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ANALYZING THE SAFETY IMPACT OF CONTAINMENT INERTING AT VERMONT YANKEE

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ABSTRACT

Post-accident hydrogen generation in BWR containments is analyzed as a function of engineered hydrogen control system, assumed either nitrogen inerting or air dilution. Fault tree analysis was applied to assess the failure probability per demand of each system. These failure rates were then combined with the probability of accidents producing various hydrogen generation rates to calculate the overall system hydrogen control probability. Results indicate that both systems render approximately the same overall hydrogen control probability (air dilution: .917 - .989; nitrogen inerting: .987 - .998). Drywell entries and unscheduled shutdowns were also analyzed to determine the impact on the total BWR accident risk as it relates to the decay heat removal system. Results indicate that inerting may increase the overall risk due to a possible increase in the number of unscheduled shutdowns due to a lessened operator ability to correct and identify "unidentified" leakage from the primary coolant system. Further, possible benefits of inerting due to reduced torus corrosion and fire risk in containment appear to be dominated by the possible operations related disbenefits.

EXECUTIVE SUMMARY

The accident at Three Mile Island (TMI) has led to a re-evaluation of federal safety regulations and utility operating procedures. Because of concern over hydrogen production from zircalloy fuel cladding oxidation in accidents where fuel temperatures rise substantially, the Nuclear Regulatory Commission (NRC) has made several recommendations for change in operating facilities. One of these recommendations would require the Vermont Yankee containment structure to be inerted with nitrogen. Although other Mark I BWRs are now inerted, it has not been quantitatively established that public health risk has been reduced by this procedure. Moreover, many utility engineers are concerned over the possibility that inerting might actually increase public health risk. They argue that a readily accessible containment may be a significant factor affecting accident mitigation. Also, utilities are concerned that inerting may increase occupational health risks. Concern over worker safety arises from the fact that nitrogen will replace oxygen in the containment causing the atmosphere to be unbreathable. To establish sound technical bases for positions taken on licensing safety issues, utilities are in need of quantitative analyses of such important matters.

This study applies the best state-of-the-art methods to assess the impact of containment entries made at the Vermont Yankee plant, combined with the established framework of WASH-1400 for the Peachbottom BWR plant, to establish as quantitatively as possible the safety impact of containment inerting. Technical alternatives to nitrogen inerting (e.g., controlled burning of hydrogen, etc.) may reduce occupational hazards while ensuring the same degree of control over hypothetical hydrogen releases. In this work, the hazards of nitrogen inerting and the post-accident hydrogen generation problem are analyzed as a function of the engineered hydrogen control system in place at the plant. Two systems are analyzed: the containment inerting system (CIS) in place at all but two BWRs in the U.S., and the containment air dilution system (CAD) operating at the Vermont Yankee nuclear power station. Fault tree analysis is applied to assess the failure probability per demand of both systems.

The probability of various accident scenarios and hydrogen generation rates are combined with the system failure probabilities to calculate the overall system hydrogen control probability. Results indicate that both systems render approximately the same overall hydrogen control probability (CAD = 0.917 to 0.989; CIS = 0.987 (Pilgrim) to 0.998 (Peachbottom).

Operating procedures and non-inerted drywell entries are also analyzed, since the possibility exists that an increase in the number of unscheduled shutdowns can increase the probability per reactor-year of an accident initiated by a loss of the decay heat removal system (HRS). The Reactor Safety Study (WASH-1400) showed that the HRS failure event scenario dominates the overall BWR accident risk by at least an order of magnitude. Assuming a two-fold increase in the number of unscheduled reactor shutdowns as a result of inerting (i.e., inability of operators to correct and identify leakage classified as "unidentified"), the overall BWR safety risk is increased in direct proportion to the increase in the number of shutdowns. It would appear, therefore, that inerting may result in an increase in the overall BWR accident risk if there is a significant increase in plant shutdowns for inerted containments.

It is therefore recommended that alternatives to inerting be seriously evaluated as possible candidates for post-accident hydrogen control in BWRs. Moreover, a preliminary analysis of the torus corrosion and fire prevention aspects of inerting does not reveal substantial benefits to counterweight the safety-related disadvantages associated with delayed maintenance and drywell leakage identification.

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CHAPTER ONE

INTRODUCTION

This thesis describes a probabilistic safety analysis of containment inerting in boiling water reactors (BWRs). The incident at Three Mile Island (TMI) resulted in a renewal of interest in the hydrogen generation problem. This chapter reviews the TMI incident, the Nuclear Regulatory Commission's (NRC) recommendations, and finally, concludes with a statement of the objective of this work.

1.1 Hydrogen Generated During the Accident at Three Mile Island

During the accident at Three Mile Island (TMI), a significant amount of hydrogen was produced through the oxidation of zirconium cladding as it interacted with steam. The amount of cladding that reacted with water has been estimated to be between 50 to 70 percent.[1] Previous design basis accidents had expected less than a .1% metal-water reaction. After the first few hours of the accident, some of the hydrogen was trapped in the upper region of the reactor vessel above the inlet and outlet nozzle (see Fig. 1.1). The hydrogen was distributed between the gas-steam bubble in the vessel, the gas that dissolved in the reactor coolant, and the gas which escaped to the reactor building through the open pressure relief valve on the pressurizer. About nine hours into the accident, a pressure pulse of 28 psig was recorded in the containment building due to the burning of hydrogen. The pressure spike was below the 60 psig design pressure of the containment building, and well below the expected burst pressure of 160 psig.

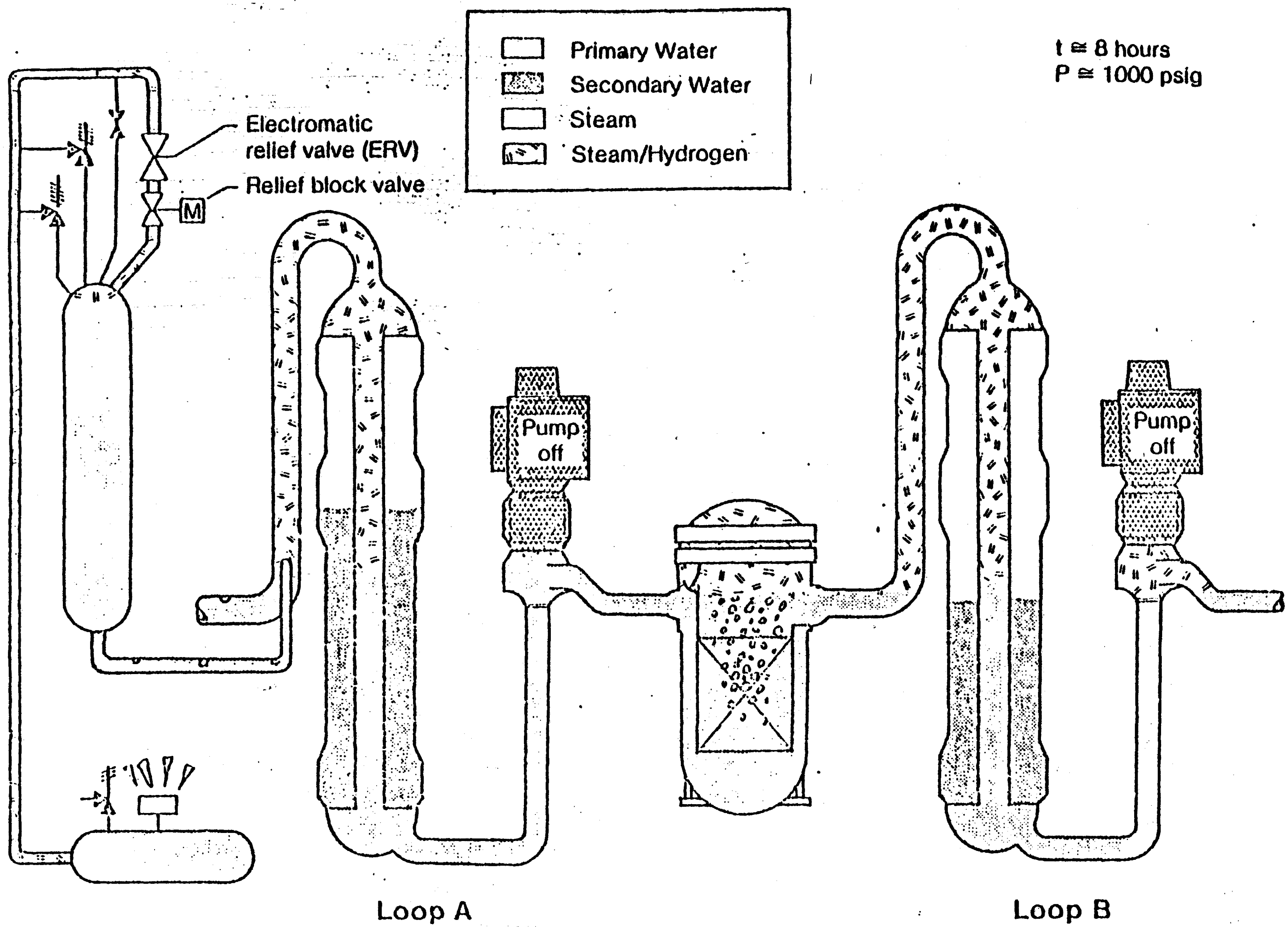


Figure 1.1 Primary system depressurizing and releasing hydrogen through the pressurizer into the containment.

The fact that a hydrogen burn occurred was later confirmed when oxygen analyzers in the containment showed a depletion of the oxygen content.[2] A report prepared by Batelle Columbus Laboratories[2] indicates that the measured 28 psig pressure pulse did not necessarily represent a uniform pressure increase because the quantity of hydrogen that burned (inferred from the oxygen depletion as 1034 lbs of hydrogen) is substantially greater than that required to explain the measured containment pressure increase (564 lbs of hydrogen). This discrepancy can be explained if it is assumed that the hydrogen formed a non-uniform distribution inside the containment. This means that much higher pressures could have existed locally. However, if such high pressures did exist, the instrumentation either did not (or could not) monitor them.

1.2 The Nuclear Regulatory Commission's Lessons-Learned Task Force

The ~~amount of~~ hydrogen generated by the large metal-water reaction at TMI and the resulting pressure increase in the containment were considered in the Nuclear Regulatory Commission's (NRC) TMI-2 Lessons-Learned Task Force.[3] In that report, recommendations were made for the control of post-accident generated hydrogen. One of the NRC's recommendations was that all Mark I and Mark II BWR containments should be inerted with nitrogen to prevent against hydrogen burns or explosions. This recommendation was made since the relatively small volume of Mark I and Mark II containments (approximately 300,000 cubic feet) have a smaller margin available to accomodate metal-water reactions as compared to larger containments. Recommendations were based on the regulatory position specified in Regulatory Guide 7.1.[4]

Requirements for hydrogen control are based on three possible hydrogen generation paths: (i) metal-water reactions involving the zirconium fuel cladding and the reactor coolant, (ii) radiolytic decomposition of the water, and (iii) corrosion of metals by solutions used for emergency cooling and containment spray.

Hydrogen generated during an accident may react with oxygen in the containment atmosphere. This reaction can take place at a rapid enough rate to lead to the rupture of the containment due to overpressurization. Government regulations specify that nuclear power plants should have the capability to: (i) monitor the hydrogen concentration in the containment, (ii) mix the containment atmosphere, and (iii) control combustible gas concentrations without relying on purging and/or repressurization of the containment atmosphere following a loss-of-coolant accident (LOCA). Inerted containments satisfy the above requirements. Only two plants with Mark I containments (Hatch 2 and Vermont Yankee) use other types of hydrogen control systems. However, the current position of the NRC task force (as of 1980) requires that these plants also inert their containments in a manner similar to other operating plants around the country.

1.3 Study Objective

The objective of this thesis is to analyze two common methods for hydrogen control used in boiling water reactors (BWRs), namely inerting and the air dilution systems, and their ability to handle combustible mixtures of hydrogen. This thesis is organized as follows: Chapter 2 describes the hazards of inerting with nitrogen, using as an example the incident at the nuclear plant in Tarapur, India. In Chapter 2, the inerting controversy is also briefly described, particularly as it relates to the Vermont Yankee nuclear power station. Chapter 3 reviews the hydrogen generation problem in terms of mechanisms for hydrogen generation, flammability limits, deflagration and detonation. Also discussed are the effects of hydrogen generation on the containment, a brief description of different types of BWR containments, and available methods for hydrogen control. Chapter 4 analyzes containment failure modes due to hydrogen generation. An event tree is also developed to illustrate the sequence of events that may lead to degraded core conditions and/or meltdown. The containment air dilution (CAD) system of Vermont Yankee and the containment inerting system (CIS) of Pilgrim I and Peachbottom II are analyzed using fault tree analysis to determine their failure rate per ~~reactor year~~ demand. Finally, the ability of each system to handle hydrogen is quantified through use of WASH-1400 probabilities on accident scenarios that produce hydrogen. Chapter 5 analyzes the impact of containment inerting on the frequency of accident initiating events. The potential impact of inerting on operational procedures are discussed using Vermont Yankee drywell entry data and the effects of additional shutdowns on overall BWR accident risk are quantified. Chapter 6 presents conclusions and recommendations.

CHAPTER TWO

THE INERTED CONTAINMENT PROBLEM

In BWRs using containment inerting systems, the primary containment is inerted with nitrogen within 24 hours after startup of the reactor and is deinerted 24 hours prior to reactor shutdown.[15] During normal operation, the primary containment has 4% oxygen and 96% nitrogen by volume. (Air contains ~21% oxygen and 79% nitrogen by volume).

2.1 Hazards of Inerting with Nitrogen

If a person breathes in an atmosphere where the oxygen concentration is below 20%, his respiration rate will increase slightly until the oxygen concentration decreases below 8% by volume. This condition is called anoxia, and increases not only the breath rate, but also the pulse rate (Fig. 2.1). The body is relatively unresponsive to small decreases in the oxygen concentration. However, sharp oxygen reductions cause the body to react strongly since the cells will need more oxygen to form CO_2.[13] The rate and depth of respiration is controlled by a respiratory center at the base of the brain. If the anoxia process is gradual, the subject's performance will deteriorate so gradually that they may not be fully aware of what is happening.[17] When anoxia is severe, the individual loses consciousness in a matter of seconds and will die unless help is obtained. If the person does not die, there is the possibility of irreversible brain damage, specifically to the central nervous system which cannot sustain a prolonged ~~loss~~ of oxygen.[13] The person who suffers a cardiac arrest before the heart is artifically caused to resume beating may spend the rest of his

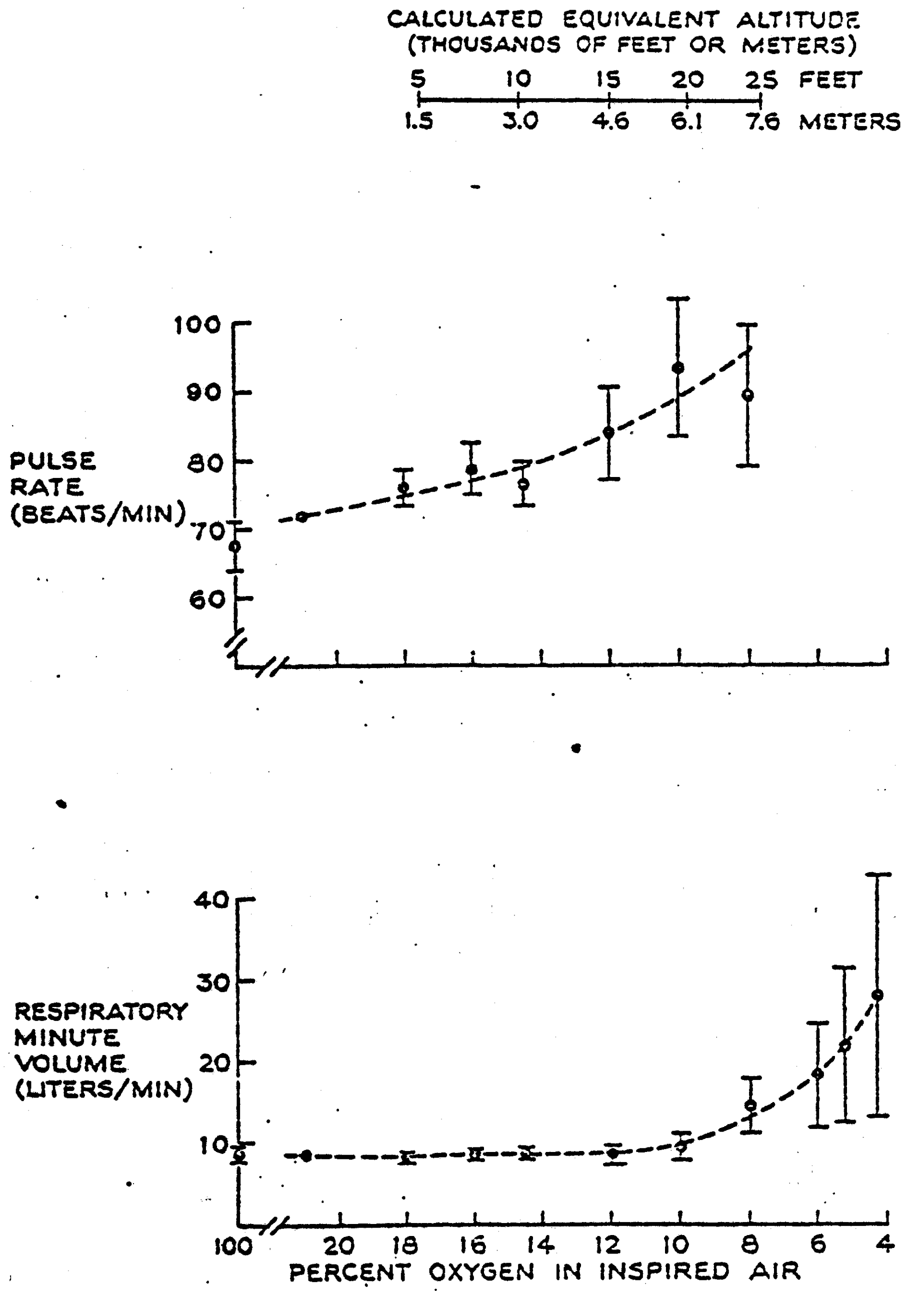


Figure 2.1 Effects of Anoxia in the Human Body[16]

life as a retarded adult due to severe brain damage.[17]

There are many recorded cases of accidents caused by anoxia, e.g. miners entering a shaft containing methane and CO_2, and workers entering a supposedly deinerted atmosphere. At the Westinghouse Astronuclear Core operational facility, a man died after entering a furnace that contained argon. The argon was being used to purge the furnace in the last stages of cooldown as part of the process of leaching fuel elements.[18] However, the most dramatic death due to anoxia to occur in the nuclear power industry was the incident at the Tarapur nuclear power station in India.

2.2 The Tarapur Atomic Power Station Incident

In July 1970, a maintenance supervisor at the Tarapur atomic power station in India died of asphyxiation (according to the autopsy) during an entry into a chamber that was supposed to be deinerted. The chamber atmosphere, which had been inerted with nitrogen some eight months before the incident, contained only 7% oxygen at the time of entry (compared to ~21% oxygen in air). Although the Indian government has been reluctant to discuss the case, it has been inferred that the man died due to the low concentration of oxygen in the chamber atmosphere. In order to have a complete picture of the incident, the physical layout of the Tarapur facility and the accident itself is now described.

The station consists of two boiling water reactors, each running its own individual turbine generator designed to produce 210 MWe, for a combined 420 MWe output for the total plant. In such a "dual" facility, each reactor has a separate reactor vessel, reactor drywell and suppression pool. Above the suppression pools, there is one large chamber, called the "common chamber", which is connected to both reactors' suppression pools by a set of diaphragms or blowout disks. The common chamber has a personnel lock, which is a double door connected at mid-height with the diaphragm. The drywell and suppression pool of unit 1, and the common chamber were inerted in November of 1969. However, due to steam and/or air leaks, the drywell pressure was found to increase frequently and had to be vented.[29]

According to Robert L. Turner, the resident warranty representative for the General Electric Company at the Tarapur facility, the incident occurred after a shutdown of one of the units during a turbine maintenance outage.[30] One of the tests that is run prior to putting a unit back in service is a containment leak test. Normally, this is performed by testing the leak tightness of the drywell and the suppression pool, which are interconnected. During this test at the Tarapur plant, operators found a higher than expected leakage between the unit and the common chamber. They concluded that it was necessary to go inside the chamber to inspect the diaphragm.

During preparation for the entry, concern was raised about whether or not the common chamber had been adequately purged of nitrogen. It was assumed that the common chamber had been purged two or three times since November 1969. The oxygen monitoring system also indicated a normal oxygen level inside the chamber. Three people-- the chief superintendent who was the ranking official at the site, and one of the shift supervisors-- went into the personnel lock to check its appearance and performance, since this was an area that had been closed for a number of months. Once inside the lock, they closed the outer door and opened the inner door to the common chamber. They looked inside the chamber with a flashlight without leaving the personnel lock. After three minutes, they went out saying that they did not see anything wrong except a smell of stale paint. The sequence of events during the next entry, as well as the personnel involved, are shown in Table 2.1. In summary, three men entered the lock and two went into the common chamber for inspection. After a few minutes all three men became unconscious due to anoxia and were pulled out. During the rescue, another person went into the common chamber and also became unconscious. He could not be revived and was the single casualty of the incident.

After the incident, the air inside the common chamber was tested and showed an oxygen level of only 7%. Apparently, oxygen samples were not taken before the entry was made; the possibility also exists that a faulty air monitoring system was used. After this incident, the primary containment vessels at the Tarapur plant have never again been inerted.

2.3 The History of Containment Inerting at Vermont Yankee

At the Vermont Yankee plant operated by the Yankee Atomic Co., the hazards of inerting with nitrogen have been emphasized in licensing hearings since 1971. On November 5, 1971 the Atomic Energy Commission (AEC) asked Vermont Yankee for information regarding the control of combustible gases which could be generated in a hypothetical loss-of-coolant accident. Vermont Yankee responded by presenting an

Table 2.1

SEQUENCE OF ENTRY INTO THE COMMON CHAMBER AT THE TARAPUR ATOMIC POWER STATION (ACCORDING TO ROBERT TURNER'S TESTIMONY).[30]

Personnel Involved

Chief Superintendent	(CS)	Maintenance Supervisor	(MS)
Operations Superintendent	(OS)	GE Warranty Representative	(WR)
Maintenance Foreman	(MF)	Shift Supervisor	(SS)
Maintenance Person	(MP)	Healt Physicist	(HP)

Event Sequence

-OS, MF, MP went into personnel lock.
-OS, MF stepped into the common chamber on its grating with flashlights, wearing no protective gear other than coveralls.
-OS, MF flashed lights on, trying to look at the diaphragms 10 feet below.
-OS told MF that he did not feel too well. He though it would be a good idea to leave the chamber.
-OS started out, looked behind him and saw that MF had collapsed on the grating.
-OS picked MF up and carried him to the personnel lock.
-MP helped OS to put MF into the lock.
-OS collapsed at the inner door.
-MP pushed OS into the lock, apparently closed the inner door and opened the outer door.
-MS who was outside at the time of the events above, saw there was a problem, jumped into the lock and closed the outer door.
-MS saw OS on the floor of the chamber; the inner door was opened.
-MS probably tried to pick OS up but collapsed.*
-WR, SS outside heard something knocking inside, opened outer door and found MF and MP in the lock.
-WR, SS took MF out the lock. MF was concious.
-MP jumped out of the lock.
-SS went into the lock followed by WR and HP.
-WR went into common chamber and found MS and OS unconcious.
-WR took both MS and OS into the personnel lock.
-MS was pulled out of the lock, and one person started giving him artificial respiration.
-OS was pulled out and WR started giving him artificial respiration until OS began to breath.
-WR tried to help MS.
-Maintenance crew carried one big oxygen cylinder to the point where MS was.
-Plant doctor arrived and started giving MS a lung pumping motion without any result.

*WR found MS with a bump on his head. It was assumed that he hit something inside the chamber and was too weak to leave quickly.

approach involving the addition of nitrogen to the containment.[37] The use of an inerted containment was accepted by the AEC as a short term method of controlling combustible gases based on technical specifications for inerting.[21] However, spokesmen for Vermont Yankee stated that, for a longer term, they would pursue alternative gas control systems to avoid the hazards of nitrogen inerting.

On October 12, 1972, the Vermont Yankee operating license was ammended by the NRC to authorize for full power operation provided that an inerted atmosphere was maintained during the normal operations (with the exception of startup test programs and demonstration of plant electrical output).[38] Startup of the plant was substantially completed by June, 1974, and plant electrical output was demonstrated by December, 1972.

High power testing was performed during February and March of 1974 when restrictions on power level were removed and the plant operated at its 100% capacity level of 540 MWe. After attaining normal operations the ACE concluded in a safety evaluation that it was necessary to inert the containment.[39] Based on this safety evaluation, the operating license was amended further giving twenty days for the plant's staff to complete the inerting operation.[40]

On July 11, 1974 after preliminary hearings were conducted before the Atomic Safety and Licensing Appeal Board (ALAB), a memorandum and order were issued ~~by them~~ stating that inerting was not justified pending the outcome of a full hearing because the evidence presented showed that inerting creates safety problems with greater consequences than those it was intended to solve.[41] On September 18, 1974, the ALAB issued its final decision: the inerting requirement at

Vermont Yankee could not legally be imposed and the factual records presented would not justify the inerting requirement.[42] Finally, the Commissioners of the Atomic Energy Commission concluded on November 7, 1974 that the non-inerted Vermont Yankee containment should be preserved pending completion of the rule-making.[43] On June 1, 1976, a description of a containment air dilution (CAD) system designed by Yankee engineers was sent to the Nuclear Regulatory Commission.[11] The system was installed in 1976 but until this day remains unlicensed pending additional rule-makings.[20]

CHAPTER THREE

THE HYDROGEN GENERATION PROBLEM

The production of hydrogen during the course of an accident presents two potential threats to containment integrity; first, by increasing the internal gas pressure in the system and secondly, by burning or exploding when combined with the oxygen present in the containment atmosphere. The additional thermal energy produced in the burning or detonation of the hydrogen raises the pressure inside the containment and eventually can result in containment failure by overpressurization.[5]

3.1 Mechanisms of Hydrogen Generation

Hydrogen can be produced during the course of a reactor accident through high temperature metal-water reactions between fuel cladding and reactor coolant, radiolytic decomposition of water, and corrosion of metals by solutions used for emergency cooling or containment sprays. The main source of hydrogen from metal-water reactions is produced through the high-temperature zircalloy-water and steel-water reactions. These reactions take place according to the following reactions:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + \text{heat} \tag{3.1}$$

$$\text{Fe(steel)} + xH_2O \rightarrow \text{Fe(steel) oxides} + xH_2 + \text{heat} \tag{3.2}$$

Reaction (3.1) represents an initial source of hydrogen in a meltdown and occurs when steam from water in the pressure vessel contacts overheated zircalloy fuel cladding. It has been estimated that the rate of consumption of zircalloy is about 10 per cent per 1000 seconds. Fig. 3.1 shows the zircalloy consumption as a function

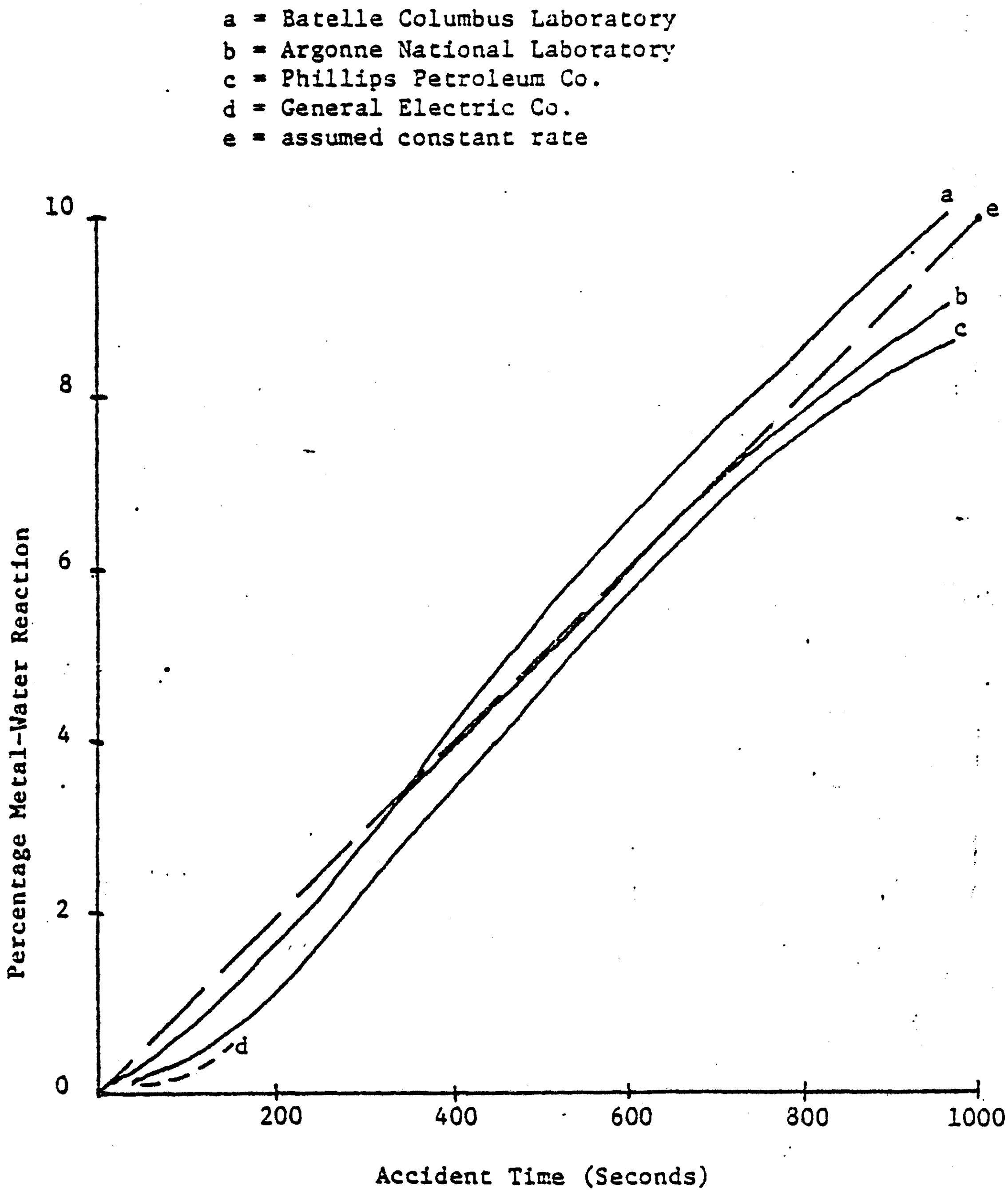


Figure 3.1 Percentage Metal-Water Reaction vs. Accident Time[44]

of time derived from a comparison of BWR core heatup calculations.[44] Assuming a conservative constant comsumption rate, all zircalloy would be consumed in less than three hours and could result in a 72% hydrogen containment concentration (Fig. 3.2). Given that the amount of steam decreases with time (Fig. 3.3), the rate of zircaloy consumption will be lower but using a conservative approach, an upper bound for the consumption rate can be assessed. Fig. 3.4 shows the concentration of containment hydrogen for each class of reactor containment as a function of the amount of metal-water reaction.

