valves will leak or fail to close (0.13 per demand)

(b) The probability that there will be leak paths through the steam line or the feedwater line (0.066 per demand) ma0381-0902a-72-52

TABLE 5.3
Probabilities of Failure To Isolate Containment

Table 5.3
PROBABILITIES OF FAILURE TO ISOLATE CONTAINMENT

System	Small Leak	½" Hole	2" Hole	12" Hole	24" Hole	Total
1. Locks	$4x10^{-4}$	$1x10^{-4}$	-	-	$1x10^{-6}$	$5.0x10^{-4}$
2. Vents	$3x10^{-2}$	-	-	-	$1x10^{-1}$	$1.3x10^{-1}$
3. Steam Line/ Feedwater	$6x10^{-2}$	$3x10^{-3}$	$2x10^{-3}$	$1x10^{-3}$	-	$6.6x10^{-2}$
4. Sumps	$3x10^{-2}$	-	$3x10^{-3}$	-	-	$3.3x10^{-2}$
5. Demin/ Waste	$1x10^{-3}$	-	$4x10^{-5}$	-	-	$1.0x10^{-3}$
6. Air Supply	$2x10^{-5}$	-	$2x10^{-7}$	-	-	$2.0x10^{-5}$
7. Fuel Pit	$2x10^{-2}$	-	$2x10^{-3}$	-	-	$2.2x10^{-2}$
8. Resin Sluice	$2x10^{-5}$	-	$1x10^{-7}$	-	-	$2.0x10^{-5}$
9. Control Rod Drive	$1x10^{-7}$	$>10^{-7}$	-	-	-	$1.0x10^{-7}$
TOTALS	$1.4x10^{-1}$	$3.1x10^{-3}$	$7.0x10^{-3}$	$1.0x10^{-3}$	$1 x10^{-1}$	$2.5x10^{-1}$

Subsequent to May 1980, the design of the system for closing the vent valves has been improved. Although the new design has not been implemented, the effect of this improvement has been determined to reduce the probability of failure of the vent valves to close from 0.1 per demand to 0.025 per demand. This would lower the total probability of failure to close or leak from 0.13 per demand to 0.055 per demand. The probability of leakage through the feedwater line or the steam line is unaffected by this modification.

It should be noted here that the release associated with the two categories of isolation failure (vent valves fail open and primary system leakage to the turbine building) are expected to be significantly different. For the first category, the releases are expected to be characterized by Sequence 2 in Table 5.1 while the second category is characterized by leakages for Sequence 11. For conservatism, releases from the second category have been assumed to be equal in severity to releases associated with open vent valves.

Primary system isolation failure is dominated by the probability that the MSIV will fail to close. This probability has been estimated to be 0.038 per demand. The releases associated with this containment isolation failure mode have been characterized by Sequences 5 and 7 in Table 5.1.

5.3.2.2 OVERPRESSURE FAILURE

As discussed earlier, the primary energy sources which can contribute to pressurization of an isolated containment include:

- Stored energy in the primary system and fuel
- Decay heating
- Zircaloy oxidation
- Hydrogen combustion
- Nuclear power prior to shutdown (ATWS events)
- Fuel debris interaction with the concrete

Analysis has been performed to characterize the containment pressure response to these energy sources. Assuming that the enclosure spray system is functional, the design pressure of 27 psig is not predicted to be exceeded for a variety of accident conditions. Only ATWS events have significant potential to challenge containment integrity as discussed in Appendix VII. For the risk assessment, however, the pressure to which containment integrity can be assured has been calculated to be 72 psid based on the ASME III, Appendix F faulted allowable stress of 0.7 times ultimate strength. This analysis is reported in more detail in Appendix IV.

5.3.2.3 VACUUM RELIEF FAILURE

For some sequences in which the containment atmosphere is significantly diluted by steam prior to enclosure isolation, it may be necessary to provide vacuum relief when the enclosure spray is activated. The allowable level of vacuum in the enclosure was calculated in the design analysis to be 0.94 psid (assuming a snow load of 0.28 psi). It should be noted that the very low probability of failure of the vacuum relief system, estimated to be 5.4×10^{-5} per demand, makes this failure mode not important from a risk perspective.

5.3.2.4 HIGH CONTAINMENT WATER LEVEL

For sequences in which cooling of the core is provided by core spray and the containment is cooled by the enclosure spray following RDS actuation, it is ultimately necessary to switch to the recirculation mode of core cooling. Should this switch-over not occur, the enclosure spray may continue to add water to the containment until the sphere fails. Based on the same allowable stress as the overpressure failure criterion, it has been determined that failure of the sphere by overfilling will not occur below a level of 634' 6". At an enclosure spray flow rate of 400 gpm, filling the enclosure to this level (which is several feel above the centerplane of the sphere) will require approximately 8 days. Thus the overfill failure mode seems quite unlikely. This analysis is reported in Appendix IV.

5.3.3 SUMMARY OF FAILURE MODES

From the analysis in the previous sections, it can be concluded that the following containment failure modes are expected to be relevant in the risk assessment of the Big Rock Point Plant:

(a) Normal Leakage - This would occur in any accident sequence in which the containment isolates and some significant pressure builds up within the containment.

(b) Containment Isolation Failure - Because of the relatively high probabilities of vent valve leakage or failure to close, and the relatively high probability of failure of the MSIV to close, the failure of containment isolation is expected to be important in the evaluation of the risk from Big Rock Point.

(c) Overpressure Failure - Although a limited number of pressure challenges to containment integrity are expected to occur, the potential severity of such sequences as ATWS events characterized by high primary system pressure may make containment overpressure failure a significant sequence.

5.4 QUANTIFICATION AND CATEGORIZATION OF RADIONUCLIDE RELEASES

For all of the sequences noted in Table 5.1, radionuclide releases from the containment to the environment have been estimated. The methodology for this release analysis has been discussed in Section 3.4 and the details are presented in Appendix V. The purpose of this section is to present the releases and to develop categories which are representative of the releases associated with the accident sequences presented in Chapter 4.0.

Table 5.4 is a summary of the calculated radionuclide releases for the eleven sequences of interest. Careful review of the releases in this table indicates that several distinct groups or categories exist. These categories, together with the largest radionuclide release fraction in each chemical group, are reported in Table 5.5. As shown, five release categories have been defined in that table ranging in severity from BRP-1, in which the primary system is open to the region outside the containment during the core meltdown process, to BPR-5 in which containment integrity is maintained during and after the core melt and the radionuclide releases occur via leakage through very small leak paths in the containment.

Table 5.5 also lists the release fractions represented in WASH-1400 for the PWR-2 release category. These release fractions are for the PWR V-Sequence which is similar in effect to the Big Rock Point sequence involving a large steam line break outside containment with failure to isolate the primary system (BRP-1). A comparison between the release fractions for BRP-1 and PWR-2 reveals the expected similarities between the two categories.

A brief description of the primary characteristics of the five release categories is presented below:

BRP-1 This release category is characterized by rapid depletion of coolant inventory in the primary system through a break in the main steam line located in the pipe tunnel. Following the rapid blowdown of the system, the primary system fails to isolate via closure of the MSIV and consequently the primary system is open to the atmosphere during the time when core melt produces significant releases of radionuclides from the fuel. In this sequence, the containment is bypassed during the time when the largest radionuclide releases would occur.

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TABLE 5.4
Summary of Important Radionuclide Release Parameters
for Big Rock Point

Table 5.4

SUMMARY OF IMPORTANT RADIONUCLIDE RELEASE PARAMETERS FOR BIG ROCK POINT

Sequence Number	Sequence Description	Initial Release Time (HR)	Release Duration (HR)	Xe-Kr	$I_{org.}$	I_2-Br	Cs-Rs	Te	Ba-Sr	Ru	La
1.	S_3E_mC-1	1	4	8.2×10^{-5}	6.0×10^{-7}	7.4×10^{-5}	7.8×10^{-6}	5.0×10^{-6}	7.9×10^{-7}	4.5×10^{-7}	6.8×10^{-8}
2.	S_3E_mC-2	4	8	6.1×10^{-1}	4.3×10^{-3}	1.9×10^{-2}	9.0×10^{-2}	1.4×10^{-1}	9.1×10^{-3}	9.8×10^{-3}	1.7×10^{-3}
3.	PEF_sC-1	6	14	1.8×10^{-3}	1.2×10^{-5}	5.3×10^{-5}	4.5×10^{-4}	1.6×10^{-4}	5.7×10^{-5}	1.5×10^{-4}	1.7×10^{-6}
4.	PEF_sC-2	6	14	8.9×10^{-1}	6.0×10^{-3}	4.3×10^{-2}	3.0×10^{-1}	1.2×10^{-1}	3.9×10^{-2}	1.2×10^{-1}	1.2×10^{-3}
5.	S_8ZC	1	19	9.0×10^{-1}	7.0×10^{-3}	9.0×10^{-1}	8.1×10^{-1}	1.5×10^{-1}	1.0×10^{-1}	3.0×10^{-2}	3.0×10^{-3}
6.	S_7C	3	17	1.1×10^{-3}	7.6×10^{-6}	4.1×10^{-5}	1.3×10^{-4}	1.3×10^{-4}	1.4×10^{-5}	1.0×10^{-5}	1.7×10^{-6}
7.	Primary System Isolation Failure to turbine BLDC Sequence PIFsYC	1	19	8.8×10^{-1}	6.9×10^{-3}	8.3×10^{-2}	1.6×10^{-1}	2.6×10^{-2}	1.7×10^{-2}	5.1×10^{-3}	5.1×10^{-4}
8.	S_3EmL-1	12	228	3.9×10^{-2}	2.7×10^{-4}	4.0×10^{-4}	2.4×10^{-4}	2.6×10^{-4}	2.6×10^{-5}	2.1×10^{-5}	3.4×10^{-6}
9.	S_3EmL-2	240	0	1.0	7.0×10^{-3}	8.7×10^{-3}	2.4×10^{-4}	2.7×10^{-4}	2.6×10^{-5}	2.1×10^{-5}	3.4×10^{-6}
10.	BRP-2 Complete Melt (early containment failure)	0.33	0.67	0.7	5×10^{-3}	6.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	1.5×10^{-2}	2.0×10^{-2}	2.0×10^{-3}
11.	Noble Gas Release			0.9	0	0	0	0	0	0	0

TABLE 5.5
Summary of Release Categories for Big Rock Point

Release Category Designation	Table 5.4 Sequence Numbers Included	Fraction of Core Inventory Released							
		X_e-K_r	I org.	I_2-B_r	C_s-R_B	T_e	B_a-S_r	R_u	L_a
BRP-1	5	9.0×10^{-1}	7.0×10^{-3}	9.0×10^{-1}	8.1×10^{-1}	1.5×10^{-1}	1.0×10^{-1}	3.0×10^{-2}	3.0×10^{-3}
BRP-2	10	7.0×10^{-1}	5.0×10^{-3}	6.0×10^{-1}	6.0×10^{-1}	1.0×10^{-1}	1.5×10^{-2}	2.0×10^{-2}	2.0×10^{-3}
BRP-3	2,4,7	8.9×10^{-1}	6.9×10^{-3}	8.3×10^{-2}	3.0×10^{-1}	1.4×10^{-1}	3.9×10^{-2}	1.2×10^{-1}	1.7×10^{-3}
BRP-4	9	1.0	7.0×10^{-3}	8.7×10^{-3}	2.4×10^{-4}	2.7×10^{-4}	2.6×10^{-5}	2.1×10^{-5}	3.4×10^{-6}
BRP-5	1,3,6,8,11	0.9	2.7×10^{-4}	4.0×10^{-4}	4.5×10^{-4}	2.6×10^{-4}	5.7×10^{-5}	1.5×10^{-4}	3.4×10^{-6}
PWR-2		9.0×10^{-1}	7.0×10^{-3}	7.0×10^{-1}	5.0×10^{-1}	3.0×10^{-1}	6.0×10^{-2}	2.0×10^{-2}	4.0×10^{-3}

BRP-2 This release category has been developed based on a subjective evaluation of the expected releases from the containment for an event similar to TMI-2 in the severity of fuel damage. In this event, however, the containment was assumed to be ineffective in both retaining and depleting the radionuclide source term released from the fuel. The radionuclide releases from containment which have been assumed for this case, are those estimated releases from the fuel for the TMI-2 accident. No radionuclide depletion was assumed because the removal processes were assumed to be ineffective in reducing the concentrations in the primary system or containment environment. No sequences were identified for which BRP-2 releases could be appropriate.

BRP-3 Three accident sequences have been analyzed in the development of this release category. Two principal types of sequences can be identified from these three sequences.

(a) Sequences involving early failure of the ability to maintain primary coolant inventory leading to the requirement to actuate RDS. Failures in either RDS or in core spray would then lead to melting of the core, and failure to close the enclosure vent valves would lead to significant releases to the environment. In the sequences of this type which were analyzed, the enclosure spray was assumed not to function. The primary effect of this failure is to cause the radionuclide sweep-out rate from the containment to be slightly higher than it would be with enclosure spray functioning. The depletion of radionuclides from the containment atmosphere resulting from the availability of enclosure spray would be expected to have a secondary effort.

(b) Sequences leading to core melt in a way similar to (a) above with a path available from the primary system to the turbine building through the main steam line and through the turbine or condenser. This sequence would result if a large path existed from the primary system into the turbine building rather than a very small leak path. The pressure retention capabilities of the turbine building have been ignored in the evaluation of releases for this type of sequence.

BRP-4 As discussed earlier in this chapter, the current best analysis of the containment pressure history in the long-term following a core melt event in which the containment isolates indicates that the design pressure of 27 psig would never be reached. This conclusion is valid even if the enclosure spray is assumed to be unavailable throughout the event. The analysis which has determined the releases in BRP-4 category has assumed that the containment integrity is maintained during the first ten days and that a rapid depressurization caused by a large enclosure leak at ten days releases all of the inventory remaining in the containment atmosphere at that time. This release category is being used to conservatively represent the radionuclide releases which would occur if seal degradation on the time frame of days caused containment leakage.

BRP-5 The final release category included in this analysis has been defined based on the predicted radionuclide releases developed for a number of sequences in which the core damage occurs in a containment which has been successfully isolated and which does not fail as a result of overpressurization or seal degradation. The sequences analyzed to determine the releases for this category include both sequences in which the enclosure spray functions and in which it fails. One of the sequences considered in this release category is a sequence in which the primary system fails to isolate via a small leak into the turbine building (Table 5.5, Sequence Number 11). Because this evaluation was intended to characterize sequences in which a very small leak existed and only a small pressure driving force for leakage existed, the associated releases have been judged to be insignificant in all chemical groups except the noble gases. Ninety percent of the noble gases were assumed to be released. The sequence categorization used in developing releases for BRP-5 has resulted in the addition of some release fraction for each of the remaining seven chemical groups.

5.5 ASSOCIATION OF RELEASE CATEGORIES WITH ACCIDENT SEQUENCES

In the evaluation of risk from Big Rock Point, the following elements have been required:

(a) Development and compilation of important accident sequences and their probabilities of occurrence;

(b) An assessment of the severity of radionuclide releases associated with the spectrum of accidents and the various possible containment failure modes;

(c) A categorization of predicted radionuclide releases by severity of release;

(d) Association of each potentially important accident sequence with the various release categories which it could lead to, and an assignment of the probability that each sequence will lead to each relevant release category;

(e) Calculation of the total probability of each release category;

(f) Calculation of the spectrum of health consequences which could result from each release category given variations in possible weather conditions at the time of the accident;

(g) Development of the composite risk curve based on the information developed in Elements (a), (e) and (f) above.

Item (a) has been discussed in Chapter 4.0, Items (b) and (c) have been developed earlier in this chapter, Items (f) and (g) are discussed in Chapter 6.0, and this section will deal with Items (d) and (e). The completion of these items leads to an estimate of the probability of occurrence of each of the release categories.

Table V.5-4 in Appendix V presents a compilation of the important accident sequences together with estimates of the probability that each sequence will lead to each relevant release category. The notes to Table V.5-4 describe the basis of the assignment of the probability that each sequence will lead to each relevant release category.

Some observations extracted from Table V.5-4 (and Table 6.4 of Chapter 6) will be discussed at this time. The first of these observations is that the total core damage probability has been split among the five release categories in such a way that the probabilities of BRP-3 and BRP-5 comprise nearly all of the probability of core damage. Because the radionuclide release fractions are of the same order for Categories BRP-1, 2 and 3, and because these fractions are significantly greater for BRP-3 than for BRP-5, the risk from the Big Rock Point Plant is expected to be dominated by accidents leading to Release Category BRP-3.

The second observation arises from a separation of the sequences leading to releases in BRP-3 into classes which are associated with their relevant containment failure mode. If sequences which contribute more than 99% of the probability of Release Category BRP-3 are considered, the results in Table 5.6 are produced. Note that fire sequences which lead to failure of containment integrity contribute over 52% to the probability of BRP-3. The next most significant contributors are sequences which terminate in the failure of the vent valves to close (19.5%) and sequences leading to leakage through the steam and feedwater lines, past the backup isolation valves, and into the turbine building (12.8%). ATWS overpressurization sequences are the next class of sequences of importance, contributing 7.3% to the probability of BRP-3.

TABLE 5.6
Containment Failure Modes Contributing to
Release Category BRP-3

Failure Mode	Failure Probability (Per Demand)	Contributing Sequence Probability Per Year (10^{-4})	Percentage Contribution To BRP-3 (Percent)
Vent Valve Fails To Close	0.1	7.13	19.5
Vent Valve Leaks	0.03	7.13	5.8
Leaks in Steam and Feedwater Lines and Backup Isolation Valves Fail To Seal	(0.066) x (1.0)	7.13	12.8
MSIV Fails to Close and Backup Isolation Valves Fail To Seal	(0.038) x (1.0)	2.25	2.3
ATWS Over Pressurization	1.0	0.266	7.3
Fire Sequence Failures	Various	2.33	52.3

The results in Table 5.6 can be employed to assess the effect of improvements in vent valve closure reliability, leakage susceptibility of the backup isolation valves, and reductions in the probability of fire-related sequences. As an example, a reduction in the probability of vent valve failure to close from 0.1 per demand 0.025 per demand will reduce the probability of the BRP-3 release category by about 15% [$\cong(\frac{0.075}{0.1})$ x 19.5%].

The information contained in Table V.5-4 in Appendix V will be employed in the final quantification of risk to be developed in Chapter 6.0 and discussed in Chapters 7.0 and 8.0.

5.6 ACCIDENTS NOT INVOLVING THE CORE

Three accidents not involving the entire reactor core have been considered in the Big Rock Point PRA study: a possible refueling accident, the potential loss of the water in the spent fuel storage pool and a dropped fuel transfer cask. The refueling accident has considered the possible damage to one fuel assembly. This may occur as the result of an error or equipment malfunction occurring as a fuel assembly is being trasferred from the reactor to the spent fuel pool. The scenario for the spent fuel pool accident assumes that the water in the pool is lost by evaporation (ie, due to decay heat generation from the spent fuel rods) so that the fuel assemblies become uncovered, overheat and fail. In the event of a dropped fuel transfer cask, several fuel assemblies may be damaged. A more detailed description of these events and the results of the risk assessment performed is presented in Appendix VIII. This section summarizes the results of the analysis presented there.

5.6.1 REFUELING ACCIDENT

During refueling, accidents can occur which result in either mechanical damage to the fuel assembly or in the inhibiting of heat transfer from the spent fuel being handled. In most BWR, the transfer is made without a cask. A gate isolating the reactor well (pool) above the reactor vessel from a separate storage pool is lifted and the fuel is removed from the reactor. It is then transferred (without a cask) underwater to the storage area. From the storage area, the fuel is transferred to the spent fuel pit via a transfer tube. There is one significant procedural difference between the refueling process used at BRP and that used in most BWR. This is the use of a 24-ton cask to transfer the fuel assemblies from the core to the spent fuel pit.

An evaluation of the consequences of a refueling accident has concluded that the resulting "equivalent fraction of core inventory released" ranges from 1.4×10^{-4} (Xe) to 7.9×10^{-10} (Se) with the values for cesium and strontium being 7.5×10^{-5} and 4.4×10^{-8}, respectively.

The impact of these releases, when compared with the consequence of core melt sequences, is insignificant. The result of such an accident would be an unhabitable containment. However, the probability of this occurring (ie, 1.5×10^{-3} per year) is much less than that resulting from RDS actuation for a variety of causes ($\sim 1 \times 10^{-2}$ per year) where the consequences are also much greater.

5.6.2 SPENT FUEL POOL ACCIDENT

The spent fuel pool at Big Rock Point is located within containment. Normally, this tank is completely filled with water and would not be expected to present a problem since the fuel assemblies would be adequately cooled without forced circulation. There is, however, one scenario in which the spent fuel pool could conceivably become a factor in the PRA study. The sequence of postulated events which comprise this scenario is given below relative to the Big Rock Point Plant.

1. A LOCA or another type of event occurs which contaminates the area (ie, the containment building) preventing access to that area.

2. As a direct result of the accident or because of some other effects, equipment which is used to provide cooling capability to the spent fuel pool malfunctions. For example, a LOCA or high water in the containment could fail the two Fuel Pit Pumps.

3. Once the normal cooling circuit is lost, the spent fuel pool begins to heat up; the water boils, and the fuel assemblies eventually become uncovered.

4. Once the water level drops below the top of the active fuel assembly, the fuel rods will become overheated, helped to some extent by the exothermic steam/Zircaloy oxidation process.

5. The cladding will eventually reach the melting point, the cladding will be breached, and radioactive products will be released to the containment.

An evaluation of the release fractions of various radionuclide groups is presented in Appendix VIII. The values relative to core values range from 29% for Group VII (La) to less than 1% for Groups I, II and IV (Xe, I and Te). Cesium (Group III) and strontium (Group V) have values of about 9% and 8%, respectively.

It is estimated that the consequences associated with such releases would average about 10% of those associated with a core melt. Moreover, the overall risk from a spent fuel pool accident is estimated to be less than 1% of that associated with the core melt sequences considering the reduced probability (factor of 10) and reduced consequences (factor of 10). It is therefore concluded that the potential for a spent fuel pool accident will have a negligible impact on the overall risk at the Big Rock Point Plant.

5.6.3 DROPPED FUEL TRANSFER CASK ACCIDENT

A third potential noncore related accident is one involving a dropped fuel transfer cask. The refueling cask could potentially be dropped over the core, over the spent fuel pool, or in the time period when the cask is being moved between the core and spent fuel pool. During the transfer between the core and the spent fuel pool, the cask is moved horizontally along the refueling floor with only a few inches of clearance. If the cask were dropped during this period, little or no damage would be expected because of the short distance that the cask could fall. Thus, the two potential areas of concern are dropping the cask over the core or while lowering or lifting it over the spent fuel pool.

If the cask were dropped over the spent fuel pool, there is the possibility that it will directly strike the bottom of the pool, strike one of the spent fuel racks, or impact on one or more of the fuel assemblies in the pool. Analyses have indicated that if a cask of this type would strike the bottom of the pool directly, the liner integrity might be lost, but loss of coolant from the pool would be limited to possible seepage through the concrete and would be well within the capability of the makeup system.

If the cask were to strike the racks, structural damage of the racks would occur. However, it is expected that the maximum possible consequence of a dropped cask accident would occur if the cask would strike a number of fuel assemblies with sufficient force to breach the cladding, thereby releasing radioactive material. Since a similar occurrence can be postulated to occur over the reactor core and since the level of activity would be much less in the pool than in the recently shutdown core, consideration of a dropped cask into the core will conservatively cover the case for the pool.

The most probable cause of a dropped cask would either be failure of the cables or failure of the crane. Because the safety sling would prevent cask drop if the cable failed, the probability of crane failure would be much higher than the combined failure of the cable and the safety sling. The probability of crane failure has been estimated to be 10^{-4} per year. This value is 15 times smaller than the 1.5×10^{-3} per year value reported in Section 5.6.1 as bounding the probability of a single assembly refueling accident. Thus, even if a large number of fuel assemblies were damaged by dropping the cask over the reactor or over the pool, the consequences would not be any worse than that described in Section 5.6.1 and are therefore considered to be insignificant when compared to the consequences from core melt sequences or RDS actuation.

Appendix IV

CORE DEBRIS TRANSPORT AND CONTAINMENT FAILURE MODE ANALYSIS

IV.1 Introduction

In evaluating the risk from the Big Rock Point plant, it has been necessary to assess, employing best estimate methods, a variety of phenomena relating to the core damage process and the integrity of the containment boundary. The purpose of this appendix is to assemble these analyses into one source for future reference. In the introductory section, a brief summary of the conclusions of the various analyses will be presented. Additional analysis of the radionuclide releases associated with a spectrum of accident sequences is reported in Appendix V.