Steel-water reactions (eqn. 3.2) could generate massive amounts of hydrogen. However, experimental studies indicate that iron or steel must be nearly molten before appreciable reaction with steam occurs. Contact between large amounts of molten steel and water might cause steam explosions before the reaction could generate hydrogen.

The radiolytic decomposition of water is a delayed but potentially significant source of hydrogen. Beta or gamma radiation can cause ionization and subsequent decomposition of water molecules resulting in hydrogen. However, the production of large amounts of hydrogen in an accident would require that high radiation doses be applied to large volumes of water; for example, in the range of 10^8 to 10^9 rads applied to the entire water supply of the reactor. Since it would require several days or weeks to accumulate such exposures, this source of hydrogen is considered a long term rather than an immediate problem. For BWR systems, it has been found that hydrogen concentrations greater than four volume percent are possible from

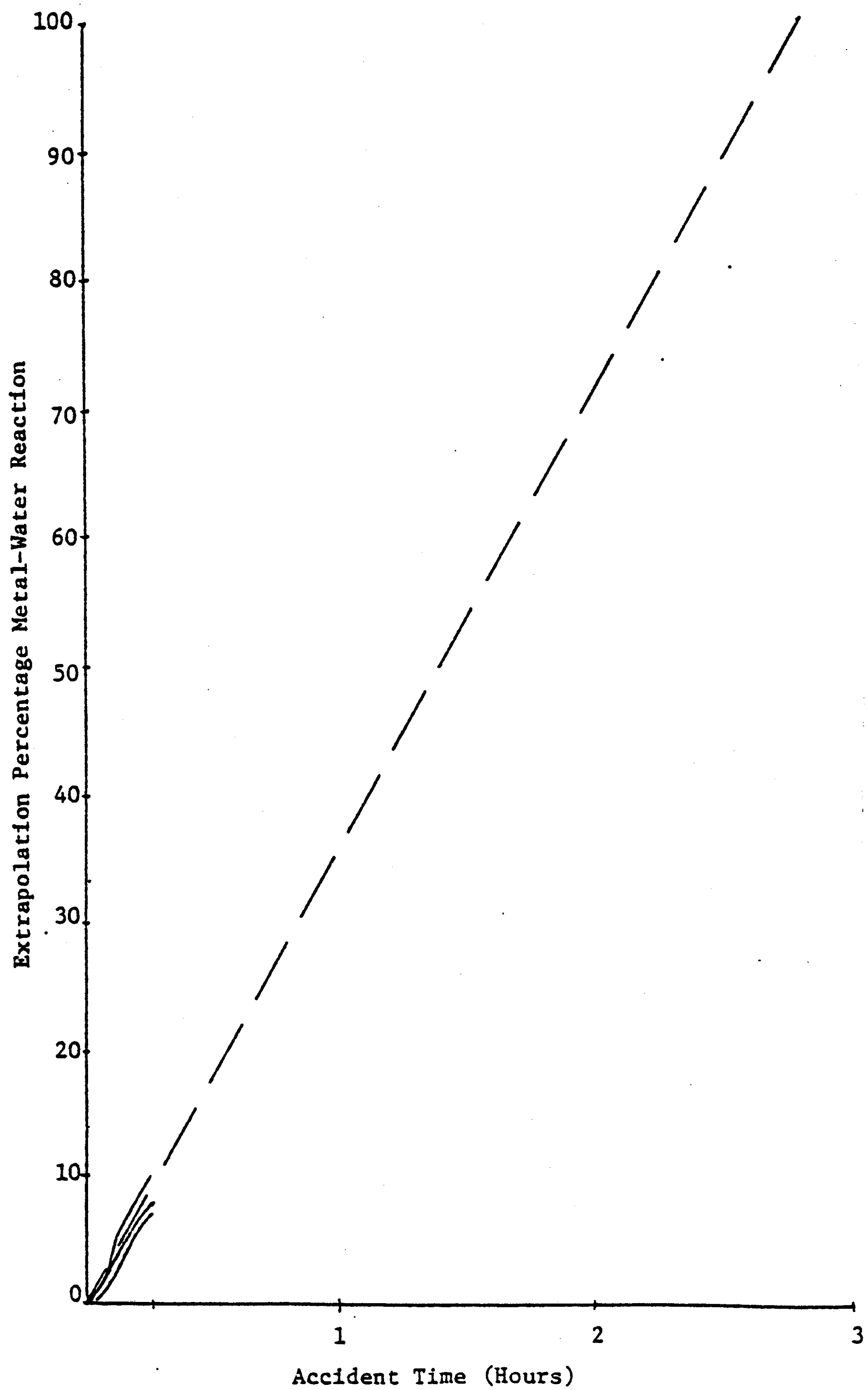


Figure 3.2 Extrapolation Percentage Metal-Water Reaction vs. Accident Time.

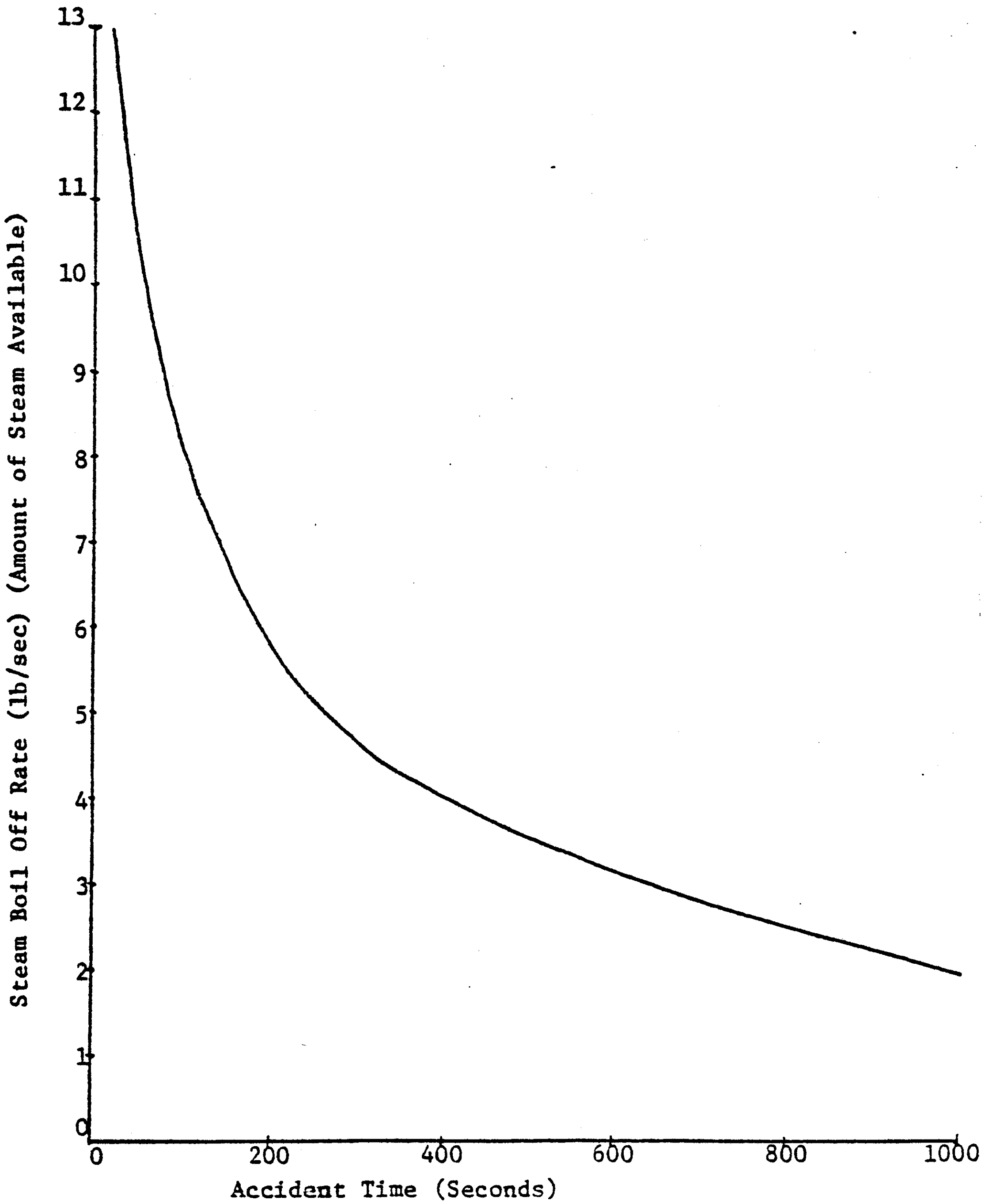


Figure 3.3 Steam Boil Off Rate vs. Accident Time[44]

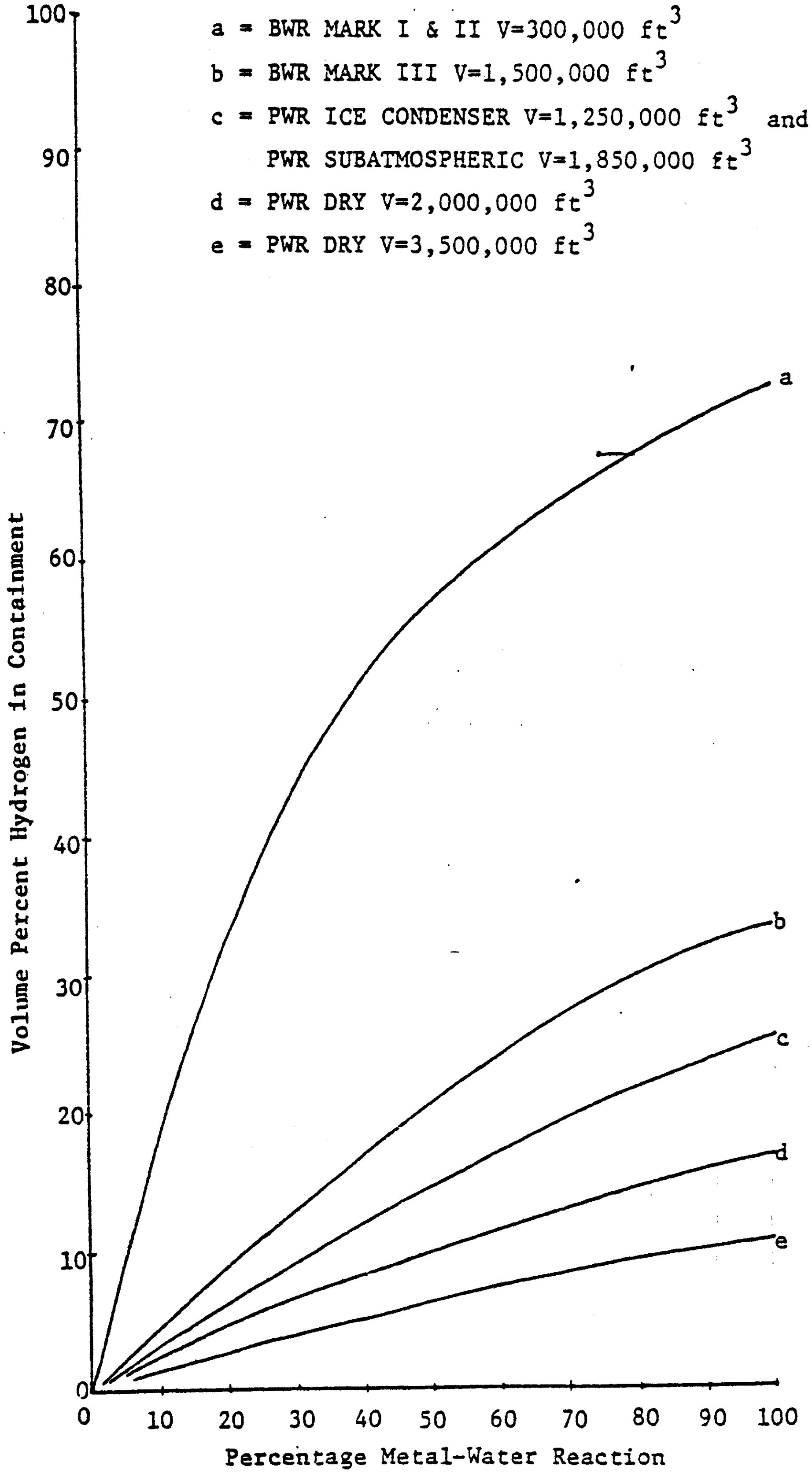


Figure 3.4 Volume Percent Hydrogen in Containment vs. Percentage Metal Water Reaction

radiolysis, with a conservative estimate of the time to reach such a limit of between 15 to 100 hours.[6]

Protective coatings applied to the interior surfaces of reactor containment facilities should react with the suppression solutions because of extreme temperatures and radiation levels after a design-basis accident. The probable reactions between the spray solutions and the zinc-rich primer coat are:[6]

$$Zn + OH^- + H_2O \rightarrow HZnO_2 + H_2 \qquad (3.3)$$

$$Zn + 2H_2O\ (steam) \rightarrow Zn(OH)_2 + H_2 \qquad (3.4)$$

$$4Zn + 5/2\ O_2 + 3H_2O + CO \rightarrow Zn_4CO_3(OH)_6 \qquad (3.5)$$

Hydrogen evolution from these reactions could represent around a 0.5 volume per cent increase in the hydrogen concentration in the containment.

3.2 Properties of Hydrogen-Oxygen Mixtures

3.2.1 Flammability Limits

A flammable mixture of gases, such as hydrogen-oxygen, may be diluted with one of its constituents or with other gases until it is no longer flammable. The marginal composition at which such a mixture becomes flammable is defined as the "flammability limit". A combustible gas mixture generally has an upper and lower flammability limit. When the composition is between these limits, the mixture will burn. Shapiro and Moffette[7] establish the lower flammability limit for hydrogen in air as 4.1, 6 and 9 volume per cent for upward, horizontal and downward propagation. Although the upper limits for horizontal and downward propagation are somewhat smaller than the upper limit for upward propagation, these limits are not well established.[7] In this study, upper flammability limits are taken as 74 volume percent as a conservative estimate. In all cases, flammability limits are assumed to apply to gases maintained at atmospheric pressure and room temperature, which may be saturated with vapor. Limits vary with direction since convection currents produced by hot expanding combustion products cause an upward gas movement rather than a uniform ignition front.

It is important to specify the direction of flame propagation when quoting flammability limits since conditions change over the range. The rate of flame propagation is less than the rate at which the flammable mixture rises due to strong convection currents. According to Shapiro and Moffette[7], the effects of water vapor mixtures in the flammability limits can be calculated (Fig. 3.5). The upper and lower flammability limits converge as the percentage of water vapor increases. As temperatures rise, the water vapor content rises such that the lower flammability limit rises slowly while the upper limit falls rapidly. When the hydrogen-air-steam mixture reaches 60 percent steam, the limits coincide at about 10 percent hydrogen (Fig. 3.5).

Recent data indicates that the flammability region may actually occur over a smaller range than previously predicted. General Electric Co. conducted a series of hydrogen flammability tests under conditions simulating post-LOCA BWR containment pressures, temperatures and water vapor content (Table 3.1; Fig. 3.6).[8] Results of these tests along with other data were used to establish more flexible flammability limits. The current version of the NRC

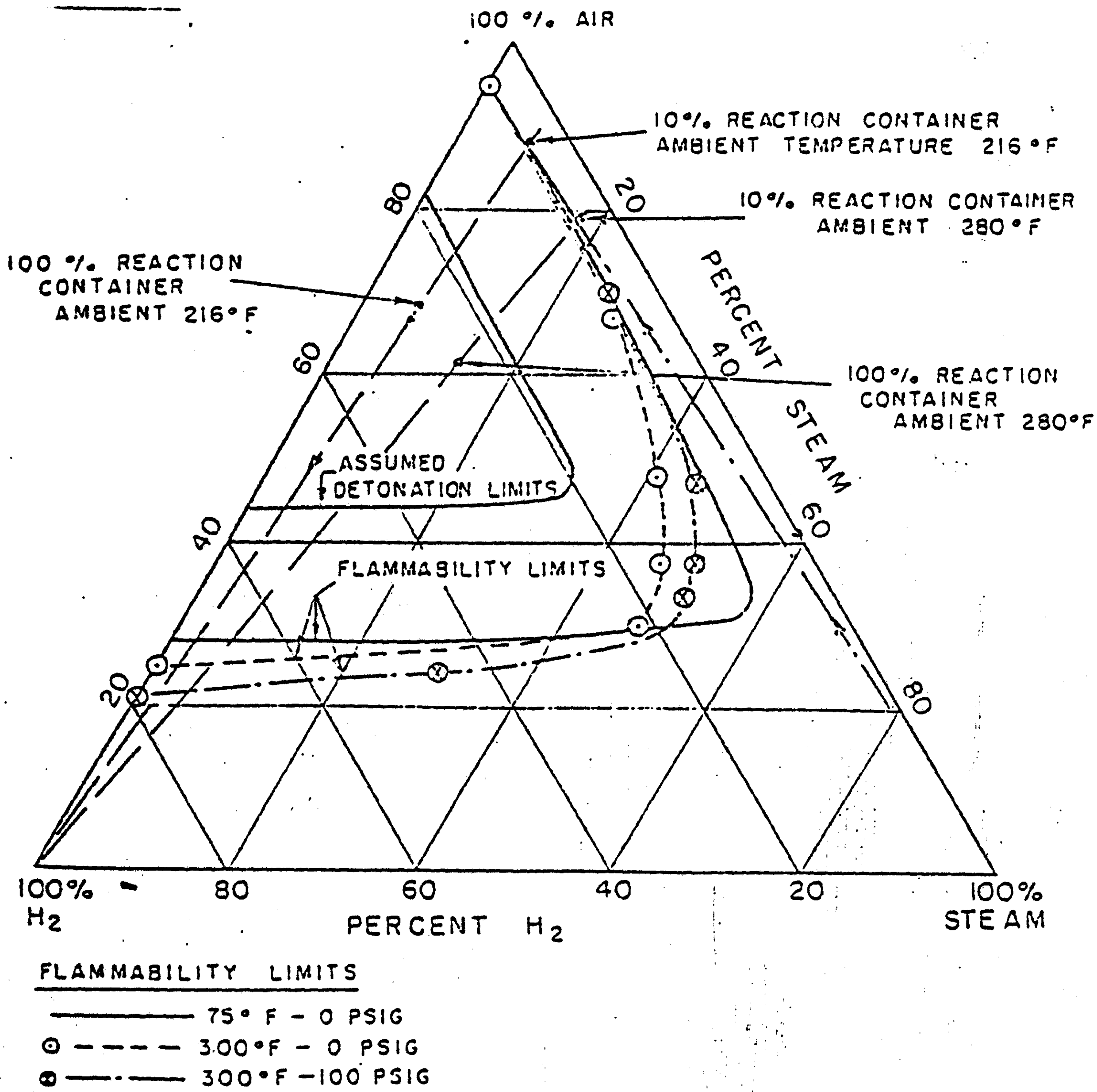


Figure 3.5 Flammability Limits of Hydrogen-Air Steam Mixtures[7]

Table 3.1

HYDROGEN FLAMMABILITY TEST RESULTS[8]

Test	Ambient Conditions: Temperature (°C)	Pressure (mm Hg)	Relative Humidity (%)	Test Temperature (°C)	Ignitor: Type	Location	H_2 + [illegible] Pressure (mm Hg)	Test Pressure (mm Hg)	H_2 Concentration: Ambient Pressure (%)	Test Pressure (%)	Test Relative Humidity (%)	P max (psig)	R av (psi/sec)	R max (psi/sec)	ΔP (psi)	Δt (sec)	Comments
1-A	8.9	752	40	60	Spk	MC	45.6	855.4	6.1	5.3	72	2.9	0.13	0.13	8.9	7.00	
1-B	8.9	750	57.6	60	Dev	MC	45.6	855.4	6.1	5.3	34	–	–	–	–	–	No ignition
1-C	8.9	750	44	60	Pyro	MC	45.6	843.1	6.1	5.4	65	11.7	1.5	4.9	9.4	6.70	
2-A	8.9	755	55	63	Pyro	Top	45.6	853.2	6.0	5.3	59	9.5	1.0	2.7	7.6	7.40	
2-C	8.9	750	40	60	Pyro	MS	45.6	844.3	6.1	4.8	53	8.9	1.0	3.3	7.0	6.90	
2-D	8.9	752	55	63	Pyro	Bot	45.6	853.3	6.1	4.8	68	11.0	2.0	4.2	9.1	4.45	
3-A	7.8	752	55	38	Pyro	MC	33.4	752.0	4.3	4.3	8.8	3.9	0.6	1.0	3.3	6.50	No ignition at first try
3-B	7.2	752	58	38	Pyro	MC	45.6	752.0	6.1	6.1	9.0	12.0	2.1	4.1	11.0	5.03	
3-C	7.0	752	55	38	Pyro	MC	60.9	752.0	8.1	8.1	8.8	30.5	21.4	25.0	27.1	1.20	
4-A	3.3	760	52	38	Pyro	MC	33.4	782.6	4.1	3.3	73	0.8	0.06	0.65	0.15	7.50	
4-B	3.3	752	50	32	Pyro	MC	45.6	771.4	6.1	5.7	99	3.0	0.3	0.55	2.2	7.00	
4-C	4.4	760	40	38	Pyro	MC	67.0	786.3	8.1	7.7	63	20.5	12.1	19.2	20.2	2.30	
4-D	15.7	751	52	71	Pyro	MC	33.4	905.1	4.1	3.3	63	–	–	–	–	–	No ignition at three tries. Transient data
4-E	17.3	750	50	71	Pyro	MC	45.6	905.1	6.1	5.0	66	5.1	0.28	6.4	7.1	7.00	
4-F	12.8	750	56	71	Pyro	MC	66.2	919.4	8.1	6.6	71	18.0	3.6	7.5	12.8	3.70	
–	–	–	–	(100)	(Pyro)	(MC)	–	–	[illegible]	–	(100)	–	–	–	–	–	Not tested
–	–	–	–	(100)	(Pyro)	(MC)	–	–	6.1	–	(100)	–	–	–	–	–	Not tested
5-A	23.3	750	45	100	Pyro	MC	60.8	1316.7	8.1	4.6	76	–	–	–	–	–	No ignition at two tries. Transient data
–	–	–	–	(100)	(Pyro)	(MC)	–	–	[illegible]	–	(100+)	–	–	–	–	–	Not tested
6-A	6.7	760	47	38	Spk	MC	60.8	781.0	8.1	7.8	79	20.3	1.0	3.3	19.0	10.0	Not a smooth [illegible]
–	–	–	–	[illegible]	(Dev)	(MC)	–	–	[illegible]	–	(100)	–	–	–	–	–	Not tested
6-D	6.7	760	47	38	Pyro	Top	67.8	786.9	8.1	7.7	66	31.3	7.8	23.4	28.5	2.950	
6-E	11.3	750	49	29	Pyro	MC	65.2	788.8	8.1	7.7	89	22.9	17.0	31.0	27.8	1.9	
6-F	17.1	750	49	38	Pyro	MS	66.4	786.2	8.1	7.7	60	29.5	9.6	16.6	25.8	3.9	
6-G	15.6	750	60	38	Pyro	Bot	60.6	786.2	8.1	7.7	91.2	29.8	15.5	26.3	21.8	2.8	

Test	Ambient Conditions: Temperature (°C)	Pressure (mm Hg)	Relative Humidity (%)	Test Conditions: Temperature (°C)	Pressure (mm Hg)	Relative Humidity (%)	Ignitor: Type	Location	Added Water Pressure (mm Hg)	H_2 Pressure (mm Hg)	H_2 Concentration: Ambient Pressure (%)	Test Pressure (%)	P max (psig)	R max (psi/sec)	R av (psi/sec)	ΔP (psi)	Δt (sec)	Comments
A-1	30.6	755	47	102.2	1363	79.3	Pyro	MC	173	91.6	11.8	8.9	17.9	2.9	1.1	6.0	3.2	
A-2	16.3	719	58	107.2	1364	71.0	Pyro	MC	133	43.0	11.3	8.5	29.9	6.1	2.0	15.0	3.2	
A-3	20.2	757	55	100.7	1436	73.5	Pyro	MC	569	105.2	13.3	7.3	75.7	5.4	2.9	18.1	6.3	
A-4	25.6	750	44	108.3	1370	63.6	Pyro	MC	577	103.0	13.6	7.6	22.5	7.3	3.2	18.3	5.0	
A-5	20.7	751	40	102.5	1460	69.0	Pyro	MC	515	114.4	15.1	8.1	23.5	7.3	3.6	23.3	6.7	
A-6	30.6	757	56	100.2	1419	67.3	Pyro	MC	636	116.3	16.3	8.3	23.3	11.0	–	75.2	–	Pressure recorder [illegible]
A-7	31.3	760	40	101.7	1404	72.8	Pyro	MC	570	170.0	15.8	8.2	29.0	12.0	8.4	25.5	4.0	
B-1	16.1	757	81	87.3	1295	58.5	Pyro	MC	259	79.5	10.4	6.5	13.0	3.7	1.5	7.2	4.0	
B-2	20.0	760	47	72.3	1438	63.9	Pyro	MC	253	72.6	10.3	6.5	19.3	3.7	1.7	10.5	8.0	
B-3	20.4	757	48	87.9	1142	69.7	Pyro	MC	325	75.3	10.0	6.3	15.2	4.5	2.0	10.4	4.4	
B-4	21.1	753	43	86.1	1134	77.0	Pyro	MC	349	80.3	11.6	7.3	20.5	12.7	4.1	20.0	3.9	
B-5	19.6	754	54	87.8	1170	70.2	Pyro	MC	323	87.5	10.9	7.9	22.5	6.3	3.0	14.3	4.0	
C-1	23.9	756	23	71.7	1920	56.7	Pyro	MC	197	117.3	15.6	5.5	44.8	11.1	4.4	21.9	3.6	
C-2	21.1	757	57	71.9	1953	72.1	Pyro	MC	163	119.0	15.7	6.0	40.6	19.3	4.4	23.7	5.4	
C-3	19.1	763	54	72.2	1646	68.5	Pyro	MC	186	120.9	17.7	7.3	56.0	17.7	6.8	33.3	5.0	
C-4	29.4	754	56	71.1	1800	68.0	Pyro	MC	153	134.9	17.8	7.5	87.0	72.3	9.2	26.8	4.8	
D-1	23.9	754	60	71.1	956	72.5	Spk	MC	133	27.0	4.0	3.9	–	–	–	–	–	No ignition
D-2	23.9	756	65	71.1	943	72.5	Spk	MC	163	46.0	6.1	4.7	–	–	–	–	–	No ignition
D-3	23.9	756	68	71.1	977	77.5	Spk	MC	163	55.3	7.3	5.6	5.7	0.1	0.08	0.0	10.0	
D-4	20.2	753	58	71.1	779	77.5	Spk	MC	175	50.3	8.5	6.5	5.5	0.2	0.11	0.5	4.4	
D-5	22.2	754	67	71.1	1071	75.9	Spk	MC	173	74.0	8.8	7.6	6.3	1.7	0.4	1.5	3.3	
D-6	31.7	754	57	71.1	1024	87.0	Spk	MC	187	86.0	11.1	8.2	8.3	1.4	0.7	2.8	3.6	
E-1	20.7	749	56	72.9	379	43.2	Dev	MC	753	33.8	5.1	3.5	–	–	–	–	–	No ignition
E-2	22.6	753	50	71.1	470	73.6	Dev	MC	177	46.5	6.1	4.7	–	–	–	–	–	No ignition
E-3	20.9	751	55	71.1	775	77.1	Dev	MC	178	56.7	7.5	5.7	–	–	–	–	–	No ignition
E-4	20.7	757	45	71.1	1025	76.7	Dev	MC	179	69.2	8.9	6.8	–	–	–	–	–	No ignition
E-5	17.7	759	50	71.1	975	63.0	Dev	MC	100	72.9	9.7	7.4	17.1	0.3	2.2	17.5	5.0	
E-6	29.4	757	40	71.1	1024	77.5	Dev	MC	178	86.1	11.1	8.2	28.5	20.3	12.5	29.8	2.0	
D-7	20.3	763	47	98.9	1362	76.3	Spk	MC	543	92.1	6.8	3.8	–	–	–	–	–	No ignition
D-8	20.3	766	47	98.3	1496	76.3	Spk	MC	543	65.1	8.6	4.6	–	–	–	–	–	No ignition
D-9	20.3	764	47	98.3	1460	76.3	Spk	MC	543	72.7	10.3	5.2	–	–	–	–	–	No ignition
D-10	20.3	760	47	98.9	1563	76.3	Spk	MC	543	81.1	10.7	5.6	–	–	–	–	–	No ignition
D-11	20.3	760	47	98.9	1534	76.3	Spk	MC	543	94.1	12.4	6.1	15.25	–	–	0.75	–	Response ΔP
D-12	20.3	763	47	98.5	1570	76.3	Spk	MC	543	107.1	14.1	6.8	15.35	–	–	0.75	–	Response ΔP
E-7	20.7	761	50	98.9	1553	63.7	Dev	MC	673	129.0	17.2	8.3	–	–	–	–	–	No ignition
E-8	22.2	753	52	100.0	1470	77.0	Dev	MC	530	105.0	14.1	7.3	–	–	–	–	–	No ignition
E-9	29.4	753	44	100.0	1444	78.6	Dev	MC	505	94.0	17.5	5.5	–	–	–	–	–	No ignition
E-10	19.3	753	57	103.0	1461	81.2	Dev	MC	503	82.0	19.5	5.6	–	–	–	–	–	No ignition
E-11	20.7	750	44	100.0	1454	62.9	Dev	MC	617	63.0	3.1	4.7	–	–	–	–	–	No ignition
E-12	30.0	708	60	103.6	1406	78.4	Dev	MC	570	53.0	7.3	3.8	–	–	–	–	–	No ignition
F-1	30.0	760	52	102.7	2257	79	Pyro	MC	616	175.7	23.1	7.5	71.2	16.5	8.6	40.5	4.8	
F-2	25.6	758	46	103.3	2372	78	Pyro	MC	536	175.7	23.1	7.4	68.5	14.7	8.9	27.3	4.7	
F-3	30.0	766	40	105.5	2414	78	Pyro	MC	662	212.3	23.2	9.2	100.0	40.0	19.5	74.0	2.8	
F-4	27.8	766	47	104.6	2450	81	Pyro	MC	743	227.3	23.2	9.1	121.0	43.5	74.5	52.3	2.6	
F-5	23.8	760	46	104.5	2430	80	Pyro	MC	905	252.3	33.3	10.4	175.0	34.0	75.8	87.9	2.4	
F-6	30.0	760	52	104.6	2444	73	Pyro	MC	941	254.5	32.8	10.7	133.0	34.8	30.3	31.0	2.3	

The four tests listed as "Not Tested" are defined as [illegible] conditions shown in parentheses. They were not conducted [illegible].

NOTES: Abbreviations

Ignition Source

- Spk = 100 Joule spark
- Dev = Electric [illegible]
- Pyro = Pyrofuze

Ignitor Location

- MC = Middle height, center
- Top = Top center ([illegible] inches down from vessel roof)
- MS = Middle height, side
- Bot = Bottom center ([illegible] inches up from vessel wall)

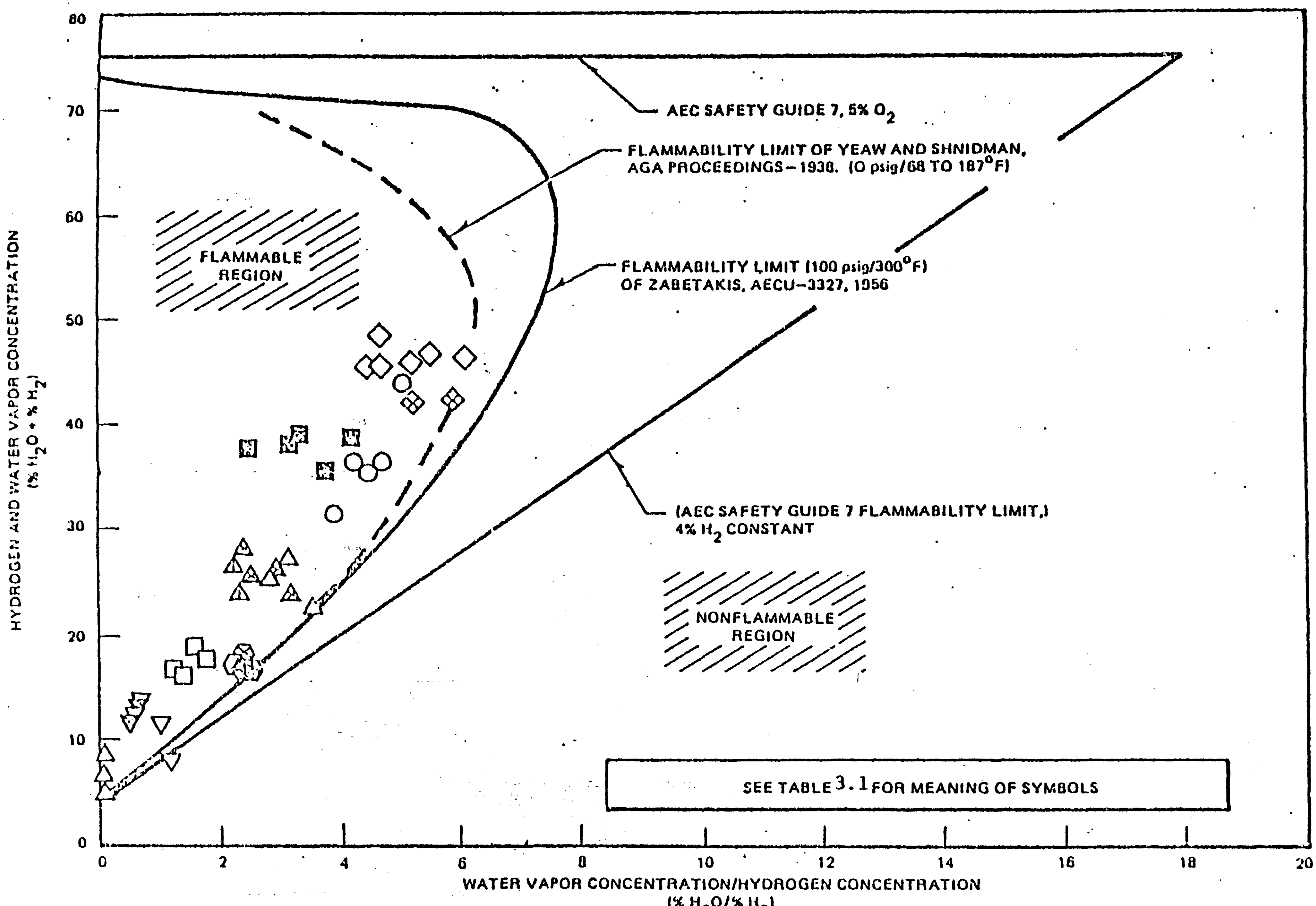


Figure 3.6 Flammable Mixtures[8]

Regulatory Guide recommends that to avoid burning, four volume percent of hydrogen should not be exceeded if more than five volume percent is present.[4] However, the percentage of hydrogen may increase to six volume percent under the assumption that the excess two volume percent hydrogen would partially burn in the containment given more than five volume percent oxygen were present. The assumptions under which these limits are applied are explained in the next section of this study.

3.2.2 Hydrogen Deflagration and Detonation

Hydrogen combustion can vary from separated flames that propagate upward, to coherent flames that propagate uniformly in all direction at sub-sonic velocities, to supersonic detonation waves.[8] Deflagration, or simple burning, can produce effects similar to those of explosions. Deflagration occurs as a chain reaction in which the principal carriers are the free radicals H, O, and OH. Ignition occurs in a hydrogen-oxygen mixture when the rate of production of the chain carriers exceeds the rate of their destruction.[6] Ignition can occur from sparks from electrical equipment or discharged accumulated static, or by temperature increases. Sparks can ignite a mixture below the flammability limit but the flames produced are not self-propagating and are extinguished when the source of ignition is removed. The spontaneous ignition temperature of a hydrogen-air mixture is 585°C although below this temperature, a self-propagating flame can be produced in a four volume percent mixture.[7]

The flame propagates at a velocity dependent upon its direction resulting from the tendency of the burned gas to rise, and from the hydrogen concentration. Mixtures with compositions close to the flammability limit will not burn all of the available hydrogen. As the proportion of hydrogen in the mixture increases, greater amounts of hydrogen are burned. For example, only half of the hydrogen in a 5.6 volume percent mixture will burn. Combustion will not be complete until the percentage of hydrogen is increased to 10 percent or more.

Detonation is a rapid and violent process characterized by a chemically supported shock wave. The velocity of wave propagation is the same as the velocity of sound in the burning mixture.[6] The destructiveness of a detonation is due primarily to the destruction of the shock front. Shapiro and Moffette show hydrogen detonation limits to occur between 19 and 45 volume percent (Fig. 3.5)[7]; hydrogen concentrations within this range will not necessarily detonate. Experiments have shown that a detonation is more likely to occur in tubes smaller than larger ones, and that a detonation wave can be converted to that of normal combustion by suddenly widening the tube. A strong initiating source is also required to produce detonation. The use of flames or sparks does not produce detonation.

3.2.3 Effects of Hydrogen Burning On BWR Containments

Post-accident hydrogen generation can threaten containment integrity if the hydrogen burns or detonates. Hydrogen burning or detonation may have a significant effect on the overall containment pressure. The pressure rise due to combustion of hydrogen can be predicted from the burning rate, which depends on the geometry of the vessel and velocity of the propagating flame. The maximum possible pressure rise in a closed vessel can be determined by assuming complete combustion of hydrogen with no heat losses to the vessel walls.[8] The combustion energy is absorbed by the mixture of combustion products. The overall energy balance is:

$$\Delta U = \overline{C}_v n_f (T_f - T_o) = n_o [H_2] \Delta u^o \tag{3.6}$$

where:

$[H_2]$ = mole fraction of hydrogen

n_o = total moles of initial mixture

T_o = initial temperature before combustion

Δu^o = combustion energy per mole of hydrogen

ΔU = internal energy difference

n_f = total moles of final mixture

C_v = average specific heat at constant volume

T_f = temperature of the final mixture

Assuming ideal gas behavior, the ratio of the final pressure P_f to the initial pressure P_o is:

$$\frac{P_f}{P_o} = \frac{n_f}{n_o} \frac{T_f}{T_o} \tag{3.7}$$

Solving for T_f from equation (3.6) and substituting into equation (3.7) gives the maximum pressure rise as:

$$\left(\frac{\Delta P}{P_o}\right)_{max} = \frac{P_f - P_o}{P_o} = \frac{\Delta u^o [H_2]}{\overline{C}_v T_o} + \frac{n_f}{n_o} - 1 \tag{3.8}$$

This result is plotted against the initial percentage of hydrogen for initial water vapor concentrations (Fig. 3.7). This model can be used to predict the pressure transients associated with burning of various concentrations. The pressure transients in a Mark I drywell for hydrogen concentrations of up to 18 volume percent is shown in Fig. 3.8.

3.3 Methods for Hydrogen Control in Boiling Water Reactors

Several systems have been proposed to control flammable hydrogen-oxygen mixtures. Systems are designed to maintain hydrogen produced in metal-water reactions below the flammability limits established by the regulatory guides (four volume percent hydrogen concentration and five percent volume oxygen).[4,14] Most Mark I and Mark II containments' atmospheres are required by the Nuclear Regulatory Commission to be inerted with nitrogen (the exceptions are the Vermont Yankee and Hatch-2 units).[3] Other methods include combinations of air dilution systems, recombiners and/or controlled venting. In order to understand the differences between BWR containment design, these designs are now described, followed by detailed descriptions of hydrogen control systems.

3.3.1 BWR Containments

Boiling Water Reactor containments have been designed using the pressure suppression concept. Three basic types have evolved, starting with the bulb-shaped Mark I, evolving to the conical-shaped Mark II, and ending finally with the multibarrier pressure containment type Mark III design.[35] The Mark I primary containment consists of a drywell, a pressure suppression chamber, and a connecting vent system between the drywell and the suppression chamber. In the event of a pipe break within the drywell, reactor water and

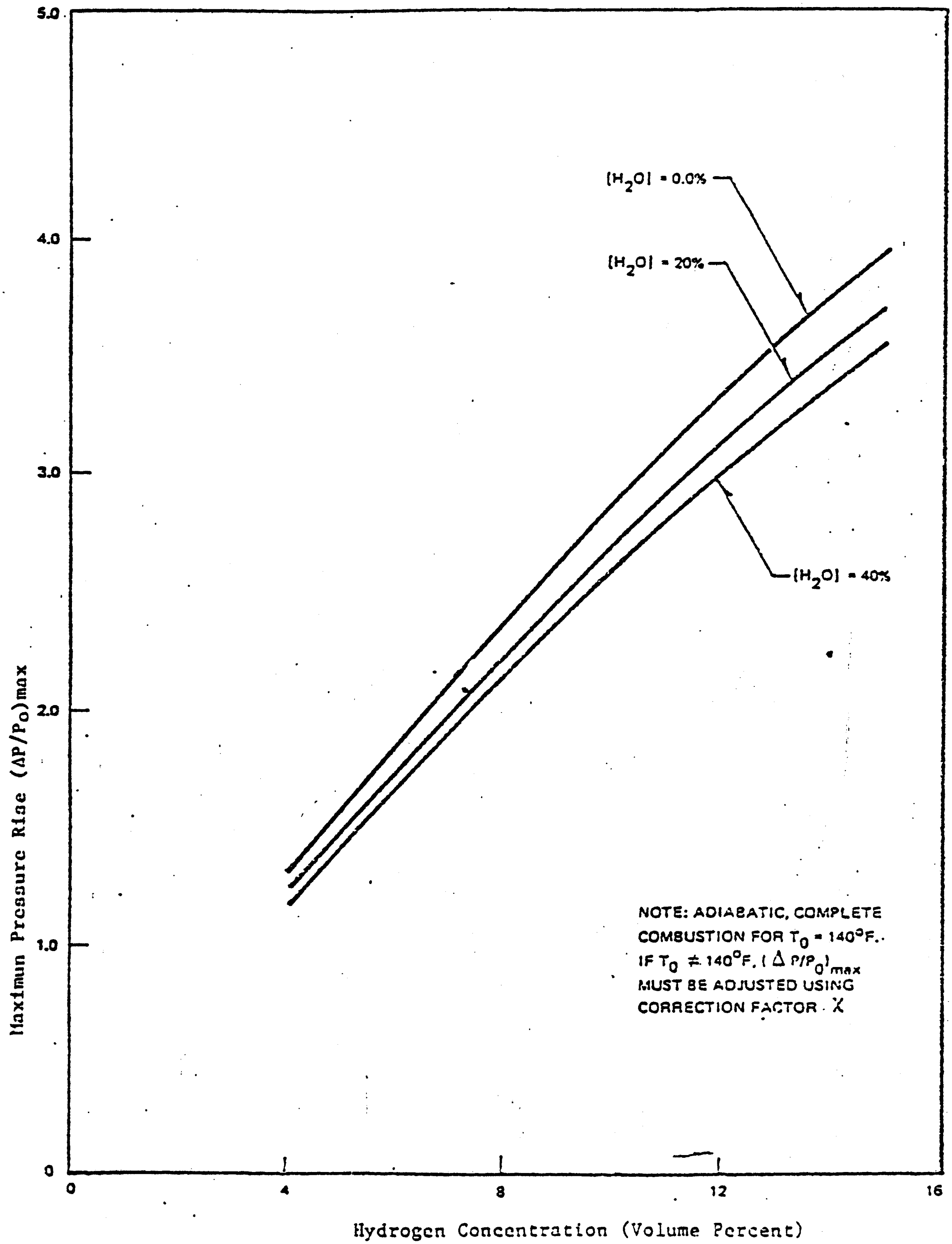


Figure 3.7 Final Pressure vs. Hydrogen Concentration

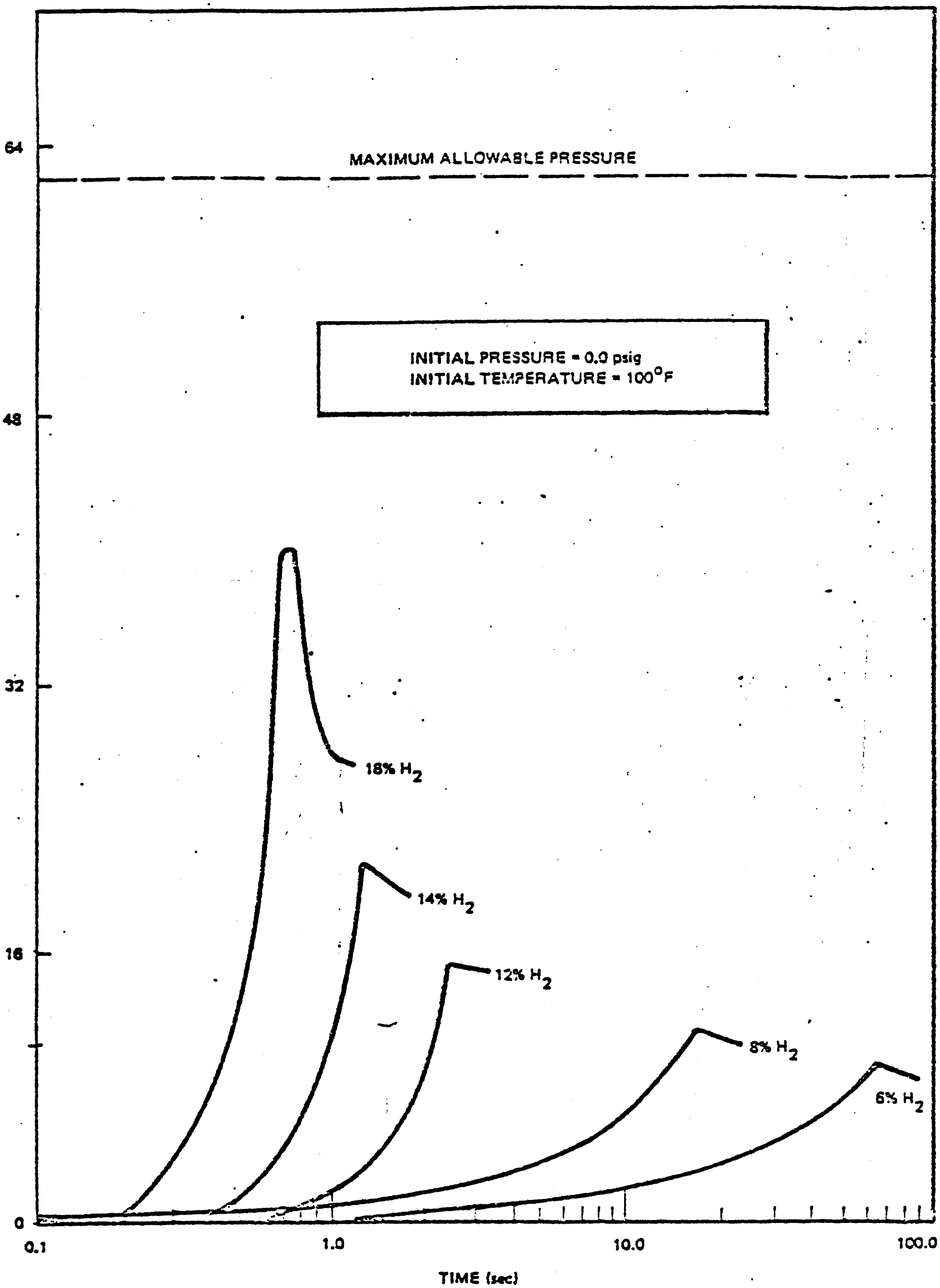


Figure 3.8 Pressure for Hydrogen Combustion in Mark I Drywell[8]

Table 3.2

CHARACTERISTICS OF LWR CONTAINMENTS

CONTAINMENT TYPE		DESIGNER	MATERIAL	DESIGN PRESSURE (PSIA)	VOLUME (FT^3)
DRY		BECHTEL, DUKE PWR.	STEEL	55-62	2.5×10^6
		BECHTEL, EBASCO	CONCRETE (R)	70	2.5×10^6
		BECHTEL	CONCRETE (P)	70	2.0×10^6
SUBATMOSPHERIC		STONE & WEBSTER	CONCRETE (R)	60	2.3×10^6
ICE CONDENSER		WESTINGHOUSE	CONCRETE (R)	27	1.25×10^6
PRESSURE SUPPRESSION					
MARK I	(DRYWELL)	GE	STEEL	74	159,000
	(WETWELL)		STEEL	74	204,000
MARK II	(DRYWELL)	GE	CONCRETE (R)	65	184,000
	(WETWELL)		W/STEEL LINER	65	209,000
MARK III	(DRYWELL)	GE	CONCRETE (R)	45	280,000
	(WETWELL)		CONCRETE (R) W/STEEL LINER	30	1.5×10^6

steam will be released into the drywell. The resulting increase in pressure will force a mixture of steam, water and air into the suppression chamber via the vents.[33]

The drywell is a steel pressure vessel with a spherical lower portion about 62 feet in diameter and a cylindrical upper portion 33 feet in diameter. The drywell houses the reactor vessel, reactor coolant recirculation system, and other pipes related to the cooling system. The drywell is enclosed in four to five foot thick reinforced concrete for shielding purposes. Design pressure and volume of typical BWR containments are shown in Table 3.2 for comparative purposes.

The pressure suppression chamber is a steel pressure vessel in the shape of a torus that encircles the base of the drywell. The torus houses approximately $78x10^3$ cubic feet of water and has a net air space of approximately $108x10^3$ cubic feet. The Mark I containment is located inside the reactor building. This building forms part of the secondary containment system, along with the standby gas treatment system (SGTS) and other auxiliary equipment.

The Mark II primary containment consists of a drywell, pressure suppression chamber, connecting vents, a venting and vacuum relief system, containment cooling systems and other service equipment.[34] The drywell forms a truncated cone, and the cylindrical pressure suppression chamber is immediately below. These two units comprise a structurally integrated reinforced concrete pressure vessel. The drywell and the suppression pool are separated by a reinforced concrete floor. The primary containment is structurally separated from the surrounding reactor building. Design pressure and volume of the Mark II primary containment are shown in Table 3.2.

The Mark III primary containment is a steel cylinder with torispherical head. The drywell is a cylindrical structure with reinforced walls and roof.[35] The functions of the drywell are to provide shielding to reduce radiation levels in the containment to levels which permit normal access, provide structure to support the upper pool, and to channel the steam release from a loss of coolant accident through horizontal vents for condensation in the suppression pool. The suppression pool is a 360° annular pool located between the weir wall inside the drywell, and the containment wall on the bottom floor of the reactor building. The suppression pool provides a heat sink for safety relief valve operation, a heat sink for hot-standby operation, a means to condense steam released in the drywell during a LOCA, and a source of water for the emergency core cooling system. The entire volume of the containment is open to the suppression pool. Design pressure and volume of the Mark III primary containment are shown in Table 3.2.

The Mark III containment provides a number of advantages over the Mark I and Mark II containments; Fig. 3.9 shows a comparison of the size and form of the three containment types. The advantages are reduced overall reactor building height, improved seismic response, improved accessibility for installation and inspection of nuclear boiler piping and equipment, and improved pipe whip protection.[35] However, the lower containment design pressure makes the Mark III containment more vulnerable to hydrogen burning.

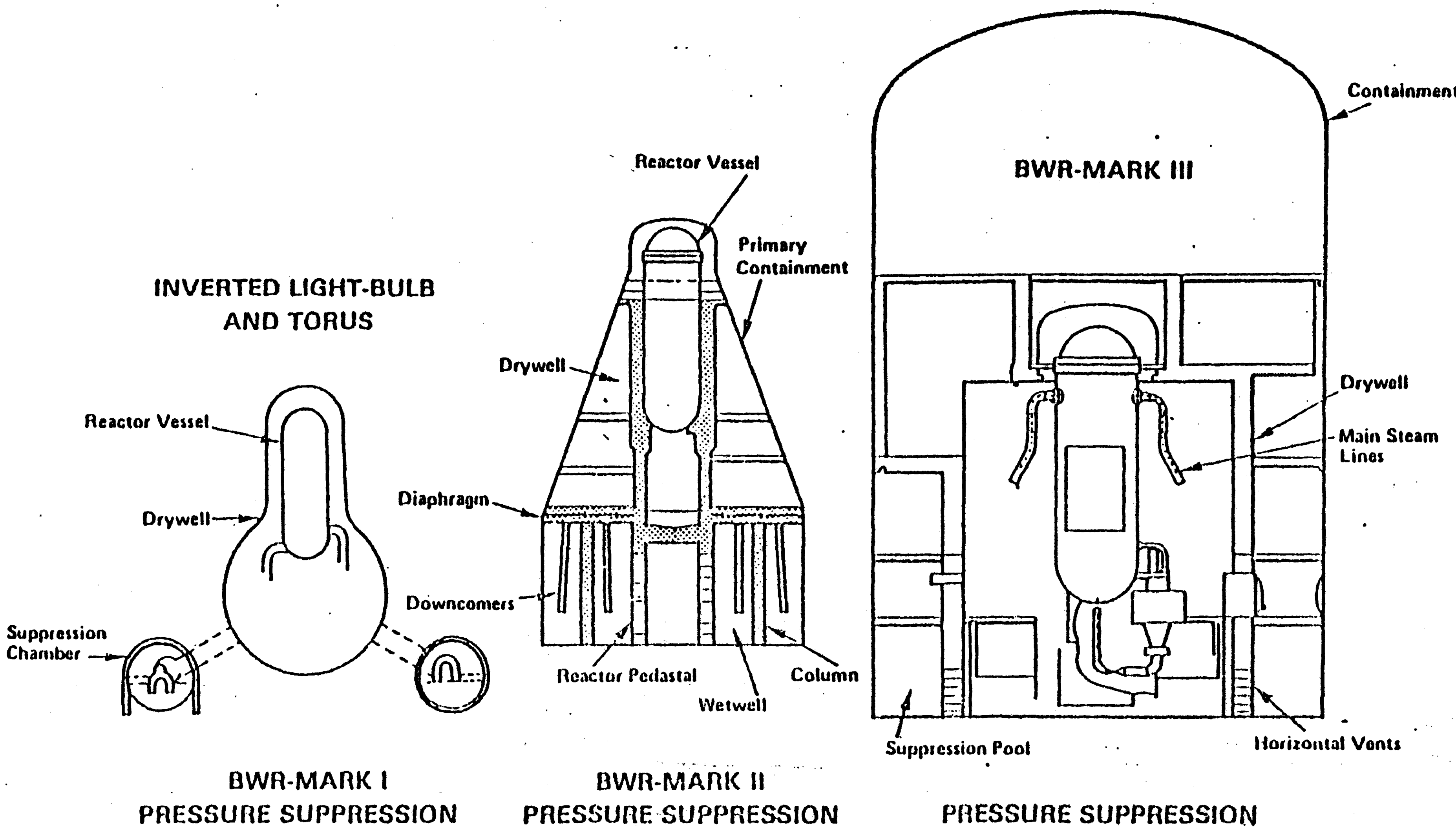


Figure 3.9 Comparison between BWR Containments

3.3.2 The Containment Inerting System (CIS)

Containment inerting consists of purging the containment atmosphere with nitrogen until the oxygen concentration is below five volume percent; the reactor is then operated under those conditions. If the hydrogen concentration is kept below this limit, hydrogen generated is unable to burn or explode. In the event of hydrogen generation during an accident, a nitrogen make-up system is activated to help reduce the hydrogen concentration to four volume percent and maintain the oxygen concentration below five volume percent. Controlled venting through the standby gas treatment system (SGTS) is provided to reduce the pressure inside the containment.[12] The containment inerting system (CIS) at Pilgrim nuclear power station is analyzed in detail in Chapter 4.

3.3.3 The Containment Air Dilution System (CAD)

In the containment air dilution system (CAD), the atmosphere in the containment is diluted with air during or after an accident.[9] In this system, hydrogen concentration becomes the parameter of concern. System design is based on the requirement that the containment atmosphere be maintained below four volume percent hydrogen in the event of an accident. The system monitors the hydrogen gas concentration and injects additional air as required to dilute the hydrogen and maintain it below the flammability limit. Controlled venting is manually initiated when, during an accident, the pressure reaches half the drywell design pressure of 28 psig.[11] The CAD system in use at the Vermont Yankee nuclear power station is analyzed in detail in Chapter 4.

3.3.4 Controlled Venting

Venting of the containment atmosphere occurs only if after an accident, the hydrogen concentration approaches four volume percent. Venting times are designed on the basis of dose acceptability.[10] Fission product releases are minimized by passing the vented gas through chemical scrubbers or charcoal filters in the standy gas treatment system. However, control of noble gas radioactivity under venting conditions is very difficult.[6]

3.3.5 Recombiners

If venting is deemed unacceptable, there are a variety of non-venting recombiner schemes available. Chemical recombination of hydrogen is a way to prevent hydrogen burning and at the same time control increases in hydrogen pressure. Applied to BWRs, recombiners would need to be more complex and expensive, requiring a supplementary oxygen supply to consume all the hydrogen that might be produced. Recombiners can be classified into flame, catalytic and electrical types.[6] The principal disadvantages of recombiners is the possibility of extinguishing the flame and having it "flash back" through the injector. Catalytic recombiners use a catalytic bed that dilutes the gas mixture below the flammability limits and are now in use in PWRs. Recent designs include nickel and nickel-chromiun oxide combinations supported on aluminum-oxide bases and platinized honeycomb ceramic disks.[6] Disadvantages include choice of diluent, condensing or non-condensing reactions, catalyst, preheat temperature, pressure-drop specifications, vessel materials and number of recombining stages. Electric recombiners use electric resistance heaters heat the continuous flow of containment atmosphere to above the hydrogen-oxygen reaction temperature (Fig. 3.10). A comparative analysis of the air dilution system and the inerting system is made in the next chapter in order to find out the influence on the probability of containment failure due to post-accident hydrogen generation.

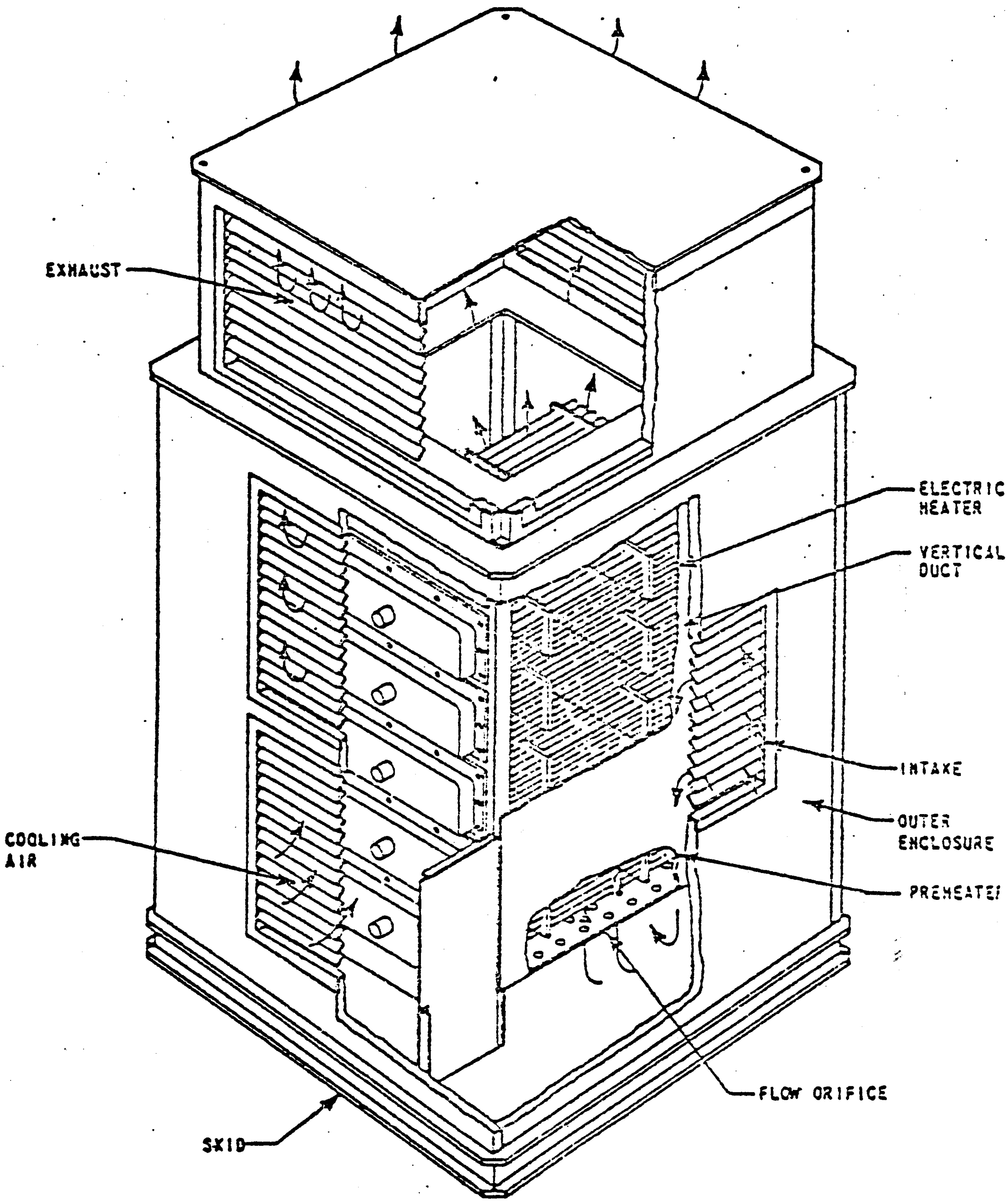


Figure 3.10 Electric Hydrogen Recombiner

CHAPTER FOUR

QUANTIFICATION OF THE PROBABILITY OF CONTROLLING POST-ACCIDENT HYDROGEN IN BWRs

In order to assess the overall probability that the CAD or the CIS systems are capable of handling a given amount of hydrogen generated during an accident, a set of probabilities need to be calculated.