Analyses on several issues are contained in this appendix. The introductory and report sections in which these analyses are presented are tabulated below:

Issue	Introductory Section	Appendix Section
Potential for Core Melt	IV.1.1	IV.2 and IV.4
Containment Isolation Failure	IV.1.2	IV.3
Challenges to Containment Integrity	IV.1.3	IV.4
Containment Natural Heat Rejection Ability	IV.1.4	IV.4 and IV.5
Containment Strength Evaluation	IV.1.5	IV.6

IV.1.1 Potential for Core Melt

Because of the small size of the Big Rock Point core, it was necessary to evaluate the potential for heat removal from the core in the absence of water. The incentive for this analysis was to determine whether natural heat transfer processes could be effective in preventing core melt. Three analyses were performed:

a. Natural circulation analyses employing pressurized steam as the transfer medium and the steam drum as the heat sink. Despite a variety of highly optimistic assumptions regarding this heat transfer (discussed further in Appendix IV.2). The conclusions reached in this study were:

   1. The calculations which have been made tend to support the assumption that melting will occur.

   2. A more detailed calculation may be required. In particular, pressure drop loss coefficients for the pump, reactor internals, etc., will need to be considered if these additional calculations are to be meaningful.

b. Primary system heat sinks were evaluated to assess their potential for removing heat from the core. This analysis (reported in Appendix IV.4) concluded that insufficient heat sinks are available even assuming a decay power level of 1% (appropriate after 6 hours).

c. Thermal radiation to the vessel wall was assessed as a mechanism for removing decay heat from the core. Again, even considering decay power levels as low as 1%, it was concluded (in Appendix IV.4) that radiative heat transfer was insufficient to prevent core melt.

IV.1.2 Containment Isolation Failure

An evaluation of the potential for failure of the containment to isolate was performed and is reported in Appendix IV.3. The conclusions of this analysis include:

a. The probability of failure of the relief valves to close plus the probability that they will leak is 1.3×10^{-1}/demand for the design as it existed in May of 1980.

b. Future modifications of the relief valve closure system will lower the combined probability of failure to close and significant leakage to 5.5×10^{-2}/demand.

c. The only other significant leakage path to the region outside containment is through the steam line and feedwater line into the condenser. This leak path has a probability existing of 6.6×10^{-2}/demand.

IV.1.3 Challenges to Containment Integrity

Various possible challenges to containment integrity resulting from the accident have been considered and are discussed in Appendix IV.4. The principal challenges considered together with the conclusions of the evaluation are listed below. The challenges associated with ATWS sequences are treated separately in Appendix VII.

a. Hydrogen Combustion: Given the large containment volume and comparatively small core, hydrogen combustion does not provide a threat to the containment integrity. This conclusion is discussed in IV.4.1 and IV.4.2.

b. In-Vessel and Ex-Vessel Steam Explosions: Neither in-vessel nor ex-vessel steam explosions provide a threat to the containment integrity. The major effect of an ex-vessel steam explosion is dispersal of the core material which enhances core coolability. This conclusion is developed in Appendix IV.4.1.

c. Decay Heat Addition Leading to Containment Overpressure: If the containment sphere is assumed to be the ultimate heat sink, the energy extraction rate through the insulation is less than the required rate at 1% decay power, but sufficient at 0.2% decay power. However, without the insulation, the steel sphere would provide the necessary heat removal capacity to prevent overpressurization by steam. This conclusion is developed in Appendix IV.4.1. In addition to the steady-state analysis in Appendix IV.4.1, an assessment was made of the transient pressure history in containment resulting from a core melt sequence with limited water in the enclosure and decay heating. This evaluation, reported in Appendices IV.4.3 and V concludes that containment design pressure is never reached.

d. Overfill With Water From the Enclosure Spray or Other Sources: Although it is possible to cause containment failure as a result of excessively high water level, the level required is several feet over the mid-plane of the sphere (see Appendix IV.6). Decay heat removal with this volume of water in the containment is discussed in Appendix IV.5.

IV.1.4 Containment Natural Heat Rejection Ability

Analyses reported in Appendices IV.4.1 and IV.4.3 have evaluated the natural heat removal capability of the enclosure under the condition that the enclosure spray system does not function (see summary in IV.1.3c above). If the enclosure spray functions, but it is not possible to operate the post incident system in the recirculation mode, then the enclosure can be filled to the mid-plane with water and the decay power will be accommodated either with the core intact or in a disrupted state with no active heat sink. The supporting analysis for this conclusion is reported in Appendix IV.5.

IV.1.5 Containment Strength Evaluation

The pressure to which containment integrity can be assured has been calculated to be 72 PSID based on the ASME III, App. F faulted allowable stress of 0.7 times ultimate strength. This pressure is further corroborated for containment cable penetrations in Consumers Power Company submittal on environmental qualification dated October 31, 1980.

For some sequences in which the containment atmosphere is significantly diluted by steam prior to enclosure isolation, it may be necessary to provide vacuum relief when the enclosure spray is activated. The allowable level of vacuum in the enclosure was calculated in the design analysis to be 0.94 PSID (assuming a snow load of 0.28 PSI).

For sequences in which cooling of the core is provided by core spray and the containment is cooled by the enclosure spray following RDS actuation, it is ultimately necessary to switch to the recirculation mode of core cooling. Should this switch-over not occur, the enclosure spray may continue to add water to the containment until the sphere fails. Based on the same allowable stress as the overpressure failure criterion, we have determined that failure of the sphere by overfilling will not occur below a level of 634'6". The analysis supporting the above conclusions is presented in Appendix IV.6.

IV.2 Preliminary Assessment of Natural Convection Cooling Capability of BRP Following RDS

IV.2.1 Introduction

One of the scenarios developed in the BRP/PRA study has the reactor depressurized without any means of providing core cooling (i.e., no core spray available). While it is presently being assumed that under these conditions the core will melt, no definitive calculations have been performed which demonstrate that this assumption is correct. In particular, the question to be answered is whether or not there is sufficient natural convection capability available to cool the core using steam, assuming that losses to the enclosure environment from the pipes and steam drum will provide the necessary heat sink?

The purpose of this study is to address that question as simply as possible. If, for example, it can be shown that melting will occur even if some gross, unconservative simplifications are made, then the assumption can be made that core melting will occur given RDS and no core spray.

IV.2.2 Model

Figure 1 shows the simplified flow diagram used in this study. The natural convection flow path for the steam is from the core to the steam drum via the six-14 inch risers; from the steam drum to the core via the four-17 inch downcomers which merge into two-20 inch pump discharge lines. In this simple model, only the frictional losses in the core, the risers, the downcomers and the pump discharge lines are considered in the first calculation. No losses were considered from the reactor internals, expansion and contractions, pipe bends, or through the non-rotating pumps. While all of these losses are important, neglecting the pump loss is very non-conservative for the purposes of this study. A second set of calculations were performed in which the pump resistance was estimated (Section 2.1). Although this estimate is crude, it does point out how important the losses through the pump will be.

IV.2.2.1 Frictional Losses in the System

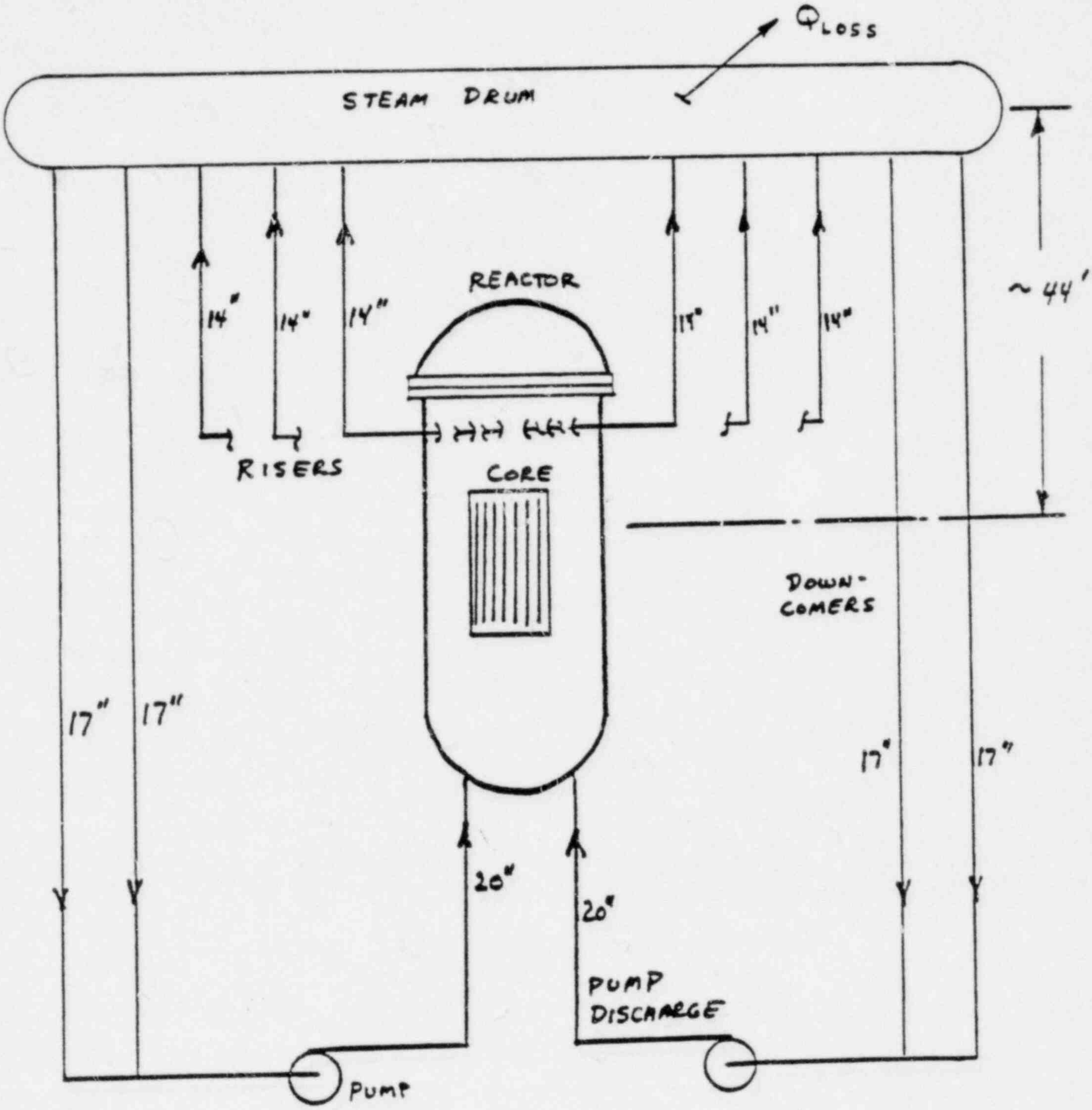
Table 1 shows the hydraulic characteristics of the four items used in the study. Average thermodynamic properties of steam in the 50-60 psia range are given below:

$$C_p \doteq 0.5 \text{ BTU/lb-}^{o}\text{F}$$

$$\mu \doteq 0.4 \text{ lb/ft-hr}$$

$$k \doteq .02 \text{ BTR/ft-hr-}^{o}\text{F}$$

$$\rho \doteq .12 \text{ lb/ft}^3$$



SIMPLIFIED FLOW DIAGRAM

Fig. 1

Table 1

Geometry of Flow Path Used in the Analyses

Item	Number	Pipe O.D. (in)	Average Length (ft)	Wall Thick (in)	Hydraulic Characteristics De (ft)	A_f (ft^2)	L (ft)	$L/(De \cdot A_f^2)$
1. Risers	6	14	58.8	0.75	1.04	5.11	58.8	2.165
2. Downcomers	4	17	60.6	0.858	1.27	5.10	60.6	1.835
3. Pump Discharge Pipe	7	20	58.7	1.009	1.5	3.53	58.7	3.140
4. Core	1	--	6.7	--	1.4×10^{-2}	13.77	6.7	0.862
								$\sum \approx 8.0$

The pressure loss equation for any pipe section can be written as:

$$P_i = \frac{fL}{D_e}\left[\frac{(W/3600)^2}{2g\,\rho A^2 \cdot 144\ in^2/ft^2}\right] \qquad (lb/in^2)$$

where W is the total mass flow in lb/hr.

Assuming a value for the friction factor (f) of 0.035, a combined expression for the pressure loss through the four items used is found to be:

$$\Delta P_T = 5.55 \times 10^{-10} W^2$$

A plot of the pressure loss as a function of steam flow rate is given on Figure 2. Also shown on this figure is an estimated curve of the pressure drop including allowance for a non-running pump. This curve was generated assuming flow area through the pump could be represented by a ½ inch annular space between a circular impeller and a circular housing. The flow area for two pumps is then approximately equal to 0.22 ft^2. The pressure drop is given by

$$\Delta P = K\,\frac{(W/3600)^2}{2g\rho A^2 \cdot 144}$$

With a combined contraction and expansion loss coefficient of 1.5, we find that

$$\Delta P = 2.2 \times 10^{-9} W^2$$

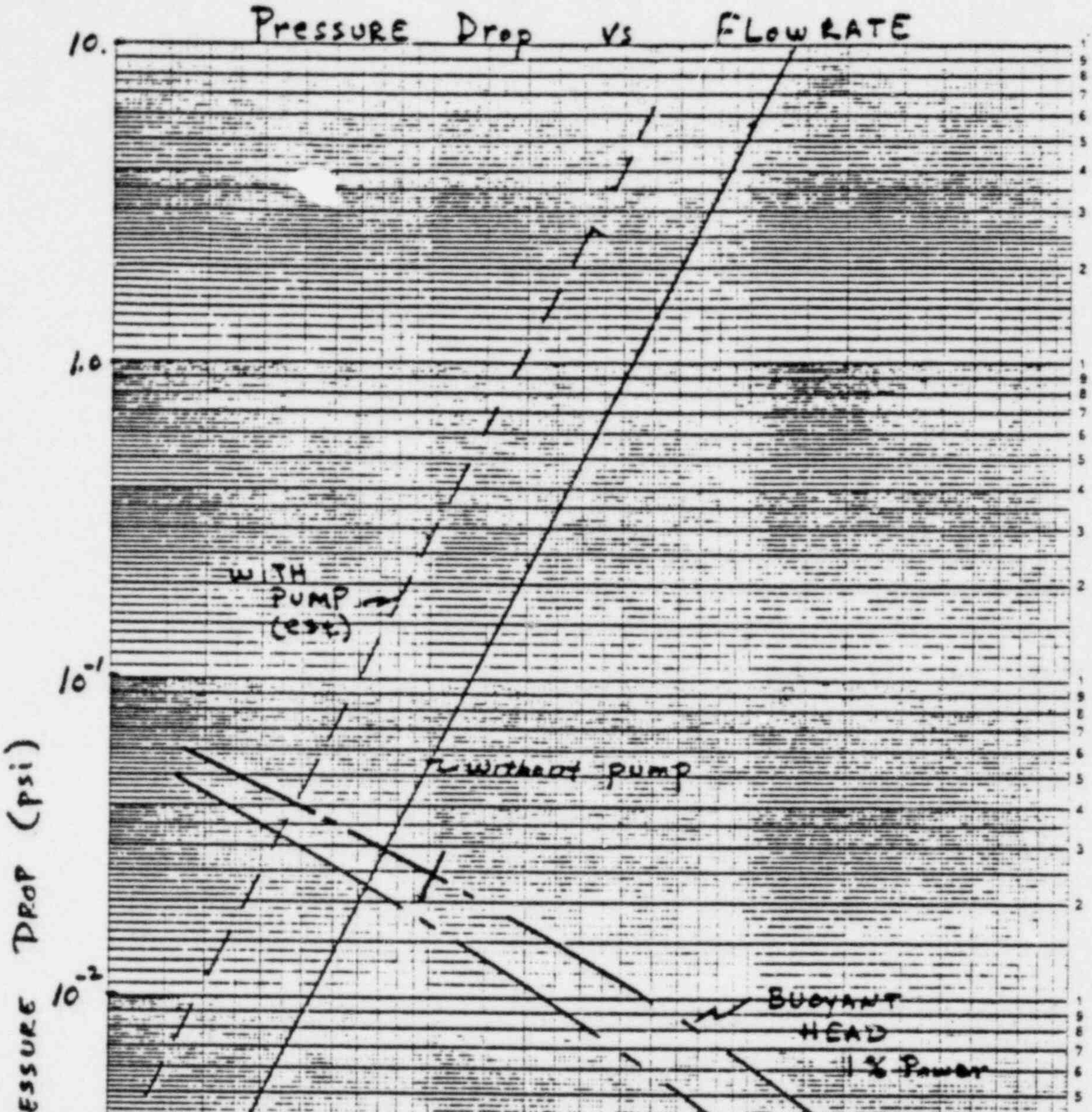
which must be added to the previous loss to give

$$\Delta P_T = 2.75 \times 10^{-9} W^2$$

(Note that the pump loss is about five times the losses in the rest of the system.)

IV.2.2.2 Buoyant Driving Force

To achieve a stable operating condition, the difference in density between the cold leg steam and the hot leg (riser) steam provides the driving force which establishes the flow rate. When this driving force is equal to the frictional losses in the system, a stable condition will occur.



Pressure Drop vs Flowrate
10.
1.0
10^{-1}
10^{-2}
10^{-3}
10^{-4}
Pressure Drop (psi)
WITH PUMP (est)
without pump
BUOYANT HEAD 1% Power
BUOYANT HEAD ½% Power
10^3
10^4
10^5
10^6
STEAM FLOW RATE (lb/hr)
K&E LOGARITHMIC 46 7602
KEUFFEL & ESSER CO.

Fig. 2

If the assumption is made that all of the heat losses occur in the steam drum (i.e., the highest point), a maximum driving force will be established. If, in fact, losses occur throughout the system, the driving head will be degraded from that obtained assuming all losses in the steam drum.

For purposes of this calculation, a heat loss equivalent to one percent power was assumed:

$$Q = (0.01)(220 \text{ MW})(3.413 \times 10^6 \text{ BTU/HR-MW})$$
$$= 7.5 \times 10^6 \text{ BTU/HR}$$

For comparison purposes, a test performed at BRP several years ago indicated that losses were on the order of 1.5×10^6 BTU/HR and the heat balance diagrams in the P & ID manual assume fixed losses (throughout the entire system) of 2.7×10^6 to 4.0×10^6 BTU/hr. The value of 7.5×10^6 BTU/hr therefore represents an optimistic heat rejection capability.

With an average value for specific heat of 0.5, the temperature rise of the steam passing through the reactor is approximately equal to

$$\Delta T_c = Q/W \cdot C_p = 7.5 \times 10^6/W \cdot 0.5$$
$$= 15 \times 10^6/W$$

Assuming that the inlet temperature to the core is 300°F (T_{sat} at 50 psia is 281°F), the outlet temperature is then given by

$$T_{out} = 300 + 15 \times 10^6/W$$

The following table shows how the outlet temperature and density change as a function of flow rate. The driving force (buoyant head) is calculated from

$$\Delta P_B = (\rho_{in} - \rho_{out}) \cdot 44 \text{ ft}/144 \text{ in}^2/\text{ft}^2$$

$$\Delta P_B = 0.306\ (.114 - \rho_{out})$$

Effect of Flow Rate on Buoyancy at 1% Power

Flow Rate (lb/hr)	T_{out} (°F)	Outlet Density (lb/ft^3)	Driving Head (psi)
5×10^5	330	.109	.0015
10^5	450	.094	.006
5×10^4	600	.080	.010
10^4	1,800	.038	.023

The values from the above table are plotted on Figure 2 (extrapolated below 10^4 lb/hr). The intersection of the curves represent the stable operating point. Pertinent information for the two cases are shown below:

Case	Flow Rate (lb/hr)	Corresponding Outlet Cooling Temp. (°F)
Without Pump	7.3×10^3	2355
With Pump (est.)	3.8×10^3	4250

It is obvious that when the pump is considered (even with the rough estimate), the cladding temperature will exceed the melting point. Without consideration of the pump, the cooling temperature is below the melting point. However, since the film drop must be added to this value (see Section 2.3), even this case has unacceptable temperatures.

A similar exercise was performed for one-half percent power. The curve is also shown on Figure 1 and the corresponding outlet coolant temperatures are 1475°F (without considering the pump) and 3700°F (considering the pump). This shows that even at one-half percent power (3.8×10^6 BTU/hr), temperatures will be unacceptable when all of the losses (e.g., pump, bends, etc.) are considered.

IV.2.2.3 Temperature Drop Between Clad and Coolant

In addition to the temperature rise of the coolant from core inlet to exit, there is a film drop between clad and coolant. The equation for the heat transfer coefficient is

$$N_u = 0.023\ Re^{0.8}\ Pr^{0.4}$$

Using the average coolant properties and the geometry of the fuel rod, we find that this reduces to

$$h = 1.4 \times 10^{-3}\ W^{0.8}\ \text{BTU/HR-ft}^2\text{-}^\circ\text{F}$$

For the one percent power case, the average heat flux is

$$\phi = (7.5 \times 10^6\ \text{BTU/hr}) \div (84\ \text{ass'y} \times 115\ \text{rods} \times 0.686\ \text{ft}^2/\text{rod})$$

$$\phi = (7.5 \times 10^6) \div (6624\ \text{ft}^2)$$

$$\phi = 1132\ \text{BTU/HR-ft}^2$$

There is also a 2.1 peaking factor so that the maximum heat flux is 2378 BTU/HR-ft^2. However, since the maximum flux will probably not occur at the point of maximum coolant temperature (i.e., at the exit), we will use the average heat flux for the calculation.

As previously noted, the equilibrium flow rate for the one percent power case without considering the pump resistance was 7.3×10^3 lb/hr and the corresponding exit coolant temperature was 2355°F. The cladding temperature is therefore given by

$$T_{clad} = 2355 + \emptyset/h$$

where $\emptyset = 1132$ BTU/HR-ft^2

$$h = 1.4 \times 10^{-3}\ W^{.8} = 1.725\ BTU/hr\text{-}ft^2\text{-}^\circ F$$

We therefore have

$$T_{clad} = 2355 + 1132/1.725 > 3000^\circ F$$

Therefore, even without considering the pump resistance, the cladding temperature reaches unacceptable values*. Oxidation will occur which will result in an exothermic reaction, thereby increasing temperatures even higher. The oxide film will also increase the thermal resistance of the cladding, thereby raising the temperature of the fuel.

IV.2.3 CONCLUSIONS

Based upon this simplified model, it is concluded that it would not be possible to remove ½ to 1 percent power from the BRP by natural losses alone. In order to achieve an equilibrium steam flow rate, the coolant temperature rise needed to support a buoyant driving force would raise the cladding temperature above 3000°F even if losses through critical items such as the pump are not considered. If such losses are considered, the resulting temperatures are well above the clad melting point.

IV.3 Probability of Failure to Isolate Containment

There are a number of pathways by which the enclosure interior can communicate with the outside environment. Should any series of events occur which result in a radiation release inside the enclosure space, these pathways, which range in size from 1/2" diameter up to 24" diameter, must be tightly closed to prevent these releases from flowing directly to the environment. In this section, the specific ways by which this enclosure isolation can fail to occur are analyzed.

IV.3.1 Methodology

This analysis is comprised of two fundamental tasks:

1. determination of the particular failure modes of the enclosure barrier, and
2. a quantification of the probability that these failure modes will occur.

Since the primary purpose of this analysis with respect to the overall risk assessment is to provide input relevant to the calculation of size and probability of source terms, the most significant features of the enclosure failure mode is the resultant size "hole" in containment. That is, the general "failure to isolate" event must be broken down into more precisely defined events which can be translated into leak rates. This was accomplished by identifying all of the existing enclosure penetrations and categorizing these penetrations according to size. It was further recognized that the failure to isolate any particular penetration can result from two basic mechanisms: (1) failure of valves within the penetration to close or remain closed, or (2) leakage past closed valve(s).

Once all of the individual penetrations are identified, logic models were developed to determine the specific combinations of component failures which must occur to produce a failure to close the penetration upon demand.

Included in these models were the valves used to isolate the containment, the actuating mechanisms for these valves, the associated control circuitry, etc. The probability of each of the events in these logic models was then quantified and an overall probability of failing to close each penetration was calculated. In addition, the probability of experiencing leaks past each of the valves was determined and translated into a probability of a leak occurring through each penetration.

IV.3.2 Results of Analysis

There were 116 separate penetrations of containment identified. A large proportion of these, however, were electrical cable or instrumentation piping penetrations which did not significantly contribute to either the probability or consequences of containment non-isolation. The remainder of the penetrations involved locks, vents, or piping penetrations. With the exception of the personnel and equipment locks, these potential pathways through containment ranged in size from 1/2" to 24" diameter.

These penetrations were grouped into nine basic system categories:

1. Locks
2. Vents
3. Steam Line/Feedwater
4. Sumps
5. Demin/Waste
6. Air Supply
7. Fuel Pit
8. Resin Sluice
9. Control Rod Drive

Each system usually involved a number of separate penetrations of various sizes.

Logic models were developed for each system to identify the combinations of failures that must occur to produce an open pathway through containment. These logic models in the form of reliability block diagrams (RBD's) are presented in Figures IV-1.1 through IV-1.9.

Each event in the RBD's details the individual component's identifier (e.g. MO7050 is the main steam isolation valve), information concerning the valves' normal position, and the particular failure mode of the component (e.g. "fails to close"). A list of the acronyms used in the logic models is presented in Table IV-1. In addition, the size of each separate pathway associated with the system is indicated on the RBD of that system.