4.1 Overview of the Probabilistic Framework of Analysis

A hydrogen related event tree is developed in order to assess the sequence of events that can lead to the uncovering of the core and the production of hydrogen, perhaps eventually leading to meltdown (see section 4.2). A fault tree analysis is used here to calculate the probabilities of failure on demand ($P_f(S)$) of the CAD and CIS hydrogen control systems (see section 4.3). Using probabilities of failure of each system, the probability that the system is available to work ($P_{CAD}(S)$ and $P_{CIS}(S)$) are defined as follows:

$$P_{CAD}(S) = 1 - P_f(S) \qquad (4.1)$$

$$P_{CIS}(S) = 1 - P_f(S) \qquad (4.2)$$

The next step in the analysis is to calculate the probability of hydrogen generation (or percent metal-water reaction) given that an accident occurs (P(A)) (Fig. 4.1). From WASH-1400, large LOCA accidents in BWRs have a probability of producing a core melt of $\sim 3\times10^{-5}$/ reactor year.[45] For these accidents, it is assumed that all the zirconium reacts with water to produce hydrogen. For small accidents with probabilities in the range of 3×10^{-3}/ reactor-year, it is assumed that the metal-water reaction linearly decreases from about 100% to almost zero and remains zero over the

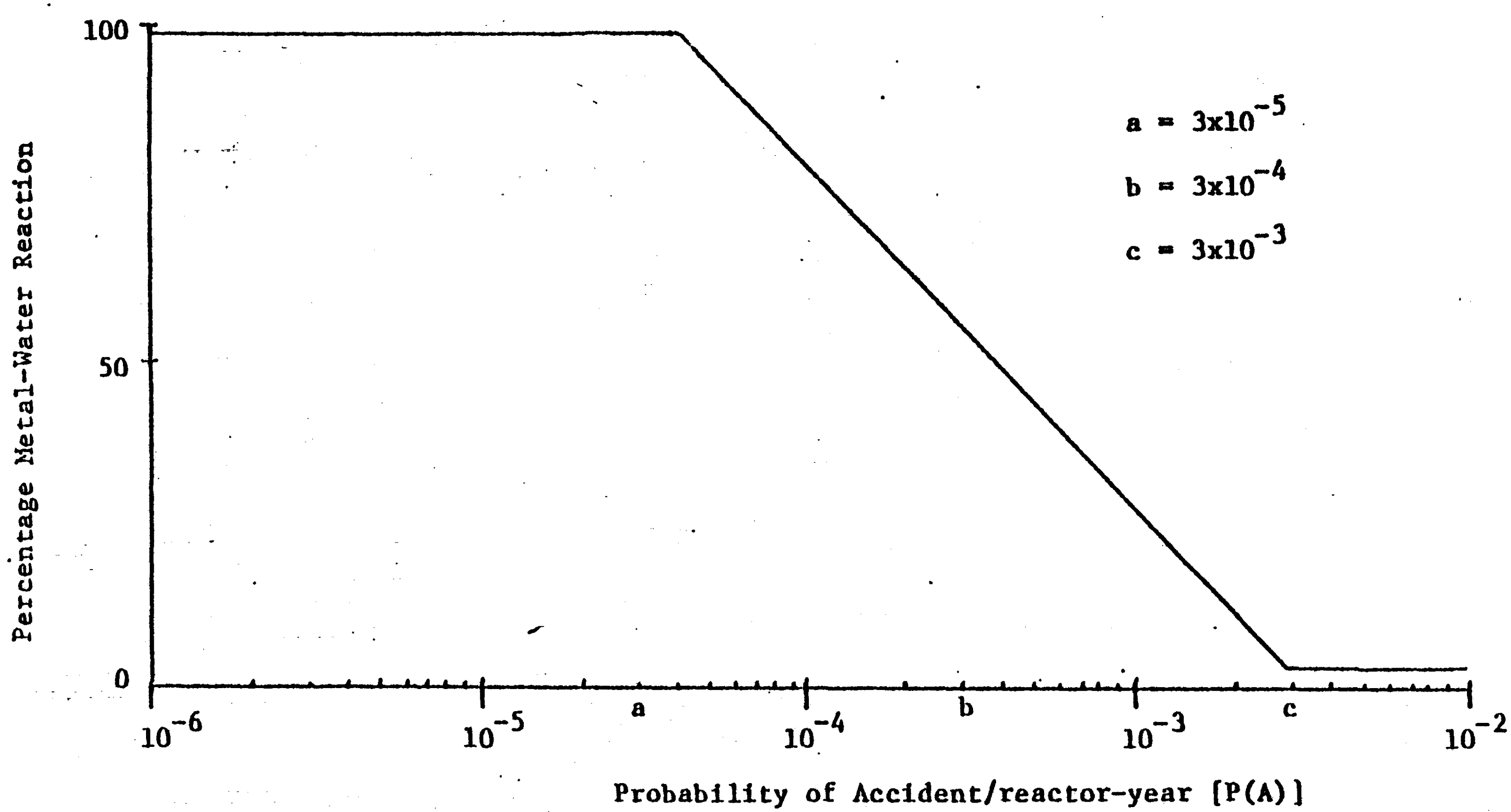


Figure 4.1 Percentage Metal-Water Reaction vs. Probability of Accident/reactor-year

range of the higher probability yet less serious accidents.

The hydrogen generation rate is important because it affects the probability that the control system is physically capable of maintaining the combustible mixture below the flammability limits. The maximum hydrogen rate physically achievable over an hour is calculated from core heatup calculations. Fig. 3.1 shows the amount of cladding that reacts as a function of time derived from comparing different heatup curves.[44] The figure shows a 10% cladding reaction per every 1000 seconds. Extrapolating this curve up to 100%, all the zirconium is consumed in about three hours (Fig. 3.2). This is a very conservative assumption since the amount of steam decreases with time according to the steam availability curve used in the above calculations (Fig. 3.3). However, this value is used here as an upper bound on the hydrogen generation rate.

The maximum amount of hydrogen produced by a metal-water reaction is shown in Fig. 3.4. For a 100% metal-water reaction, the maximum volume percent in a BWR Mark I containment is 72%. Using the same figure, the percent metal-water reaction required to have four volume percent hydrogen concentration achieved in four or five minutes, implying a generation rate between 144×10^3 and 180×10^3 cubic feet per hour. These values are the upper bound of the generation rate plotted in Fig. 4.2. The accident at Three Mile Island generated hydrogen at approximately 100×10^3 cubic feet per hour (Fig. 4.2).

The CAD system is designed to work when the hydrogen

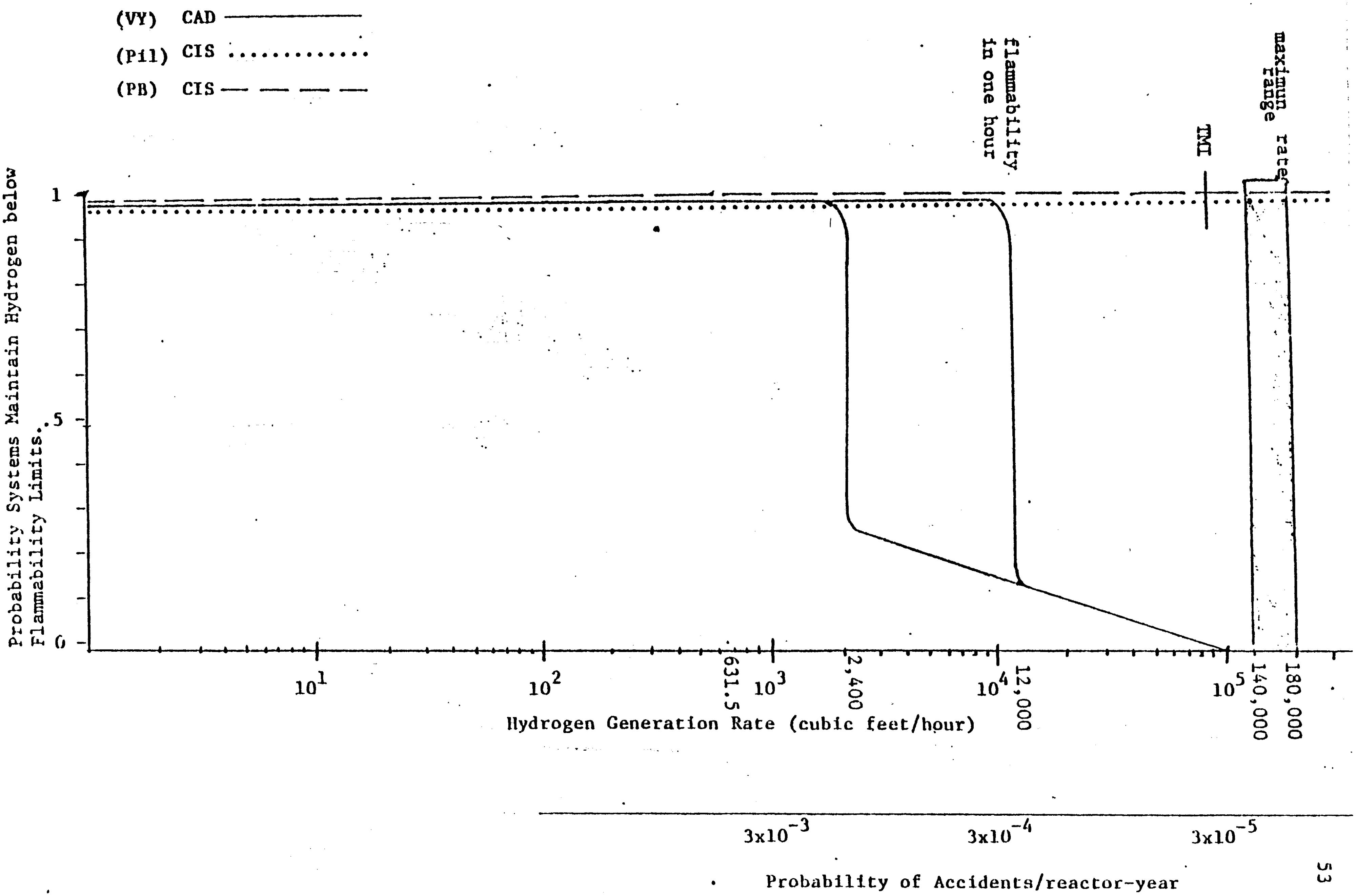


Figure 4.2 Probability of CAD and CIS Systems to Control Hydrogen vs. Hydrogen Generation Rate

concentration reaches four volume percent which in the design basis accident occurs in approximately nineteen hours. If a generation rate of 631.5 cubic feet/hour is assumed, up to this point the probability of success of the CAD is the one from equation 4.1 ($P_f(S)$ from fault tree analysis). During normal operations, the CAD system pressurizes the containment to reduce the hydrogen concentration, and then vents through the Standby Gas Treatment System (SGTS) to reduce the pressure. (The maximum venting rate is 2400 cubic feet/ hour). If a generation rate reaches four volume percent in one hour (i.e. 12,000 cubic feet/hour), this corresponds to a probability of accident of $3x10^{-4}$/reactor year. As the hydrogen generation rate increases, the probability of accident decreases. For these low probability accidents, the probability of the CAD system being able to handle high H_2 rates drops almost to zero. For the CIS, the probability of controlling the hydrogen remains about constant since having no oxygen to react, the hydrogen would not burn. (This assumes that the reactor is inerted during operations). However, during the 24 hour period prior to shutdown and after startup, the reactor is not inerted, which means that the probability of having a combustible mixture is decreased during this time because the oxygen concentration is above five volume percent.

4.2 Hydrogen Related Event Tree

The design basis LOCA in a BWR is defined as a double-ended rupture of the primary coolant recirculation line (WASH-1400)[5]. A small LOCA is defined as a break in the cooling system of about 1/2 to 2 inches in diameter. The sequences of events for both large and small LOCAs is very similar; the differences are in the emergency coolant injection requirements. The event tree that is developed here is a reduced event tree with emphasis on those sequences that lead to hydrogen generation and eventually to failure of the containment due to hydrogen overpressurization (Fig. 4.3).

In this study, the initiating event is assumed a random rupture in the reactor coolant system. The next event in the sequence is failure of the electric power followed by failure of the reactor protection system that provides the reactor trip in case of an accident. The next event is failure of the vapor suppression system. If the vapor suppression system fails, the primary containment fails due to overpressurization. The next event is failure of the emergency coolant recirculation systems. Failure of these systems would leave the core uncovered long enough to produce significant amounts of hydrogen.

A separate event tree is developed showing the sequence of events required to handle the hydrogen. Fig. 4.4 shows the event tree for the containment air dilution system and Fig. 4.5 shows the event tree for the containment inerting system. Both trees are basically the same; the difference is in the probability of both systems to handle the hydrogen generated. The first column in the event tree is the control system

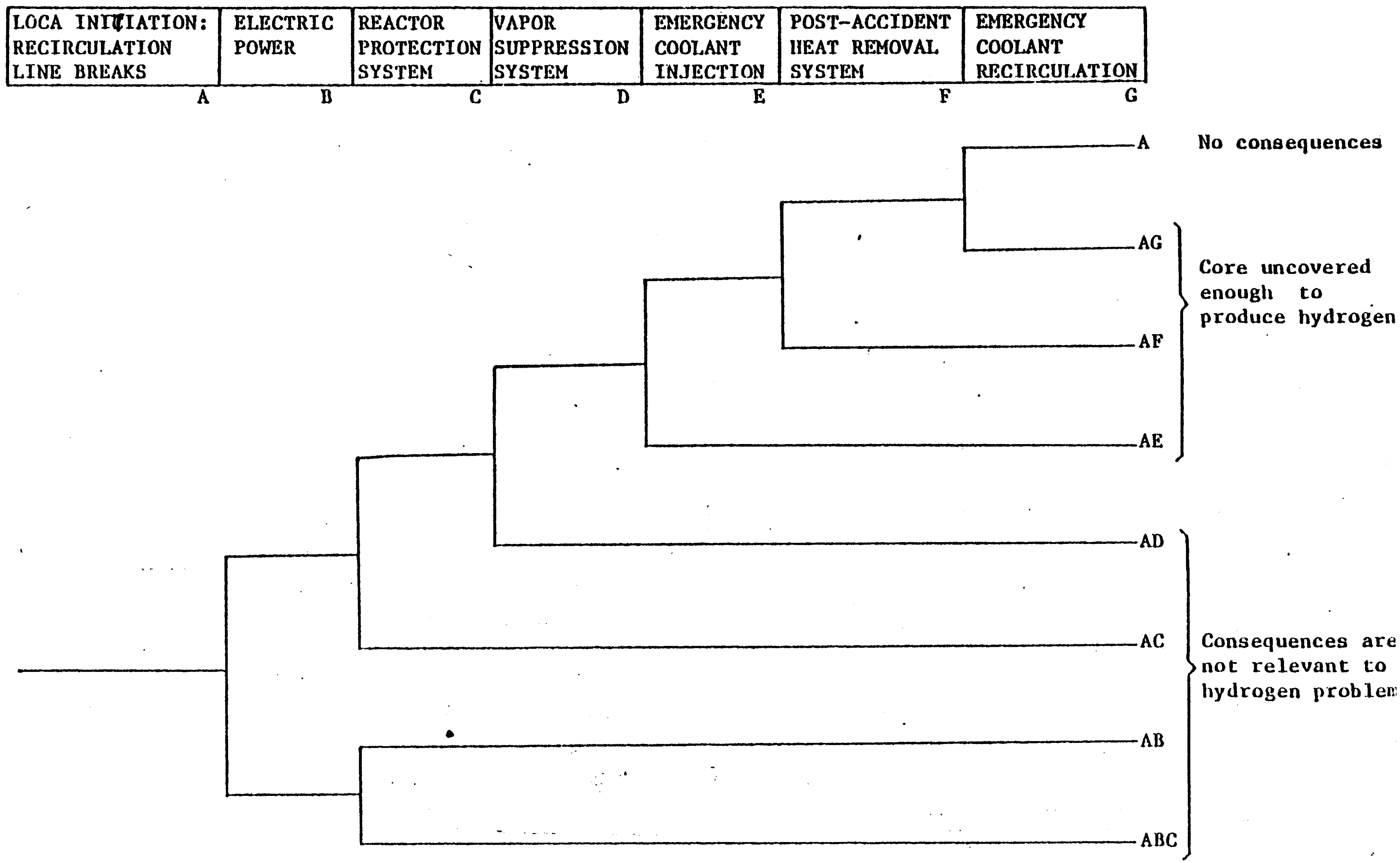


Figure 4.3 Hydrogen Related BWR LOCA Event Tree

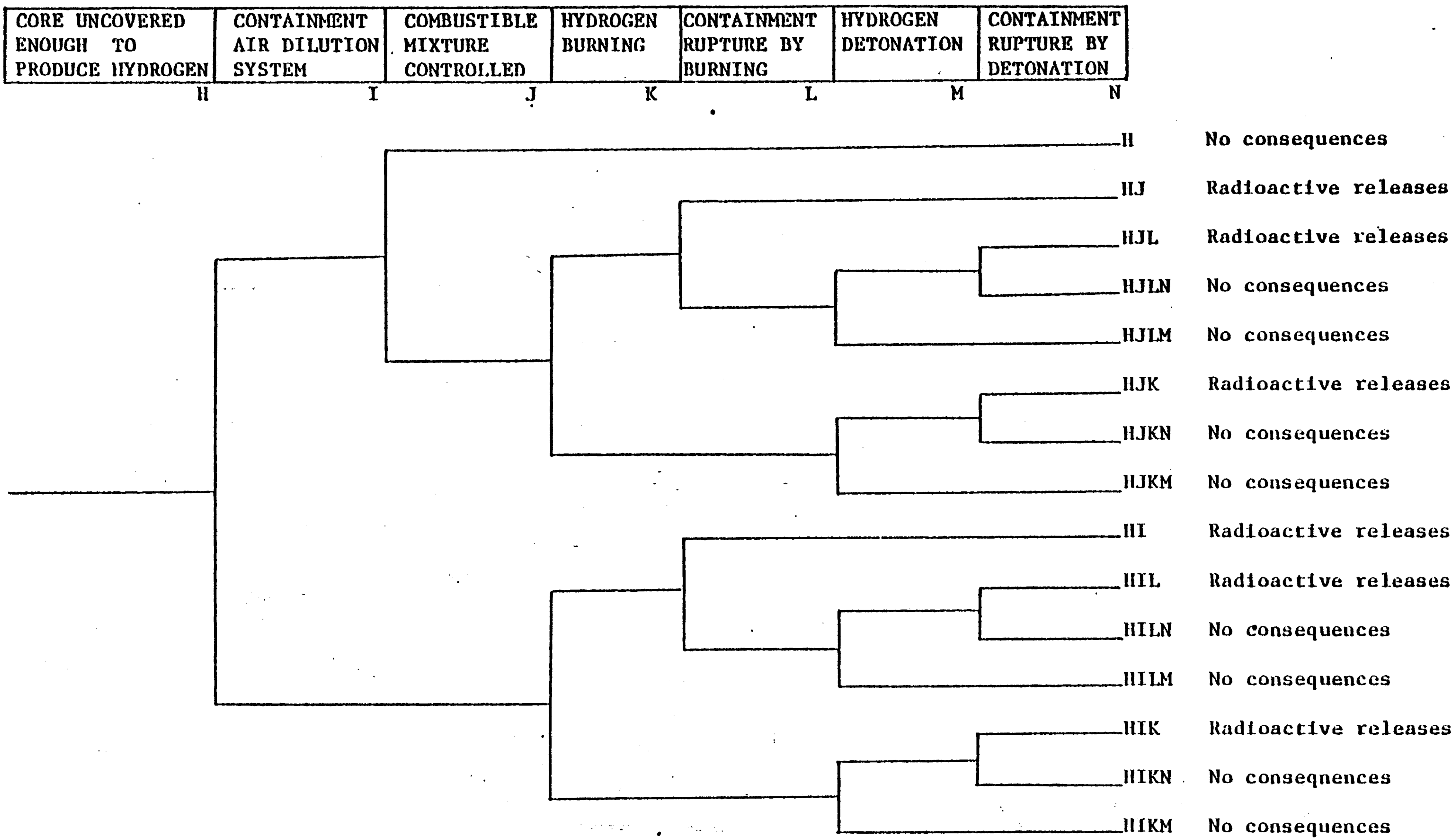


Figure 4.4 Hydrogen Control Event Tree (Containment Air Dilution System)

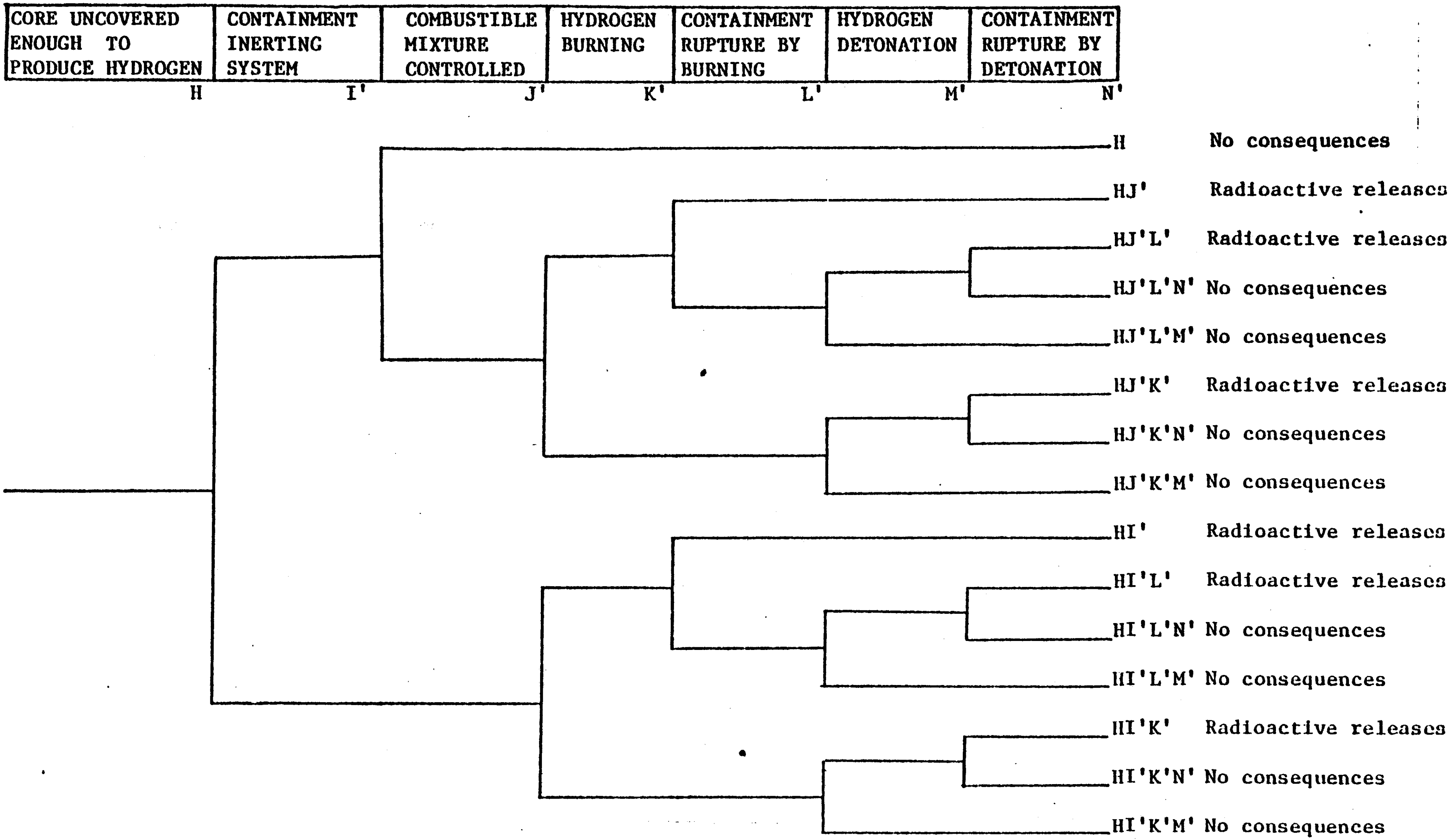


Figure 4.5 Hydrogen Control Event Tree (Containment Inerting System)

(CAD or CIS); if this system is not used or fails to work, it is assumed that the hydrogen is not controlled. If the system works and is used, the possibility exists that the system can control the gas mixture. If the combustible mixture is not controllable, the next event in the sequence will be hydrogen burning. The following sequences apply to both branches (use of the system or not). If hydrogen burns, the next sequence is containment failure due to over pressurization by hydrogen burning. If there is no rupture or no hydrogen burning, the hydrogen concentration could increase to the detonation limits (20 percent) and explode. The final event is containment rupture by detonation. Fig. 4.4 and Fig. 4.5 show the sequence of events that could lead to radioactive releases in the case of containment failure.

Assuming the combustible mixture is controlled and there is no containment failure, the core could remain uncovered increasing the rate of hydrogen production building up radiation. Fig. 4.6 shows the event tree related to the uncovered core. If the core continues to stay uncovered, eventually it will start to melt and other events will dominate the hydrogen problem (i.e. steam explosions, etc.). The different stages in Fig. 4.6 affect the probabilities of the hydrogen control event trees (Fig. 4.4 and Fig. 4.5). In order to assess the probability that the CAD or CIS systems can handle hydrogen generated in an accident, a detailed analysis of both systems is next attempted.

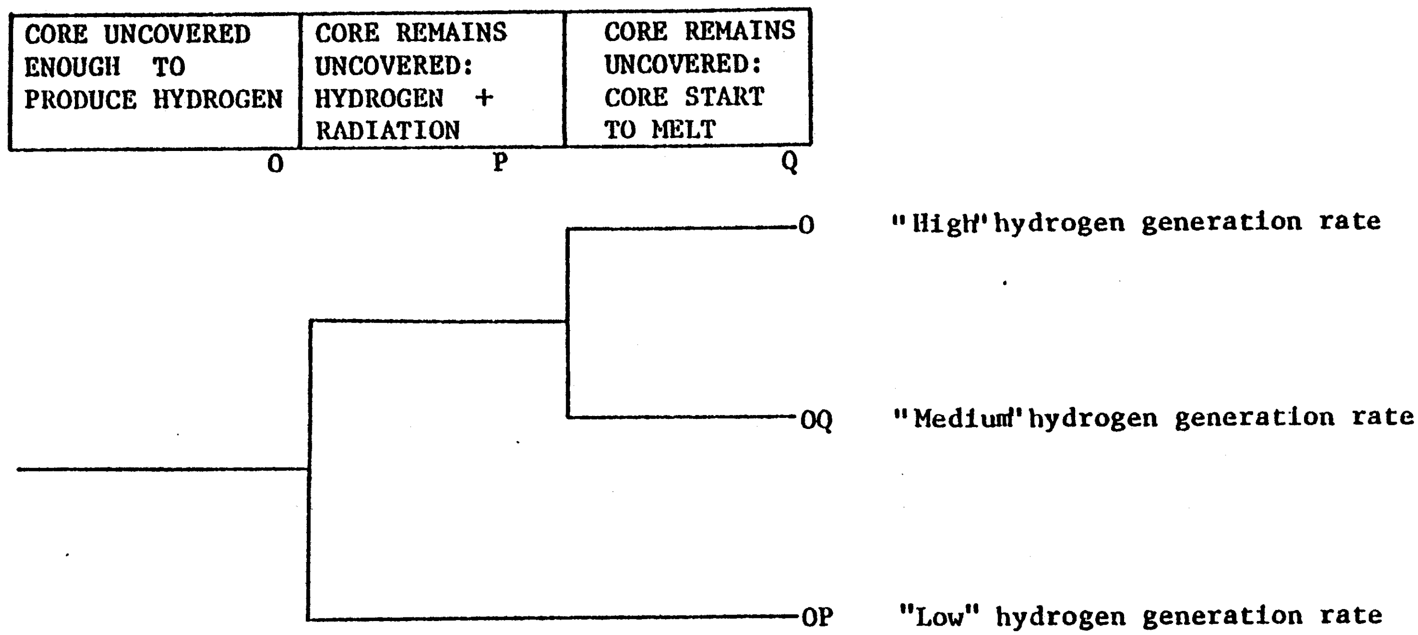


Figure 4.6 Uncovered Core Event Tree

4.3 Fault Tree Analysis of Hydrogen Control System

4.3.1 Analysis of the Containment Air Dilution System

The purpose of this analysis is to identify and evaluate the potential failure modes of the CAD system shown in Fig. 4.7. The weak points of the system are examined. Failure probabilities associated with the CAD system were derived using data supplied by Vermont Yankee and the failure data used in WASH-1400.[31,32] Fault trees are used in this analysis because they provide a convenient and efficient method for the computation of system failure probability and also lead to discovery of all possible failure combinations. The failure probability of the CAD system is used in the calculation of the overall probability of the control system to handle hydrogen. Any other information from the fault tree analysis is useful in evaluating the weak points of the system, permitting redesign to improve system reliability. In order to understand the development of the fault tree, the important aspects of the CAD system are now described.