The next step in the analysis was to quantify the logic models. This entailed determining the probability of occurrence of each event in the logic models and algebraically combining these probabilities according to the boolean logic reduction of the RBDs. The component failure rate data is listed in Table IV-2. These failure rates are based on Big Rock point plant specific experience over the last 10 years. Table IV-3 presents a summary of Big Rock Point containment isolations problems during the ten-year period 1970-1979 as reported in the Licensee Event Reports (LERs). For those cases where BRP data was insufficient, unavailable, or very uncertain, generic industry data was used (see Appendix III for a more detailed discussion of failure data). The results of this quantification task are presented in Table IV-4.

IV.3.3 Discussion of Results

The significant conclusions which can be derived from the calculations summarized in Table IV-3 are:

1. The overall probability of failure to isolate containment upon demand is 0.25.
2. This total probability is comprised of two basic failure modes - failure-to-close and leakage. The failure to close mode occurs with a probability of 0.11 and the leakage mode occurs with a slightly higher probability of 0.14.

3. The failure-to-close mode is dominated by the probability of failing to close the vent valves. This event, which results in a 24" diameter pathway through the side of the containment, contributes over 90% of the total failure-to-close probability.

4. The total leakage failure mode probability is comprised of significant contributions from four major systems: the vents (24%), steam line/feedwater (43%), sumps (21%), and fuel pit (14%).

The primary reason that the vent valves contributed such a high percentage of the overall failure-to-close probability is the particular configuration of the solenoids used to actuate the vents at Big Rock Point. The existing design is such that, for both the supply and exhaust vents, a single solenoid failure can result in failure of both vent valves. Specifically, if either SV9151 or SV9152 binds or shorts, then both the supply valves CV4097 and CV4096 will remain open; similarly, if either SV9153 or SV9154 fails then the exhaust valves CV4905 and CV4094 will fail in the open position.

This particular design deficiency has been recognized and plans exist to change the solenoid configuration. If the solenoids on both the exhaust and supply sides were made redundant, the probability of failing to close the vent valves on demand (based on BRP plant specific data) would be reduced to 0.025 and the total failure-to-close probability would be reduced to 0.035/demand.

An important point to note concerning the leakage failure mode probability is that only one contributor, the vent valves, results in a flowpath directly to the environment. Both the sump and fuel pit penetrations drain into radwaste tanks. The steam drum/feedwater containment penetrations lead back into the feedwater or condensate system inside the turbine building. Because of the rather tortuous routes involved and the number of barriers which must be breached outside of containment before any radioactive releases through these penetrations can ultimately reach the environment, the source terms associated with these contaiment penetrations (except for possibly the noble gases) are expected to be much less than those associated with the vent valves.

The vent valve leakage probability was based on Big Rock Point plant specific data which included several failures of the butterfly vent valve (CV4097). Modifications to this valve were performed two years ago and there have been no subsequent leakage failures. However, there have only been five leak tests of this valve since the modification; this number of tests cannot support a lower estimate of the valveleakage probability than reported in Table IV-4.

Table IV-1

ACRONYMS USED IN ANALYSIS

OF	=	Operates Falsely, Spurious Operation
FTO	=	Failure to Operate
FTC	=	Failure to Close
FTRC	=	Failure to Remain Closed
XL	=	External Leakage
R	=	Rupture
IL	=	Internal Leakage
CV	=	Control Valve
SV = SOV	=	Solenoid Valve
MO = MOV	=	Motor Operated Valve
RMC	=	Remote Manual Control
NO	=	Normal Open
MAN	=	Manual Isolation Valve
AOV	=	Air Operated Valve
IPR	=	Initial Pressure Regulator
HS	=	Hand Switch (remote)
RPS	=	Reactor Protection Signal
Q	=	Cyclic Failure Rate, Failures/demand
λ	=	Hourly Failure Rate, Failures/hour
P	=	Probability of Failure, based on one demand

Table IV-2

BRP CONTAINMENT ISOLATION FAILURE RATES

FAILURE RATE DESCRIPTION		PROBABILITY
Q (LOCK - DOOR LEAK)	=	10^{-2}/d
Q (LOCK - DOOR FTC)	=	10^{-2}/d
Q (LOCK - DOOR FTRC)	=	10^{-4}/d
Q (LOCK - TEST VALVE FTRC)	=	10^{-2}/d
Q (LOCK - EXTERNAL LEAK)	=	.0000022/d
Q (FAN- OF)	=	10^{-3}/d
Q (VENT - VALVE FTC)	=	.0837/d
Q (VENT - VALVE LEAK)	=	.1/d
Q (VENT - EXTERNAL LEAK)	=	.000022/d
Q (SOV - VALVE FTO)	=	.02/d
Q (SOV - VALVE LEAKS)	=	.01/d
Q (SOV - VALVE OF)	=	.00022/d
Q (MSIV - MOV FTC)	=	.0384/d
Q (MOV - VALVE FTRC)	=	.0019/d
Q (MOV - VALVE FTC)	=	.0156/d
λ (MSIV - MOV FTRC)	=	.881/10^6 hr.
Q (CONTROL VALVE - FTC)	=	.0001/d

Table IV-3

BRP LICENSEE EVENT REPORTS OF
CONTAINMENT ISOLATION PROBLEMS ('70-'79)

Date	Event
10/30/79	Transistor failure in power supply in containment vacuum relief control system. Would have caused CCF of supply vent valves to FTC (CV4096 & CV4097)--power was reinstated.
9/11/79	Test of fuel pit drain, CV4027 leaked but CV4117 (redundant) was operable. Seat was machined
3/2/79	Resin sluice valves CV4093 and CV4092 had excessive leakage, but CV4091 provided integrity. Seats were replaced on two valves.
3/2/79	Check valves in 3/4" lub line to CRD Pump 1 both leaked--degraded containment. Manual valving re-established containment.
2/4/79	(ibid 3/2/79)
2/1/79	CV4097 was leaking but CV4096 was operable--Disc seal on 4097 was adjusted; improved seal planned.
9/6/78	MO7050 (MSIV) failed to close due to hardened valve stem packing. Packing lubed and new packing ordered.
9/4/78	CV4097 leaked--seal adjusted (15-25 mil. openings--several)
8/29/78	CV4027 (in fuel pit drain line) leaked, CV4117 ok. Foreign material flushed out.
8/19/78	Demin check valve reverse leak over a three week period. Valve repaired. Redundant valve CV4105 was closed to isolate.
2/17/78	SOV SV4879 in cleanup resin sluice was deemed inoperative in accident environment. Put into safe mode for isolation.
1/20/78	CV4097 leaked; CV4096 ok. Realigned valve disc/disc ring.

Table IV-3 (cont.)

Date	Event
9/16/77	Both check valves in 10" fW line (VFW9 and VFW304) leaked. Line isolated outside containment. Scale on seat of check valve--grinding removed scale.
9/4/77	CV4093 in resin sluice FTC for ILRT. Leakage terminated by manual valving. Binding caused failure; tab was re-oriented.
8/14/77	Check valve in Demin leaked; redundant CV4105 was ok. Deposits were cleaned off valve.
8/12/77	Set of integral poppet check valves to one CRD pump leaked. Changed to second pump. Second 2" check valve in common to both pumps installed.
8/12/77	CV4093 leaked. Flushing resolved problem. Backup CV4091 was ok.
6/19/76	CV4096 ok, CV4097 leaked. Both were cleaned. Minor seat adjustments made to CV4097.
4/28/76	Resin sluice valves (CV4091, 2, 3) all failed to close. Manual valve closure provided containment integrity. Defective plunger in SV4879 in the air operating line to the three valves.
4/17/76	CV4097 leaked (18 x T.S.); CV4096 ok; disc was adjusted on CV4097.
9/17/74	Emergency lock test fixture was left installed following test. Inner door open since 6/21/74. One half inch diameter opening to outside since that date (90 days). Test fixture installed 4/29/74 and left open. Procedures changed so that doors always closed. Detailed steps to return to normal after test.
4/26/74	CV4097 flange leak. Flange bolting was tightened. (No redundancy to flange--leakage was 2 scfm.) This valve was replaced earlier in 1974.
10/17/73	Loose fitting on SV9153 (exhaust). Foreign material under seat on SV9152 (supply). Valves were repaired. No failure of vent valves to close.
11/23/72	SV4876 defective (to CV4027) and replaced by new design.
8/31/72	SV4876 defective (to CV4027) and replaced.
'71 & '70	No reports on containment.

Table IV-4

PROBABILITIES OF FAILURE TO ISOLATE CONTAINMENT

System	Small Leak	½" Hole	2" Hole	12" Hole	24" Hole	Total
1. Locks	$4x10^{-4}$	$1x10^{-4}$	-	-	$1x10^{-6}$	$5.0x10^{-4}$
2. Vents	$3x10^{-2}$	-	-	-	$1x10^{-1}$	$1.3x10^{-1}$
3. Steam Line/ Feedwater	$6x10^{-2}$	$3x10^{-3}$	$2x10^{-3}$	$1x10^{-3}$	-	$6.6x10^{-2}$
4. Sumps	$3x10^{-2}$	-	$3x10^{-3}$	-	-	$3.3x10^{-2}$
5. Demin/ Waste	$1x10^{-3}$	-	$4x10^{-5}$	-	-	$1.0x10^{-3}$
6. Air Supply	$2x10^{-5}$	-	$2x10^{-1}$	-	-	$2.0x10^{-5}$
7. Fuel Pit	$2x10^{-2}$	-	$2x10^{-3}$	-	-	$2.2x10^{-2}$
8. Resin Sluice	$2x10^{-5}$	-	$1x10^{-7}$	-	-	$2.0x10^{-5}$
9. Control Rod Drive	$1x10^{-7}$	$\sim 10^{-7}$	-	-	-	$1.0x10^{-7}$
TOTALS	$1.4x10^{-1}$	$3.1x10^{-3}$	$7.0x10^{-3}$	$1.0x10^{-3}$	$1 x10^{-1}$	$2.5x10^{-1}$

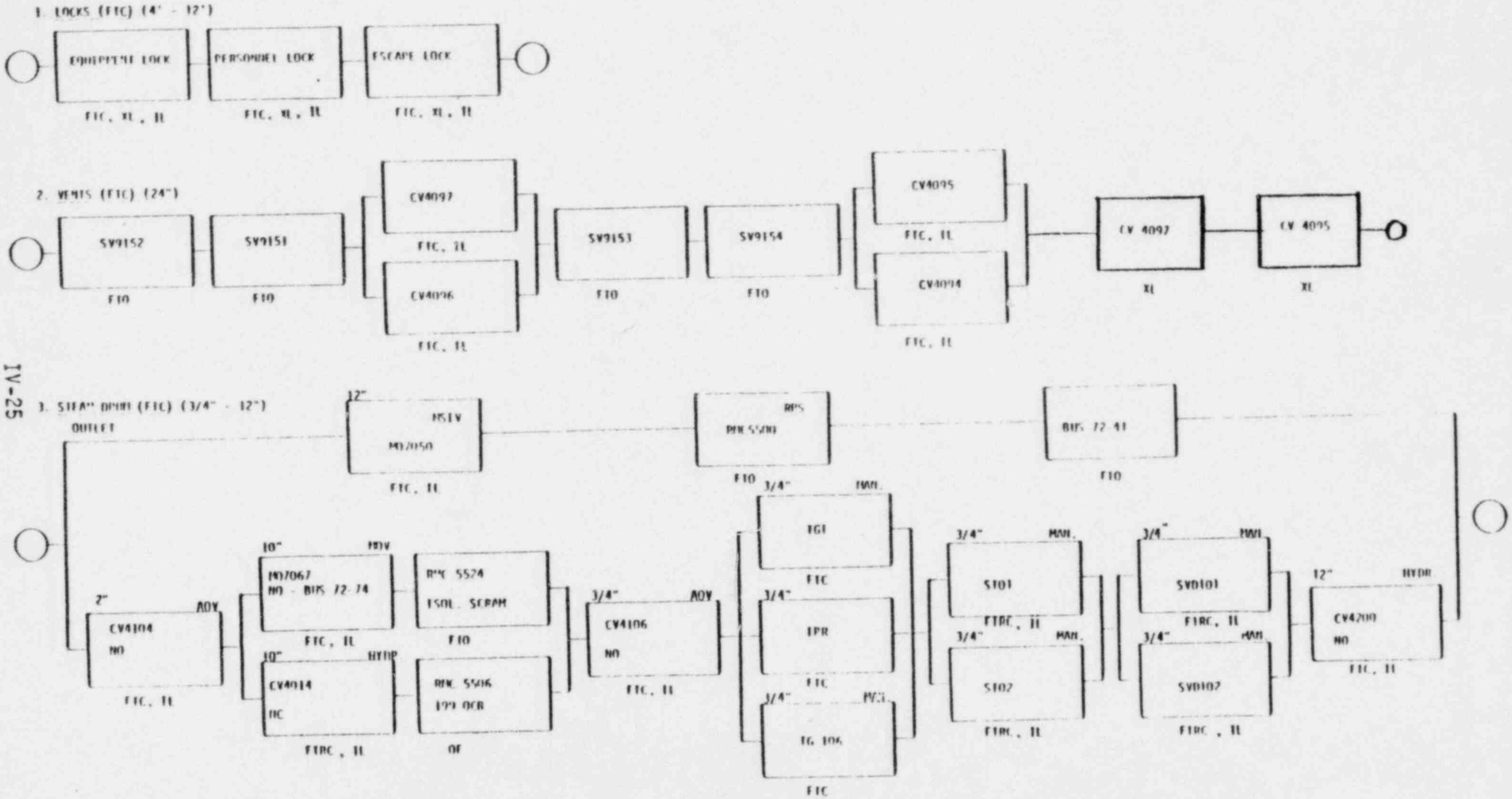


Figure 1 Failure to Isolate Containment Reliability Block Diagrams

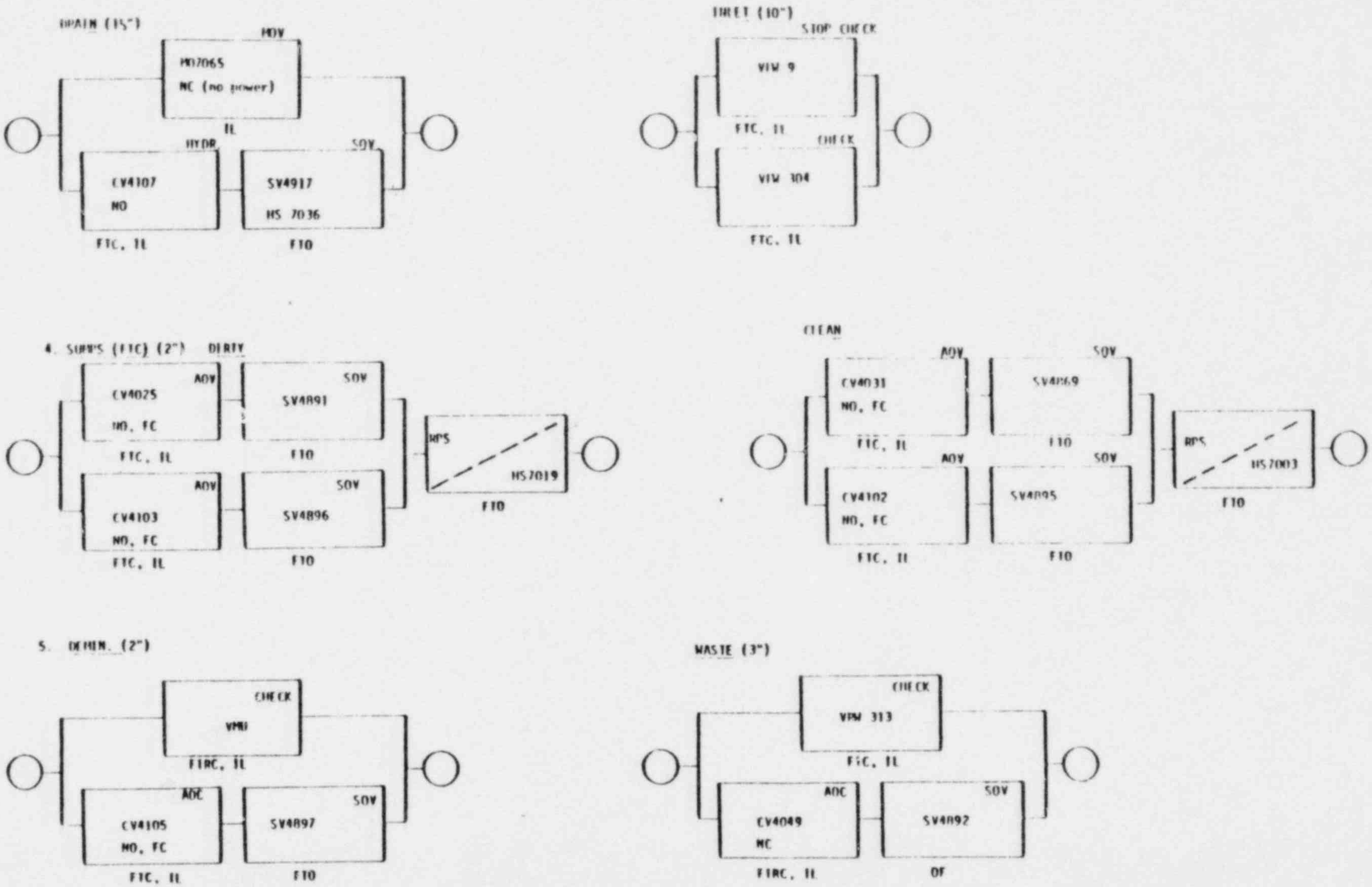


Figure 1 (cont.)

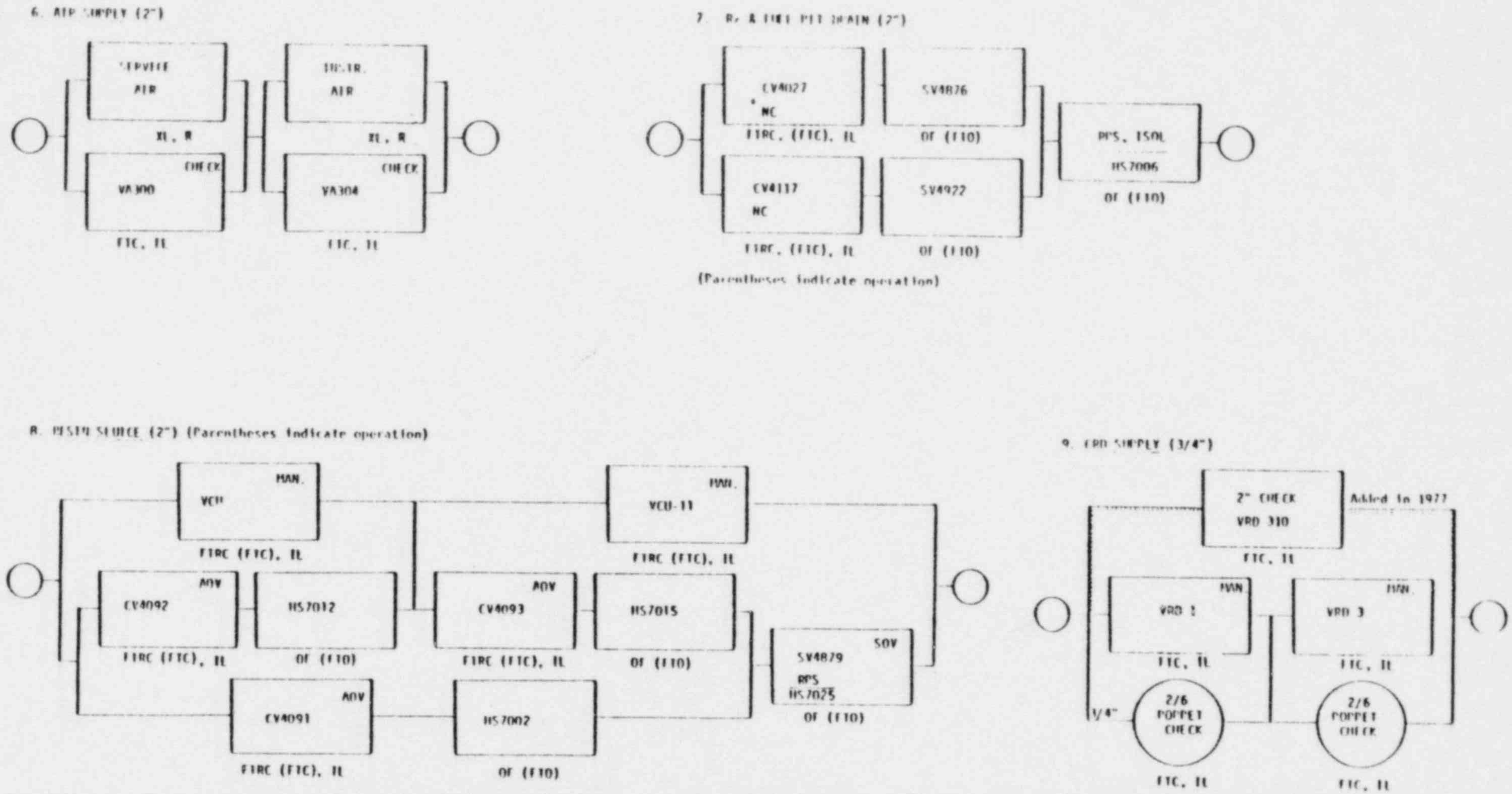


Figure 1 (cont.)

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COPY

July 20, 1982

Dennis M Crutchfield, Chief
Operating Reactors Branch No 5
Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 -
BIG ROCK POINT PLANT - RESPONSE TO
DRAFT SER ON SEP TOPIC VI-4,
CONTAINMENT ISOLATION SYSTEM

By letter dated June 11, 1982 the NRC submitted a draft safety evaluation report (SER) of SEP Topic VI-4, "Containment Isolation System". This letter requested Consumers Power Company responses to four (4) outstanding items which identify staff concerns about the adequacy of isolation provisions on selected piping penetrations. In summary, the outstanding items are as follows:

1. Adequacy of isolation provisions in test, vent and drain lines;
2. Adequacy of isolation or leak detection provisions on ECCS subsystems;
3. Adequacy of isolation provisions on instrument lines; and
4. Adequacy of isolation provisions on closed systems.

In addition to the above concerns, the SER indicates that in the case of item 1 there is a lack of administrative controls.

A review of valve check-off sheets has shown that those test, vent and drain line valves identified in the SER (VFW-138, VFW-171 and VPI-101) are included on valve checklists and thus are administratively controlled. A breach in a test, vent or drain line leading to a failure of containment isolation would necessarily involve a passive failure such as a pipe break. Analyses performed in conjunction with the Big Rock Point PRA have indicated that passive failures are a negligible contributor to the overall containment isolation failure probability.

Those valves identified under item 2 as being part of ECCS subsystems and maintained in a locked open position are kept in such a position to assure availability of the ECCS system if called upon to function. As is the case with the test, vent and drain line valves, the ECCS system would be required to experience a passive failure in order for containment isolation to be lost.

Items 3 and 4 involve the isolation of instrument lines and systems designated as closed inside containment. As with cases 1 and 2 described above, a loss of containment isolation would have to be the result of a passive failure.

Reference is made in the draft SER to analyses performed as part of the Big Rock Point PRA to determine the probability of containment isolation failure. Because of the small source term applicable to Big Rock Point, a very conservative approach was used in the original PRA which resulted in a failure probability of .25/demand. Because this value was somewhat high, a reanalysis was undertaken to more realistically assess the issue. The reanalysis concluded that the failure probability was actually about .06/demand. It is believed this reanalysis is still conservative in that only valves subject to periodic leak testing are considered suitable for containment isolation purposes. The magnitude of this containment failure probability is dominated by active failures such as the failure of a valve to close when called upon to do so. The reanalysis, which is to be incorporated into the Big Rock Point PRA report, is included as an attachment to this letter. Those parts of the PRA report to be revised by the attached material include Appendix IV, Section IV.3, Probability of Failure to Isolate Containment, and Appendix V, Table V.5-4, Summary of Important Accident Sequences for Big Rock Point. This revision will be formally transmitted to holders of controlled copies of the PRA under separate cover.

The Consumers Power Company review of the technical details in the draft SER is currently in progress. Based on the review to date, however, it is clear that comments will be extensive and that numerous corrections to the SER will be required. These comments will be transmitted to the NRC upon completion of our review.

Robert A Vincent (Signed)

Robert A Vincent
Staff Licensing Engineer

CC Administrator, Region III, USNRC
NRC Resident Inspector-Big Rock Point

Attachment - 62 pages

oc0782-0014b142

BIG ROCK POINT PLANT

PRA

REVISED
SECTION IV.3
APPENDICES IV & V
& TABLE V.5-4

For the 1% power case, the average heat flux is

$$\phi = (7.5 \times 10^6 \text{ Btu/h}) \div (84 \text{ assembly} \times 115 \text{ rods} \times 0.686 \text{ ft}^2\text{/rod})$$

$$\phi = (7.5 \times 10^6) \div (6624 \text{ ft}^2)$$

$$\phi = 1132 \text{ Btu/h-ft}^2$$

There is also a 2.1 peaking factor so that the maximum heat flux is 2378 Btu/h-ft^2. However, since the maximum flux will probably not occur at the point of maximum coolant temperature (ie, at the exit), we will use the average heat flux for the calculation.

As previously noted, the equilibrium flow rate for the 1% power case without considering the pump resistance was 7.3×10^3 lb/h and the corresponding exit coolant temperature was 2355°F. The cladding temperature is therefore given by

$$T_{clad} = 2355 + \phi/h$$

where $\phi = 1132 \text{ Btu/h-ft}^2$

$$h = 1.4 \times 10^{-3} W^{.8} = 1.725 \text{ Btu/h-ft}^2\text{-°F}$$

We therefore have

$$T_{clad} = 2355 + 1132/1.725 > 3000°F$$

Therefore, even without considering the pump resistance, the cladding temperature reaches unacceptable values. Oxidation will occur which will result in an exothermic reaction, thereby increasing temperatures even higher. The oxide film will also increase the thermal resistance of the cladding, thereby raising the temperature of the fuel.

IV.2.3 CONCLUSIONS

Based upon this simplified model, it is concluded that it would not be possible to remove 1/2 to 1% power from the BRP by natural losses alone. In order to achieve an equilibrium steam flow rate, the coolant temperature rise needed to support a buoyant driving force would raise the cladding temperature above 3000°F even if losses through critical items such as the pump are not considered. If such losses are considered, the resulting temperatures are well above the clad melting point.

IV.3 PROBABILITY OF FAILURE TO ISOLATE CONTAINMENT

An analysis was undertaken to analyze the probability that the BRP containment will fail to provide isolation. The specific systems examined were determined based upon work previously done as part of The Big Rock Point Probabilistic Risk Assessment. The following systems were considered those in which containment isolation failures were most likely to occur:

1. Locks
2. Vent Valves
3. Steamline
4. Feedwater
5. Sumps
6. Demin Water
7. Treated Waste
8. Fuel Pit Drain
9. Resin Sluice

Each system was analyzed by first developing a fault tree in which the paths to containment isolation failure were identified. Containment isolation was found to fail as the result of 1 of 2 mechanisms:

1. Leakage through a valve, or
2. Failure of a valve to close.

The leakage history of containment isolation valves was obtained by reviewing leak test reports. Valve failure to close probabilities were determined from reliability data presented in Appendix III of the PRA report. The leakage and failure to close probabilities were then incorporated into the fault trees to obtain the probability that a particular system would fail to provide containment isolation.

IV.3.1 RESULTS OF ANALYSIS

The overall probability of failure to isolate containment was calculated to be 6.1×10^{-2}. Table IV.4 indicates each system's contribution to this failure probability. Primary contributors to containment isolation failure are flow through the vent valves, steamline and feedwater lines.

For this analysis, the probability that the main steamline would not provide containment isolation was taken to be equal to the probability that the MSIV (MO-7050) would fail to close with no credit being taken for valves located downstream of the MSIV. A leak test program being developed for main steamline valves (CV-4014, CV-4104, CV-4106, ST-01 and SVD-101) could reduce the likelihood of containment isolation failure.

Check valves in the feedwater line (VFW 9 and 304) have failed all recent leak tests making the current probability of containment isolation failure due to flow through the feedwater lines equal to ~1.0. New check valves are being installed in the feedwater line however, replacing valves VFW-300 and 301. The containment failure analysis was performed assuming the new check valves were in place and that generic failure data was applicable.

An important point to note concerning the leakage failure mode probability is that only one contributor, the vent valves, results in a flowpath directly to the environment. Both the sump and fuel pit penetrations drain into radwaste tanks. The steam drum/feedwater containment penetrations lead back into the feed-water or condensate system inside the turbine building. Because of the rather tortuous routes involved and the number of barriers which must be breached outside of containment before any radio-active releases through these penetrations can ultimately reach the environment, the source terms associated with these contain-ment penetrations (except for possibly the noble gases) are expected to be much less than those associated with the vent valves.

The analysis presented here has considered only active failures. However, for completeness, it was necessary to examine passive failures of containment systems and structures.

Accidents such as an ATWS with unsuccessful poison injection could result in containment overpressurization and failure. Such mechanistic passive failures have been addressed on a sequence-by-sequence basis as part of Appendix V, "Radionuclide Release and Consequence Analysis." Failure mechanisms were not identi-fied for containment penetrations such as those through which electrical cable and instrumentation piping pass. These penetra-tions were therefore not considered susceptible to a passive failure. Passive failure of containment structures such as the service water system due to high-energy line breaks is possible. The likelihood of such a line break causing rupture of contain-ment or piping structures depends on the accessibility of the high-energy lines to those components and the probability that these sections of high-energy lines will rupture. It is felt that the contribution of this mechanism to the containment failure probability to be negligible when compared to the magnitude of the failure rate of active components.

TABLE IV.4
Probability of Failure to Isolate Containment

Locks	1.95×10^{-4}
Vents	1.142×10^{-2}
Steamline	3.84×10^{-2}
Feedwater	1.34×10^{-2} (a)
Sumps	3.15×10^{-4}
Demin Water	1.142×10^{-7}
Treated Waste	5.75×10^{-5}
Fuel Pit Drain	1.1×10^{-4}
Resin Sluice	3.425×10^{-4}
	6.42×10^{-2}

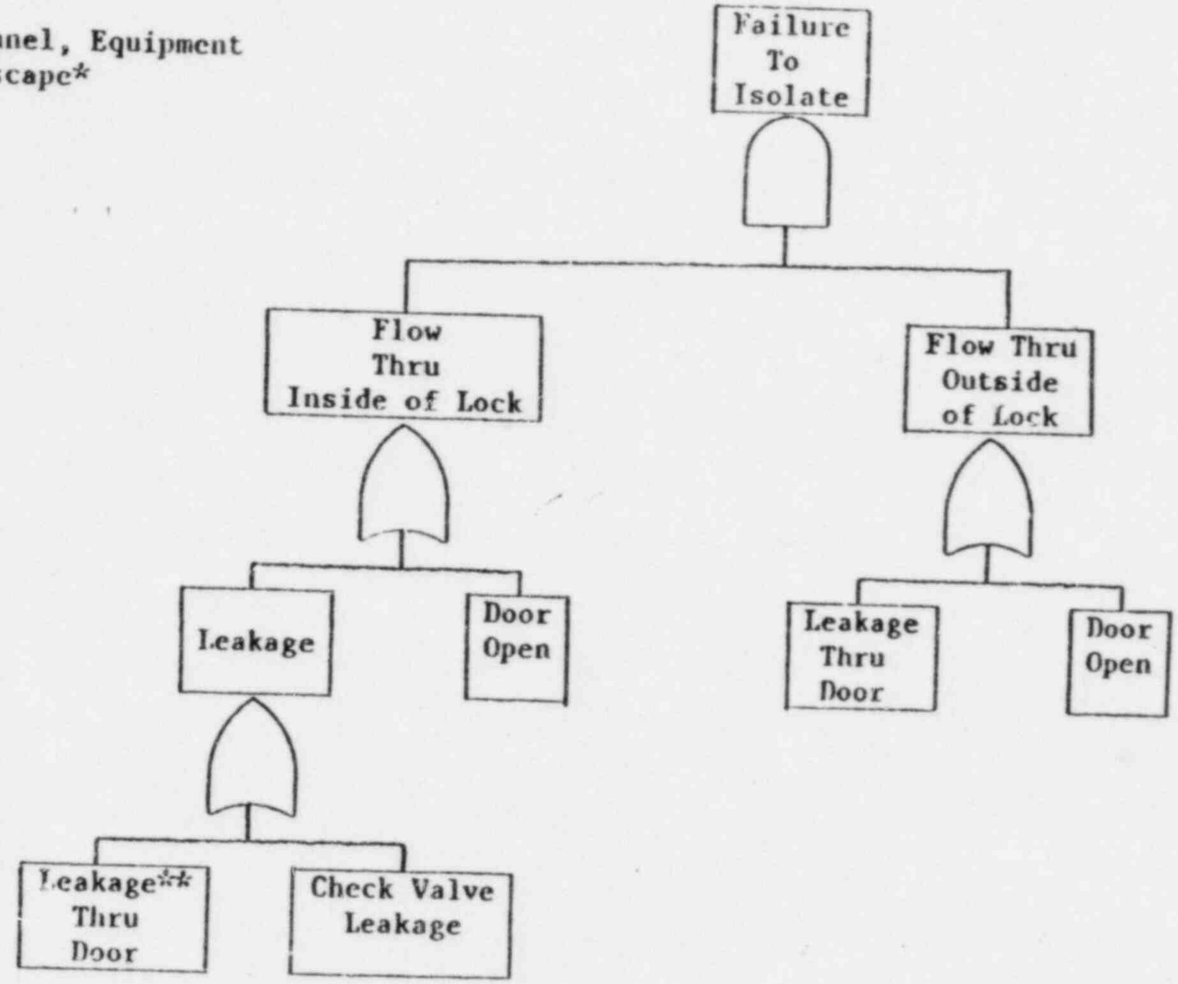
*The feedwater failure to isolate probability was determined by assuming that Valves VFW6 and VFW2 are made motor operated and generic failure data is applicable.

APPENDIX IV.4.2

Failure to Isolate Containment Fault Trees, Component Leakage Data and Calculation of Failure to Isolate Containment Probabilities

1. Locks:

Personnel, Equipment and Escape*



*Escape lock does not contain a check valve.
**Leakage thru door includes leakage thru door and leakage thru equalizing value (leakage thru the equalizing valve was not independently determined).

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TABLE IV.4.2.1.a

Personnel Lock

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope ($\frac{Lb/24\ Hr}{Day}$)
8/21/64	10.14		
2/ 9/64	23.78	172	.0793
8/24/65	.7997	196	-.1172
2/ 5/66	2.3	165	.0091
8/13/66	1.34	189	-.0051
2/21/67	17.55	192	.0844
8/18/67	7.9	178	-.0542
3/ 4/68	0.0	199	-.0397
9/ 9/68	34.2	189	.181
4/ 2/69	7.73	205	-.1291
10/14/69	6.75	195	-.005
4/14/70	6.08	182	-.0037
10/11/70	1.44	180	-.0258
4/14/71	6.95	185	.0298
10/15/71	0.0	184	-.0378
3/ 7/72	3.38	143	.0236
11/ 6/72	4.95	213	.0074
4/25/73	8.5	170	.0209
10/ 8/73	4.83	166	-.0221
3/22/74	4.84	165	.00006
9/30/74	6.7	192	.0097
3/15/75	52.07	166	.2733
3/15/75	17.42	(Retest)	
8/23/75	17.42	161	0.0
9/22/75	4.9	30	-.4173
4/21/76	1.7	211	-.0152
*7/20/76	1.7	90	0.0
11/23/76	0.0	126	-.0135
5/31/77	.26	189	.0014
1/11/78	5.12	225	.0216
9/ 4/78	2.12	257	-.0117

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Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope $\left(\frac{\text{Lb/24 Hr}}{\text{Day}}\right)$
1/20/79	19.54	138	.1262
8/28/79	10.048	220	-.0431
4/29/80	3.125	245	-.0283
10/26/80	.0018 (TS Fraction)	150	-.0154

Total Time Period = 5910

Avg Slope = .0043

Assume Leakage Limit = 225 lb/24 hr (~1/2 Tech Spec Limit)

Time to Failure =
$$\frac{225}{.0043}$$
= 52517.763 Days

Test Periods =
$$\frac{52517.763}{182.5 \text{ Day}}$$

= 287.76856

Leakage Probability =
$$\frac{288-287.76856}{288} =$$

8.036×10^{-4}

TABLE IV.4.2.1.b

Personnel Lock Check Valve

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope ($\frac{Lb/24\ Hr}{Day}$)
9/22/75	.01		
4/21/76	.00099	211	.000043
7/20/76	.00099	90	0.0
11/23/76	.0059	126	.000038
5/31/77	0.0	189	-.000031
3/14/78	.0078	287	.000027
1/23/79	.0005	315	-.000023
8/29/79	.068	218	.00031
4/15/80	.0108	230	.00025
10/12/80	0.0	180	.00006
		Total Time Period = 1848	Avg Slope = 7.81×10^{-5}

Leakage Limit $\approx$ 225 lb/24 hr

Time to Failure = $\frac{225}{.0000781}$ = 2,880,921.9 Days

Test Periods = $\frac{2880921.9}{182.5}$

= 15785.873

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope $(\frac{Lb/24\ Hr}{Day})$
			Leakage Probability = $\frac{15786-15785.873}{15786}$
			8.02×10^{-6}

TABLE IV.4.2.1.c

Equipment Lock

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope $(\frac{Lb/24\ Hr}{Day})$
8/21/64	16.58		
2/ 9/65	12.8	172	-.022
8/24/65	87.467	196	.381
2/ 5/66	0.0	165	-.5301
8/13/66	16.0	189	.0847
2/21/67	48.55	192	.1695
8/18/67	0.0	178	-.2728
3/ 4/68	0.0	199	0.0
9/ 9/68	0.0	189	0.0
4/ 2/69	32.3	205	.1576
10/14/69	61.6	195	.1503
4/14/70	1.44	182	-.3305
10/11/70	70.865	180	.3857
4/14/71	121.99	185	.2764
10/15/71	133.11	184	.0604
3/ 7/72	42.52	143	-.6335
11/ 6/72	26.40	213	-.0757
4/25/73	32.9	170	.0382
10/ 8/73	27.66	166	-.0316
3/22/74	18.84	165	-.0535
9/30/74	13.9	192	-.0257
3/15/75	31.74	166	.1075
3/15/75	8.69	(RETEST)	
4/27/75	8.74	43	.0012
8/23/75	8.74	118	0.0
9/22/75	20.55	30	.3937
4/21/76	12.63	211	-.0375
*7/20/76	12.63	90	0.0
11/23/76	1.3	126	-.0899

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Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope ($\frac{\text{Lb/24 Hr}}{\text{Day}}$)
5/31/77	2.83	189	.0081
3/14/78	30.1	287	.095
1/27/79	2.506	319	-.0865
8/29/79	2.762	214	.0012
4/29/80	35.65	244	.1348
10/26/80	.0018 (TS Fraction)	180	-.1936

Total Time Period = 5910 Days

Avg Slope <0.0

Assume Equip Lock Leakage Probability = Personnel Lock Probability = 8.036×10^{-4}

TABLE IV.4.2.1.d

Equipment Lock Check Valve

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope $\left(\frac{\text{Lb/24 Hr}}{\text{Day}}\right)$
9/22/75	.15		
4/21/76	.0242	211	-.0000596
7/20/76	.0242	90	0.0
11/23/76	.0079	126	-.00013
5/31/77	.01	189	.0000111
3/14/78	.0052	287	-.0000167
1/30/79	.0031	322	-.0000065
8/30/79	.024	212	.0000986
4/15/80	.0229	229	-.0000048
10/12/80	0.0	180	-.0000127

Total Time Period = 1848

Avg Slope <0.0

Assume Leakage Probability $\approx$ 0.0

TABLE IV.4.2.1.c

Escape Lock

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope ($\frac{\text{Lb/24 Hr}}{\text{Day}}$)
8/21/64	20.07		
2/ 9/65	7.2	172	-.0748
8/24/65	7.852	196	.0033
2/ 5/66	0.0	165	-.0476
8/13/66	7.91	189	.0419
2/21/67	4/87	192	-.0158
8/18/67	0.0	178	.0274
3/ 4/68	0.0	199	0.0
9/ 9/68	1.88	189	.0099
4/ 2/69	17.1	205	.0742
10/14/69	22.6	195	.0283
4/14/70	57.6	182	.1923
10/11/70	15.585	180	-.2334
4/14/70	57.6	182	.1923
10/11/70	15.585	180	-.2334
4/14/70	0.0	185	-.0842
10/15/71	0.0	184	0.0
3/ 7/72	0.0	143	0.0
11/ 6/72	17.7	213	.0831
4/25/73	9.9	170	-.0459
10/ 8/73	0.0	166	-.0596
8/23/75	0.0	684	0.0
9/22/75	6.61	30	.2203
*4/21/76	10.15	211	.0168
11/23/76	12.35	216	.0102
5/31/77	7.19	189	-.0273
3/15/78	.235	288	-.0241
10/ 7/78	.53	207	.0014
1/23/79	1.6281	108	.0102

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope $(\frac{Lb/24\ Hr}{Day})$
8/30/79	2.954		
4/30/80	0.0	219	.0061
10/26/80	0.0	244	-.012
		179	0.0
		Total Time Period = 5910 Days	Avg Slope <0.0
			Assume Escape Lock Leakage Probability = Personnel Lock Leakage Probability = 8.036×10^{-4}

*Prior to 11/76 the volumes assigned to the personnel, equipment, and escape locks were approximately four times greater than their actual volumes. Therefore leak rates determined prior to this time are over estimated by a factor of four.

LOCKS

A. Personnel

Leakage probability through door (assume leakage through inner and outer doors are equal): 8.036×10^{-4}
Leakage through check valve = 8.02×10^{-6}
Assume inner door open 10%
outer door open 10%
both doors closed 80%

Leakage through inside:
(Leakage through door) + (check valve leakage)
$(8.036 \times 10^{-4}) + (8.02 \times 10^{-6}) = 8.116 \times 10^{-4}$

1. Inside Door Open:
Flow through inside = 1.0
Flow through outside = (leakage through outside) + (outside door open) = 8.036×10^{-4} + 0.0 = 8.036×10^{-4}

Failure to isolate = (flow through inside)(flow through outside) = $(1.0)(8.036 \times 10^{-4}) = 8.036 \times 10^{-4}$

2. Outside Door Open:
Flow through outside = 1.0
Flow through inside = (leakage through inside) + (inside door open) = $(8.116 \times 10^{-4}) + 0.0$ = 8.116×10^{-4}

Failure to isolate = $(1.0)(8.116 \times 10^{-4}) = 8.116 \times 10^{-4}$

3. Both Doors Closed:
Flow through inside = leakage through inside = 8.116×10^{-4}
Flow through outside = leakage through outside = 8.036×10^{-4}

Failure to isolate = (leakage through inside)(leakage through outside) = $(8.116 \times 10^{-4})(8.036 \times 10^{-4})$ = 6.522×10^{-7}

Total personnel lock failure to isolate = $.1(8.116 \times 10^{-4}) + .1(8.036 \times 10^{-4}) + .8(6.522 \times 10^{-7})$ = 1.62×10^{-4}

B. Equipment

Leakage probabilities assumed same as for Personnel Lock (except check valve leakage probability ≈ 0.0)
Assume inner and outer door open ~ 1 week/year.

1. Inside Door Open:
   (Same as Personnel Lock)
   Failure to isolate = 8.036×10^{-4}

2. Outside Door Open:
   (Flow through inside)(flow through outside) = $(1.0)(8.036 \times 10^{-4})$
   Failure to isolate = 8.036×10^{-4}

3. Both Doors Closed:
   (Flow through inside)(flow through outside) = $(8.036 \times 10^{-4})^2$

   Failure to isolate = 6.458×10^{-7}

   Total equipment lock failure to isolate =
   $.02(8.036 \times 10^{-4}) + .02(8.036 \times 10^{-4}) + .96(6.458 \times 10^{-7}) = 3.28 \times 10^{-5}$

C. Escape Lock

Leakage probabilities assumed same as for equipment lock inner door tested once/day, assume inner door open ~.1%

1. Inside Door Open:

   Failure to isolate = $(8.036 \times 10^{-4})(1.0) = 8.036 \times 10^{-4}$

2. Outside Door Open:

   Failure to isolate = $(8.036 \times 10^{-4})(1.0) = 8.036 \times 10^{-4}$

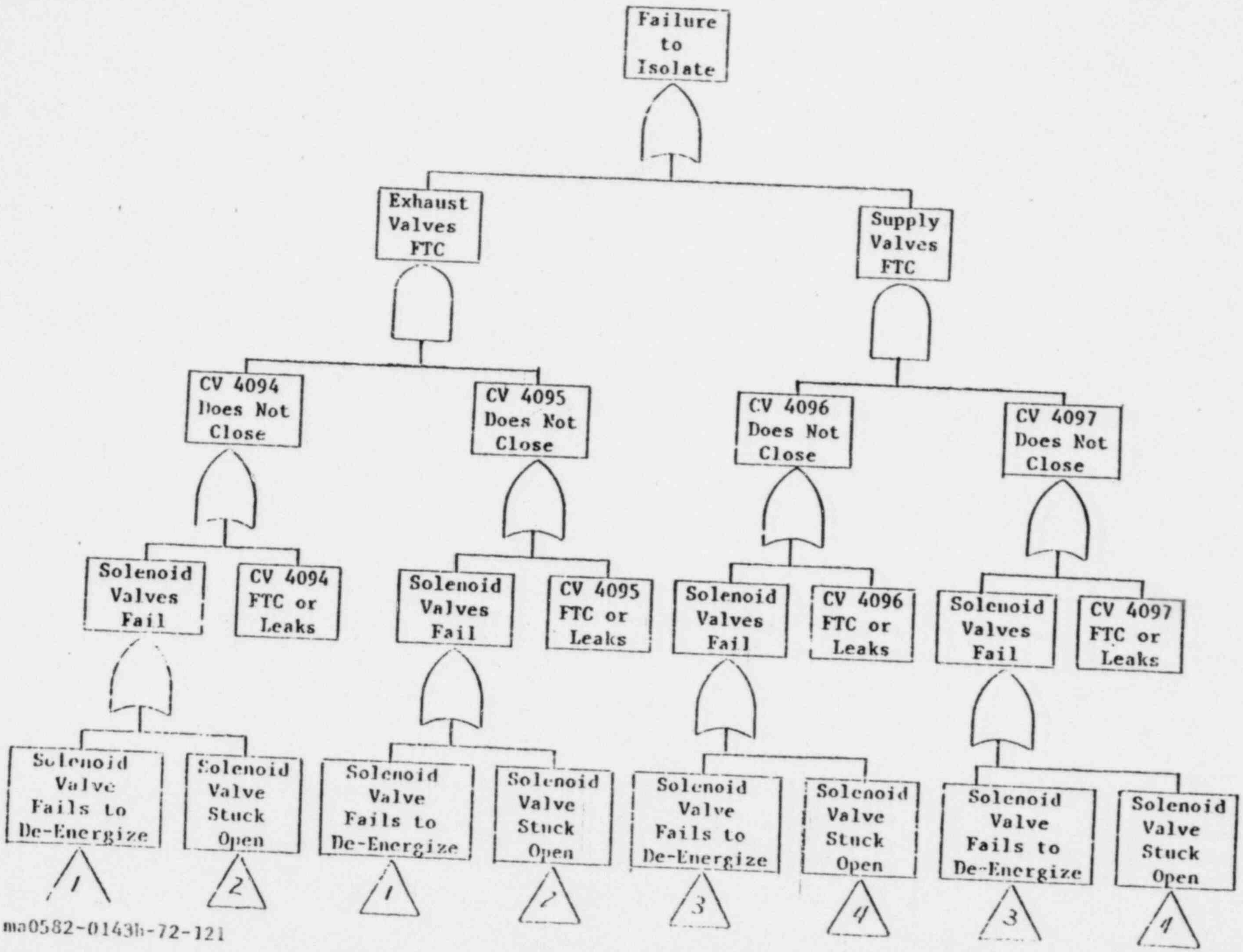
3. Both Doors Closed:

   Failure to isolate = $(8.036 \times 10^{-4})(8.036 \times 10^{-4}) = 6.458 \times 10^{-7}$

   Total escape lock failure to isolate = $.001\ (8.036 \times 10^{-4}) + 0.0\ (8.036 \times 10^{-4}) + .999\ (6.458 \times 10^{-7}) =$
   1.449×10^{-6} outer door open ~0%

Locks total failure to isolate =
(Personnel lock failure to isolate) +
(Equipment lock failure to isolate) +
(Escape lock failure to isolate) =
$(1.61 \times 10^{-4}) + (3.28 \times 10^{-5}) + 1.449 \times 10^{-6} = 1.95 \times 10^{-4}$