The CAD system is designed to limit hydrogen concentration in the containment to less than four volume percent following a LOCA.[11] The CAD system consists of three systems: a sampling subsystem, air injection subsystem, and a venting subsystem. A schematic diagram of the sample subsystem is shown in Fig. 4.8. It consists of two hydrogen analyzer cells, redundant air pumps, an air-to-air heat exchanger, pipes and valves. These components are connected to the drywell and torus at four different sampling points including a common return line to the torus.[11] The primary analyzer is located inside the reactor building and is remotely

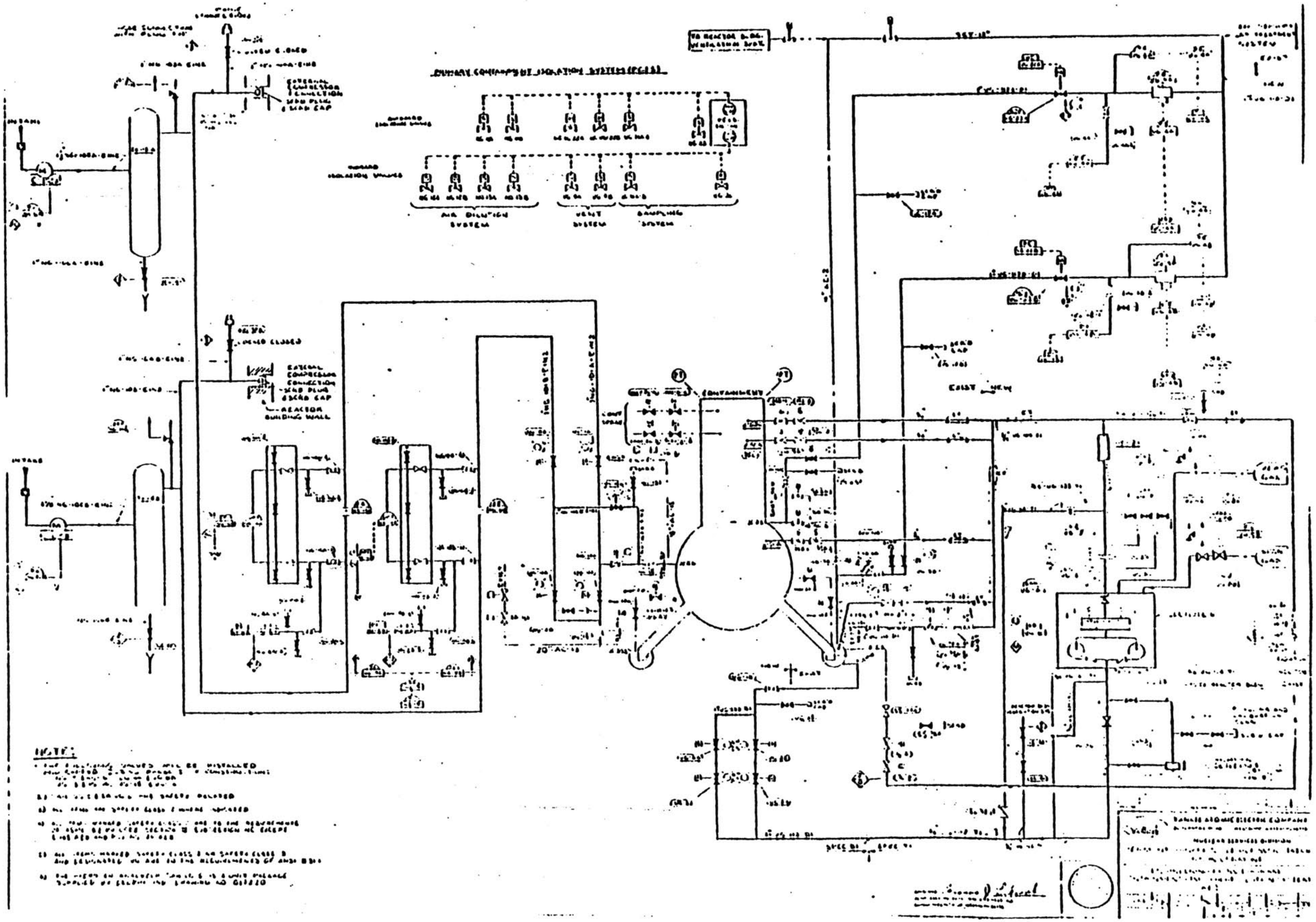


Figure 4.7 Diagram of the Containment Air Dilution System

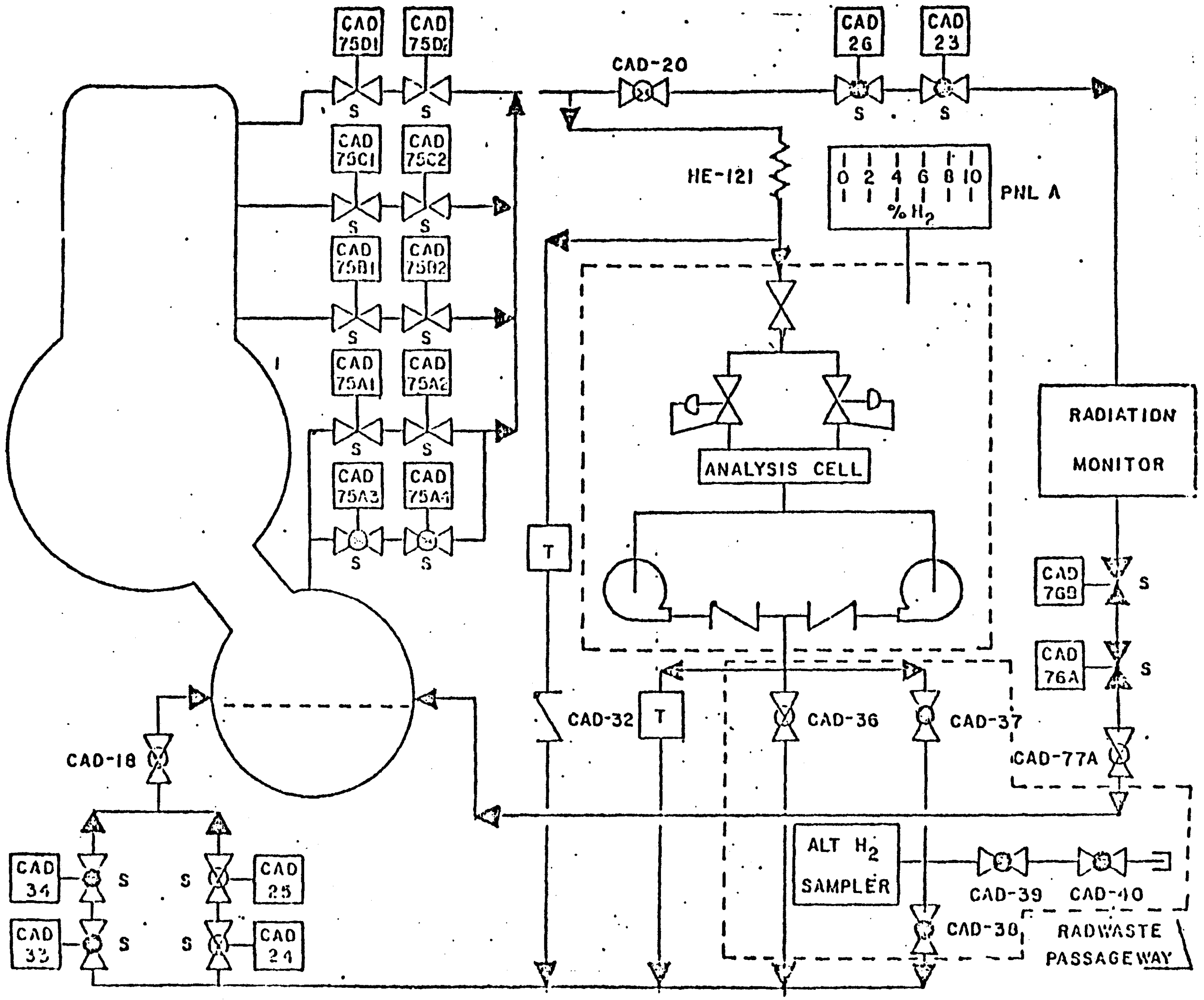


Figure 4.8 Diagram of the Sample Subsystem (Containment Air Dilution System)

operated from a read-out from the CAD system panel located in the back of the control room. The floor arrangement of a typical BWR control room is shown in Fig. 4.9, indicating the location of the CAD system panels. The CAD system panel A is shown in Fig. 4.10. The read-out from the primary analyzer is given in percent hydrogen concentration in the containment. A redundant analyzer is mounted on a wall outside the reactor building in an area which is accessible following a LOCA[11]. The read-out is expressed in "percent LEV" and the start mechanism is located on this wall at such a position that it has to be reached using a portable ladder that is reclined against the wall.

The heat exchanger, a passive component, conditions the sample for analysis. The sample flow force is provided by two redundant air pumps for both hydrogen analyzers. The accuracy of the analyzers is provided by manual calibration equipment. The valves of the sample supply and sample return are controlled from the CAD panels. These are normally open, and if they do not receive a Primary Containment Isolation Signal (PCIS), they fail close on loss of power. These valves are used only to select an appropriate sample point. A second line leads to a radiation monitor which is isolated fron the subsystem and the containment by solenoid valves that close on receipt of a PCIS signal or loss of power.

A schematic diagram of the injection subsystem is shown in Fig. 4.11. It consists of two completely independent injection

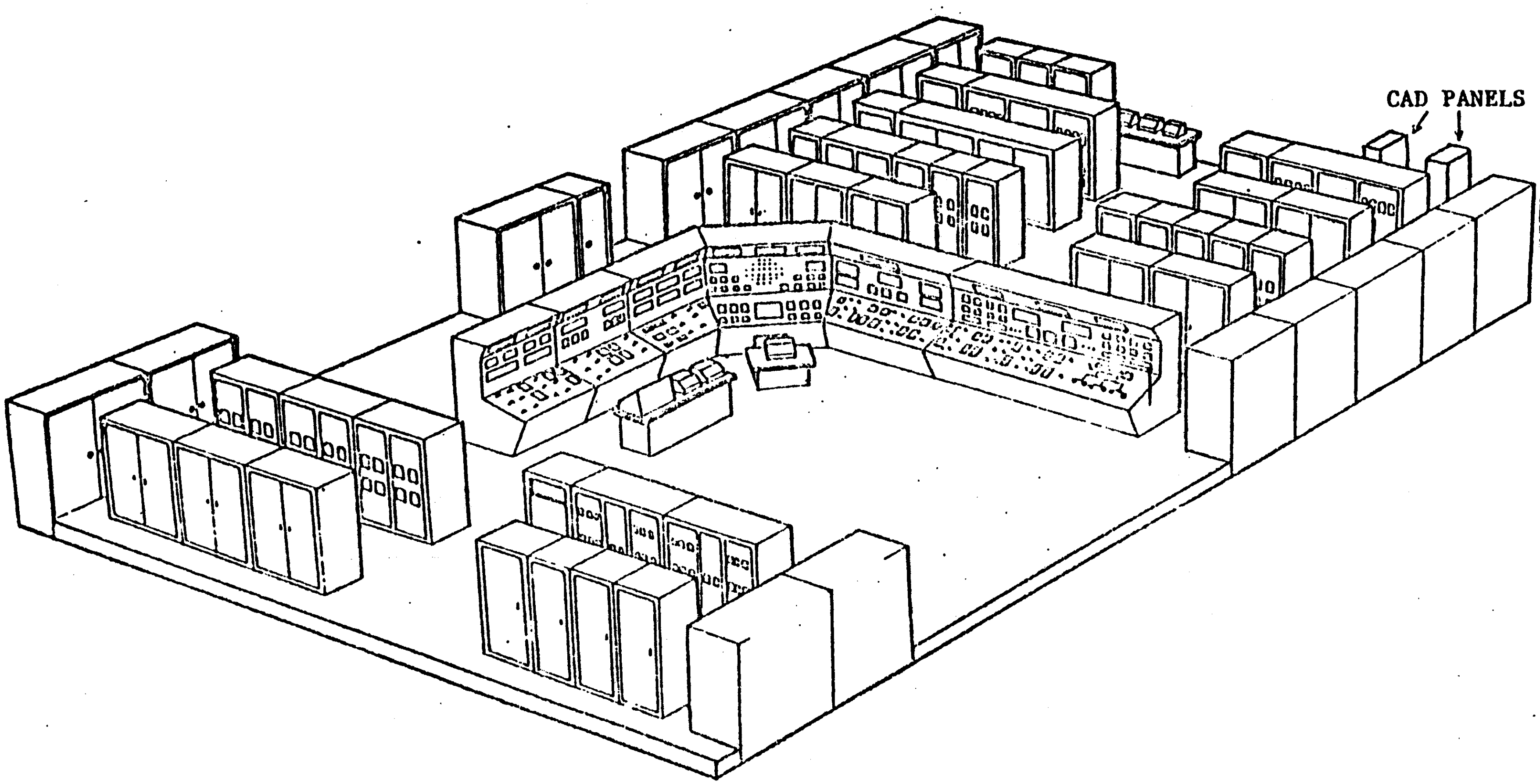


Figure 4.9 Floor Arrangement of a Typical BWR Control Room with Location of CAD System Panels Indicated

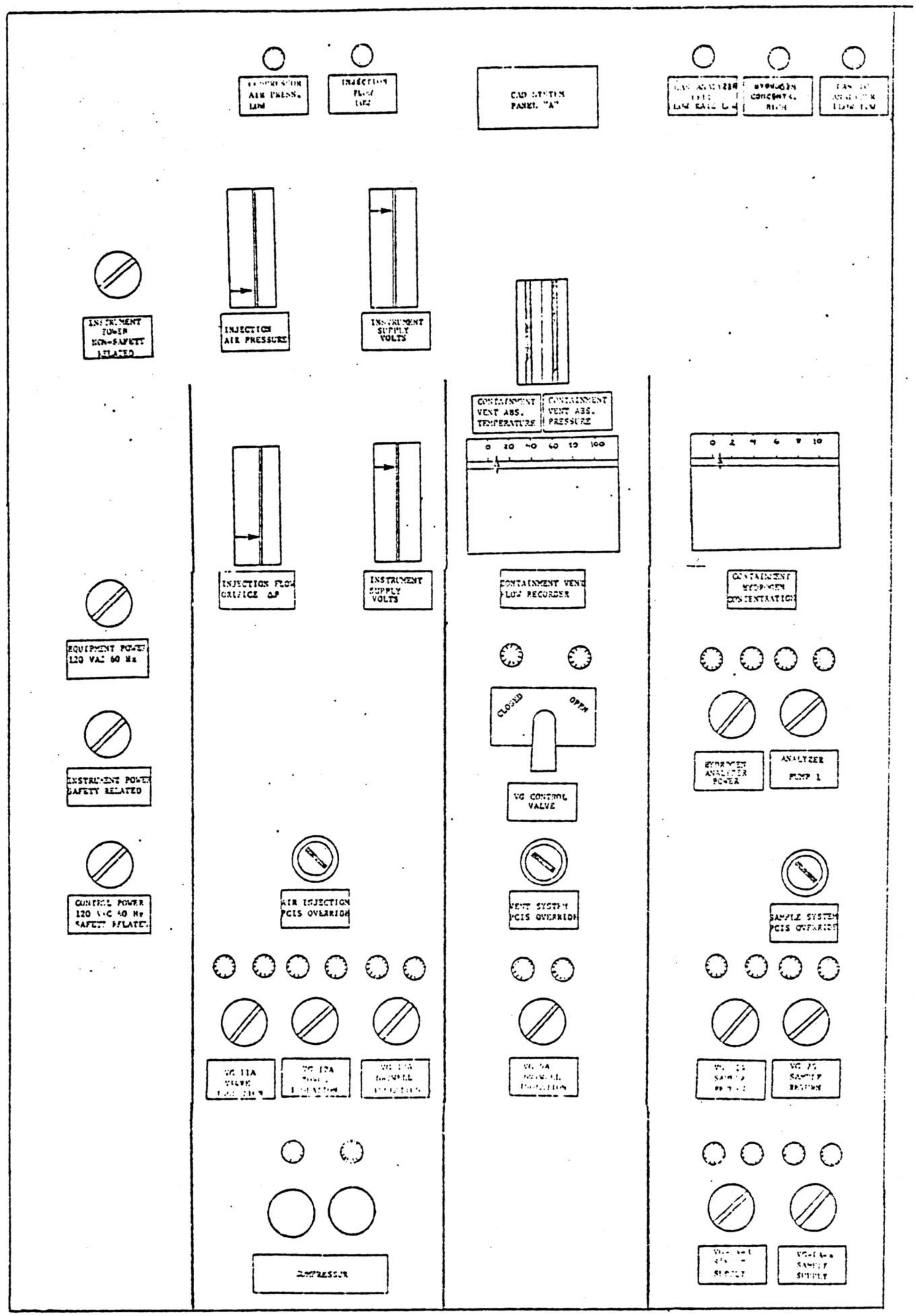


Figure 4.10 Containment Air Dilution System Control Panel "A"

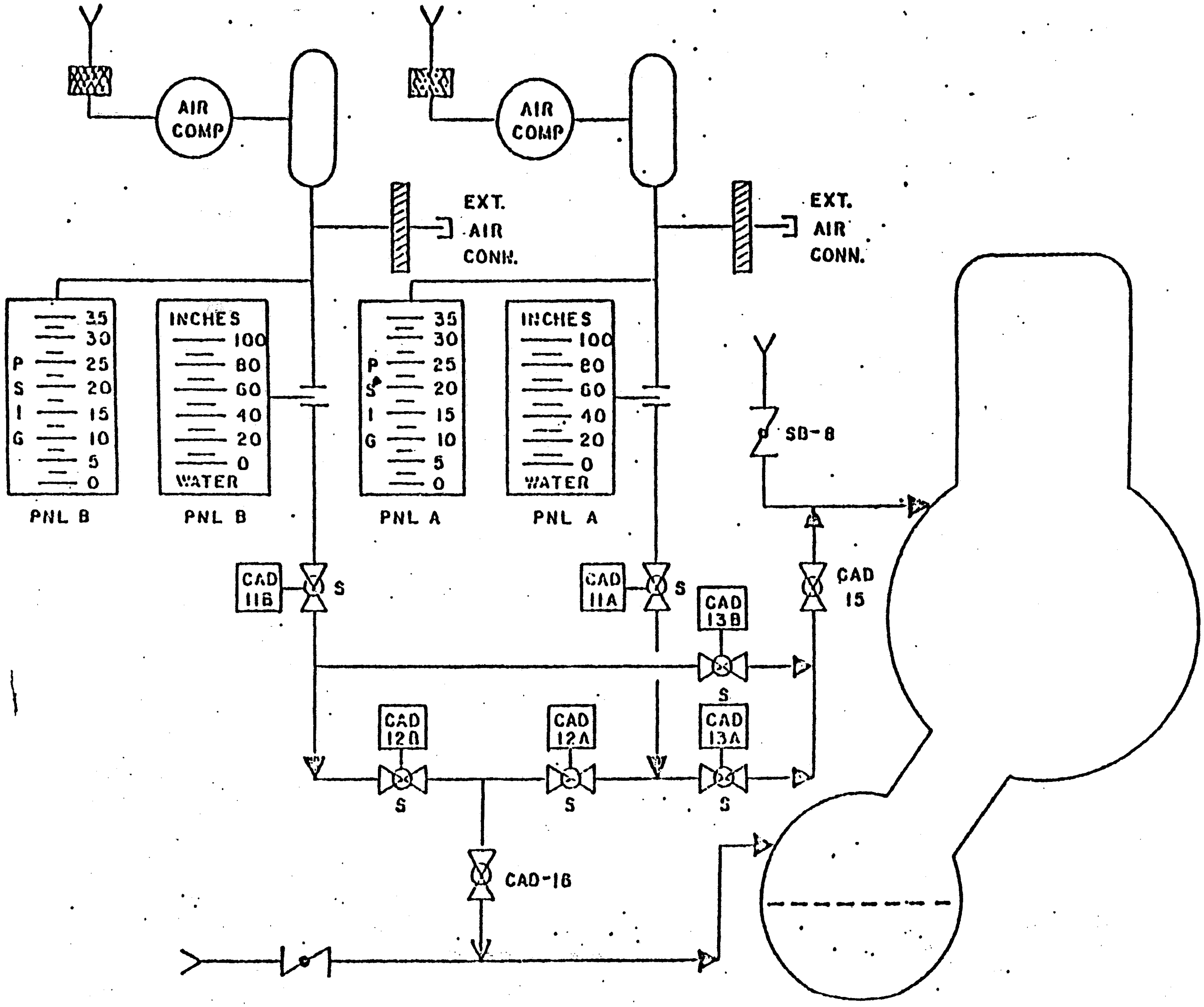


Figure 4.11 Diagram of the Injection Subsystem (Containment Air Dilution System)

flow paths. Each path consists of a motor driven air compressor located inside the reactor building. Air can be pumped into either the drywell and/or the torus[11]. Each compressor is rated at 41 standard cubic feet per minute at a discharge pressure of 30 psig. Power requirements for the compressor, valve operators, and instrumentation constitute a train which is power supplied by one of the plant emergency diesel generators; the other diesel generator supplies the redundant train. Each train has a piping connection outside the reactor building to allow the use of portable compressors in case of failure of the main compressors. All valves and compressors are manually operated from the CAD system panels located in the control room. During normal plant operation, injection valves are closed and receive a PCIS signal. On loss of power, the solenoid operated isolation valves fail in the closed position.[26]

The vent subsystem consists of piping, instrumentation, motors, and solenoid-operated valves arranged to provide two independent flow paths, one from the drywell and one from the torus (Fig. 4.12). Each path is connected to the SGTS. Flow in each path is regulated by motor operated valves from the control room. All valves are normally closed until receiving a PCIS signal.[11] On loss of power, the solenoid valves in the subsystem fail in the closed position whereas the motor operated valves fail in the position they are at the moment of loss of power.

The CAD system was designed assuming accident conditions that would involve hydrogen generation from only 1.3% of the active zirconium in the core interacting with steam. This value was assumed by applying a factor of five to the projected

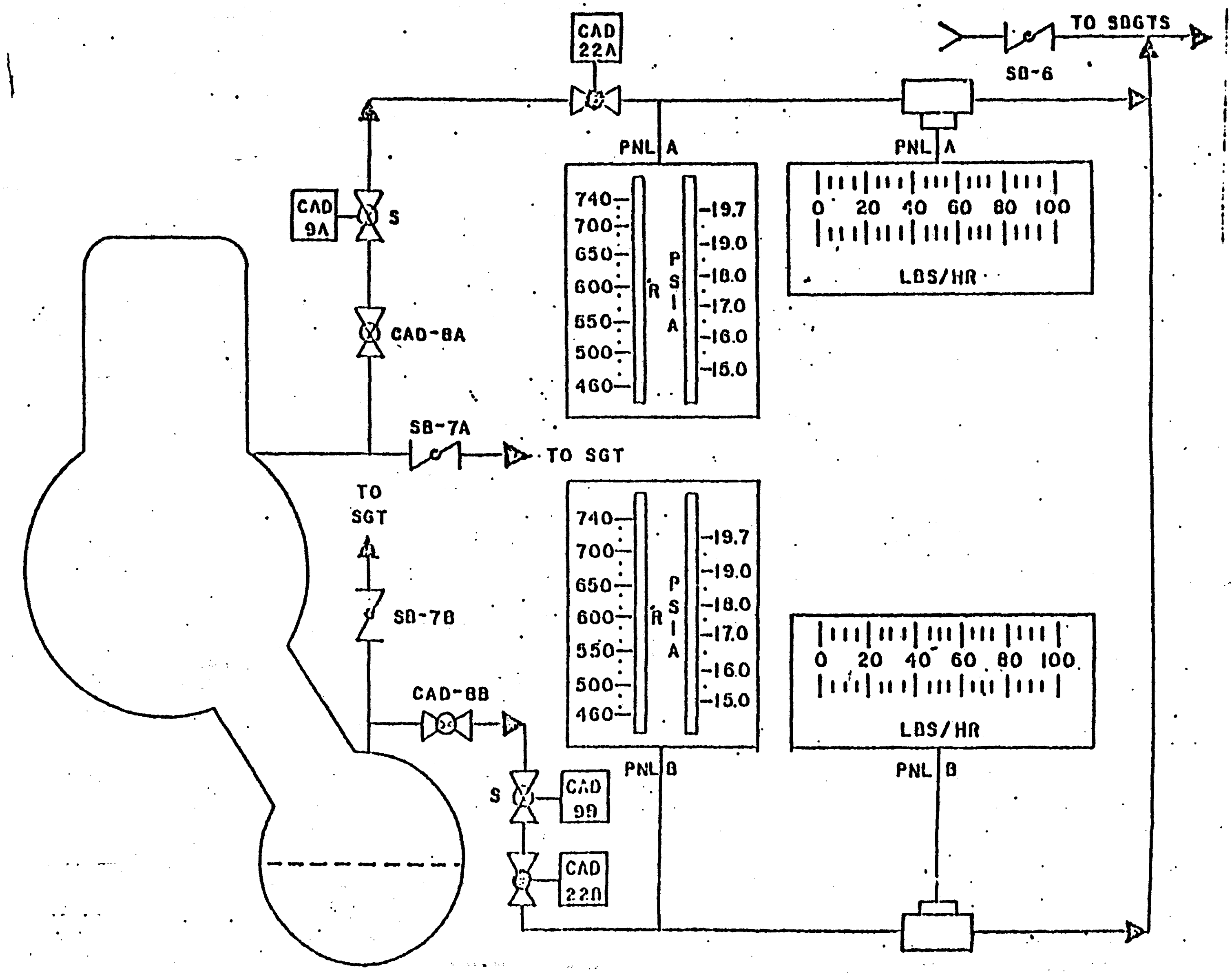


Figure 4.12 Diagram of the Vent Subsystem (Containment Air Dilution System)

calculated value of 0.26%[9]. The maximum hydrogen concentration calculated as a function of time is shown in Fig. 4.13 based on Vermont Yankee calculations. After a design basis LOCA, the hydrogen concentration could reach four volume percent in about 19 hours, assuming a 1.3% metal-water reaction. In order to prevent the increase in hydrogen concentration, a hydrogen analyzer would be activated. After a thirty minute warm-up period, the hydrogen concentration can be measured.[20] The analyzer is used to monitor the hydrogen concentration in the drywell. Once the concentration reaches 3.2 volume percent in either the drywell or torus, an air compressor is activated which can dilute the containment atmosphere at a rate of 40 cubic feet per minute.

The hydrogen concentration will continue to increase due to radiolysis approaching the four volume percent limit and will level off two days after pressurization has begun.[26] The compressor will continue to run until the pressure inside the containment reaches 28 psig, half of the containment design pressure. Following this procedure, the hydrogen concentration should decrease to about 3.5 volume percent. As hydrogen continues to be generated, the concentration will approach 3.7 volume percent and the containment will be vented. The venting rate is 20 cubic feet per minute. Venting does not decrease the hydrogen concentration because the hydrogen generated by radiolysis increases faster than it can be removed by venting. When hydrogen reaches four volume percent and the pressure has decreased to less than 28 psig, pressurization will begin and continue until the pressure increases to 28 psig and the hydrogen concentration decreases. The pressurization and

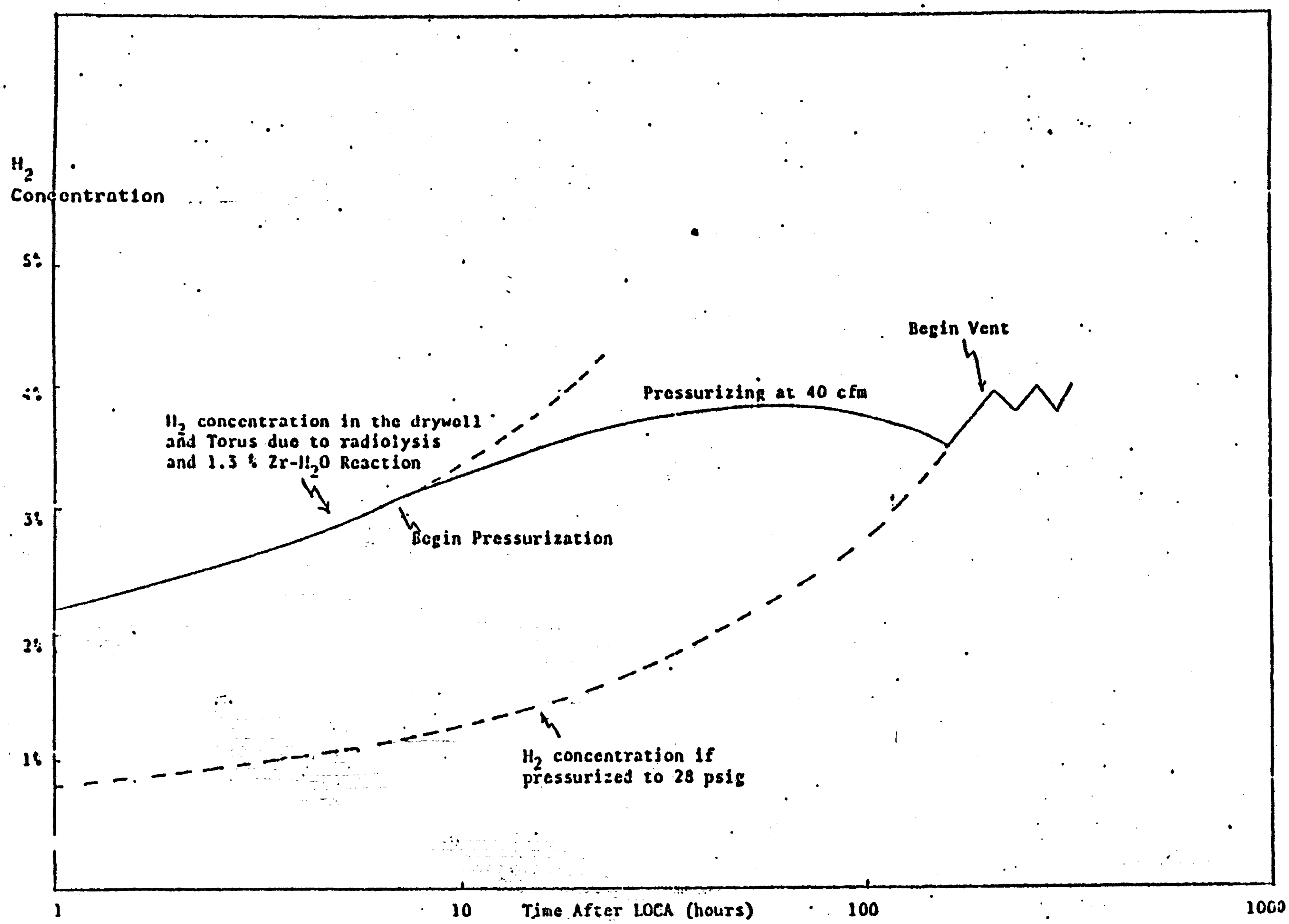


Figure 4.13 Hydrogen Concentration vs. Time After an Accident (Containment Air Dilution System)

continuous venting cycle will continue as long as necessary.

The fluctuation of hydrogen concentration during the process is shown in Fig. 4.13. The CAD system can also be operated without hydrogen analysis. The compressor can be started within seven hours following a design basis LOCA and continue to run until the pressure reaches 28 psig. At this point, the compressor is stopped and one day later, the containment is vented. After a 24 hour venting period, the compressor cycle can begin again and continue as long as necessary; venting also continues during this time.

The fault tree for the CAD system describes the different ways the system could fail to control post-accident generated hydrogen (Fig. 4.14). It is used to calculate the probability of failure of the CAD system to work when required, which is used later in the overall probability calculation. The following assumptions were made in the analysis: (i) independent component failures were considered (except where noted); (ii) electric power is assumed during the time of the accident; events where electric power or diesel-generators are required, refer to local electric service; (iii) the SGTS is assumed operational when required; (iv) the probability of air compressor failure refers to initial usage, with availability assumed to decrease during the cycling process; (v) the rare event approximation is used except where noted; (vi) the work "containment" refers to the primary containment (the drywell plus torus); (vii) all failure probabilities are placed on a "per demand" basis and refer to component unavailability or human error; these values are assumed independent of time; and (viii) point values are used from fixed data and error propagation

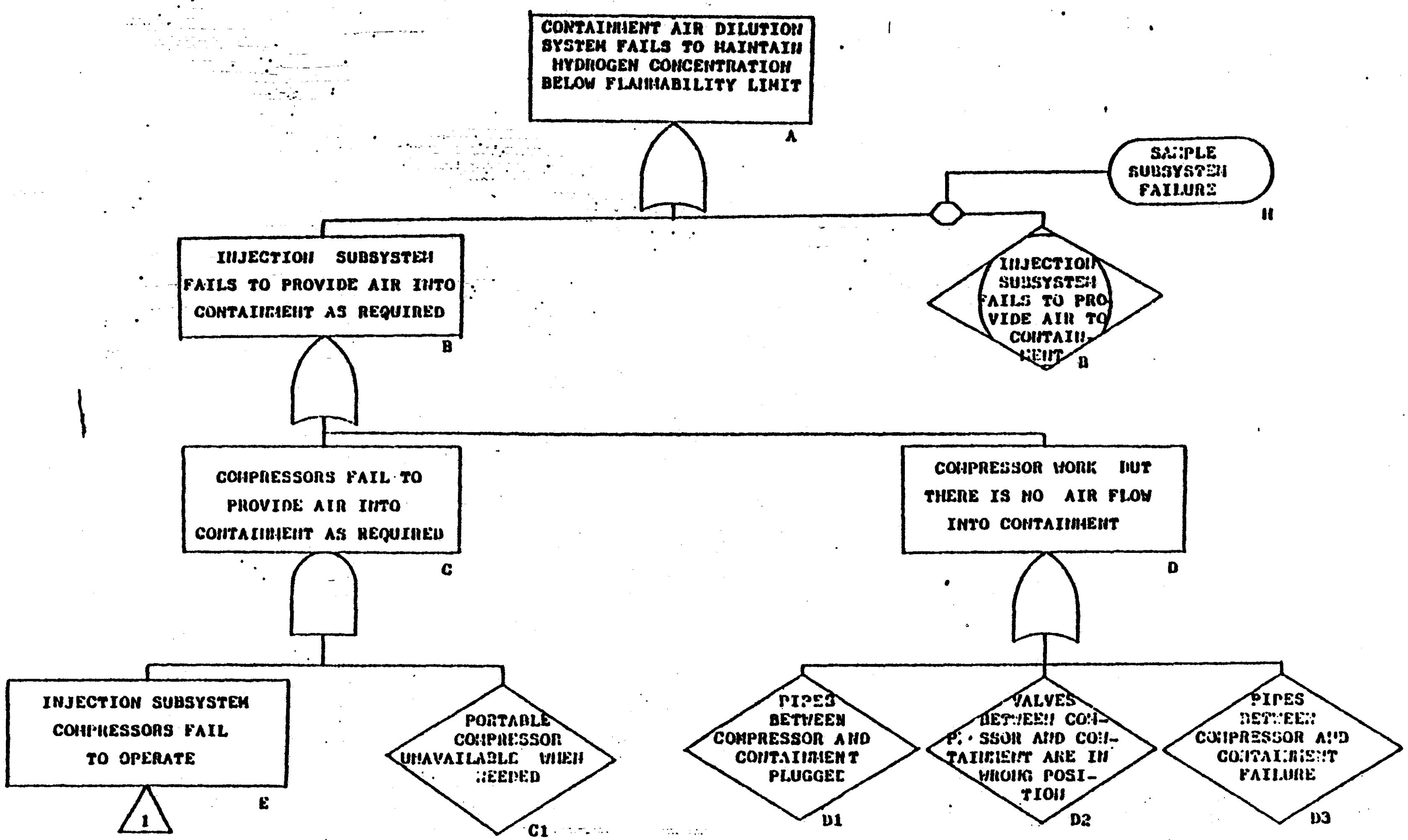


Figure 4.14 Fault Tree of the Containment Air Dilution System

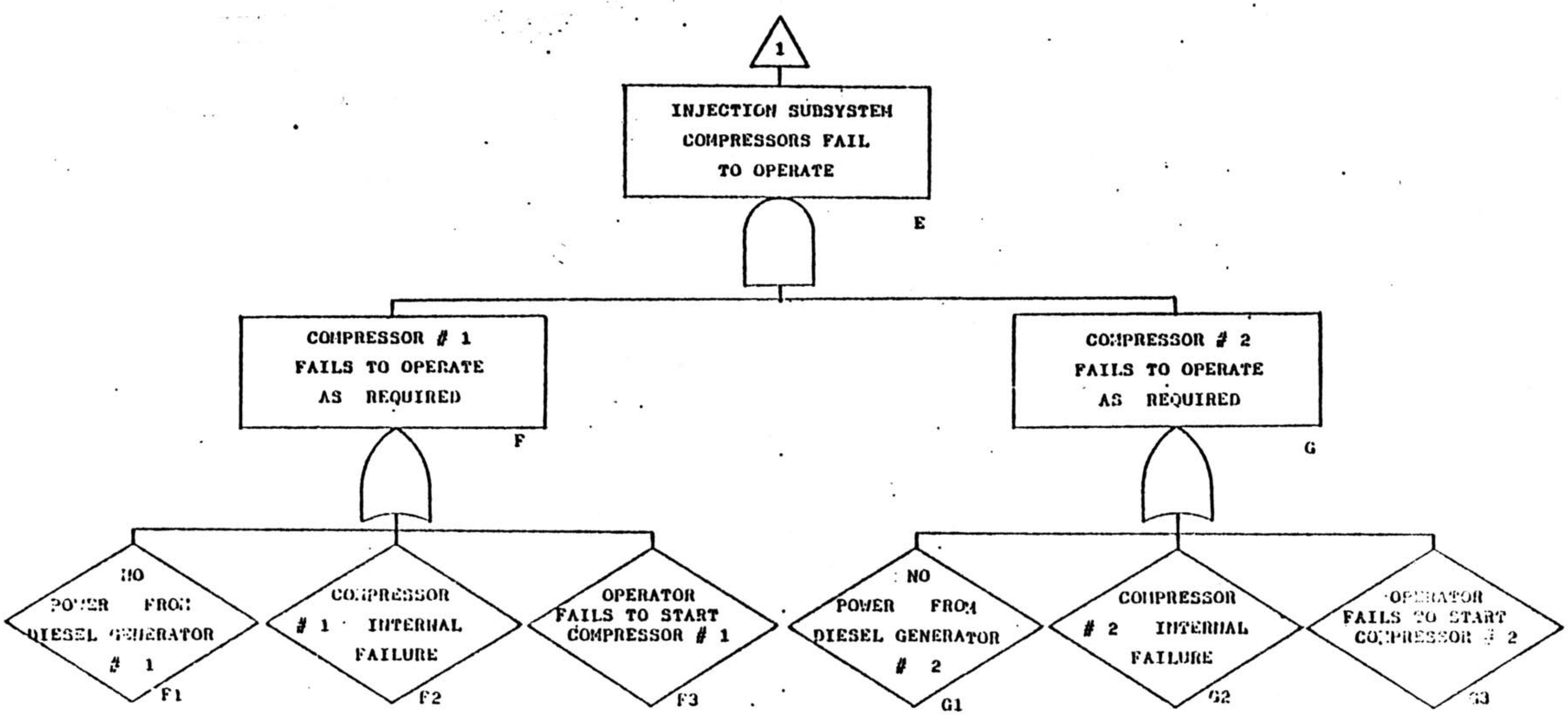


Figure 4.14 Fault Tree of Containment Air Dilution System (Continuation)

is from log normal distribution following the procedures of WASH-1400.[31]

The first step in developing a fault tree is to define a top event, which is the most undesirable event postulated to occur. For the CAD system, the top event is defined as the failure of the system to maintain the hydrogen concentration below the NRC mandated flammability limit of 4 volume percent. This event may occur if the injection subsystem fails to provide air to the containment as required after the generation of hydrogen in an accident. The failure probability of the injection subsystem could be increased if the sample subsystem fails to detect correctly the hydrogen concentration. In this case, the operator would not know when to correctly start the air compressors, since the hydrogen concentration must reach 3.3 volume percent before operator action can be initiated. The compressors could physically be run without knowing the correct hydrogen concentration, but the effectiveness of maintaining the hydrogen concentration below the flammability limit would significantly decrease. The failure of the sample subsystem is due to the failure of the hydrogen analyzers to detect the hydrogen concentration (Fig. 4.15), or to the failure of the component pipes, valves, pumps due to malfunction or operator error. The two redundant hydrogen analyzers could fail due to improper calibration. In this case, the concentration recorded would be incorrect. Another failure mode is the failure of the analyzer to start due to malfunction or operator error to start the analyzer.

The injection system can fail because of failure of the system air compressors, and the unavailability of a portable compressor that could be connected to the system in the case of failure of the

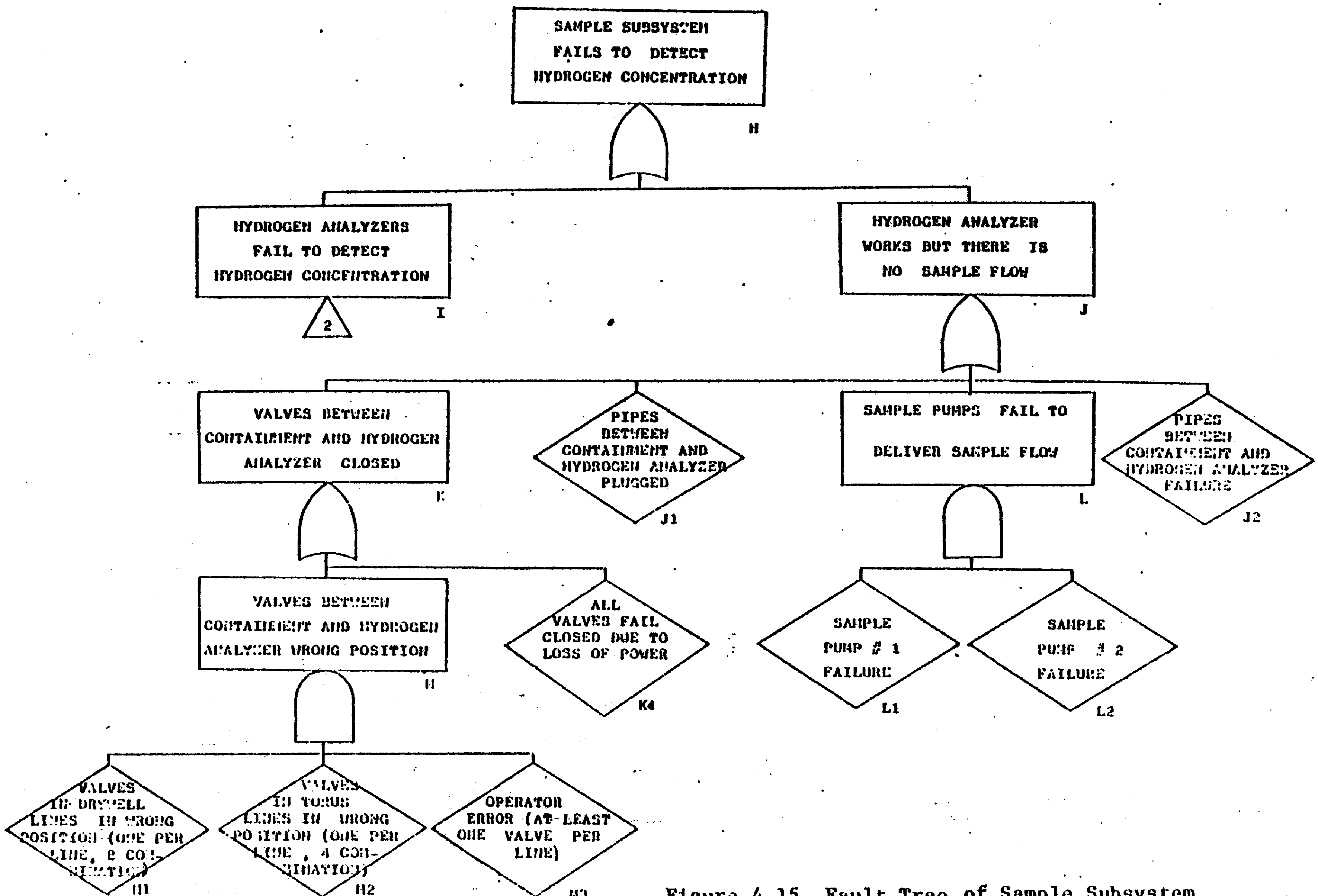


Figure 4.15 Fault Tree of Sample Subsystem (Containment Air Dilution System)

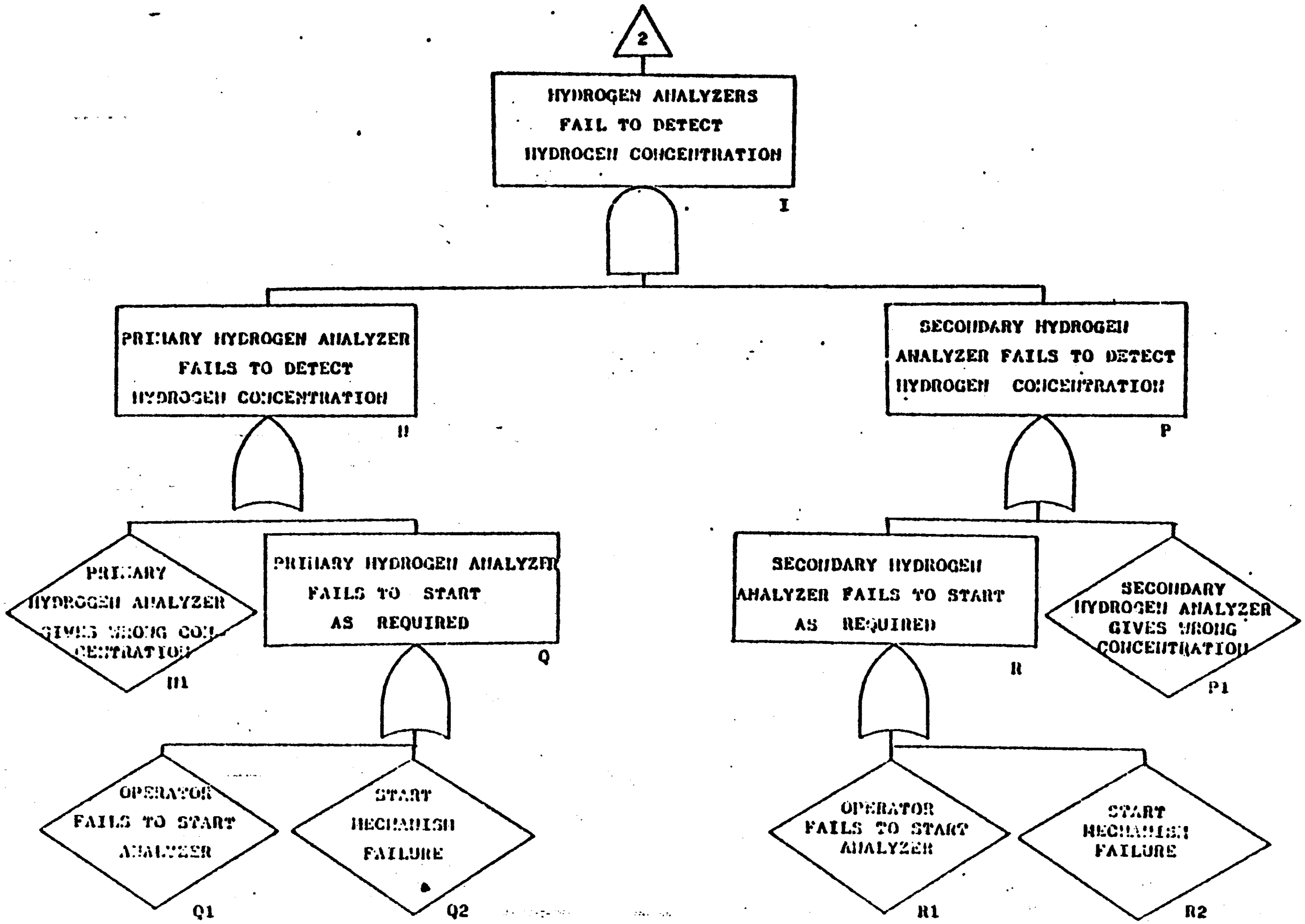


Figure 4.15 Fault Tree of Sample System (Containment Air Dilution System) (Continuation)

Table 4.1

EVENT PROBABILITIES USED IN CONTAINMENT AIR DILUTION SYSTEM FAULT TREE

EVENT DESCRIPTION	FAILURE PER DEMAND	ERROR FACTOR*
Loss of Power	$1x10^{-6}$	30
Valves drywell wrong position	$>1x10^{-10}$	--
Valves torus wrong position	$>1x10^{-10}$	--
Operator error: at leat one valve per line	$>1x10^{-10}$	--
Operator fails to stop compressor	$1x10^{-2}$	10
Compressor fails to stop	$1x10^{-4}$	10
Sample pump failure	$1x10^{-3}$	3
Hydrogen analyzer wrong concentration	$1x10^{-6}$	10
Operator fails to start primary hydrogen analyzer	$1x10^{-2}$	10
Operator fails to start secondary hydrogen analyzer	$3x10^{-1}$	10
Hydrogen analyzer start mechanism mechanism failure	$1x10^{-4}$	10
Portable compressor unavailable when needed	$1x10^{-1}$	10
No power from diesel-generator	$3x10^{-2}$	10
Compressor fails to start	$1x10^{-3}$	10
Operator fails to start compressor 1	$1x10^{-2}$	10
Operator fails to start compressor 2	$1x10^{-1}$	10

*Error factor is to be used to multiply failure per demand to obtain the upper bound, and to divide principal compressors. However, if at least one of the available compressors work, failure could still occur due to failure of the air to flow into the containment due to a rupture or plug of the connecting air pipes, or valves in wrong position. The failure of the main compressors is dependent upon one or more of the following events occurring: power failure, malfunction of the compressors, operator failure to start the compressors, or failure of the analyzers to perform on demand.

Using component failure data from WASH-1400[31], the fault tree was used to quantify the CAD system probability of failure of demand (Table 4.1). The results show that the CAD system has a median probability of failure on demand of 1.6×10^{-3} with a lower bound of 1.6×10^{-4} and an upper bound of 1.6×10^{-2}. This means that there exists an approximate 99.8% probability that the CAD system would be able to maintain the hydrogen concentration below the flammability limit for those accident sequences that result in a design basis hydrogen generation rate corresponding to a 1.3% metal-water reaction (approximately 1000 cubic feet hydrogen per hour).

4.3.2 Analysis of Containment Inerting System

The purpose of the containment inerting system (CIS) is to provide nitrogen into the primary containment in order to reduce or maintain the oxygen concentration of the drywell and the suppression pool below five percent during normal plant operation, and to reduce post-accident hydrogen concentration below four percent. The containment inerting system to be analyzed here is that which is installed at the Pilgrim 1 Nuclear Power Station in Plymouth, Massachusetts operated by the Boston Edison Electric Company. The purpose of this analysis is the same as that of the containment air dilution (CAD) system analyzed earlier; that is, to identify and evaluate potential failure modes, system weak points, and how they can lead to undesirable events. Failure probabilities of the CIS were derived using the information supplied by Boston Edison personnel and the fault tree methodology and failure data used in WASH-1400.[31,32] The probability of failure of the containment inerting system was used in the containment event tree to determine its contribution to the overpressurization failure of the containment due to hydrogen generation, and for comparison with the containment air dilution system. In order to understand the development of fault tree, the subsystems of the containment inerting system are described including inerting and deinerting and operation during potential accidents.

The inerting system consists of three subsystems designed to function as follows: (i) initial purging of the primary containment within 24 hours after startup, (ii) providing a supply of make-up nitrogen during accidents that produce hydrogen and (iii) providing a way to sample the drywell and torus for oxygen concentration and the drywell for hydrogen concentration. These subsystems are described below;

A schematic diagram of the purging and make-up system is shown in Fig. 4.16. The purging subsystem consists of two connections for liquid nitrogen supplied by trucks, a nitrogen

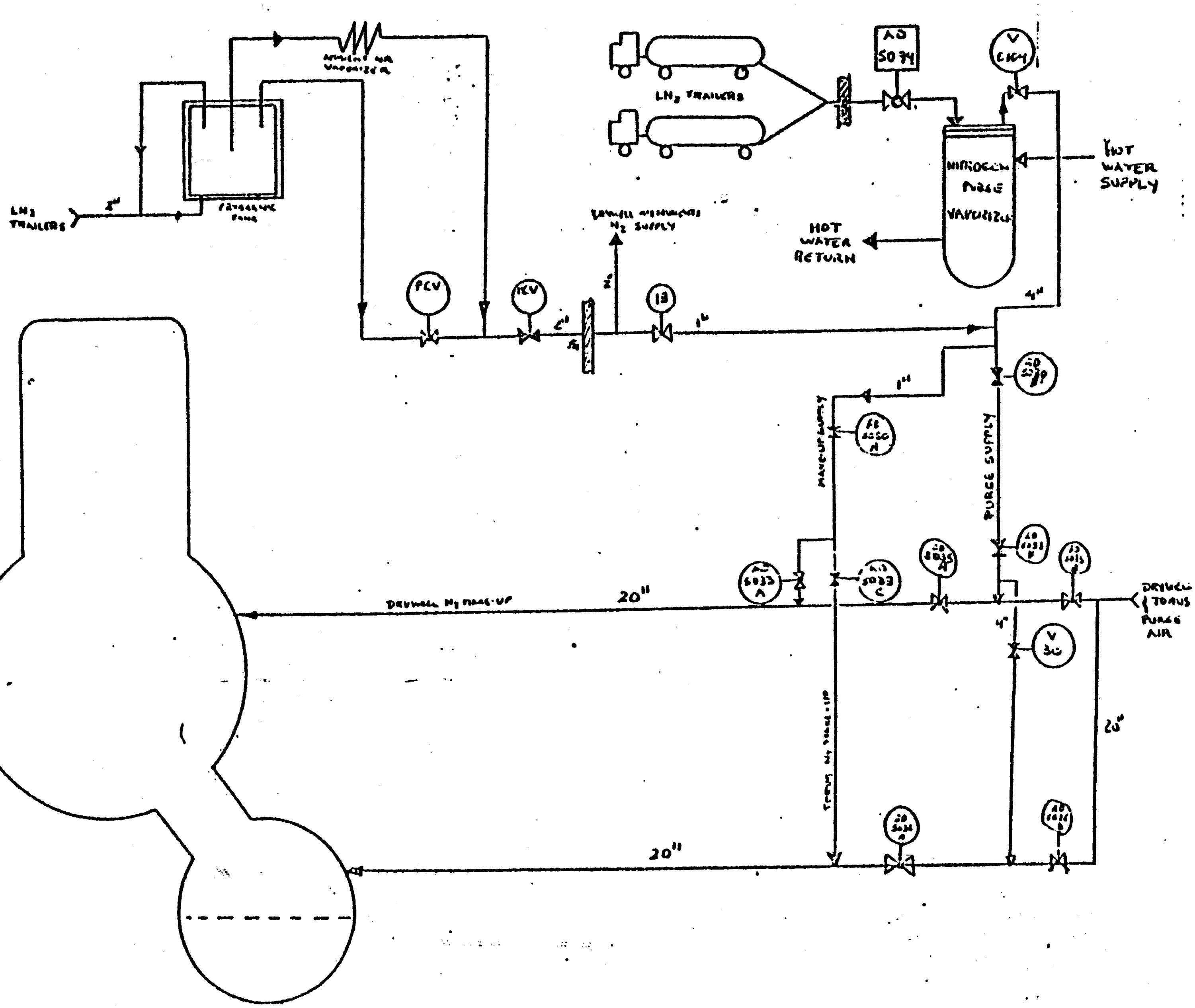


Figure 4.16 Diagram of the Purge and Make-up Subsystem (Containment Inerting System)

purge vaporizer, a set of valves, pipes to the drywell and torus and an air supply system. The make-up subsystem consists of valves a cryogenic tank of liquid nitrogen, a connection for the liquid nitrogen, and an ambient air vaporizer. The subsystem is connected to the drywell and torus via the purge subsystem pipes (Fig. 4.16). The sample subsystem consists of an oxygen analyzer connected to seven sampling points located in the drywell (3), torus (2), drywell exhaust (1), and torus exhaust (1) and to two return lines, one from the drywell and one from the torus. The oxygen analyzer is monitored from the control room back panels, and is in continuous service during plant operations because of a cold-start eight hour warm-up period. Two hydrogen analyzers provide samples from the drywell and are in standby condition at all times during operation (Fig. 4.17).

During the inerting process, the suppression pool is purged until the oxygen concentration is reduced to below four volume percent. The vent valves remain open until the purge valves are closed. The nitrogen flow is regulated to keep the lines from freezing, which happens when the flow of the cooled gas is increased too rapidly. Once the oxygen concentration in the suppression chamber is set below four volume percent, nitrogen flow is established to the drywell until a four volume percent oxygen content is reached. During this process, the nitrogen make-up system valves are closed and a SGTS train is activated.

Nitrogen can be added to the drywell or suppression chamber during normal operations. In case of hydrogen generation, the

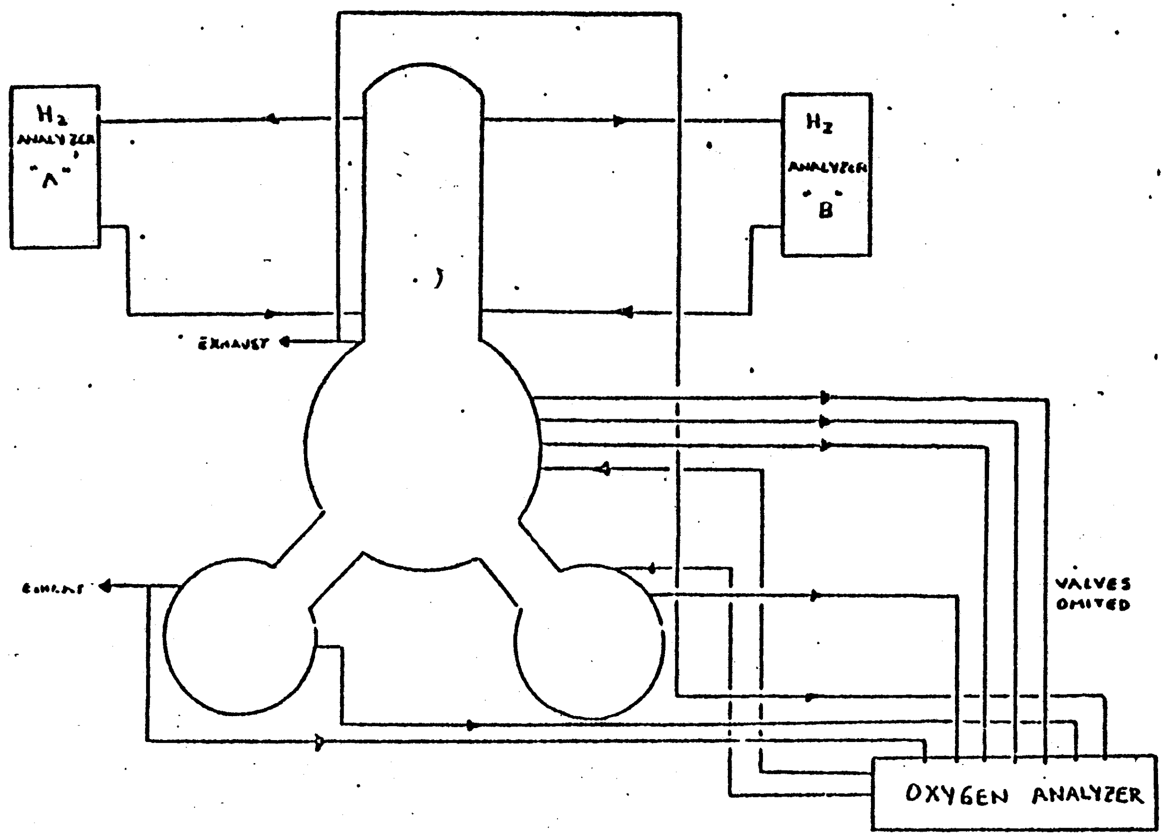


Figure 4.17 Diagram of Sample Subsystem (Containment Inerting System)

primary containment must be isolated and the SGTS placed in service (Fig. 4.18). When the hydrogen concentration approaches four volume percent, the nitrogen make-up system valves are open to add nitrogen to the primary containment. The nitrogen addition is stopped when the hydrogen concentration falls below two volume percent. When and if the hydrogen concentration should increase again to the four volume percent level and when the containment pressure reaches the design pressure of 45 psig, the containment is vented through the SGTS to lower the primary containment pressure.* During venting, the nitrogen make-up system adds nitrogen to the containment to maintain low oxygen levels. Venting and nitrogen addition are stopped when the hydrogen concentration is reduced to below three volume percent.

In order to deinert the primary containment after reactor shutdown, the nitrogen make-up system and purge system are closed. Then, the purge exhaust valves are opened as well as the air purge inlet valves to allow air flow. When the drywell air sample shows an oxygen content greater than 19 volume percent, drywell entry can then safely be made by plant personnel in order to make necessary repairs and reconnaissance.

The fault tree for the containment inerting system (CIS) was developed in the same fashion as that for the CAD system

*The decision to vent the containment during an accident would be left to upper-management utility officials, according to operators at Pilgrim 1.

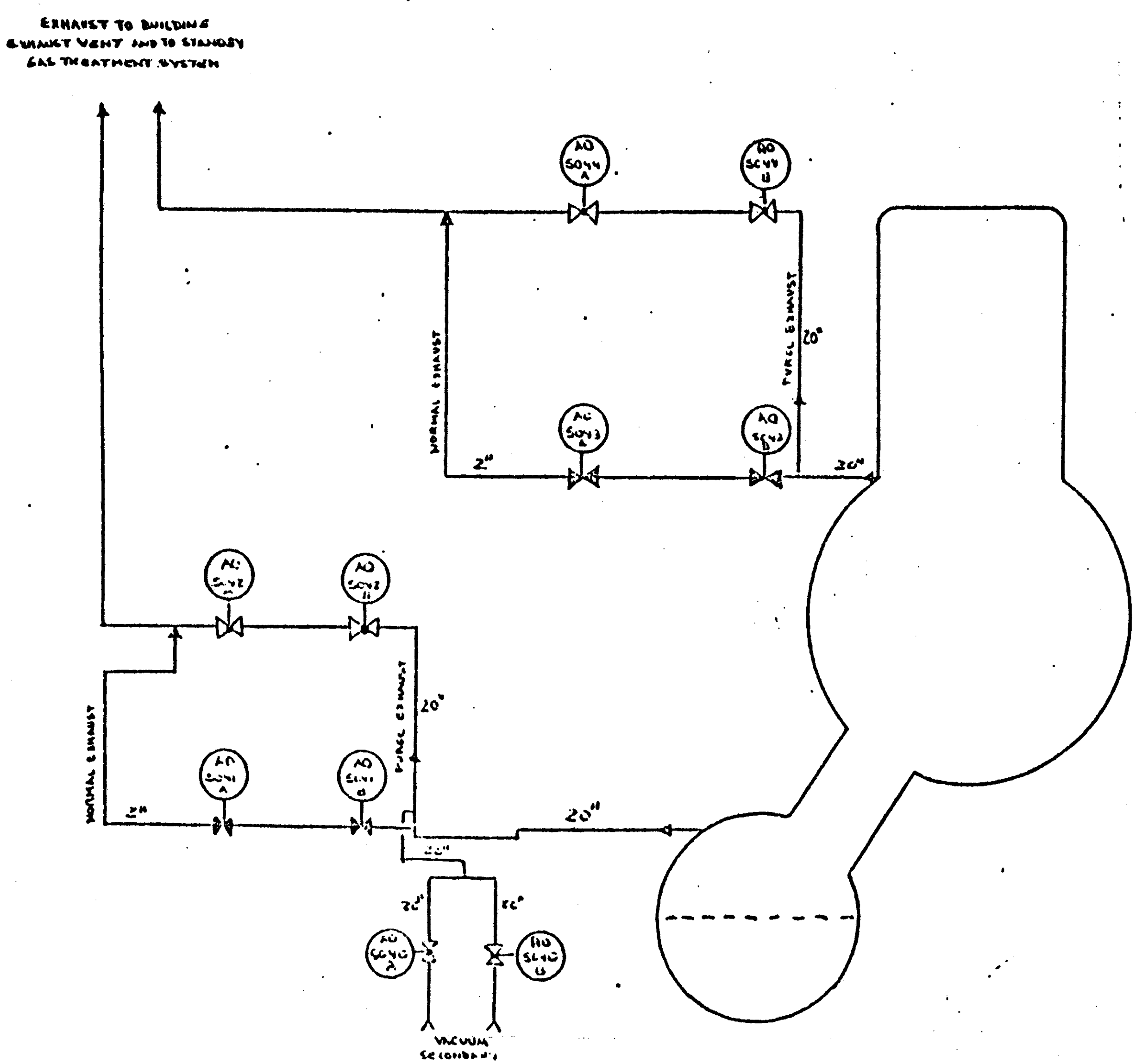


Figure 4.18 Diagram of the Vent Subsystem (Containment Inerting System)

(see section 4.3.1). The fault tree describes the different ways in which the inerting system could fail to control post-accident generated hydrogen (Fig. 4.19, 4.20). During the construction of the tree, the assumptions made followed those noted (i) to (vii) for the CAD system, with the change in assumption (iv), i.e.; the containment is inerted at the time of the accident, reflecting the attributes of the CIS system.