2. Vent Valves:



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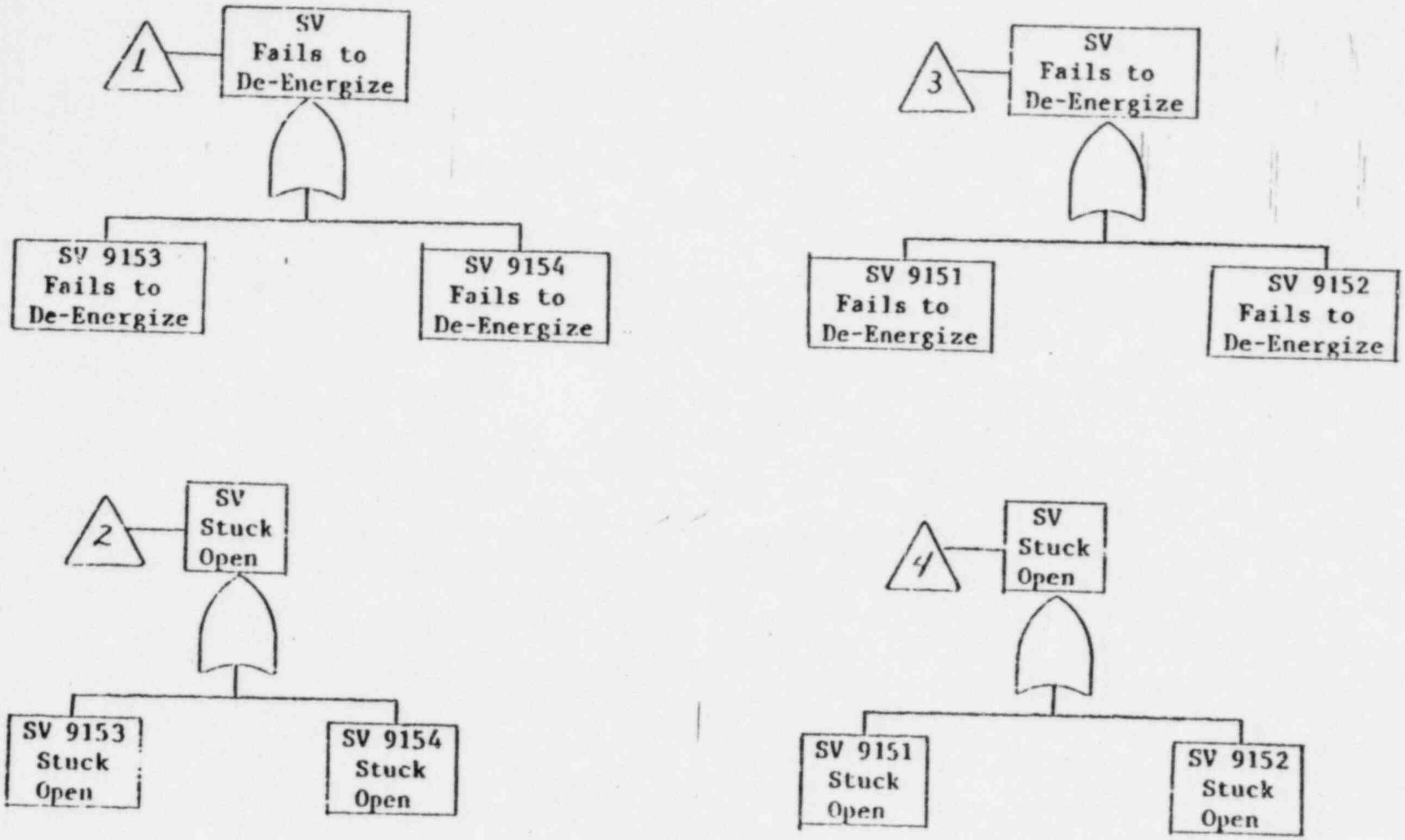


1
SV Fails to De-Energize
SV 9153 Fails to De-Energize
SV 9154 Fails to De-Energize
3
SV Fails to De-Energize
SV 9151 Fails to De-Energize
SV 9152 Fails to De-Energize
2
SV Stuck Open
SV 9153 Stuck Open
SV 9154 Stuck Open
4
SV Stuck Open
SV 9151 Stuck Open
SV 9152 Stuck Open

TABLE IV.4.2.2.a

Exhaust Vent Valves

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope ($\frac{Lb/24\ Hr}{Day}$)
8/21/64	6.61		
2/ 9/65	2.33	172	-.0249
8/24/65	8.512	196	.0315
2/ 5/66	28.679	188	.1073
8/13/66	96.0	189	.3562
2/21/67	3.09	192	-.4839
8/18/67	15.8	179	.0710
3/ 4/68	26.278	199	.0527
9/ 9/68	34.8	189	.0451
4/ 2/69	47.3	205	.061
10/14/69	33.2	195	-.0723
4/14/70	36.84	182	.02
10/11/70	17.291	180	-.1086
4/14/71	33.22	185	.0861
10/11/71	19.83	180	-.0744
3/ 7/72	25.45	148	.038
11/ 6/72	15.15	244	-.0422
4/25/73	15.5	170	.0021
10/ 8/73	21.12	166	.034
3/22/74	29.12	165	.0485
9/30/74	16.0	192	-.0683
3/15/75	90.37	166	.4480
4/27/75	77.06	43	-.3095
8/23/75	77.06	118	0.0
9/22/75	90.90	30	.4613
4/21/76	101.23	212	.0487
7/20/76	101.23	90	0.0
11/23/76	31.89	126	-.5503
5/31/77	31.89	189	0.0

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope $\left(\frac{\text{Lb/24 Hr}}{\text{Day}}\right)$
7/ 3/77	22.58	33	-.2821

Total Time Period = 4699 Days

Avg Slope = .0034 $\frac{\text{lb/24 hr}}{\text{day}}$

Assume Leakage Limit = 200 lb/24 hr

Time to Failure = $\frac{200}{.0034}$ = 58847.761 Days

Testing Periods = $\frac{58847.761}{182.5}$

= 322.45348

Probability of Leakage = $\frac{323-322.45348}{323}$ =

1.692×10^{-3}

This leakage probability is assigned to the exhaust butterfly (CV 4095), exhaust check (CV 4094) and supply check (CV 4096) valves.

TABLE IV.4.2.2.b

Supply Vent Valve (Butterfly Valve CV 4097)

Testing Date	Leak Rate Lb/24 Hr	Leak Rate Limit (1/2 TS Limit)	Slope $\left(\frac{Lb/24\ Hr}{Day}\right)$	Valve Leakage Time (Days)
8/20/64				
11/ 6/72	77.8			No Excess Leakage
4/25/73	225.0	205	.8608	17.425
9/30/74	8.69			No Excess Leakage
3/31/75	1428.5	230	7.8012	153.631
4/27/75	76.83			No Excess Leakage
5/26/75	825.3	211.85	25.8083	23.768
9/22/75	.97			No Excess Leakage
4/21/76	7454.4	208.565	35.1577	206.095
5/19/76	84.997			No Excess Leakage
6/19/76	381.69	208.93	9.5707	18.0508
9/16/77				No Excess Leakage
1/23/78	Failed to Hold Pressure			129
7/11/78				No Excess Leakage
9/ 4/78	Failed to Hold Pressure			55
10/. 5/78				No Excess Leakage
2/ 5/79	Failed to Hold Pressure			122

Leakage Time = (Testing Interval)-(Time to Leakage)

Time to Leakage = $\frac{\text{Leak Rate Limit-Leakage}(t_1)}{\text{Slope}}$

Total Leakage Time = 724.97 Days

Total Time (8/20/64-5/16/81) = 6109 Days

Leakage Probability = $\frac{724.97}{6109}$ = .1187

VENT VALVES

Supply Butterfly Valve (CV 4097) - .1187
Supply Check Valve (CV 4096) - 1.7×10^{-3}
Exhaust Butterfly Valve (CV 4095) - 1.7×10^{-3}
Exhaust Check Valve (CV 4094) - 1.7×10^{-3}

Leakage Probability

CV 4094, CV 4095, CV 4096, CV 4097 FTC - .001 (Table III-5, App III)

Solenoid Valve FTC - 4 @ .001 = .004

Flow through CV 4094: $.0017 + (.001).9 = .0026$
Flow through CV 4095: $.0017 + (.001).9 = .0026$
Flow through CV 4096: $.0017 + (.001).9 = .0026$
Flow through CV 4097: $.1187 + (.001).9 = .1196$

Flow through exhaust valves (CV 4094 and CV 4095) : $(.0026)(.0026) + (1 \times 10^{-4}) = 1.068 \times 10^{-4}$

Flow through supply valves (CV 4096 and CV 4097): $(.0026)(.1196) + 1 \times 10^{-4} = 4.11 \times 10^{-4}$

Failure to Isolate =

Probability of flow through exhaust valves +
Probability of flow through supply valves +
Probability of solenoid valves failure to close +
Probability of assorted relay or bus failures =
$(1.068 \times 10^{-4}) + (4.11 \times 10^{-4}) + (4.0 \times 10^{-3}) + (6.9 \times 10^{-3}) = 1.142 \times 10^{-2}$

VENT VALVES

Supply Butterfly Valve (CV 4097) - .1187
Supply Check Valve (CV 4096) - 1.7×10^{-3}
Exhaust Butterfly Valve (CV 4095) - 1.7×10^{-3}
Exhaust Check Valve (CV 4094) - 1.7×10^{-3}

Leakage Probability

CV 4094, CV 4095, CV 4096, CV 4097 FTC - .001 (Table III-5, App III)

Solenoid Valve FTC - 4 @ .001 = .004

Flow through CV 4094: .0017 + (.001).9 = .0026
Flow through CV 4095: .0017 + (.001).9 = :0026
Flow through CV 4096: .0017 + (.001).9 = .0026
Flow through CV 4097: .1187 + (.001).9 = .1196

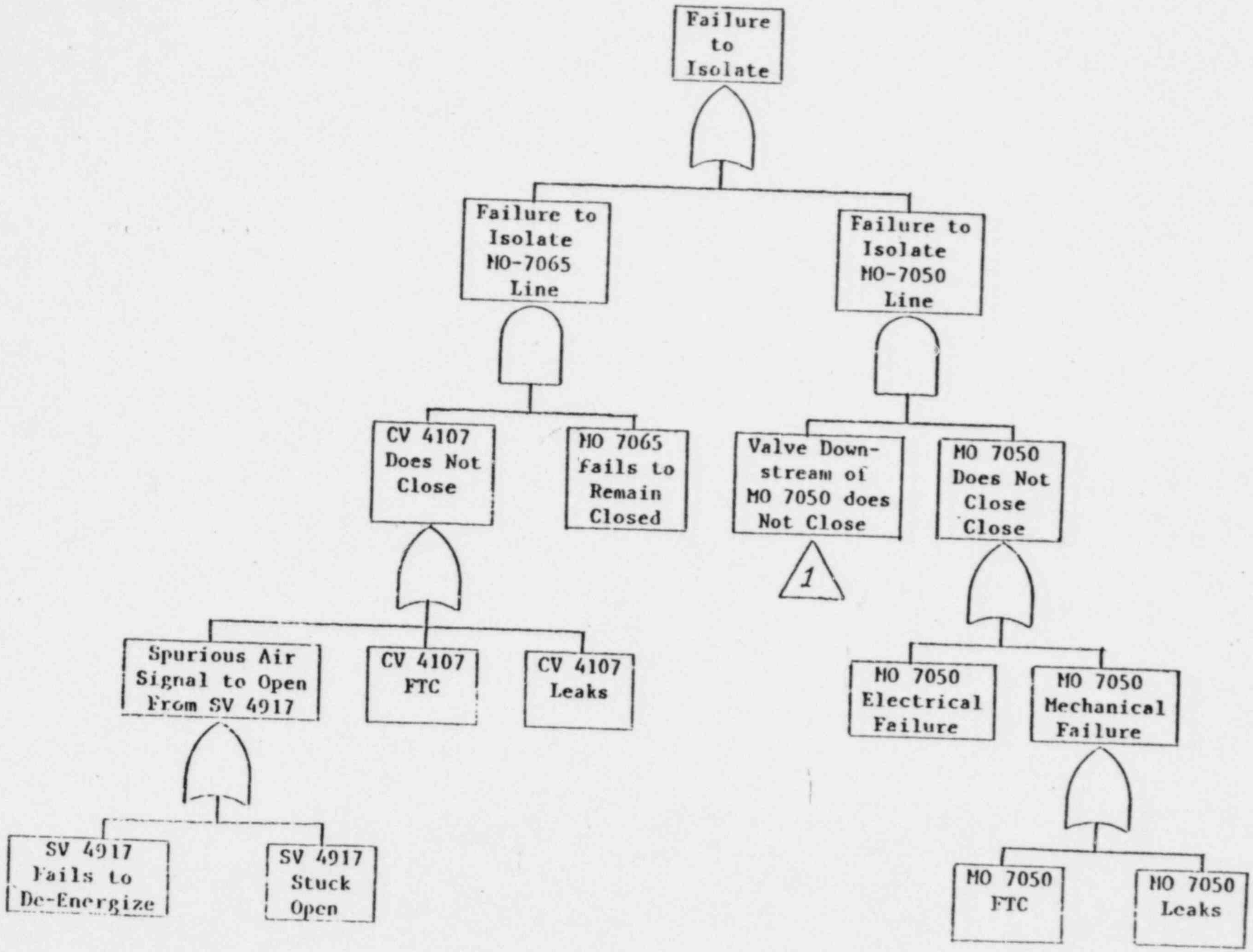
Flow through exhaust valves (CV 4094 and CV 4095) : $(.0026)(.0026) + (1 \times 10^{-4}) = 1.068 \times 10^{-4}$

Flow through supply valves (CV 4096 and CV 4097): $(.0026)(.1196) + 1 \times 10^{-4} = 4.11 \times 10^{-4}$

Failure to Isolate =

Probability of flow through exhaust valves +
Probability of flow through supply valves +
Probability of solenoid valves failure to close +
Probability of assorted relay or bus failures =
$(1.068 \times 10^{-4}) + (4.11 \times 10^{-4}) + (4.0 \times 10^{-3}) + (6.9 \times 10^{-3}) = 1.142 \times 10^{-2}$

3. Steamline:



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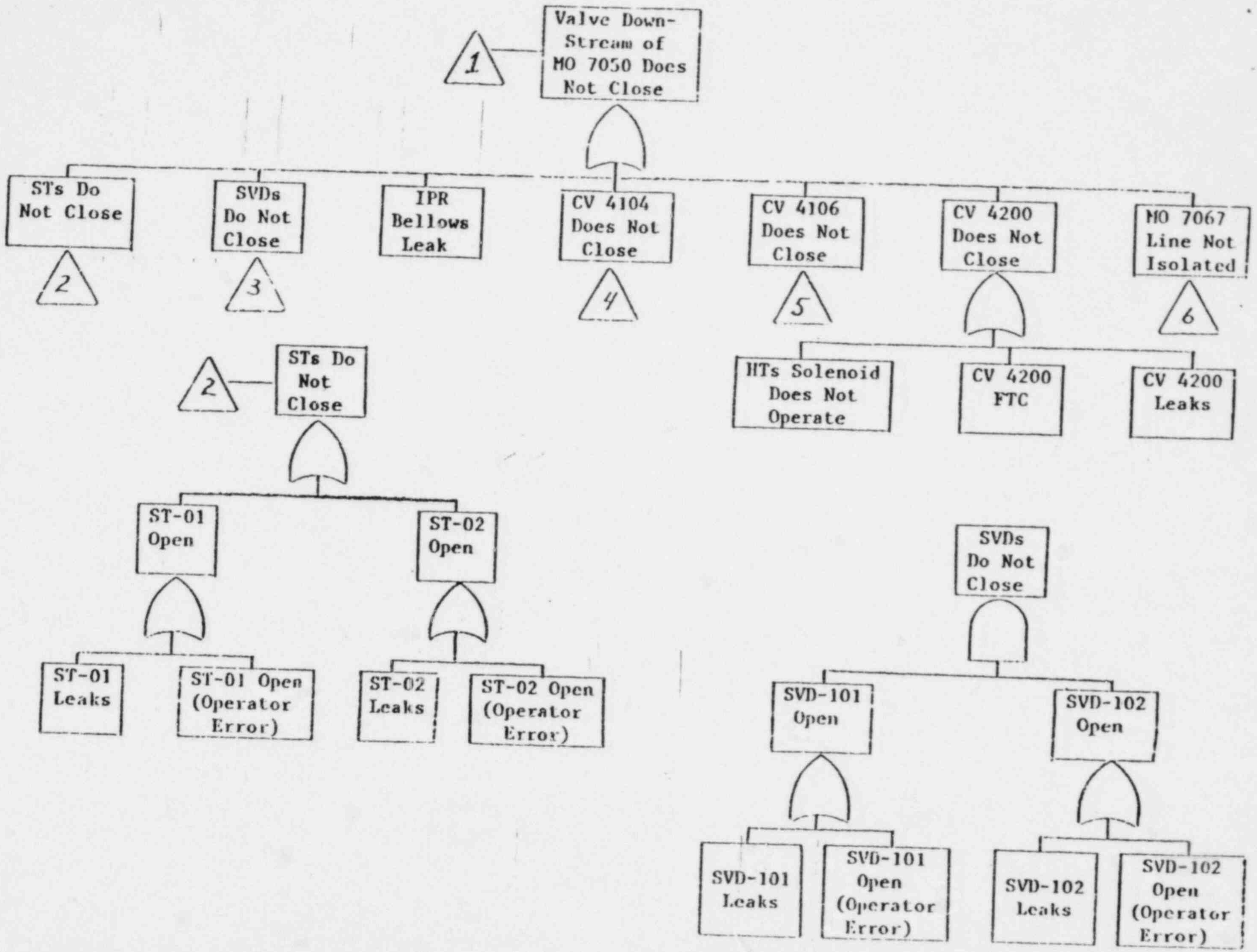

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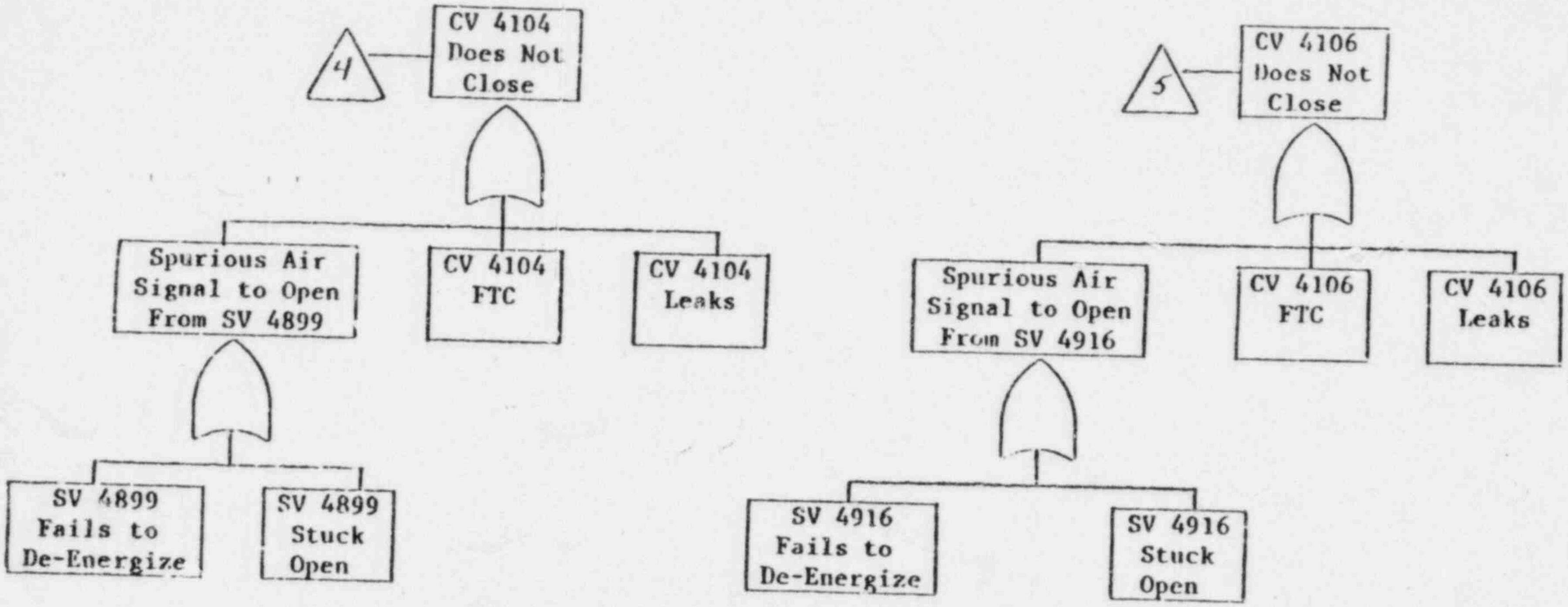
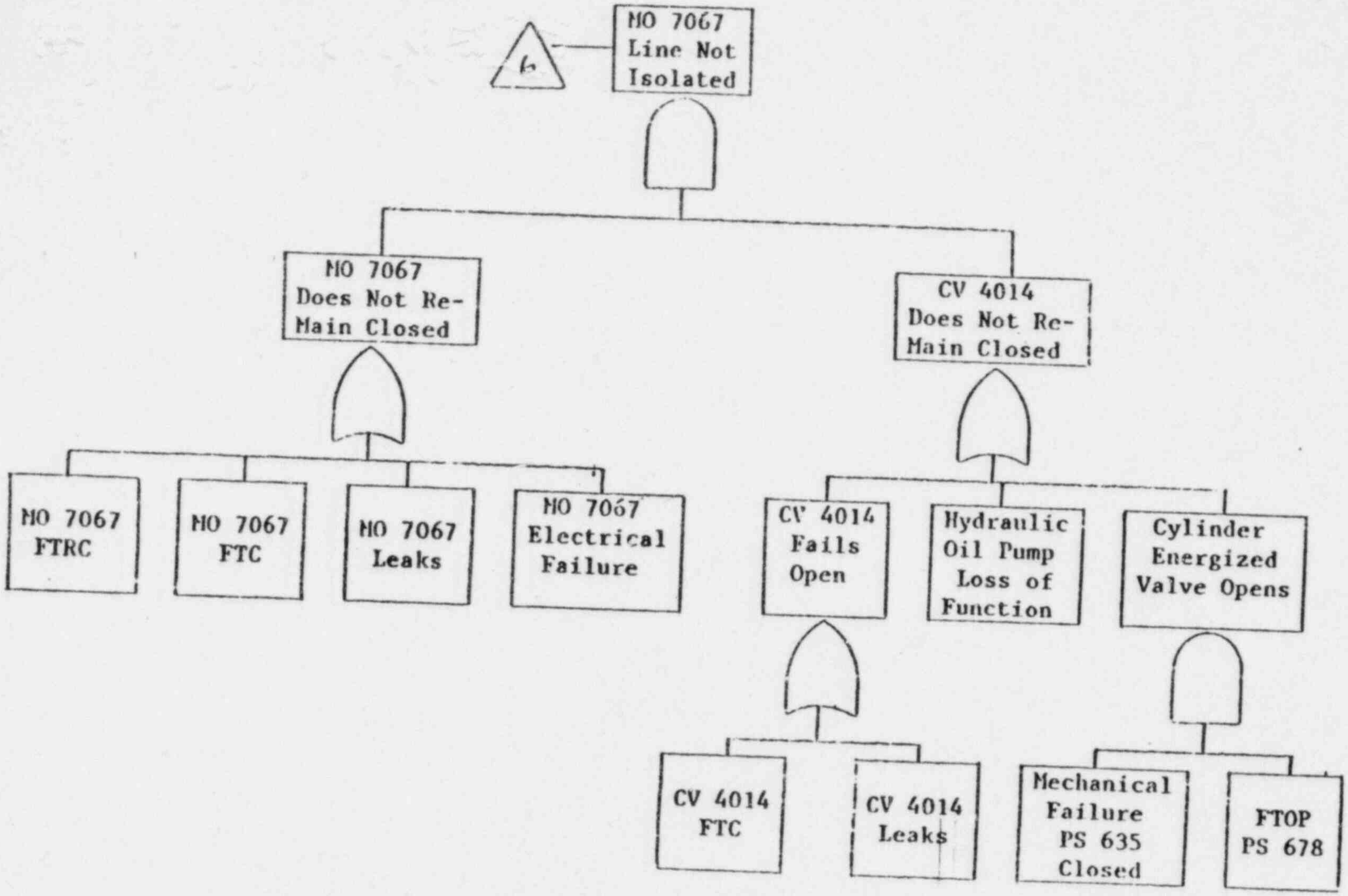
Steamline:



4
CV 4104 Does Not Close
Spurious Air Signal to Open From SV 4899
CV 4104 FTC
CV 4104 Leaks
SV 4899 Fails to De-Energize
SV 4899 Stuck Open
5
CV 4106 Does Not Close
Spurious Air Signal to Open From SV 4916
CV 4106 FTC
CV 4106 Leaks
SV 4916 Fails to De-Energize
SV 4916 Stuck Open