The first step in developing the fault tree is the definition of the most undesirable event or top event. For the inerting system, the top event is the failure to maintain the oxygen and hydrogen mixture below NRC mandated flammability limits. The inerting system prevents a flammable mixture from developing by maintaining the oxygen concentration below five volume percent; a make-up system is used during an accident to maintain the oxygen below five volume percent. Failure of the make-up system can therefore lead to the top event occurring. A failure of the sample subsystem to detect both oxygen or hydrogen concentration conditions the probability of failure of the makeup system since the operator will not be able to open the valves of the make-up system when required because the gas concentrations are unknown. Failure of the operator to open the make-up valves or failure of the valves themselves leads to the event that the nitrogen make-up valves fail to open as required. This event, along with the unavailability of nitrogen in the system, leads to the failure of the make-up system to deliver nitrogen as required to the containment. Nitrogen can also become unavailable due to the rupture of the

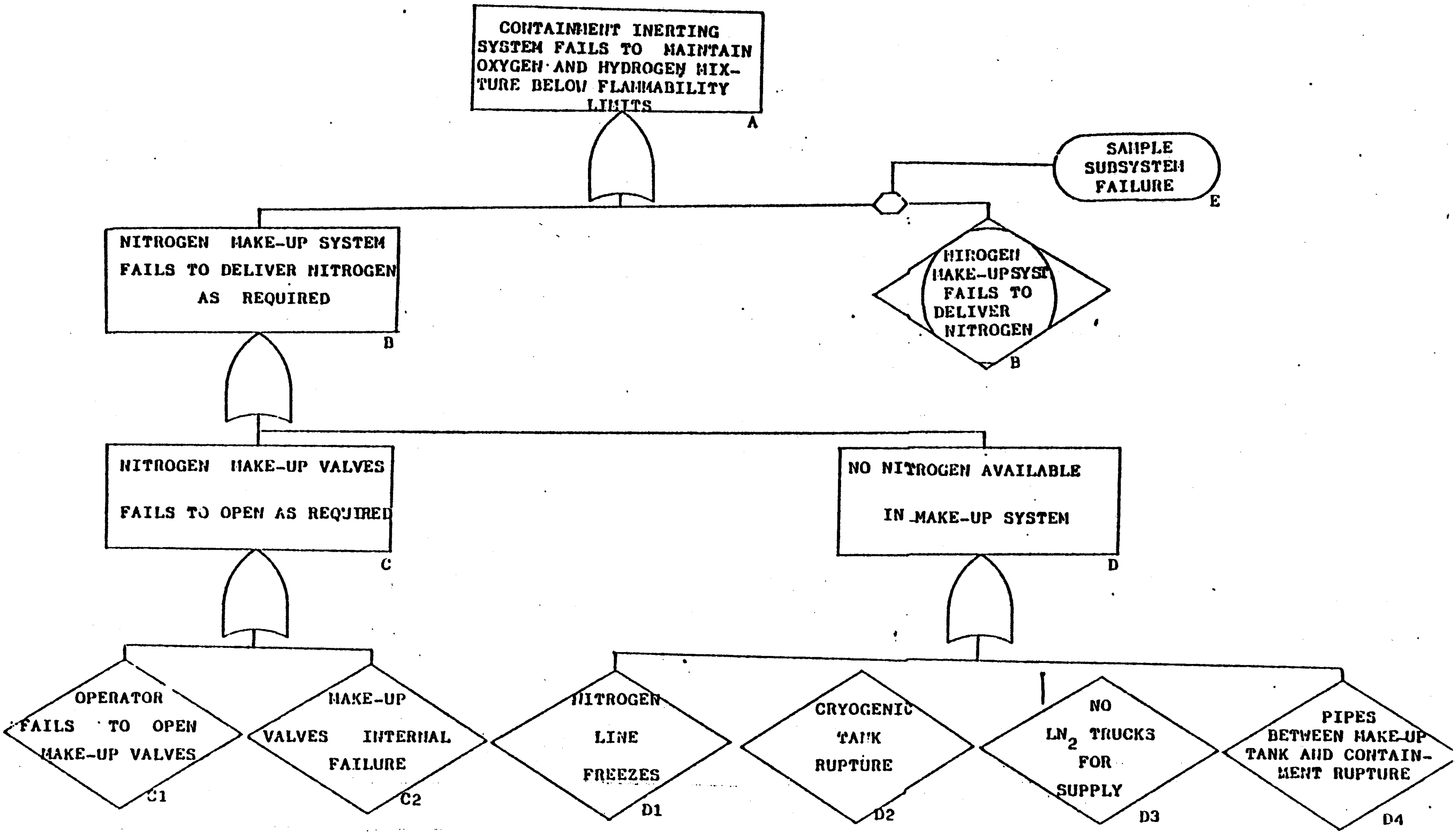
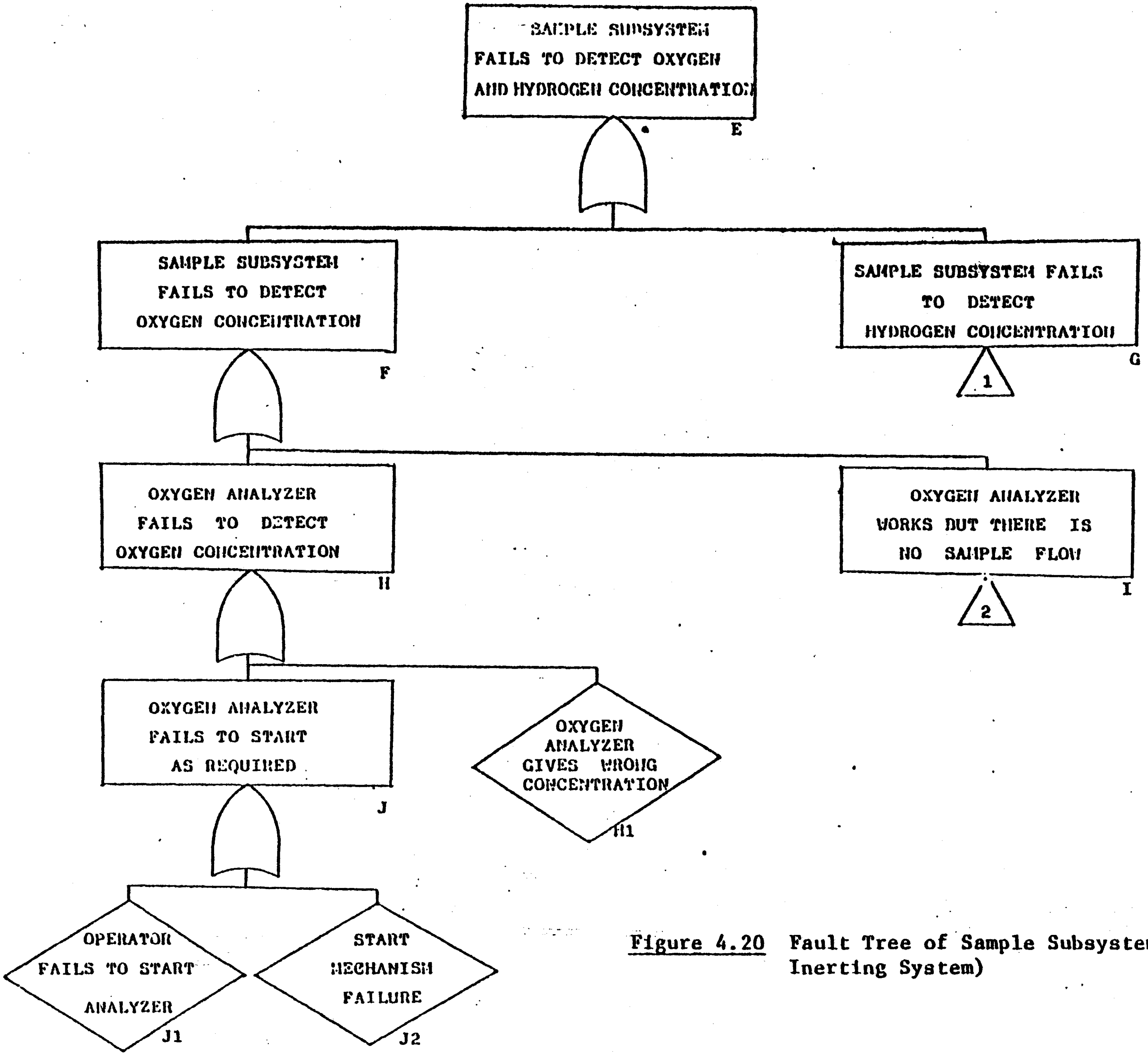


Figure 4.19 Fault Tree of the Containment Inerting System



<u>Figure 4.20</u> Fault Tree of Sample Subsystem (Containment Inerting System)

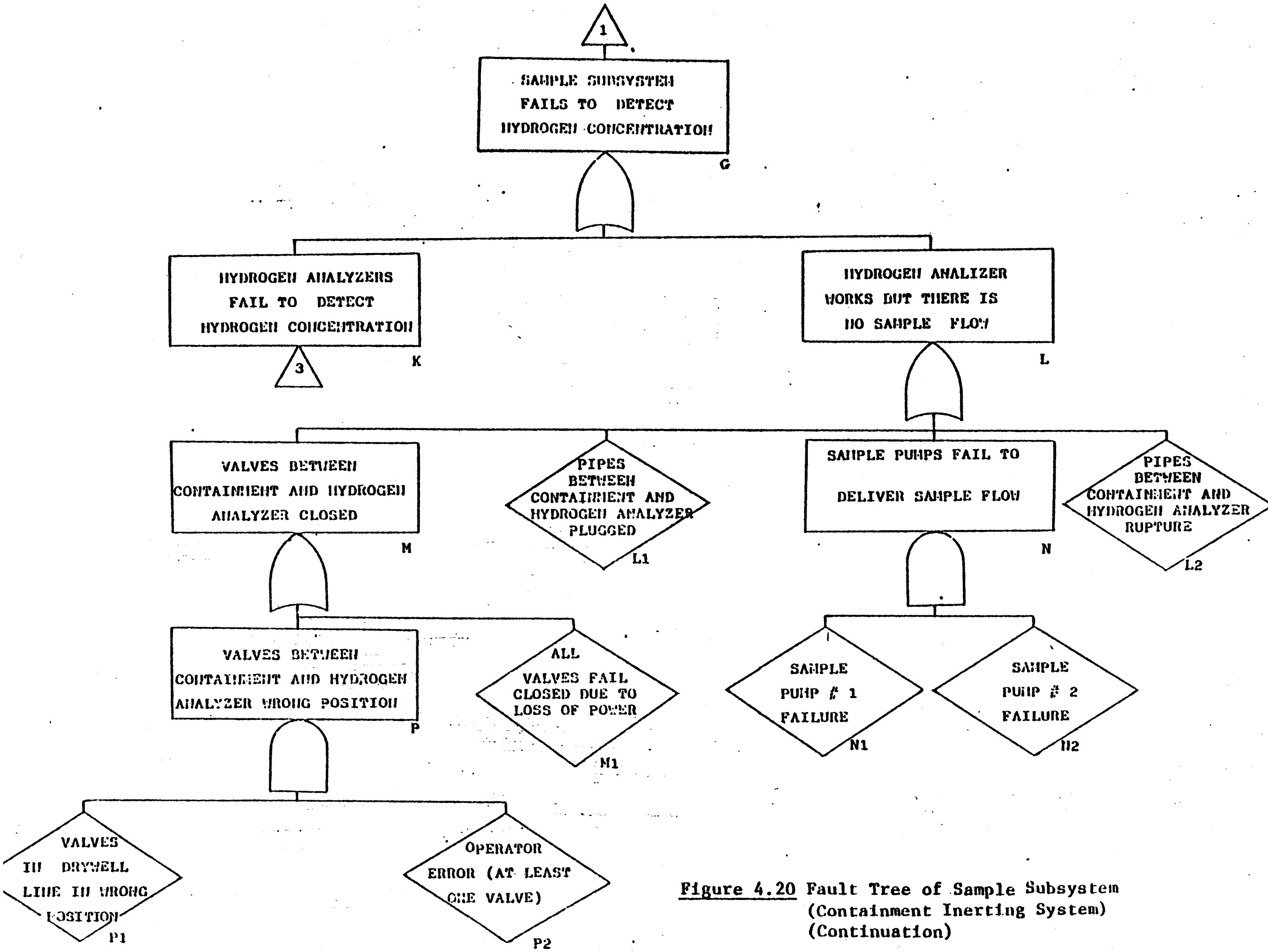


Figure 4.20 Fault Tree of Sample Subsystem (Containment Inerting System) (Continuation)

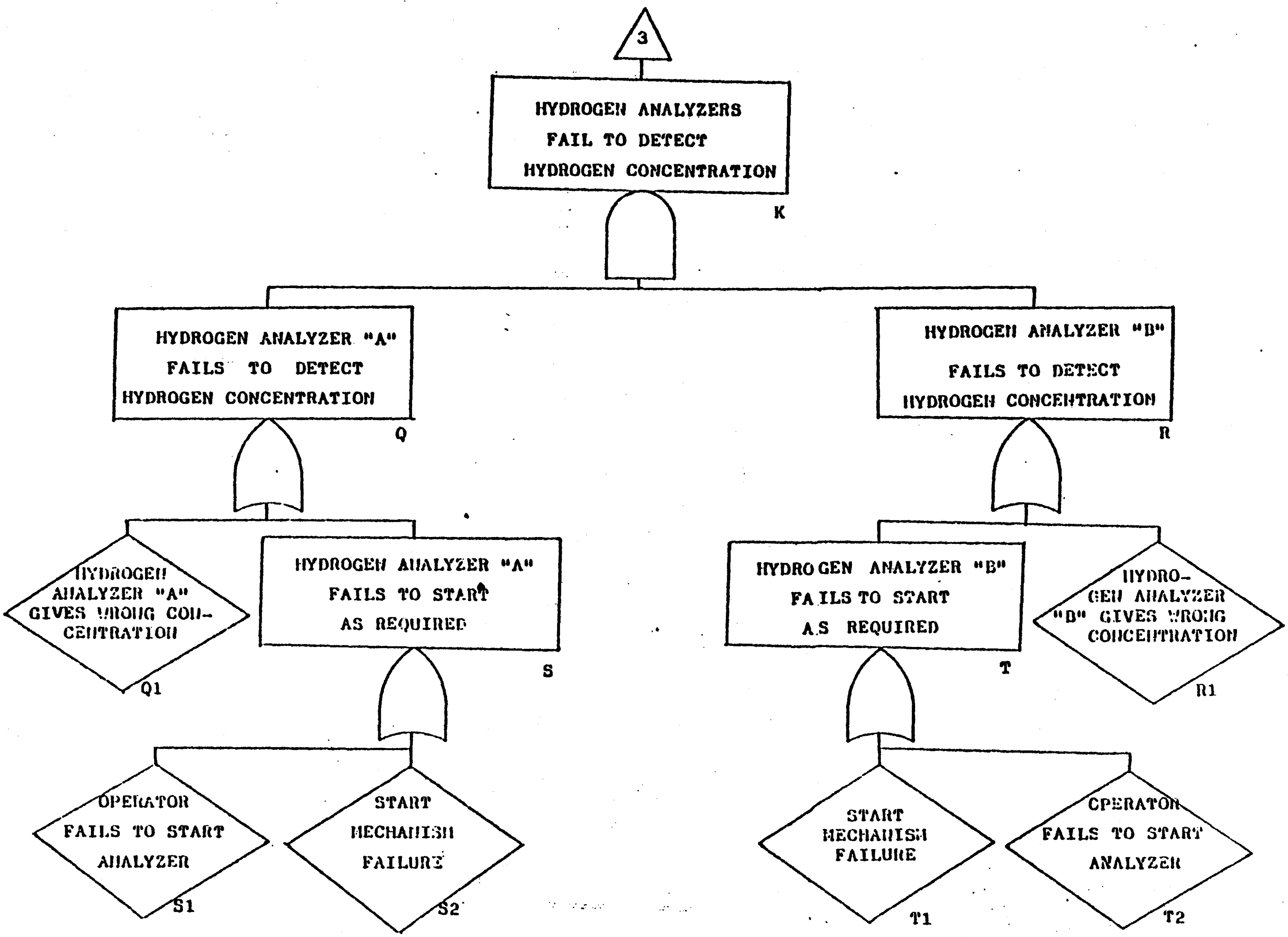
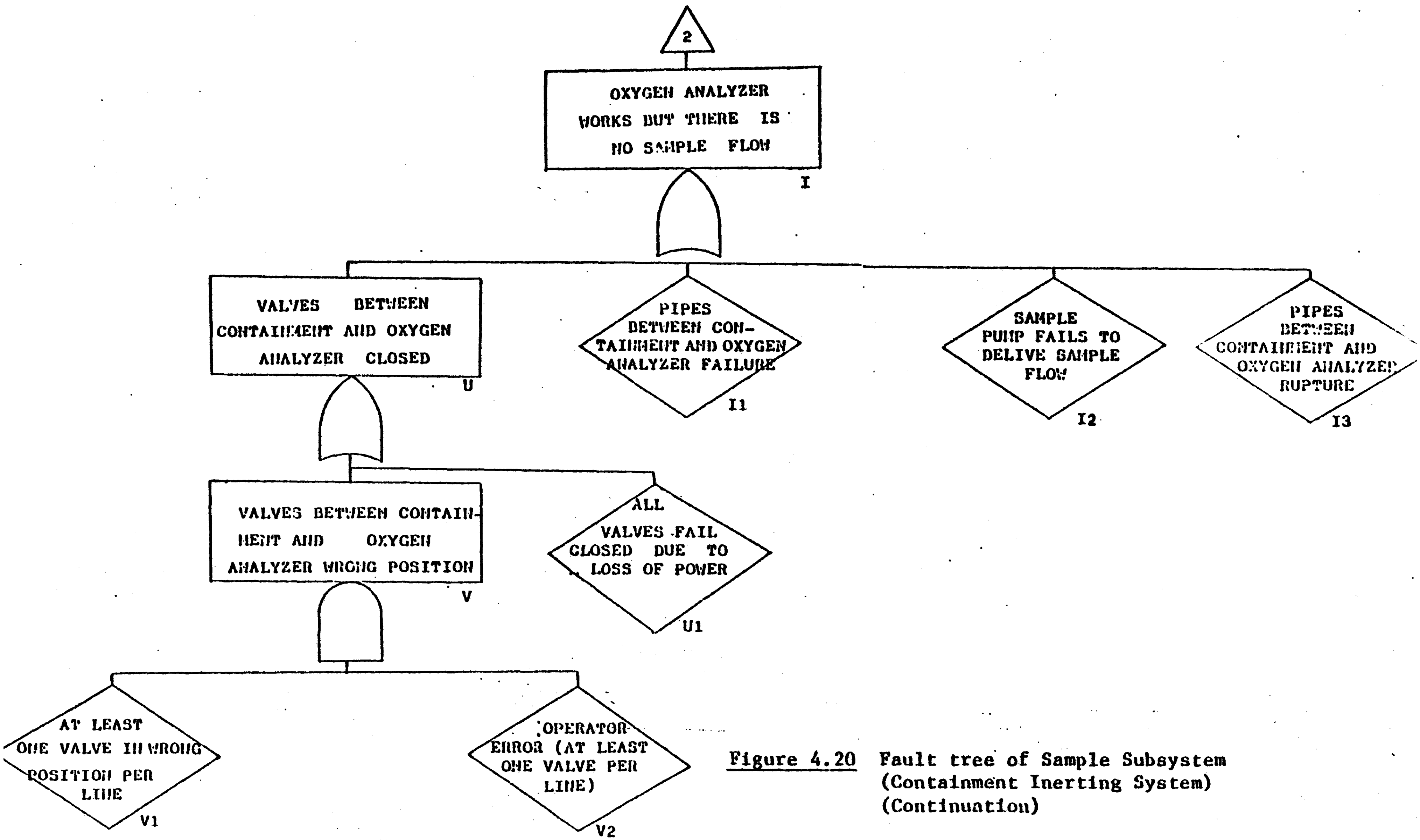


Figure 4.20 Fault Tree os Sample Subsystem (Containment Inerting System) (Continuation)



<u>Figure 4.20</u> Fault tree of Sample Subsystem (Containment Inerting System) (Continuation)

cryogenic make-up tank, a break in the pipes connecting the tank to the containment, freezing (plugging) of the pipes, and/or lack of nitrogen due to unavailability of delivery trucks.

Using the failure data from WASH-1400[32] (Table 4.2) the tree was quantified. The results show a median probability of failure on demand of $1.3x10^{-2}$, with a lower bound of $1x10^{-3}$ and an upper bound of $1x10^{-1}$. If the CIS has a redundant nitrogen make-up system, as in the case of the Peach Bottom nuclear power plant[25], the mean probability of failure on demand is reduced to $1.04x10^{-3}$ with an upper bound of $1x10^{-2}$ and lower bound of $1x10^{-4}$.

4.4 Final Results: Probability of Post-Accident Hydrogen Control

Using the probabilities calculated in previous sections, the final overall probability that the CAD and the CIS systems are capable of handling a given amount of hydrogen can be assessed. From the fault tree analysis (Section 4.3), the probability of failure of the systems ($P_f(S)$) are:

$$P_f(S)_{CAD} = 1.6x10^{-3} \quad \text{(Vermont Yankee)} \qquad (4.3)$$

$$P_f(S)_{CIS} = 1.3x10^{-2} \quad \text{(Pilgrim 1)} \qquad (4.4)$$

$$P_f(S)_{CIS} = 1.04x10^{-3} \quad \text{(Peach Bottom)} \qquad (4.5)$$

Where the probability that the system is available to work ($P_{CAD}(S)$ and $P_{CIS}(S)$) is:

$$P_{CAD}(S) = 1 - 1.6x10^{-3} = 0.9984 \quad \text{(Vermont Yankee)} \qquad (4.6)$$

$$P_{CIS}(S) = 1 - 1.3x10^{-2} = 0.9870 \quad \text{(Pilgrim 1)} \qquad (4.7)$$

$$P_{CIS}(S) = 1 - 1.04x10^{-3} = 0.9989 \quad \text{(Peach Bottom)} \qquad (4.8)$$

Table 4.2

EVENT PROBABILITIES USED IN CONTAINMENT INERTING SYSTEM FAULT TREE

EVENT DESCRIPTION	FAILURE PER DEMAND	ERROR FACTOR*
Valves between containment and make-up subsystem closed	1×10^{-6}	--
Oxygen analyzer failure	1×10^{-6}	10
Operator error: at least one valve per line	$>1 \times 10^{-10}$	--
Loss of Power:all valves closed	1×10^{-6}	30
Operator fails to open make-up valves	1×10^{-2}	10
Make-up valves fails to open as required	3×10^{-4}	10
Nitrogen line frezzes	1×10^{-8}	10
Cryogenic tank breaks	1×10^{-8}	10
No LN_2 trucks supply	3×10^{-3}	10
Hydrogen analyzer failure	1×10^{-2}	10

*Error factor is to be used to multiply failure per demand to obtain the upper bound, and to divide it to obtain the lower bound.

Table 4.3 indicates the probabilities assumed in this calculation. The hydrogen generation rate is discretized into "high", "medium" and "low" categories ($180x10^3$ cu.ft./hr.). For the "high" generation rate (upper bound of Fig. 4.6), $12x10^3$ cu.ft./hr. for "medium" (generation rate to reach four volume percent hydrogen in one hour in a Mark I containment). (The TMI hydrogen generation rate of ~ $100x10^3$ cu.ft./in. is located between the "high" and "medium" category.) For the low H_2 generation rate, 631.5 cu.ft./hr. based on the Vermont Yankee CAD design basis accident is used. In order to assess the probability that the CAD system can control the hydrogen generated in an accident, a probability of zero is assumed for the "high" generation rates because of the CAD system's physical inability to dilute such large amounts of hydrogen. For "medium" generation rates, the probelm can be analyzed from two points of view: (1) if it is assumed that the hydrogen is generated in one hour at $12x10^3$ cu.ft./hr., the H_2 concentration will be just under the four volume percent flammability limit so it is assured that the CAD system will be able to maintain the H_2 concentration below flammability with a probability of success equal to its availability (0.9984 -- see Fig. 4.2). On the other hand, (2) if the hydrogen is produced at a rate of $12x10^3$ cu.ft./hr. over a period longer than one hour, the breaking point will be equal to the maximum injection and venting capacity of the CAD system. In this case, the probability is assumed to be ~ 0.2 of being able to control the

Table 4.3

PROBABILITIES OF POST-ACCIDENT HYDROGEN CONTROL

HYDROGEN PRODUCTION RATE	PROBABILITY OF ACCIDENT P(A) (per reactor-yr)	WEIGHTING FUNCTION OF P(A)	ASSUMED HYDROGEN PRODUCTION RATE (cubic feet per hr)	PROBABILITY SUCCESS CAD SYSTEM P(S) (per design demand)
"HIGH"	$3x10^{-5}$	0.00901	180,000	0.00
"MEDIUM"	$3x10^{-4}$	0.09009	12,000	0.199 - 0.9984
"LOW"	$3x10^{-3}$	0.90090	631.5	0.9984

	CIS PILGRIM I P(S) (per design demand)	CIS PEACH BOTTOM P(S) (per design demand)	CAD SYSTEM P(S|A) (per accident)	CIS PILGRIM I P(S|A) (per accident)	CIS PEACH BOTTOM P(S|A) (per accident)
H -	0.9870	0.9989	0.00	0.00889	0.0090
M -	0.9870	0.9989	0.0179-0.0899	0.08892	0.0899
L -	0.9870	0.9989	0.89946	0.88919	0.8999
			0.9174-0.9894	0.9870	0.9988

Note: $$P(S|A) = \frac{P(S)_i \times P(A)_i}{\Sigma\, P(A)_i}$$

hydrogen from reaching the flammability limit. For "low" generation rates, the system availability (0.9984) is used. For the CIS, probabilities of success of 0.9870 (Pilgrim 1) and 0.9989 (Peach Bottom) are used for all three H_2 generation categories. These values are derived from the equations 4.6, 4.7, and 4.8.

The final failure probabilities for the CAD system to prevent hydrogen flammability over the range of theoretically possible hydrogen generation rates are (based on the Vermont Yankee CAD system design):

$$P_f\ (S/A)_{CAD} = 1 - 0.91739 = 8.26\text{x}10^{-2}\ /\text{demand} \qquad (4.9)$$

to

$$P_f\ (S/A)_{CAD} = 1 - 0.98936 = 1.06\text{x}10^{-2}/\text{demand} \qquad (4.10)$$

For the CIS system, the final failure probabilities are

$$P_f\ (S/A)_{CIS} = 1 - 0.9870 = 1.3\text{x}10^{-2}/\text{demand} \quad \text{for Pilgrim 1} \quad (4.11)$$

and

$$P_f\ (S/A)_{CIS} = 1 - 0.9988 = 1.2\text{x}10^{-3}/\text{demand} \quad \text{for Peach Bottom} \quad (4.12)$$

These results indicate that both systems have approximately similar overall probabilities of controlling any amount of hydrogen generated during reactor accident. Since the "low" hydrogen generation rates have higher propabilities of occurrence, both systems depend on the reliability of the system design. When comparing the probability of success of the Vermont Yankee CAD system with that of the CIS of Pilgrim 1 (see Table 4.3) for "low" hydrogen generation rates, the CAD is more reliable than the CIS. When CAD is compared with the CIS of Peach Bottom which has a redundant nitrogen make-up subsystem, both systems have almost the same overall H_2 control probability.

For "medium" generation rates, depending on the assumptions applied to the CAD system, it can be compared with the two CIS systems in the same way as for the "low" hydrogen generation rates, or will be 20% less reliable than the two CIS systems. For low probability accidents with high hydrogen generation rates, it is assumed that the CAD system cannot prevent hydrogen deflagration. If burning does not occur, the CAD system could help to reduce the time to reach detonation limits. Inerting can handle larger amounts of hydrogen due to maintaining oxygen concentration below five volume percent. However, the inerting system is not in operation 24 hours prior to shutdown and 24 hours after startup. During this period, the containment has no protection against hydrogen generation reducing the overall probability of handling the hydrogen. In this case, the final result depends on the number of reactor shutdowns during the year combined with the individual probabilities of the CIS to handle the different amounts of hydrogen generated.

CHAPTER FIVE

IMPACT OF CONTAINMENT INERTNG ON REACTOR SAFETY

The hazards of inerting were discussed in Chapter 2. The consequences of lack of oxygen affects operational procedures with regard to correcting leakage inside the primary containment, thus impacting upon the probability of leakage developing into accident initiating events and increasing the number of unscheduled shutdowns.