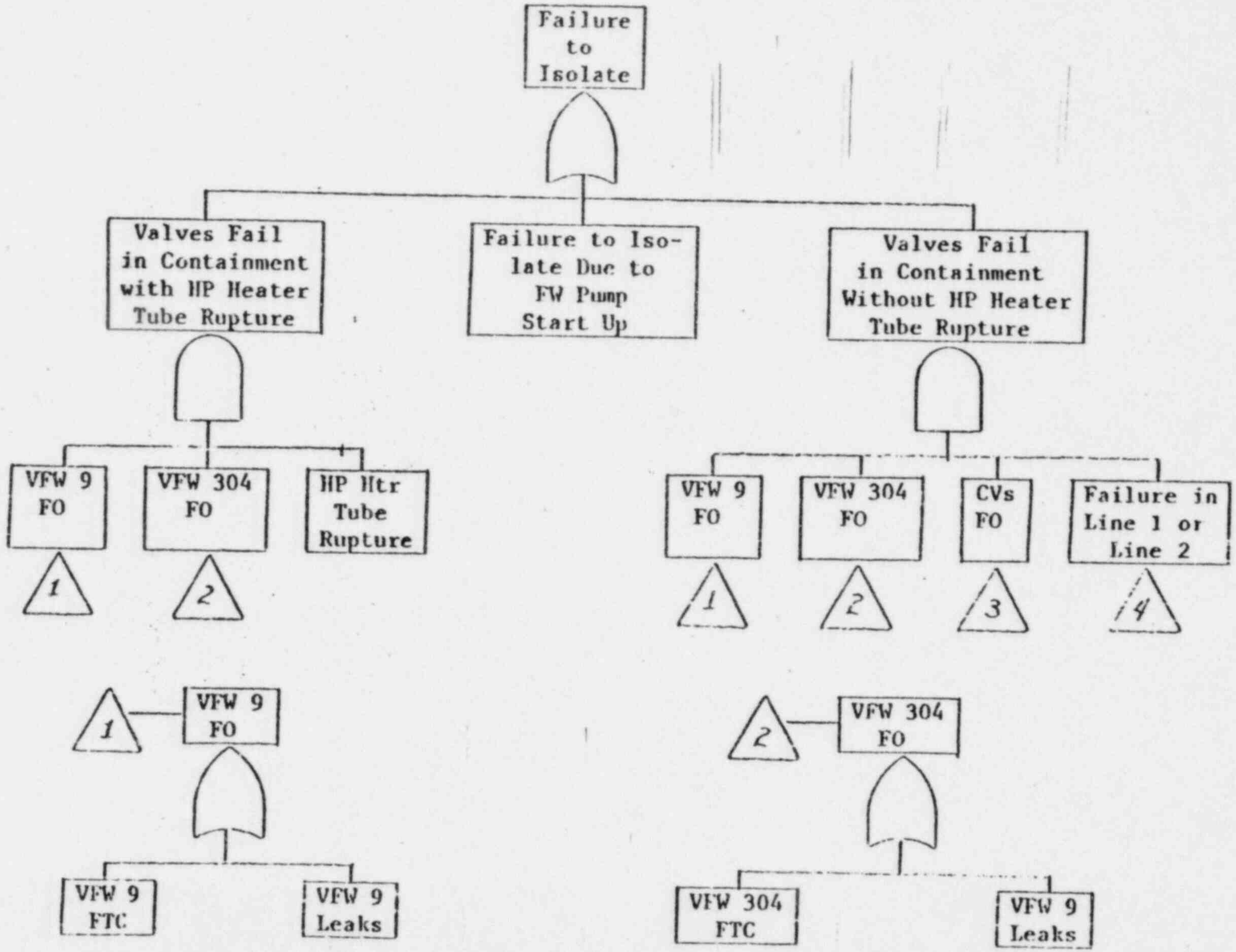

6
MO 7067
Line Not
Isolated
MO 7067
Does Not Re-
Main Closed
CV 4014
Does Not Re-
Main Closed
MO 7067
FTRC
MO 7067
FTC
MO 7067
Leaks
MO 7067
Electrical
Failure
CV 4014
Fails
Open
Hydraulic
Oil Pump
Loss of
Function
Cylinder
Energized
Valve Opens
CV 4014
FTC
CV 4014
Leaks
Mechanical
Failure
PS 635
Closed
FTOP
PS 678

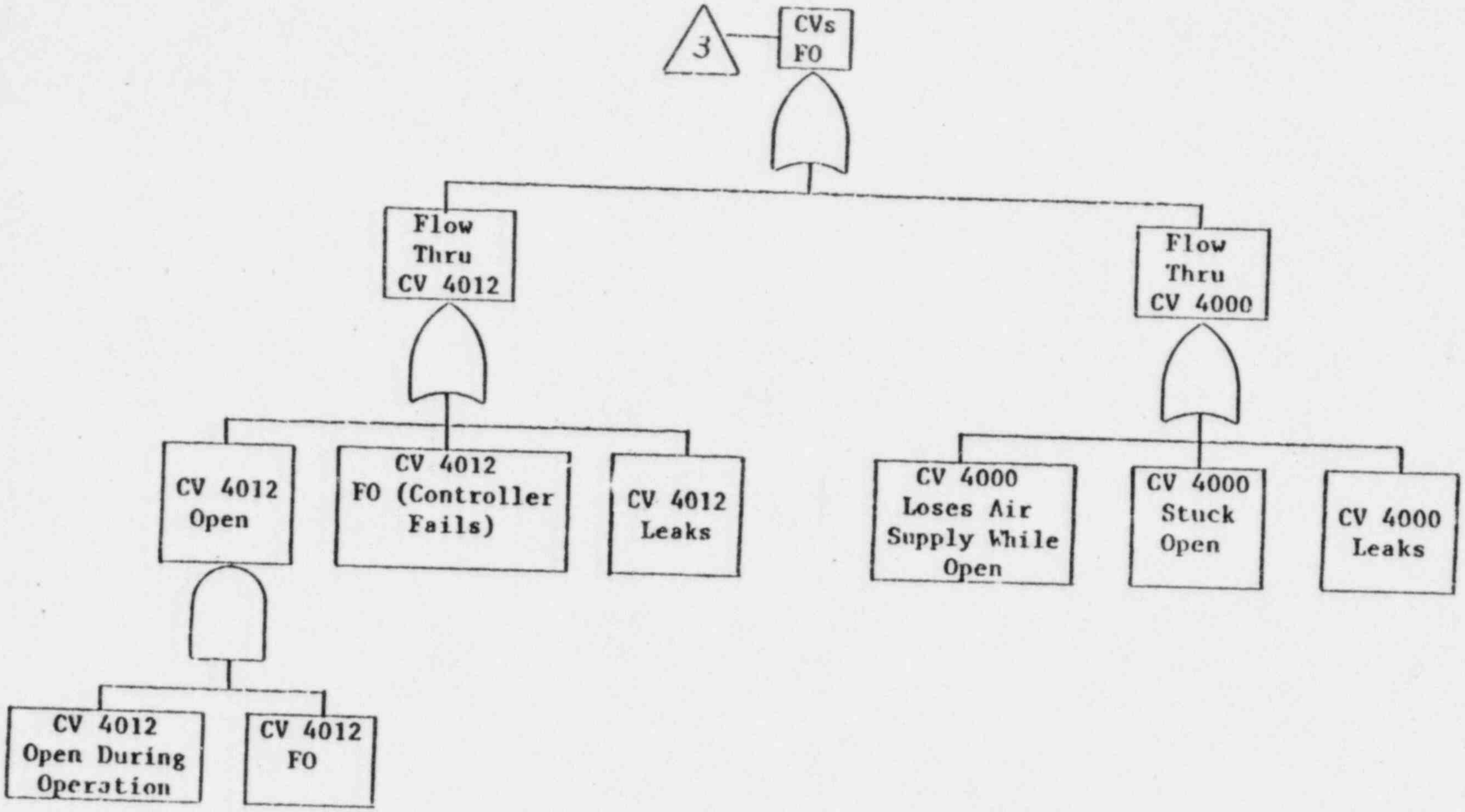
STEAMLINE

Assume probability of leakage of valves downstream of MSIV (MO 7050) = 1.0

MO 7050 FTC of 3.84×10^{-2} dominates tree, therefore, failure to isolate steamline = 3.84×10^{-2}

4. Feedwater:


Failure to Isolate
Valves Fail in Containment with HP Heater Tube Rupture
Failure to Isolate Due to FW Pump Start Up
Valves Fail in Containment Without HP Heater Tube Rupture
VFW 9 FO
VFW 304 FO
HP Htr Tube Rupture
VFW 9 FO
VFW 304 FO
CVs FO
Failure in Line 1 or Line 2
1
2
1
2
3
4
1
VFW 9 FO
VFW 9 FTC
VFW 9 Leaks
2
VFW 304 FO
VFW 304 FTC
VFW 9 Leaks



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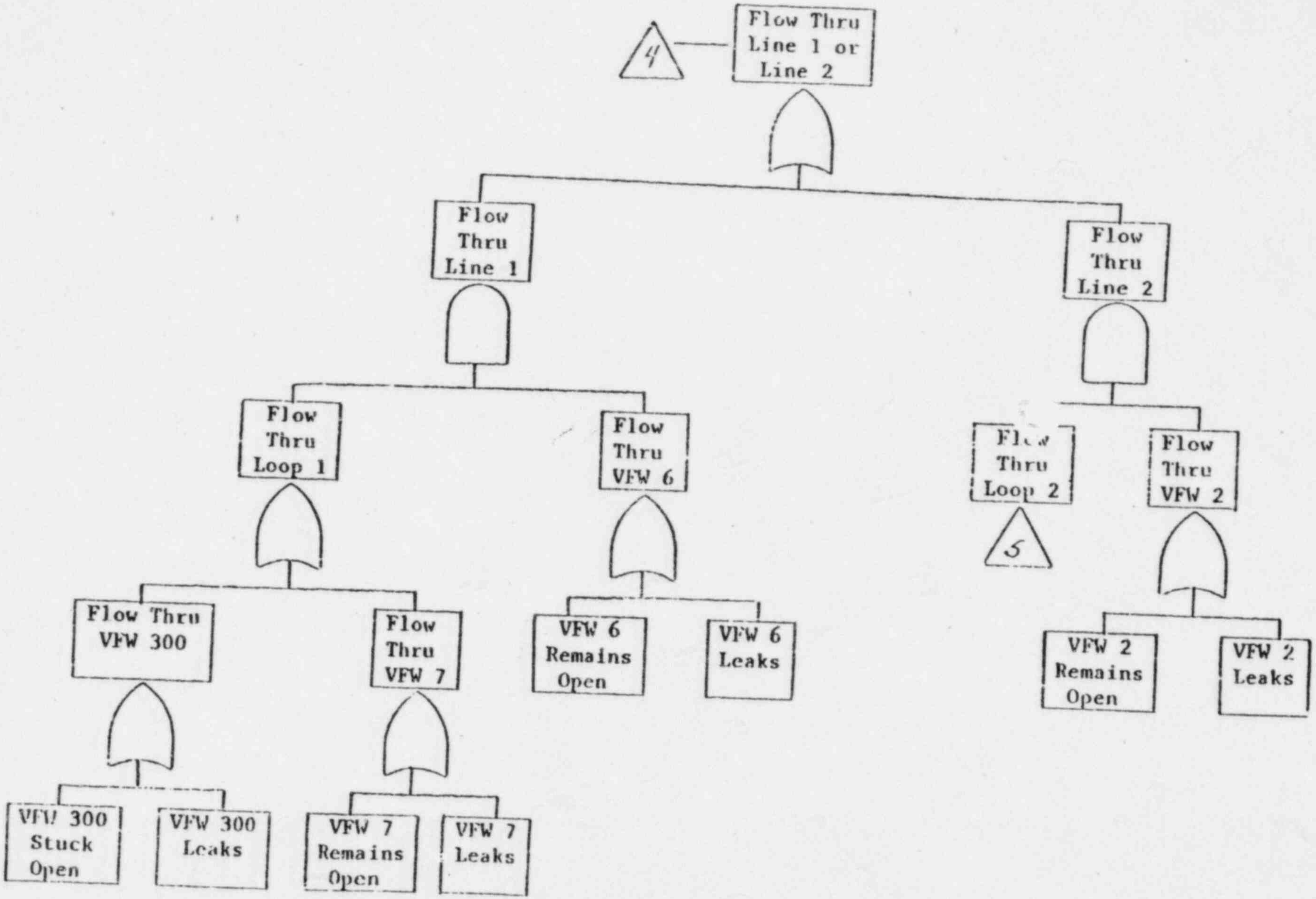

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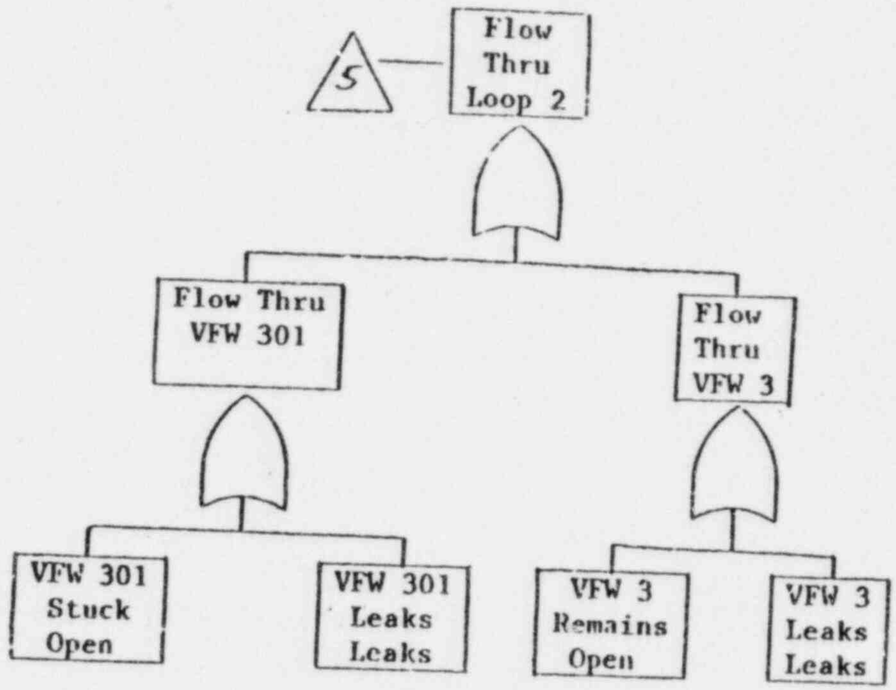

5
Flow Thru Loop 2
Flow Thru VFW 301
Flow Thru VFW 3
VFW 301 Stuck Open
VFW 301 Leaks Leaks
VFW 3 Remains Open
VFW 3 Leaks Leaks

FEEDWATER

Assume Check Valves VFW-300 and 301 are replaced and that generic leakage and failure to close probabilities are applicable.

Check valve leakage probability = $.5/10^6$ hr (Table III-4a of PRA Report)

$$\text{VFW-300 and 301 Leakage Probability} = .5/10^6 \text{ (Test Interval)}$$
$$= .5/10^6 \text{ (24·365)}$$
$$= 4.38 \times 10^{-3}$$

VFW-300 and 301 FTC = 5×10^{-4} (Table III-4a)

$$\text{Failure to Isolate FW Line} = P(\text{flow through VFW-300}) + P(\text{flow through VFW-301})$$
$$= (4.38 \times 10^{-3} + 5 \times 10^{-4}) + (4.38 \times 10^{-3} + 5 \times 10^{-4})$$
$$= 9.8 \times 10^{-3}$$

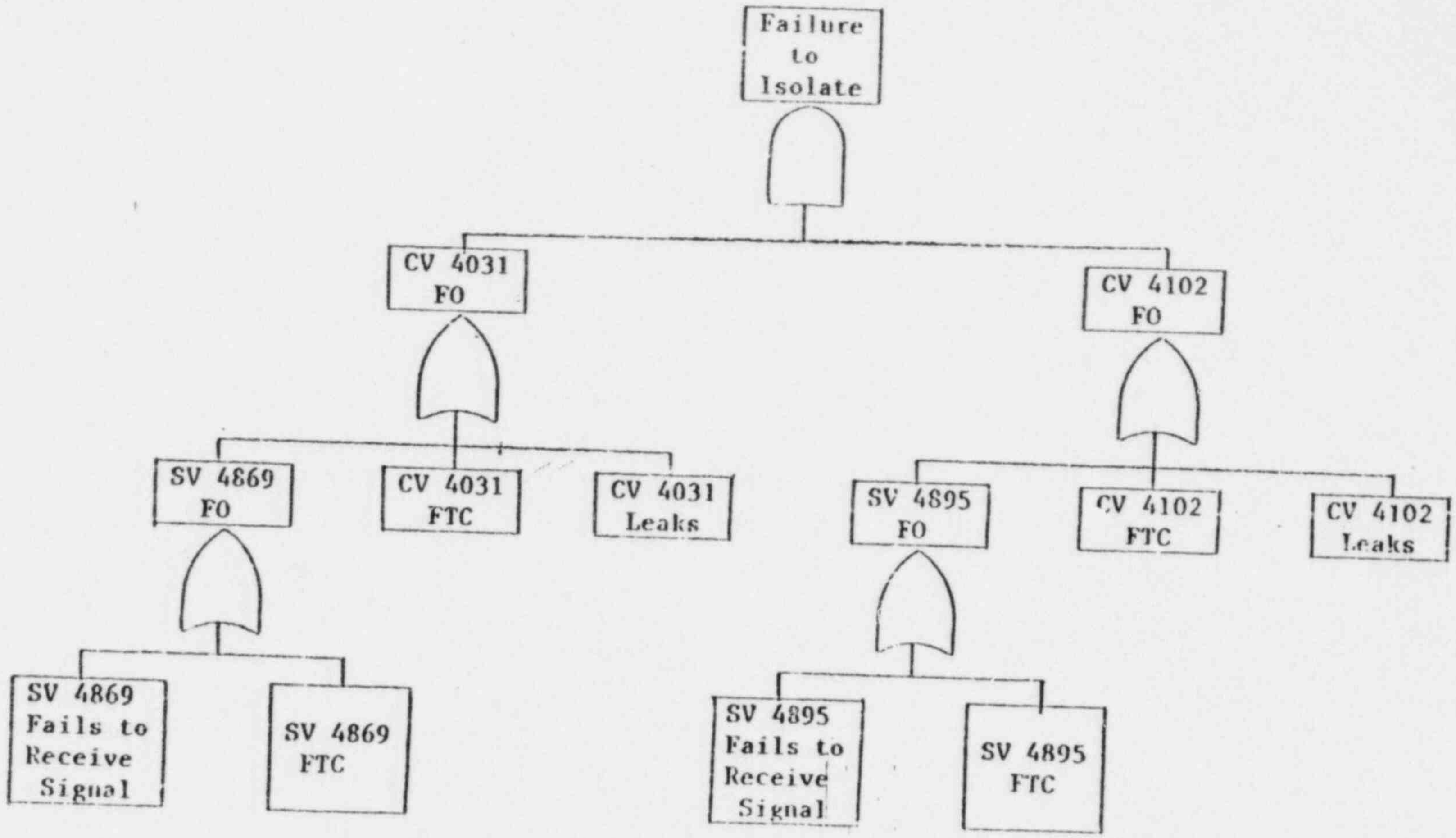
Start-up of FW pumps requires use of heat up loops which cause loss of containment isolation. Heat up during a pump start is assumed to take ~4 hr (see SOP-16). Number of shutdowns and FW pump outages requiring use of heat up loop during start-up from 1970-1979 is 79 (PRA Table III-2 and XIII-1).

Time containment was not isolated = 4(79) = 316 hours
Total time 1970-1979 = 87648 hours
Probability containment is not isolated due to FW pump start-up = $316/87648 = 3.6 \times 10^{-3}$

Total probability of failure to isolate containment due to flow through feedwater lines = $9.8 \times 10^{-3} + 3.6 \times 10^{-3}$
$= 1.34 \times 10^{-2}$

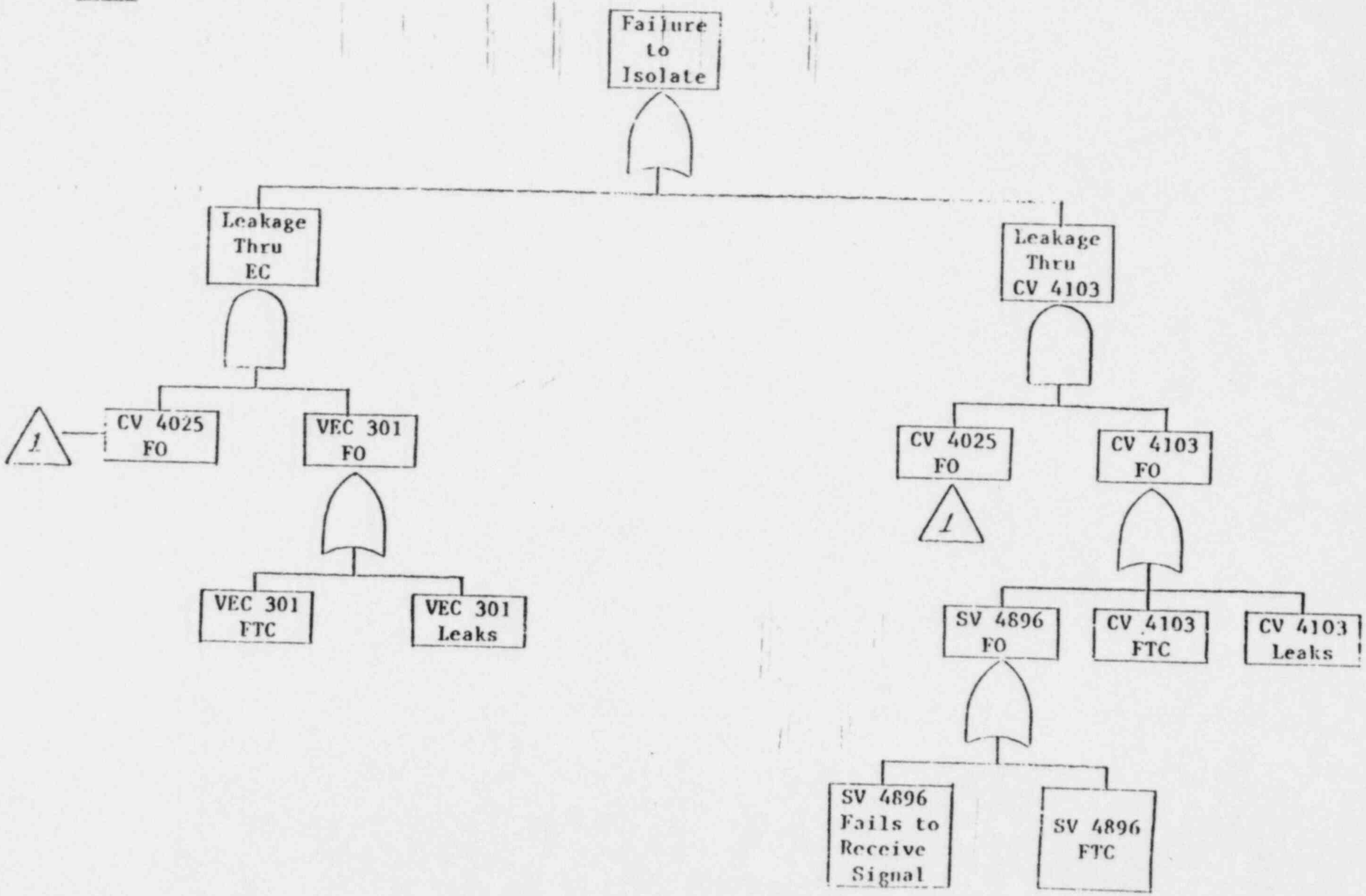
5. A. SUMP:

Clean



B. SUMP:

Dirty



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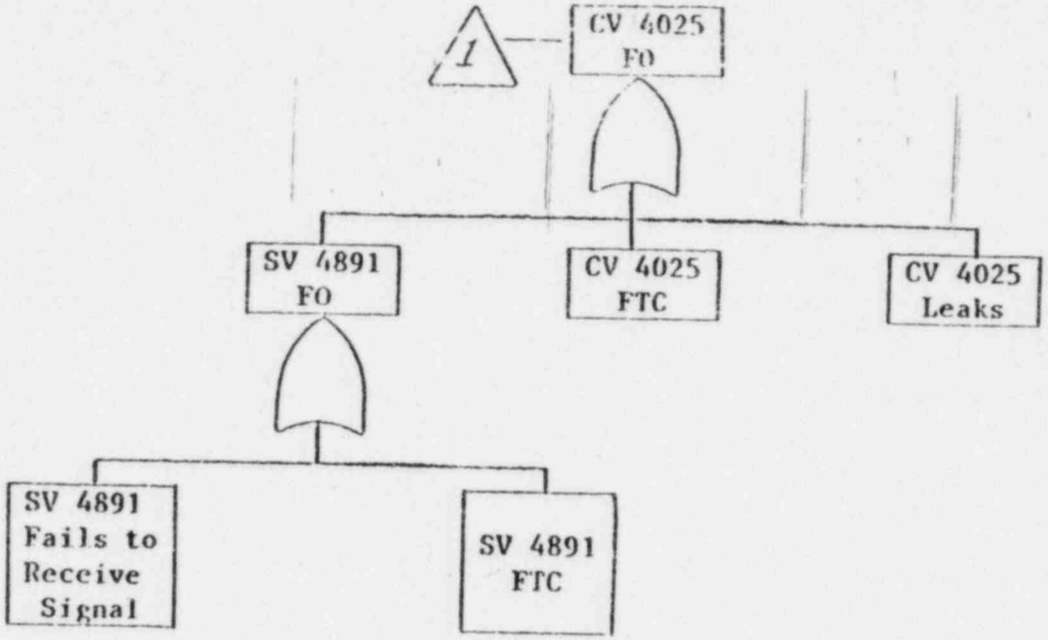

1
CV 4025
FO
SV 4891
FO
CV 4025
FTC
CV 4025
Leaks
SV 4891
Fails to
Receive
Signal
SV 4891
FTC

TABLE IV.4.2.5.a

Clean Sump

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope $\left(\frac{\text{Lb/24 Hr}}{\text{Day}}\right)$
8/ 2/77	0.0		
10/ 4/78	2.35	428	.0055
2/25/79	0.0	144	-.0163
10/15/79	1.1162	232	.0048
12/ 2/80	.447	414	-.0016

Total Time Period = 1218 Days

Avg Slope = .000367

Assume Leakage Limit = 1/2 TS
Limit = 223.5 Lb/24 hr

Time to Failure =

$$\frac{223.5}{.000367} = 608991.83$$

\# Testing Periods =

$$\frac{608991.83}{365}$$

= 1668.4708

Leakage Probability =

$$\frac{1669-1668.4708}{1669} =$$

3.171×10^{-4}

TABLE IV.4.2.5.b

Dirty Sump

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope $\left(\frac{\text{Lb/24 Hr}}{\text{Day}}\right)$
8/ 2/77	.28		
10/ 4/78	11.8	428	.0269
2/25/79	3.52	144	-.0575
10/15/79	1.3425	232	-.0094
11/25/80	3.0	407	.0041

Total Time Period = 1211 Days

Avg Slope = .00225

Assume Leakage Limit = 1/2 TS
Limit = 223.5 Lb/24 hr

Time to Failure =
(223.5-.28)   .00225
= 99208.889 Days

Testing Periods =
$\frac{99208.889}{365}$

= 271.805

Leakage Probability =
$\frac{(272-271.805)}{272}$ =

7.163×10^{-4}

SUMPS

Clean Sump:

CV 4031 - 3.171×10^{-4} Leakage
CV 4102 - 3.171×10^{-4} Probability

CV 4031 - .001 FTC
CV 4102 - .001 (App III of PRA Report)

SV 4869 - .001 FTC
SV 4895 - .001 (App III of PRA Report)

Flow through CV 4031: $3.171 \times 10^{-4} + .9(1.0 \times 10^{-3}) + 1 \times 10^{-3} = 2.217 \times 10^{-3}$

Flow through CV 4102: $3.171 \times 10^{-4} + .9(1.0 \times 10^{-3}) + 1.0 \times 10^{-3} = 2.217 \times 10^{-3}$

Failure to isolate = (flow through CV 4031)(flow through CV 4102)
$= (2.217 \times 10^{-3})^2 + .1(1 \times 10^{-3})$
$= 1.049 \times 10^{-4}$

Dirty Sump:

CV 4025, CV 4103, VEC 301 - 7.163×10^{-4} Leakage Probability

CV 4025, CV 4103 - 1.0×10^{-3} - FTC (App III, PRA Report)

VEC 301 - 5×10^{-4} - FTC (App III, PRA Report)

SV 4891, SV 4896 - 1×10^{-3} - FTC (App III, PRA Report)

Flow through emergency condensor =
(Flow through VEC 301)(Flow through CV 4025) =
$(7.163 \times 10^{-4} + .9(5 \times 10^{-4}))(.9(1 \times 10^{-3}) + (1 \times 10^{-3}) + (7.163 \times 10^{-4})) + .1(1 \times 10^{-3}) = 1.031 \times 10^{-4}$

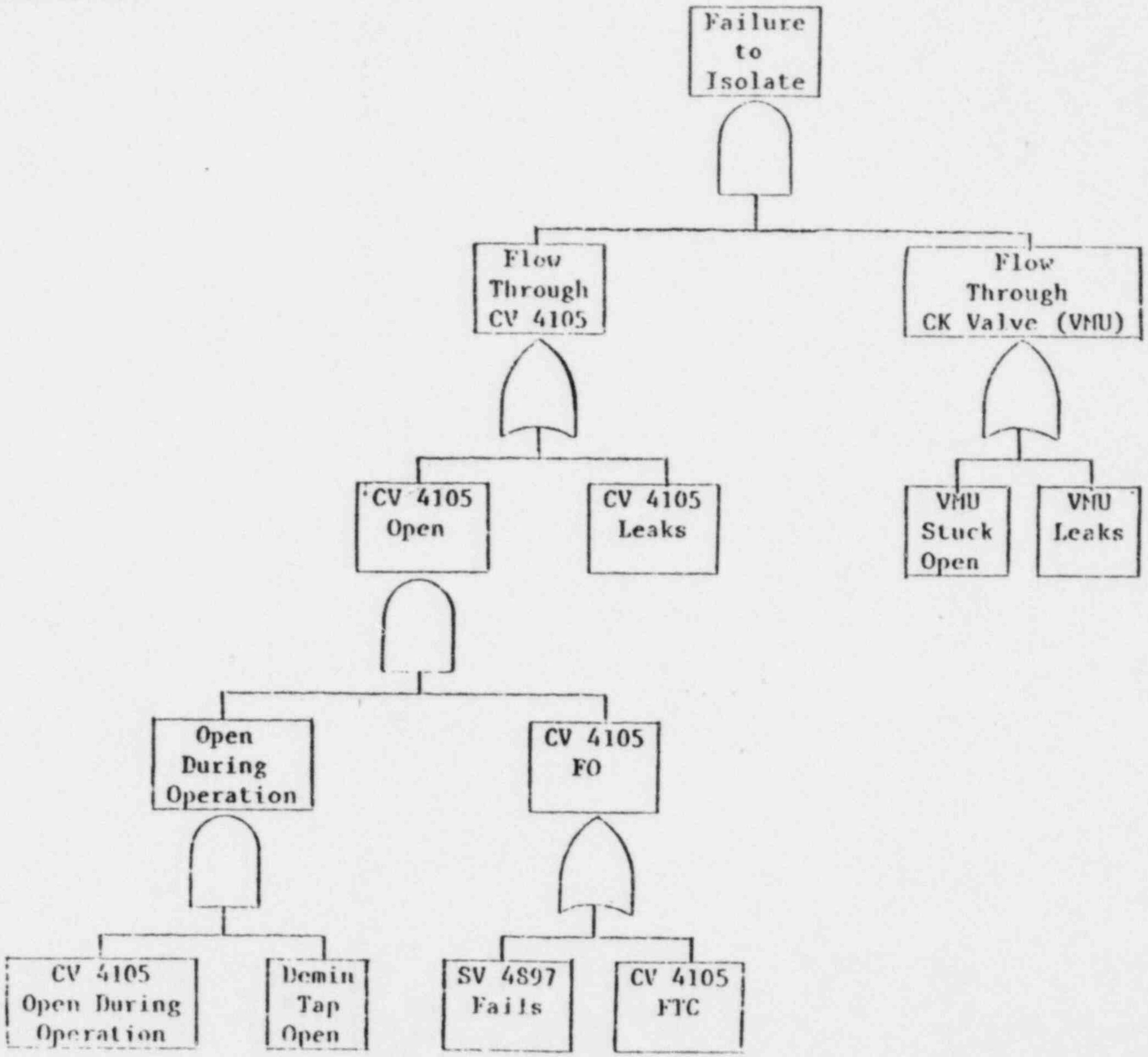
Flow through CV 4103 =
(Flow through CV 4025)(Flow through CV 4103) =
$(.9(1 \times 10^{-3}) + 1 \times 10^{-3} + 7.163 \times 10^{-4})(.9(1 \times 10^{-3}) + 1 \times 10^{-3} + 7.163 \times 10^{-4}) + .1(1 \times 10^{-3}) = 1.068 \times 10^{-4}$

Failure to Isolate:
(Flow through emergency condenser) + (flow through CV 4103) =
$(1.031 \times 10^{-4}) + (1.068 \times 10^{-4}) = 2.1 \times 10^{-4}$

Failure to isolate (clean and dirty sump) =

$(1.049 \times 10^{-4}) + (2.1 \times 10^{-4}) = 3.15 \times 10^{-4}$

B. DEMIN WATER SUPPLY:



7. Treated Waste:

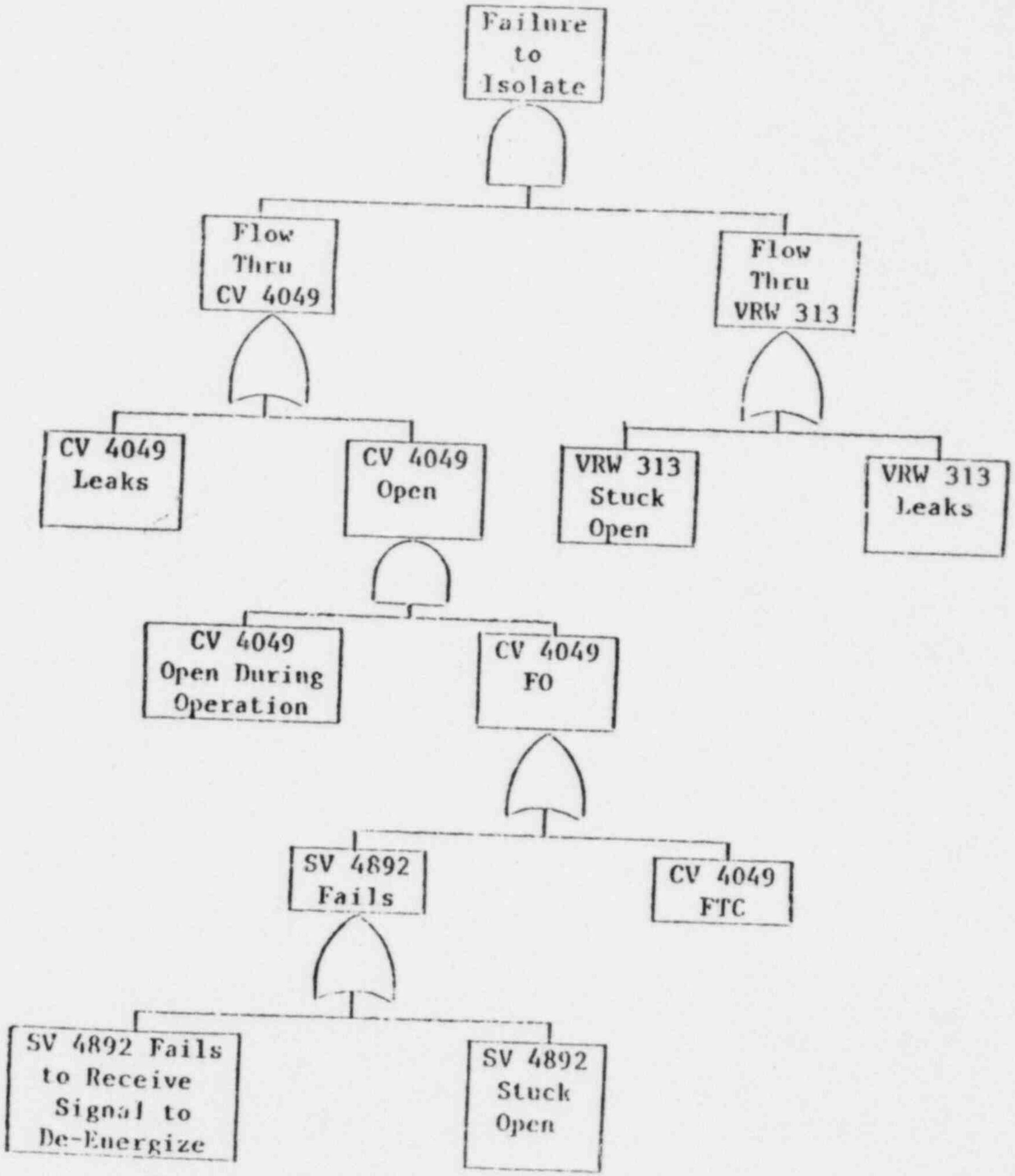


TABLE IV.4.2.6.a

Demin Water Supply

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope $\left(\frac{\text{Lb/24 Hr}}{\text{Day}}\right)$
8/14/77	2.52		
3/ 5/79	1.001	568	-.0027
11/27/80	2.07	633	.0017
		Total Time Period = 1201 Days	Avg Slope <0.0
			Assume Leakage Probability $\approx$0.0

TABLE IV.4.2.7.a

Treated Waste (CV 4049)

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope $\left(\frac{Lb/24\ Hr}{Day}\right)$
8/11/77	.001		
10/26/78	0.0	441	-0.0
3/ 1/79	0.0	126	0.0
3/ 1/80	.72	366	.00197

Total Time Period = 933 Days

Avg Slope = .000772

Leakage Limit per Tech Specs $\approx 447\ \frac{Lb}{24\ Hr}$

Time to Failure = $\frac{447}{.000772}$ = 579240.64 Days

Testing Periods = $\frac{579240.64}{365}$

= 1586.9607

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope ($\frac{Lb/24\ Hr}{Day}$)
			Leakage Probability = $\frac{1587-1586.9607}{1587}$ = 2.48×10^{-5}

Check Valve (VRW 313) failed leak test 11/27/80. Assume valve had leaked for the entire period between tests (3/1/80-11/27/80).

$$\text{Leakage Probability (VRW 313)} = \frac{\text{Leakage Time}}{\text{Total Time}}$$

$$= \frac{271\ \text{Days}}{1204\ \text{Days}}$$

$$= 2.25 \times 10^{-1}$$

DEMIN WATER SUPPLY

CV 4105 Leakage Probability ≈0.0
Check Valve Leakage Probability ≈1.0
CV 4105 FTC = 1.0×10^{-3} (Table 3.5, App III)
SV 4897 FTC = 1.0×10^{-3} (Table 3.4, App III)
Demin tap open during operation = 15 x 1 yr for 2 min each
= .5 hr/24 hr (365)
= 5.71×10^{-5}
CV 4105 open during operation = 1.0

CV 4105 Open:
[CV 4105 open during oper)(demin tap open during oper)]·[(SV 4897 FTC) + (CV 4105 FTC)]
= $(1.0)(5.71 \times 10^{-5})\cdot[(1.0 \times 10^{-3}) + (1.0 \times 10^{-3})]$
= 1.142×10^{-7}

Flow through CV 4105:
(CV 4105 Open) + (CV 4105 Leaks)
$(1.142 \times 10^{-7}) + (\sim 0.0) = 1.142 \times 10^{-7}$

Failure to isolate:
(Flow through CV 4105)(flow through check valve)
$(1.142 \times 10^{-7})\ (1.0) = 1.142 \times 10^{-7}$

TREATED WASTE

CV 4049 Leakage Probability - 2.48×10^{-5}
VRW 313 Leakage Probability - 2.25×10^{-1}
CV 4049 FTC - 1.0×10^{-3} (Table III.5, App III)
VRW 313 FTC - 5×10^{-4} (Table III-4a, App III)
SV 4892 FTC - 1×10^{-3} (Table III-4a)
CV 4049 open during operation - open 6 wks/year = 42 days/365 days
= .115

CV 4049 fails open:
(SV 4892 fails) + (CV 4049 FTC)
$(1 \times 10^{-3}) + (1 \times 10^{-3}) = 2 \times 10^{-3}$

CV 4049 open:
(CV 4049 open during oper)(CV 4049 fails open)
$(.115)(.002) = 2.3 \times 10^{-4}$

Flow through CV 4049:
(CV 4049 Leaks) + (CV 4049 open)
$2.48 \times 10^{-5} + 2.3 \times 10^{-4} = 2.548 \times 10^{-4}$

Flow through VRW 313:
(VRW 313 leaks) + (VRW FTC)
$2.25 \times 10^{-1} + 5 \times 10^{-4} = 2.255 \times 10^{-1}$

Failure to isolate:
(Flow through CV 4049)(flow through VRW 313)
$(2.548 \times 10^{-4})(2.255 \times 10^{-1}) = 5.75 \times 10^{-5}$

8. Fuel Pit Drain:

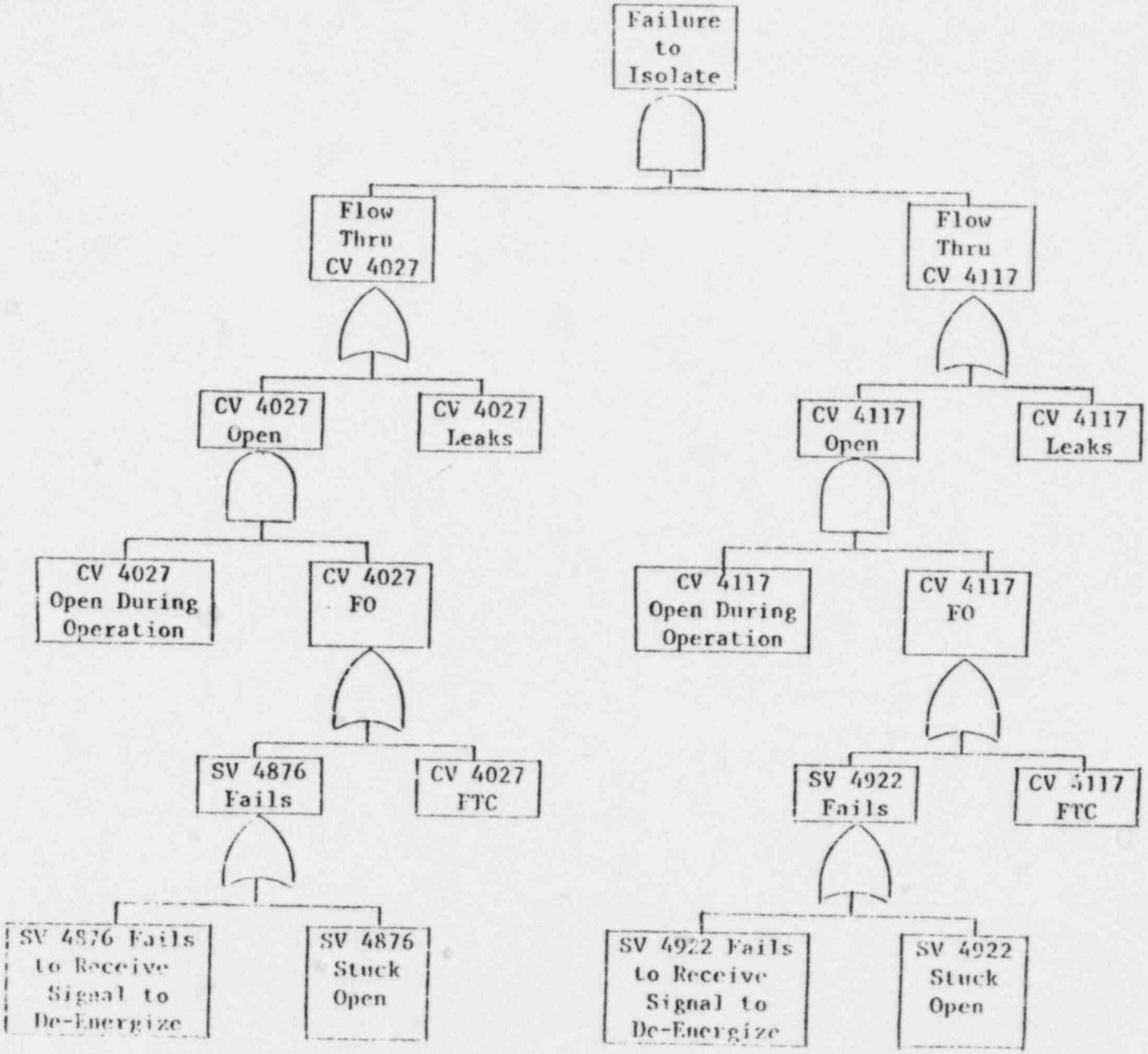


TABLE IV.4.2.8.a

Fuel Pit Drain

Testing Date	Leak Rate Lb/24 Hr	Testing Interval (Days)	Slope $\left(\frac{\text{Lb/24 Hr}}{\text{Day}}\right)$
8/21/77	11.77		
3/ 3/79	10.923	568	-.00149
9/19/79	0.0	200	-.0546
11/26/80	17.1	431	.0397

Total Time Interval = 1199 Days

Avg Slope = .00445

Leak Limit ≈ 447 lb/24 hr (TS Limit)

Time to Failure = $\frac{447-11.7}{.00445}$ = 97905.7 Days

Testing Periods = $\frac{97905.7}{365}$

= 268.2348

Leakage Probability = $\frac{269-268.2348}{269}$

$= 2.8446 \times 10^{-3}$

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FUEL PIT DRAIN

CV 4027, CV 4117 - 2.8446×10^{-3} Leakage Probability

CV 4027, CV 4117 - 1.0×10^{-3} - FTC (App III, PRA Report)

CV 4027, CV 4117 open during operation ~ 6 wk/yr = 42 days/365 days = 1.15×10^{-1}

SV 4876, SV 4922 - 1.0×10^{-3} FTC (App III, PRA Report)

Probability of CV 4027 failing open:
(SV 4876 FTC) + (CV 4027 FTC) = $1.10^{-3} + .9(1.0 \times 10^{-3})$
= 1.9×10^{-3}

Probability of CV 4117 failing open:
(SV 4922 FTC) + (CV 4117 FTC) = $1 \times 10^{-3} + .9(1 \times 10^{-3})$
= 1.9×10^{-3}

Probability of CV 4027 being open:
(CV 4027 open during operation)(CV 4027 FO) = $.115\ [(1.9 \times 10^{-3})]$
= 2.185×10^{-4}

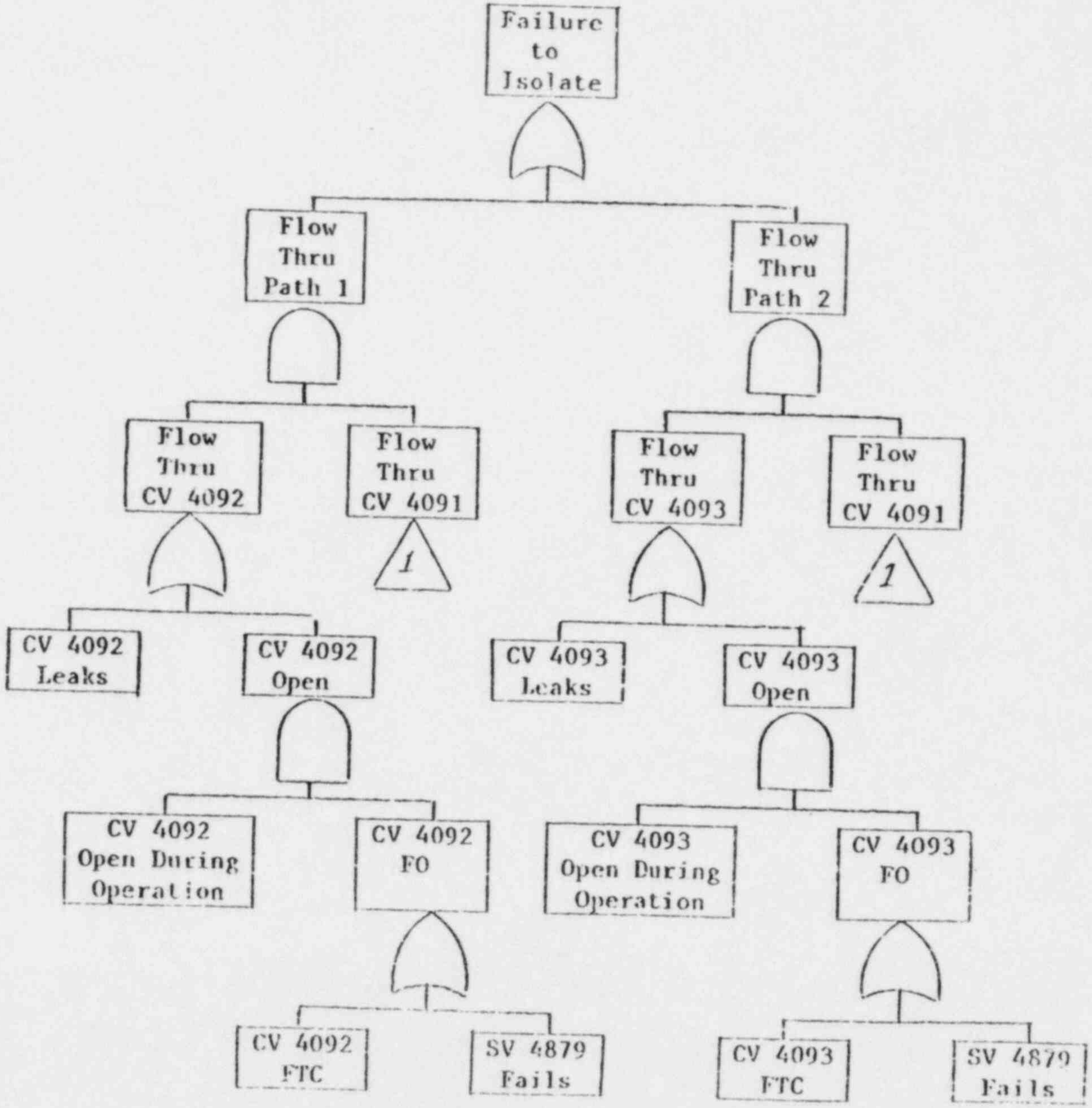
Probability of CV 4117 being open:
(CV 4117 open during operation)(CV 4117 FO) = $.115\ (1.9 \times 10^{-3})$
= 2.185×10^{-4}

Flow through CV 4027:
(CV 4027 open) + (CV 4027 leaks) = $2.185 \times 10^{-4} + 2.8446 \times 10^{-3}$
= 3.0631×10^{-3}

Flow through CV 4117:
(CV 4117 open) + (CV 4117 leaks) = $2.185 \times 10^{-4} + 2.8446 \times 10^{-3}$
= 3.0631×10^{-3}

Failure to isolate:
(Flow through CV 4027)(flow through CV 4117) = $(3.0631 \times 10^{-3})^2 + .1(1 \times 10^{-3})$
= 1.094×10^{-4}

9. Resin Sluice:



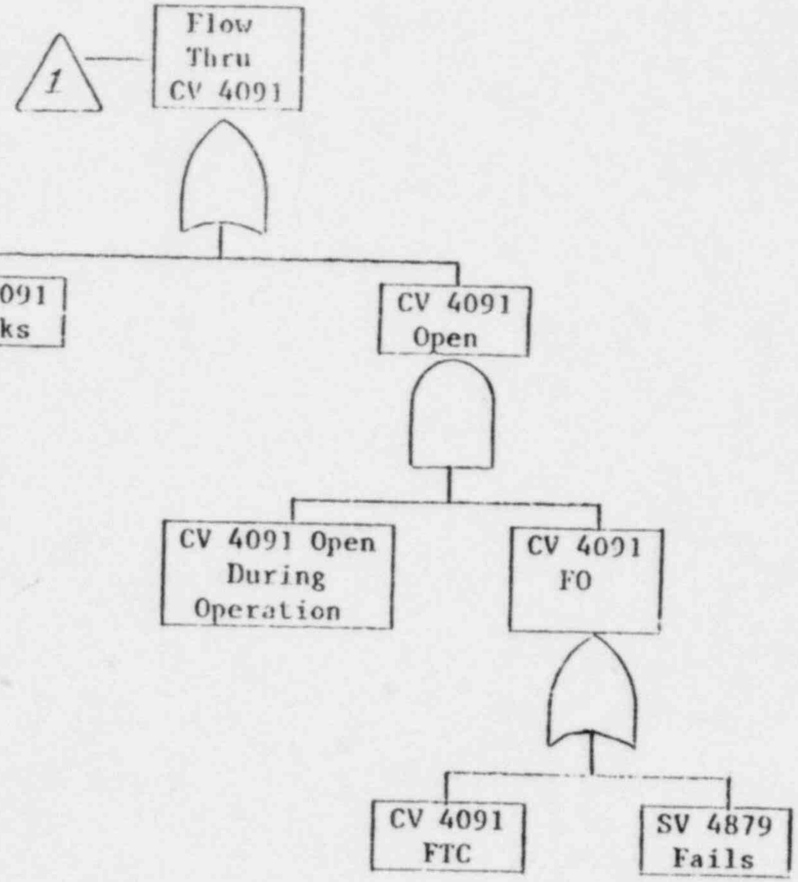


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TABLE IV.4.2.9.a

Resin Sluice (CV 4091)

Testing Date	Leak Rate 16/24 Hr	Testing Interval (Days)	Slope ($\frac{Lb/24\ Hr}{Day}$)
8/12/77	1.42		
3/ 8/79	1.11	573	-.000541
10/19/79	1.5358	225	.00189
11/26/80	.45	404	-.00269

Total Time Period = 1202 Days

Avg Slope = <0.0

Assume Leakage Probability ≈ 0.0

CV 4092, 4093
8/12/77- 3/ 8/79 Leaked
3/ 8/79-10/19/79 No Leakage
10/19/79-11/23/80 Leaked

$$\frac{\text{Leakage Time}}{\text{Total Time}} = \frac{974}{1199}$$

$$= .812$$

Leakage Probability = .812

RESIN SLUICE

CV 4091 Leak Probability ≈ 0.0
CV 4092, 4093 Leakage Probability = .812
CV 4091, 4092, 4093 FTC = 8.37×10^{-2} (App III)
CV 4091, 4092, 4093 open during operation = 3.425×10^{-4}
SV 4879 FTC ≈ 1.0

Flow through CV 4091:
(CV 4091 leaks) + (CV 4091 open during operation)(CV 4091 FO)
$\sim 0 + (3.425 \times 10^{-4})(\sim 1.0) = 3.425 \times 10^{-4}$

Flow through CV 4092:
(CV 4092 leaks) + (CV 4092 open during operation)(CV 4092 FO) ≈ 1.0

Flow through CV 4093:
(CV 4093 leaks) + (CV 4093 open during operation)(CV 4093 FO) ≈ 1.0

Failure to isolate:
(Flow through CV 4092 or CV 4093)(flow through CV 4091)
$(1.0)(3.425 \times 10^{-4}) = 3.425 \times 10^{-4}$ p

NOTE (a): For sequences in which the MSIV has closed as part of the sequence, the probability that the sequence will lead to releases is equal to the sum of the failure to isolate probabilities listed in the accompanying table excluding the probability that the steam line fails to isolate.

NOTE (b): For sequences in which the MSIV has not closed during the sequence because it has not been called upon to close, rather than as a result of inability to close caused by mechanical problems, the probability that the sequence will lead to releases is equal to the sum of the failure to isolate probabilities listed in the accompanying table.

NOTE (c): For LOSP sequences in which offsite power is restored before core damage occurs, instrument air is assumed available and instrument air lines are assumed not to be a leakage path. The failure to isolate containment probability was determined as described in footnote (a).

NOTE (d): For LOSP sequences in which core damage occurs before offsite power is restored but the diesel generator is working, it is assumed that the operator fails to restore instrument air 10% of the time making instrument and service air lines a leakage path (F_1 = .10). This leakage probability was added to the value of the failure to isolate probability determined in footnote (a).

NOTE (e): For LOSP sequences in which core damage occurs before offsite power is restored and the diesel generator fails, it is assumed the standby diesel is not put in service 10% of the time and given that it is, the air compressors are not restored 10% of the time (f_1 = .1 + .9 (.1) = .19). This leakage probability was added to the value of the failure to isolate probability determined in footnote (a).

NOTE (f): For sequences in which the MSIV fails to close on demand as part of the sequence, the probability that the sequence will lead to release is equal to the probability that the backup isolation valves fail to close on demand (Probability = 1.0).

NOTE (g): For BRP as presently designed, all accident sequences which lead to RDS actuation are also expected to produce significant core damage and containment failure. The primary factor in this expectation is that following RDS actuation, no assurance can be provided that the liquid poison will mix with the core spray water even if it is injected prior to RDS actuation. Estimates of the radio-nuclide releases from containment have been made by assigning ATWS sequences to Release Category 3, for which core damage and early containment failure are key characteristics.

NOTE (h): As described in Section III, 5.2.12, Appendix III of the PRA Report, the probability that the instrument air system is not repaired in a timely manner is .01. This probability was added to the value of the failure to isolate probability as determined in footnote (b).

NOTE (i): For sequences involving fires in the cable penetration area inside containment which are severe enough to cause cable damage leading to core melt, it is judged that the probability the fire will also cause containment isolation failure is 1.0. This failure could occur either by combustion or melting of the epoxy which seals the cables to the containment penetration, or by fire-caused failure of the MSIV to close.

For sequences involving fires in the cable spreading room outside containment which are severe enough to cause cable damage leading to core melt, it is judged that the probability of containment isolation failure is 1.0. Despite the different density of cables in the vicinity of the penetrations, a fire in the cable spreading room would also very likely disable the MSIV open. Given the base case assumption that the backup isolation valves do not close completely, open failure of the MSIV is equivalent to containment isolation failure.

For sequences involving fires in the station power room, the operator is instructed to shut down and close the MSIV. Because of some ambiguities in the procedures, it is not obvious that these actions would be implemented immediately for the Plant as presently operated. For this reason, station power room fires of sufficient severity to cause cable failures and eventual core melt are judged to cause failure of the operators ability to close the MSIV in advance of his taking this action one time in three. However, because instrument air lines are assumed to be a leakage path for this sequence F_ℓ - 1.0.

NOTE (j): As described in Section III, 5.2.12, Appendix III of the PRA Report, the probability that the instrument air system is not repaired in a timely manner is .01. This probability was added to the value of the failure to isolate probability as determined in footnote (a).

FAILURE TO ISOLATE CONTAINMENT PROBABILITIES

Locks	1.95×10^{-4}
Vents	1.142×10^{-2}
Steam Line	3.84×10^{-2}
Feedwater	1.34×10^{-2}*
Sumps	3.15×10^{-4}
Demin Water	1.142×10^{-7}
Treated Waste	5.75×10^{-5}
Fuel Pit Drain	1.1×10^{-4}
Resin Sluice	3.425×10^{-4}
	6.42×10^{-2}

*The feedwater failure to isolate probability was determined by assuming that Valves VFW6 and VFW2 are made motor operated and generic failure data is applicable.

V.6 REFERENCES

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2. CONTEMPT-LT, A Computer Program for Predicting Containment Pressure-Temperature Response to a Loss-of-Coolant Accident, ANCR-1219, Aerojet Nuclear Co (June 1975)

3. BOIL - See Appendix A of Appendix VIII of WASH-1400

4. INTER: A Preliminary Model for Core/Concrete Interactions, by W B Murfin, SAND77-0370 (August 1977)

5. PVMELT - A Model for Pressure Vessel Melt-Through, Section III of RACAP-1 (Volume II) (Reactor Accident Consequence Analysis Program), DRAFT Version, SAI (July 1980)

6. The Effects on Populations of Exposure to Low Levels of Ionizing Radiation, Report of the Advisory Committee on the Biological Effects of Ionization Radiations, BEIR Report, National Academy of Science - National Research Council, Washington, DC (1972)

7. "Some Observations on Near-Limit Flames," by Furno, A L, et al, Thirteenth International Symposium on Combustion, pp 593-599 (1971)

Summary of Important Accident Sequences for Big Rock Point

Sequence	Probability of Sequence (Per Year)	Probability of Release Category: (1) Given the Sequence, (2) Considering the Sequence Probability									
		BRP-1(1)	BRP-1(2)	BRP-2(1)	BRP-2(2)	BRP-3(1)	BRP-3(2)	BRP-4(1)	BRP-4(2)	BRP-5(1)	BRP-5(2)
S_1E_mC	4.0×10^{-6}					.026 (a)	1.04×10^{-7}	0.01 (b)	4.0×10^{-8}	0.79 (c)	3.2×10^{-6}
S_2L	3.7×10^{-6}					.0642 (b)	2.38×10^{-7}	0.01 (b)	3.7×10^{-8}	0.76 (c)	2.8×10^{-6}
S_2C	4.0×10^{-7}					.0642 (b)	3.57×10^{-8}	0.01 (b)	4.0×10^{-9}	0.76 (c)	3.0×10^{-7}
S_7L	3.7×10^{-7}					.0642 (b)	2.38×10^{-8}	0.01 (b)	3.7×10^{-9}	0.76 (c)	2.8×10^{-7}
S_3E_mL	1.0×10^{-5}					.026 (a)	2.6×10^{-7}	0.01 (b)	1.0×10^{-7}	0.79 (c)	7.9×10^{-6}
S_3E_mC	4.0×10^{-6}					.026 (a)	1.04×10^{-7}	0.01 (b)	4.0×10^{-8}	0.79 (c)	3.2×10^{-6}
S_4C	1.0×10^{-4}					.0642 (b)	6.42×10^{-6}	0.01 (b)	1.0×10^{-6}	0.76 (c)	7.6×10^{-5}
UL	1.5×10^{-7}					.0742 (h)	1.11×10^{-8}	0.01 (b)	1.5×10^{-9}	0.76 (c)	1.1×10^{-7}
UE_vUL	1.7×10^{-6}					.0742 (h)	1.26×10^{-7}	0.01 (b)	1.7×10^{-8}	0.76 (c)	1.3×10^{-6}
UE_mUL	1.9×10^{-5}					.0742 (h)	1.41×10^{-6}	0.01 (b)	1.9×10^{-7}	0.76 (c)	1.4×10^{-5}
UE_vUC	6.7×10^{-7}					.0742 (h)	4.97×10^{-8}	0.01 (b)	6.7×10^{-9}	0.76 (c)	5.1×10^{-7}
UE_mUC	7.4×10^{-6}					.0742 (h)	5.49×10^{-7}	0.01 (b)	7.4×10^{-8}	0.76 (c)	5.6×10^{-6}
UE_mUJ	5.7×10^{-6}					.0742 (h)	4.23×10^{-7}	0.01 (b)	5.7×10^{-8}	0.76 (c)	4.3×10^{-6}
WE_vL	1.7×10^{-6}					.026 (a)	4.42×10^{-8}	0.01 (b)	1.7×10^{-8}	0.79 (c)	1.3×10^{-6}
WE_mL	6.0×10^{-7}					.026 (a)	1.56×10^{-8}	0.01 (b)	6.0×10^{-9}	0.79 (c)	4.7×10^{-7}
WE_vC	6.7×10^{-7}					.026 (a)	1.74×10^{-8}	0.01 (b)	6.7×10^{-9}	0.79 (c)	5.3×10^{-7}
WE_mC	2.4×10^{-7}					.026 (a)	6.24×10^{-9}	0.01 (b)	2.4×10^{-9}	0.79 (c)	1.9×10^{-7}
BB_cE_vL	9.3×10^{-7}					.026 (a)	2.42×10^{-8}	0.01 (b)	9.3×10^{-9}	0.79 (c)	7.3×10^{-7}
BB_cE_mL	3.3×10^{-7}					.026 (a)	8.58×10^{-9}	0.01 (b)	3.3×10^{-9}	0.79 (c)	2.6×10^{-7}
BB_cE_vC	3.7×10^{-7}					.026 (a)	9.62×10^{-9}	0.01 (b)	3.7×10^{-9}	0.79 (c)	2.9×10^{-7}
BB_cZY_fL	4.9×10^{-5}					.0642 (b)	3.15×10^{-6}	0.01 (b)	4.9×10^{-7}	0.76 (c)	3.7×10^{-5}
BB_cZY_fC	2.0×10^{-5}					.0642 (b)	1.28×10^{-6}	0.01 (b)	2.0×10^{-7}	0.76 (c)	1.5×10^{-5}
$T_1AY_fL_r$	1.4×10^{-6}					.99 (g)	1.4×10^{-6}	0.01 (b)	1.4×10^{-8}		
$T_1AY_fOL_r$	4.2×10^{-7}					.99 (g)	4.2×10^{-7}	0.01 (b)	4.2×10^{-9}		
$T_1AB_oL_r$	8.8×10^{-7}					.99 (g)	8.8×10^{-7}	0.01 (b)	8.8×10^{-9}		
$T_2AB_oL_r$	3.2×10^{-6}					.99 (g)	3.2×10^{-6}	0.01 (b)	3.2×10^{-8}		

TABLE V.28

Summary of Important Accident Sequences for Big Rock Point

Sequence	Probability of Sequence (Per Year)	BRP-1(1)	BRP-1(2)	BRP-2(1)	BRP-2(2)	BRP-3(1)	BRP-3(2)	BRP-4(1)	BRP-4(2)	BRP-5(1)	BRP-5(2)
		Probability of Release Category: (1) Given the Sequence, (2) Considering the Sequence Probability									
TE_vNL	3.7×10^{-7}					.026 (a)	9.62×10^{-9}	0.01 (b)	3.7×10^{-9}	(c)	2.9×10^{-7}
TE_vNC	1.5×10^{-7}					.026 (a)	3.9×10^{-9}	0.01 (b)	1.5×10^{-9}	(c)	1.2×10^{-7}
TZL	3.8×10^{-7}					.0642 (b)	2.44×10^{-8}	0.01 (b)	3.8×10^{-9}	(c)	2.9×10^{-7}
Y_tL	4.3×10^{-7}					.026 (a)	1.12×10^{-8}	0.01 (b)	4.3×10^{-9}	(c)	3.4×10^{-7}
MNL	1.6×10^{-7}					.026 (a)	4.16×10^{-9}	0.01 (b)	1.6×10^{-9}	(c)	1.3×10^{-7}
ME_vNL	1.7×10^{-6}					.026 (a)	4.42×10^{-8}	0.01 (b)	1.7×10^{-8}	(c)	1.3×10^{-6}
ME_mNL	6.0×10^{-7}					.026 (a)	1.56×10^{-8}	0.01 (b)	6.0×10^{-9}	(c)	4.7×10^{-7}
ME_vNC	6.7×10^{-7}					.026 (a)	1.74×10^{-8}	0.01 (b)	6.7×10^{-9}	(c)	5.3×10^{-7}
ME_mNC	2.4×10^{-7}					.026 (a)	6.24×10^{-9}	0.01 (b)	2.4×10^{-9}	(c)	1.9×10^{-7}
ME_vNJ	6.0×10^{-7}					.026 (a)	1.56×10^{-8}	0.01 (b)	6.0×10^{-9}	(c)	4.7×10^{-7}
PE_vF_sL	3.6×10^{-6}					.026 (c)	9.36×10^{-8}	0.01 (b)	3.6×10^{-8}	(c)	2.8×10^{-6}
PE_mF_sL	1.4×10^{-5}					.026 (c)	3.64×10^{-7}	0.01 (b)	1.4×10^{-7}	(c)	1.1×10^{-5}
$PE_mF_sF_1L$	2.0×10^{-6}					.126 (d)	2.52×10^{-7}	0.01 (b)	2.0×10^{-8}	(c)	1.6×10^{-6}
PE_vF_sC	3.1×10^{-6}					.126 (d)	3.91×10^{-7}	0.01 (b)	3.1×10^{-8}	(c)	2.4×10^{-6}
PE_mF_sC	1.3×10^{-5}					.126 (d)	1.64×10^{-6}	0.01 (b)	1.3×10^{-7}	(c)	1.0×10^{-5}
PE_dF_sC	5.5×10^{-7}					.126 (d)	6.93×10^{-8}	0.01 (b)	5.5×10^{-9}	(c)	4.3×10^{-7}
PE_vF_sJ	1.3×10^{-6}					.126 (d)	1.64×10^{-7}	0.01 (b)	1.3×10^{-8}	(c)	1.0×10^{-6}
PE_mF_sJ	4.8×10^{-6}					.126 (d)	6.05×10^{-7}	0.01 (b)	4.8×10^{-8}	(c)	3.8×10^{-6}
PIF_sYL	9.9×10^{-7}					1.0 (f)	9.9×10^{-7}	0.01 (b)	9.9×10^{-9}		
PIF_sYC	8.5×10^{-7}					1.0 (f)	8.5×10^{-7}	0.01 (b)	8.5×10^{-9}		
PQE_mF_sL	6.7×10^{-7}					.026 (a)	1.74×10^{-8}	0.01 (b)	6.7×10^{-9}	(c)	5.3×10^{-7}
PQE_vF_sC	2.4×10^{-7}					.216 (e)	3.02×10^{-8}	0.01 (b)	2.4×10^{-9}	(c)	1.9×10^{-7}
PQE_mF_sC	2.5×10^{-6}					.216 (e)	3.15×10^{-7}	0.01 (b)	2.5×10^{-8}	(c)	2.0×10^{-6}
$PQIF_sL$	1.8×10^{-7}					1.0 (f)	1.8×10^{-7}	0.01 (b)	1.8×10^{-9}	(c)	
$PQIF_sC$	1.5×10^{-7}					1.0 (f)	1.5×10^{-7}	0.01 (b)	1.5×10^{-9}	(c)	
S_1E_mL	3.7×10^{-5}					.026 (a)	9.62×10^{-7}	0.01 (b)	3.7×10^{-7}	0.79 (c)	2.9×10^{-5}

Summary of Important Accident Sequences for Big Rock Point

Sequence	Probability of Sequence (Per Year)	BRP-1(1)	BRP-1(2)	BRP-2(1)	BRP-2(2)	BRP-3(1)	BRP-3(2)	BRP-4(1)	BRP-4(2)	BRP-5(1)	BRP-5(2)*
		Probability of Release Category: (1) Given the Sequence, (2) Considering the Sequence Probability									
$T_3AY_fL_r$	1.2×10^{-6}					.99 (g)	1.2×10^{-6}	0.01 (b)	1.2×10^{-8}		
$T_3AY_fOL_r$	3.7×10^{-7}					.99 (g)	3.7×10^{-7}	0.01 (b)	3.7×10^{-9}		
$T_3AB_oL_r$	7.8×10^{-7}					.99 (g)	7.8×10^{-7}	0.01 (b)	7.8×10^{-9}		
$T_4AB_oL_r$	2.9×10^{-7}					.99 (g)	2.9×10^{-7}	0.01 (b)	2.9×10^{-9}		
$T_5AY_fL_r$	4.1×10^{-6}					.99 (g)	4.1×10^{-6}	0.01 (b)	4.1×10^{-8}		
$T_5AB_oL_r$	2.9×10^{-6}					.99 (g)	2.9×10^{-6}	0.01 (b)	2.9×10^{-8}		
$T_6AB_oL_r$	6.3×10^{-6}					.99 (g)	6.3×10^{-6}	0.01 (b)	6.3×10^{-8}		
$T_7AB_oL_r$	4.6×10^{-6}					.99 (g)	4.6×10^{-6}	0.01 (b)	4.6×10^{-8}		
T_8AL_r	1.7×10^{-7}					.99 (g)	1.7×10^{-7}	0.01 (b)	1.7×10^{-9}		
RR_oL	1.2×10^{-5}					.0642 (b)	7.7×10^{-7}	0.01 (b)	1.2×10^{-7}	0.76 (c)	9.1×10^{-6}
RR_oC	4.8×10^{-6}					.0642 (b)	3.08×10^{-7}	0.01 (b)	4.8×10^{-8}	0.76 (c)	3.6×10^{-6}
H_1Z	1.1×10^{-6}	0.1 (g)	1.1×10^{-7}			.026 (a)	2.86×10^{-8}	0.01 (b)	1.1×10^{-8}	0.86 (c)	9.5×10^{-7}
H_2	3.9×10^{-7}	0.1 (g)	1.1×10^{-7}			.0642 (b)	2.5×10^{-8}	0.01 (b)	3.9×10^{-9}	0.76 (c)	3.0×10^{-7}
I_1E_mL	8.3×10^{-5}					.026 (a)	2.16×10^{-6}	0.01 (b)	8.3×10^{-7}	(c)	6.6×10^{-5}
I_1E_mC	7.9×10^{-6}					.026 (a)	2.05×10^{-7}	0.01 (b)	7.9×10^{-8}	(c)	6.2×10^{-6}
$D_\rho\alpha$	1.8×10^{-4}					1.0 (i)	1.8×10^{-4}	0.01 (b)	1.8×10^{-6}		
$D_\rho X$	5.6×10^{-6}					1.0 (i)	5.6×10^{-6}	0.01 (b)	5.6×10^{-8}		
$D_s\alpha Y$	4.1×10^{-5}					1.0 (k)	4.1×10^{-5}	0.01 (b)	4.1×10^{-7}	(c)	2.8×10^{-5}
D_sXY	2.3×10^{-6}					1.0 (k)	2.3×10^{-6}	0.01 (b)	2.3×10^{-8}	(c)	1.6×10^{-6}
$D_{cs}\alpha Y$	1.1×10^{-6}					1.0 (j)	1.1×10^{-6}	0.01 (b)	1.1×10^{-8}		
$D_{cs}X$	2.8×10^{-6}					1.0 (j)	2.8×10^{-6}	0.01 (b)	2.8×10^{-8}		
PE_vF_sKC	2.6×10^{-5}					.126 (d)	3.28×10^{-6}	0.01 (b)	2.6×10^{-7}	(c)	2.0×10^{-5}
PE_mF_sKC	9.7×10^{-5}					.126 (d)	1.22×10^{-5}	0.01 (b)	9.7×10^{-7}	(c)	7.7×10^{-5}
ME_vNKC	1.2×10^{-5}					.026 (a)	3.12×10^{-7}	0.01 (b)	1.2×10^{-7}	(c)	9.5×10^{-6}
ME_mNKC	3.7×10^{-6}					.026 (a)	9.62×10^{-8}	0.01 (b)	3.7×10^{-8}	(c)	2.9×10^{-6}
UE_vUKC	1.2×10^{-5}					.036 (j)	4.32×10^{-7}	0.01 (b)	1.2×10^{-7}	(c)	9.5×10^{-6}

TABLE V.28

Summary of Important Accident Sequences for Big Rock Point

Sequence	Probability of Sequence (Per Year)	Probability of Release Category: (1) Given the Sequence, (2) Considering the Sequence Probability									
		BRP-1(1)	BRP-1(2)	BRP-2(1)	BRP-2(2)	BRP-3(1)	BRP-3(2)	BRP-4(1)	BRP-4(2)	BRP-5(1)	BRP-5(2)
UE_mUKC	1.2×10^{-4}					.036 (j)	4.32×10^{-6}	0.01 (b)	1.2×10^{-6}	(c)	9.5×10^{-5}
WE_vKC	1.2×10^{-5}					.026 (a)	3.12×10^{-7}	0.01 (b)	1.2×10^{-7}	(c)	9.5×10^{-6}
WE_mKC	3.7×10^{-6}					.026 (a)	9.62×10^{-8}	0.01 (b)	3.7×10^{-8}	(c)	2.9×10^{-6}

Failure to isolate containment probabilities were determined assuming a value of .0312 for failure to isolate the feedwater line. This number was obtained by assuming Valves VFW6 and VFW2 are made motor operated, and generic failure data is applicable.