5.1 Potential Adverse Effects of Inerting on Reactor Safety.

During normal operation, the control room monitors conditions in the drywell. Symptoms requiring immediate and subsequent corrective actions can thus be identified (Fig. 5.1). The major symptom of a developing problem is an increase in the unidentified (or identified) leakage rate. Such leaks are annunciated in the control room by the drywell unit cooler annunciators drywell air cooler high drain flow and radiation leak detector, to name a few.[19,20] Changes in drywell humidity and/or significant changes in pressure, along with excessive sump pump operation can also indicate the evolution of such a problem.[19,20] In order to control leakage, operator actions must be iniciated such as: (i) monitoring the reactor vessel power, pressure and water level, (ii) referral to the pipe break procedure if appropriate (iii) monitoring the drywell floor and equipment sump readings, and (iv) determining the location of the leak. When the total unidentified leakage reaches 25 gallons per minute in both inerted and non-inerted BWR containments, technical specifications

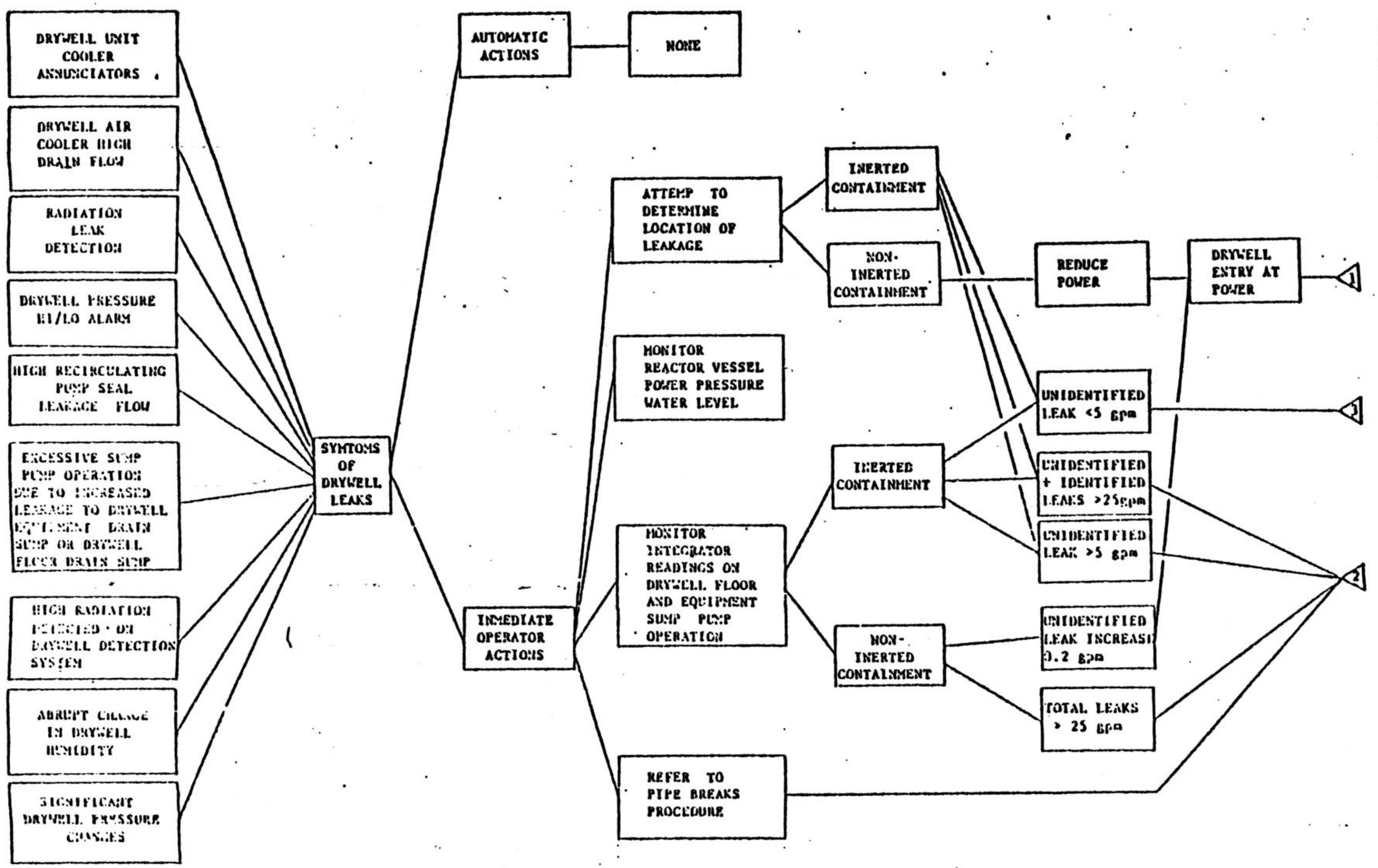


Figure 5.1 Leaks Inside Primary Containment. Operator Procedures in Inerted and Non-inerted Containments

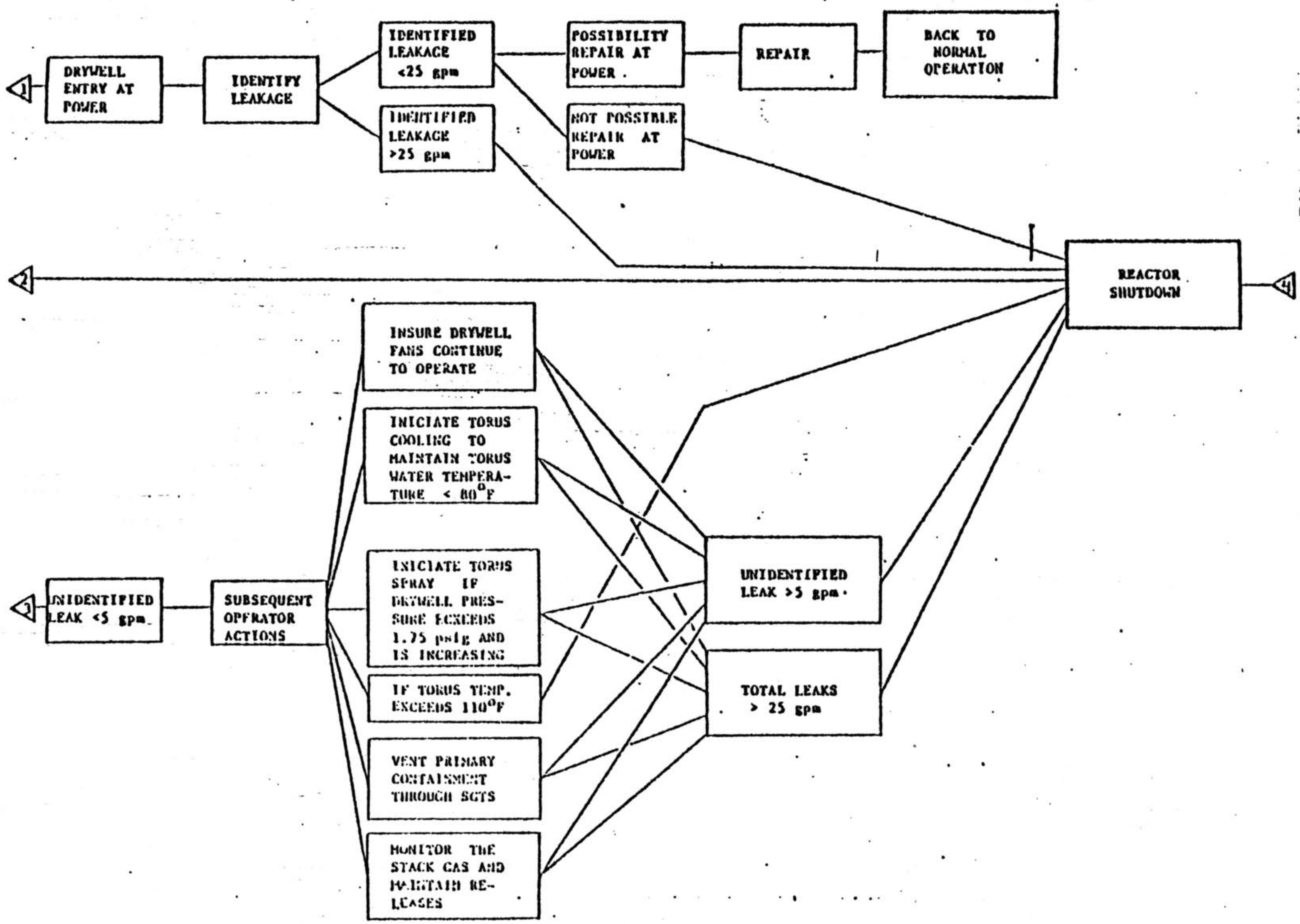
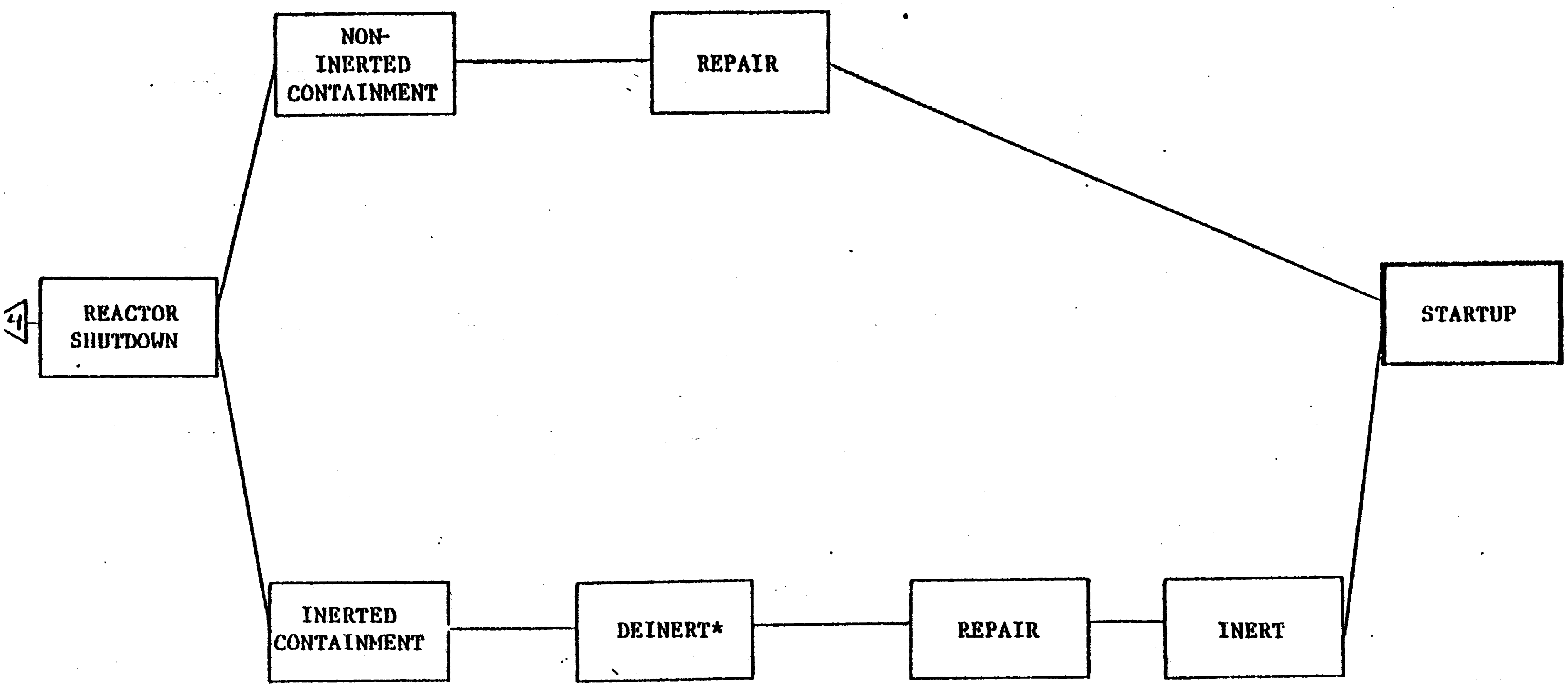


Figure 5.1 Leaks Inside Primary Containment. Operator Procedures in Inerted and Non-Inerted Containments (Continuation)



*Deinerting can be done 24 hours before shutdown but the drywell entry has to be performed after shutdown

Figure 5.1 Leaks Inside Primary Containment. Operator Procedures in Inerted and Non-inerted Containments (Continuation) require the operators to shutdown the reactor.[21,22] In the non-inerted case, drywell entries at power can take place if the power level is sufficiently reduced to between 50-70% full power. Entry can take place without recourse to the use of bulky breathing apparatus. Inspection permits the operators to determine the seriousness of the problem, aiding them in their decision as to whether to continue operation or to shutdown to make major repairs. This option means that unnecessary plant shutdown can be avoided in many cases, reducing the stress placed on the system that occurs from shutdown and affecting the probability of failure of the heat removal system. Also, in those cases where shutdown is deemed necessary, unnecessary delays in startup can be avoided since the inerting procedure is not required.

During inerted containment operation, drywell entries at power are not permitted by industry practice because the excessive danger such entry would represent to plant personnel.[20] Leakage identification is therefore made more difficult. Technical specifications require that the operator insures drywell fans are operating at all times, and that the torus temperature be maintained below 80°F. The torus spray system is initiated if torus pressure should exceed 175 psig and venting of the primary containment through the standby gas treatment system is also initiated.[20] No attempts are taken to stop leakage, which is allowed to increase to the five gallon per minute criterion whereupon the reactor is shutdown. Entry usually requires that the containment be purged until oxygen concentration reaches 20 volume percent, which usually requires 24 hours. However, during emergencies, entry could be permitted as early as one hour after shutdown but with some risk to the plant personnel making such entry.[23]

Deinerting consists of injecting air into the containment while venting nitrogen through the standby gas treatment system. The decision to vent depends upon the activity level of the gas and the mandated limits placed on such discharges to the atmosphere. If the flow rate is restricted, the concentration of oxygen will increase more slowly requiring more than 24 hours to deinert.[24] Under normal conditions, the oxygen concentration must reach 20 volume percent before entry can take place. Even so, breathing apparatus (Scott packs) are used. The breathing apparatus is an open circuit apparatus with a high-pressure cylinder of air or oxygen, a cylinder valve, a demand regulator, a facepiece, and tube assembly with an exhalation valve. The use of this apparatus limits access to the problem area inside the containment, because of its bulk (50 lbs. weight), the discomfort of the facemask, and the possibility of leaks or rupture of the equipment.

After repairs are complete, the containment is inerted within 24 hours after startup of the plant. At the Nine Point nuclear station, Unit 1, the reactor was shutdown to repack a recirculation valve. After the repairs, the unit was restarted while waiting for the delivery of two trucks of nitrogen. The trucks did not arrive causing a 24 hour delay in the normal scheduled startup of the unit.[28] The cost of the nitrogen used for inerting is approximately $50,000 per year, which is not an insignificant cost.[25]

All of these considerations (e.g., time to detect, evaluate, and repair a component, number of unscheduled shutdowns and nitrogen supply availability) affects the capacity factor of the plant. Also, components are affected due to extra stress during shutdown and startup increasing the probability of failure of the equipment (see Section 5.3). Early detection and corrective action may affect the likelihood of a small lead evolving into a significant LOCA initiating event. This likelihood is now analyzed using data on drywell entries from the Vermont Yankee plant.

5.2 Analysis of Drywell Entries at Power

In order to evaluate the safety aspects involved in the location, evaluation and isoluation of a leakage inside the drywell, it is necessary to know the circumstances under which an entry is made and its effects on the overall safety of the plant.

There are four different circumstances in which an entry to the drywell has been made; (i) entries to perform preventive maintenance during scheduled shutdown, (ii) emergency situations wherein the reactor is shutdown due to malfunction of equipment inside the drywell, (iii) entries during an unscheduled shutdown not associated with anything in the containment but useful for inspection purposes and (iv) entries after reduction of power as a consequence of monitoring a malfunction inside the drywell that does not require an immediate shutdown.[24] The last three types of entries are affected by containment inerting. For example, in an emergency situation requiring an immediate shutdown, entry would be delayed because of the need to deinert the containment.*

*Estimates of the delay range from 3 - 10 hours at minimum.

During the period from May 5, 1974 to August 28, 1979, several drywell entries at power were performed at Vermont Yankee in order to identify and repair potential leaks in drywell equipment. According to plant data on entries, most of the repairs were made in packing or bonnet leaks on recirculation valves (Fig. 5.2). One of these valves (RV-43A) was backseated four times during the period as follows:

(i) May 5, 1974: drywell equipment drain sump leakage was observed to increase to 4.6 gpm due to a packing leakage (according to a plant operator,[20] the normal leakage rate is 1.4 gpm);

(ii) May 11, 1976: drywell equipment sump leakage indicates sharp increase from 1.8 gpm to 3.3 gpm due to packing leakage;

(iii) May 6, 1977: drywell equipment drain sump leakage increases from 1.8 gpm to 3.9 gpm due to packing leakage; and

(iv) Nov. 7, 1978: drywell equipment drain sump leakage increase 0.25 gpm due to packing leak.

The same valve in loop B of the recirculation system (RV-43B) was backseated three times during the period as follows:

(i) Feb. 5, 1977: drywell equipment sump leakage increased from 1.8 gpm to 2.0 due to packing leakage;

(ii) Dec. 25, 1977: drywell equipment drain sump leakage increased from 1.8 gpm to 2.1 gpm due to packing leakage; and

(iii) Aug. 28, 1979: drywell equipment drain sump leakage increased by 1.5 gpm due to packing leakage.

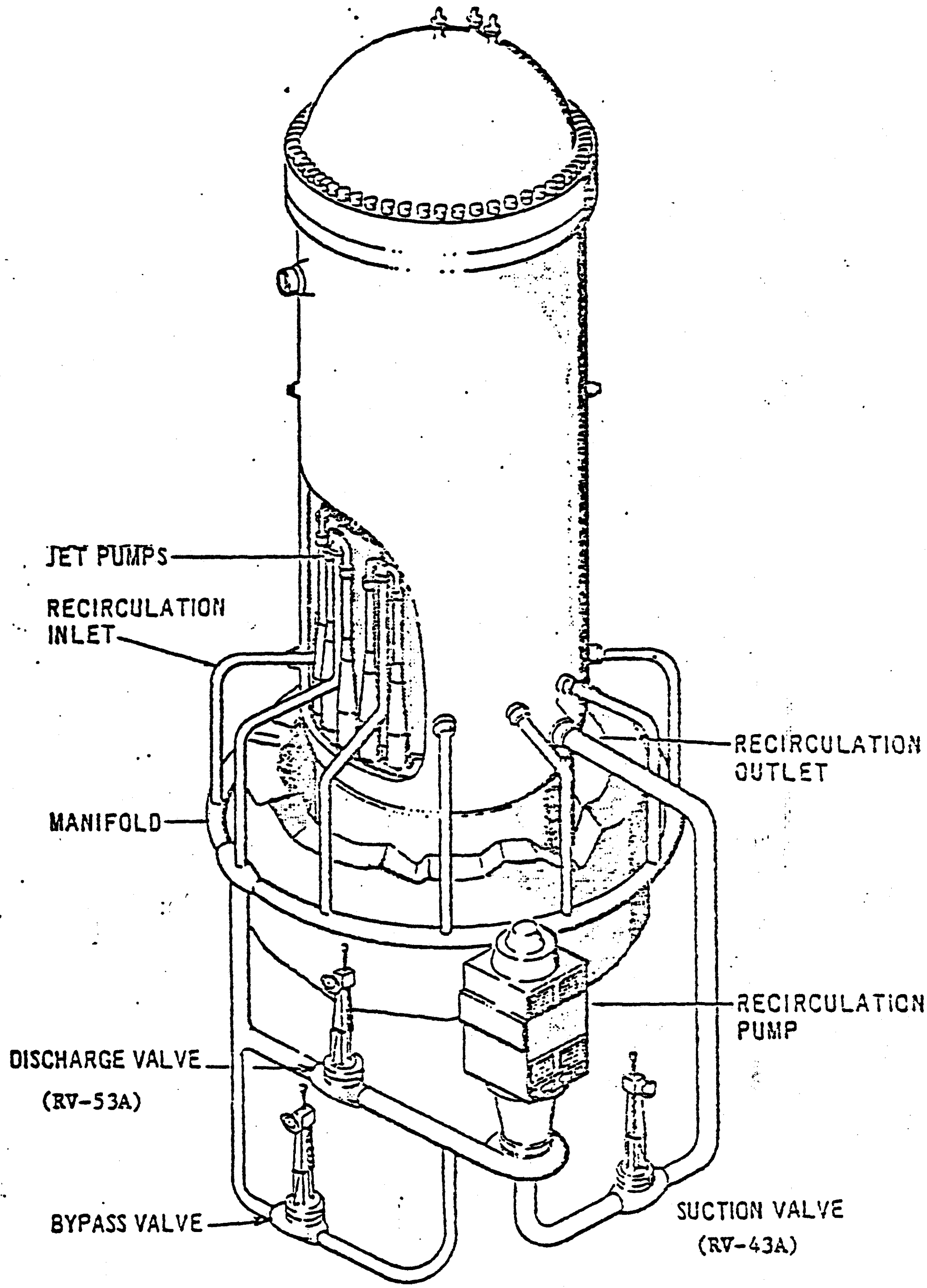


Figure 5.2 Schematic of BWR Reactor Coolant System[45]

Bonnet leaks in recirculation valves were found on March 23, 1975 in RV-43A and on August 13, 1979 in RV-53A. Other entries to the drywell included an inspection of one of the two recirculation pumps to check a possible water-to-oil cavity leak. The pump was secured and the reactor shutdown for repairs. During these entries, other malfunctions such as loose belts, stuck valves, fan failures, were discovered and the problem solved before resuming full-power operation.[20]

These entry data are used here as a conservative way to estimate the leakage failure rate in order to assess the possibility of a break in the recirculation system. Where data is not available, WASH-1400[32] failure data is used. The leakage failure rate for the individual valves are shown in Table 5.1. This individual leakage rate means that any of the valves or pumps of the recirculation system can leak above the normal leakage rate (1.4 to 1.8 gpm) up to 5 gpm (limit of unidentified leakage to shutdown the reactor).

The sequence of events that can lead to a loss-of-coolant accident in the recirculation system is shown in Fig. 5.3. In order to estimate the rate at which the leaks become breaks, a complete fatigue study would need to be completed for the individual valves. A rough estimate by Professor N. C. Rasmussen based on WASH-1400[32] establishes a factor of 2000 to 20000 between the probability of leakage and the probability of valve rupture. Leakage rate from the pumps is assumed to be negligible based on Vermont Yankee experience.

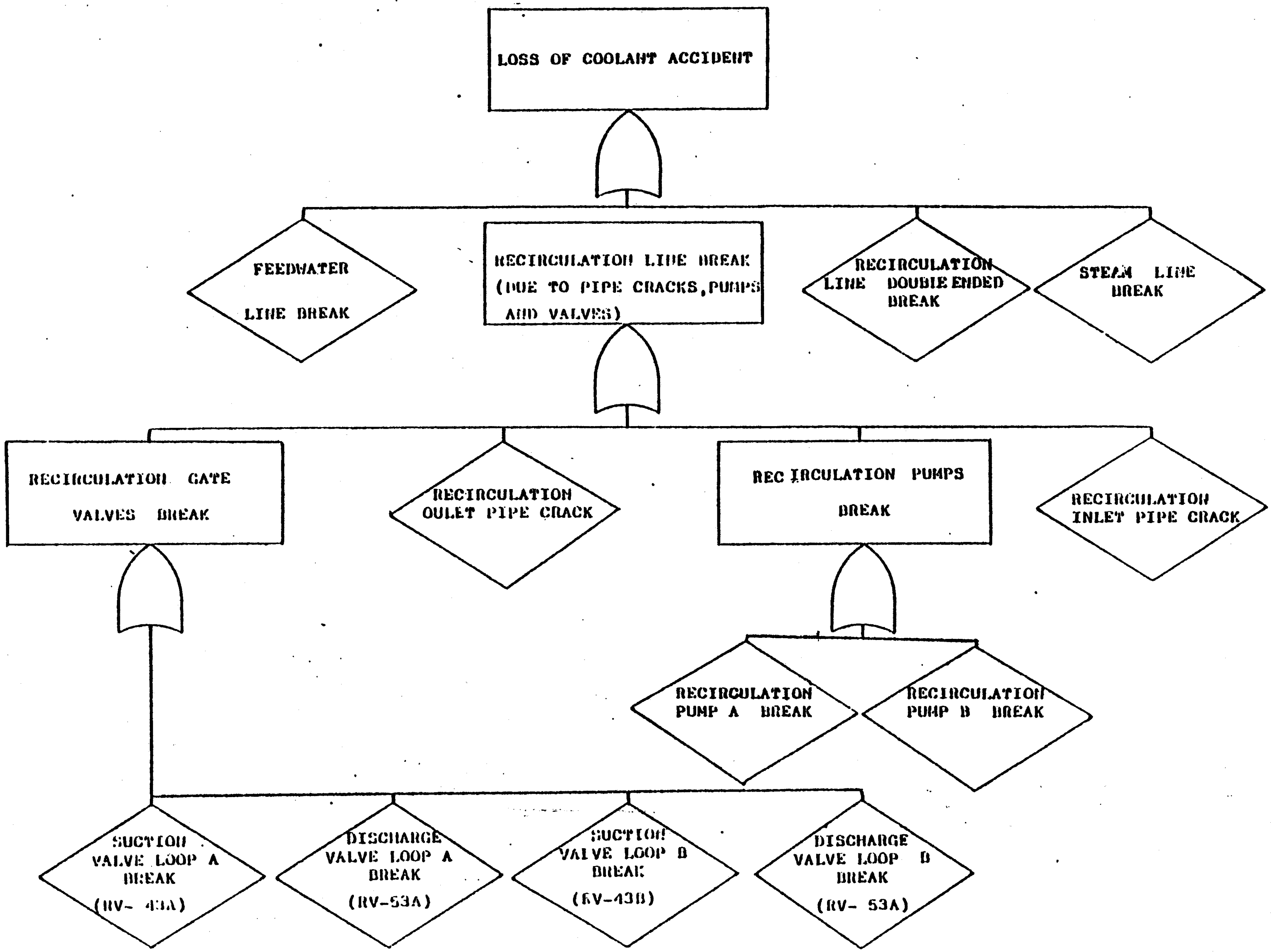
Table 5.1

LEAKAGE FAILURE RATES FOR VALVES OF THE RECIRCULATION SYSTEM
(65 months period)

VALVE	LEAKAGE FAILURE RATE (leak increase/hour)	
RV-43A	1.07×10^{-4}	*
RV 53A	2.13×10^{-5}	*
RV-43B	6.41×10^{-5}	*
RV-53B	1.00×10^{-8}	⊕

* Data from drywell entries at Vermont Yankee nuclear power station.

⊕ Data from WASH-1400.



<u>Figure 5.3</u> Fault Tree of LOCA Iniciating Events

Using the probabilities from Table 5.1, the fault tree of Fig. 5.3 is quantified to find an estimate of the contribution of valve leakage to the initiation of a small medium sized LOCA. From drywell entry experience, the probability of valve rupture from valve leakage is estimated to vary between 10^{-8} and 10^{-7}/hr. From WASH-100 failure data, the same probability is 4×10^{-8}/hr. The contributions from circumferential break, feedwater line break and steam line break are around 3×10^{-9}/hr. Using entry data and the propagation factors, the importance of stopping leaks in the recirculation system before a major problem develops can be estimated. Further experimental testing of the recirculation valves needs to be done in order to assess the effect of fatigue and thermal stress on the propagation factor, particularly those experienced during unscheduled reactor shutdowns.

The analysis shows that one might expect a reduction in the LOCA initiation rate of approximately one order of magnitude (from, say, 6×10^{-8}/hr. to 6×10^{-9}/hr.). This could theoretically be achieved by following the Vermont Yankee operating procedures for citing and correcting those problems accessible to drywell entries at power. Moreover, such practices reduce the shutdown frequency per year, thus reducing the probability of failure of the heat removal system, a major contributor to the total overall BWR accident risk.

5.3 Effects of Additional Shutdowns on Overall BWR Accident Risk

In previous sections, the effects of inerting on operating procedures and leakage rates were discussed. One of the points considered included the increase in the number of unscheduled shutdowns due to inerting during a shutdown, decay heat removal systems are required to operate to prevent core melt.[45] This condition is included in the transient events that dominate the releases in almost all the BWR risk categories (Table 5.2). The probability of failure of the decay heat removal system was determined in WASH-1400 to be ~ 1.6×10^{-6}/yr., which can be combined with the number of total shutdowns for reactor year. The difference between the number of shutdowns in a BWR operating with a CAD system and the number of shutdowns in a BWR operating with an inerted containment will directly affect the transient events that are dominant in BWR accident sequences. To determine this number, it is necessary to investigate the operational histories of BWR inerted containment shutdowns in order to investigate the shutdowns that could have been avoided if the containment had not been inerted. For example, about ten shutdowns per year can be expected in a BWR with an inerted containment.[45] Combining the probability of failure of the decay heat removal system with the ten transients per reactor year, this yields 1.6×10^{-5}/r-yr. for the sequence. Table 5.3 shows how the probability increases as the number of shutdowns increases. If the number of shutdowns is reduced to 5 per year by using a non-inerted containment, the probability decreases to 8×10^{-6}/r-yr. Due to the fact that this sequence is a dominant one, this reduction in probability affects directly the overall BWR accident risk. According to WASH-1400, the unavailability of the delay heat removal system is

Table 5.2

BWR DOMINANT ACCIDENT SEQUENCES OF EACH EVENT TREE vs. RELEASE CATEGORY[45]

	Core Melt				No Core Melt
	RELEASE CATEGORIES				
	1	2	3	4	5
LARGE LOCA DOMINANT ACCIDENT SEQUENCES (A)	$AE-\alpha$ $2x10^{-9}$ $AJ-\alpha$ $1x10^{-10}$ $AHI-\alpha$ $1x10^{-10}$ $AI-\alpha$ $1x10^{-10}$	$AE-\gamma'$ $3x10^{-8}$ $AI-\delta$ $1x10^{-8}$ $AJ-\gamma'$ $2x10^{-9}$ $AI-\gamma'$ $2x10^{-9}$ $AHI-\gamma'$ $2x10^{-9}$	$AE-\gamma$ $1x10^{-7}$ $AJ-\gamma$ $1x10^{-8}$ $AI-\gamma$ $1x10^{-8}$ $AHI-\gamma$ $1x10^{-8}$	$AGJ-\delta$ $6x10^{-11}$ $AEG-\delta$ $7x10^{-10}$ $AGHI-\delta$ $6x10^{-11}$	A $1x10^{-4}$
A Probabilities	$8x10^{-9}$	$4x10^{-8}$	$2x10^{-7}$	$2x10^{-8}$	$1x10^{-4}$
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_1)	$S_1E-\alpha$ $2x10^{-9}$ $S_1J-\alpha$ $3x10^{-10}$ $S_1I-\alpha$ $4x10^{-10}$ $S_1HI-\alpha$ $4x10^{-10}$	$S_1E-\gamma'$ $4x10^{-8}$ $S_1E-\delta$ $1x10^{-8}$ $S_1J-\gamma'$ $7x10^{-9}$ $S_1I-\gamma'$ $7x10^{-9}$ $S_1HI-\gamma'$ $6x10^{-9}$	$SE-\gamma$ $1x10^{-7}$ $S_1J-\gamma$ $3x10^{-8}$ $S_1I-\gamma$ $4x10^{-8}$ $S_1HI-\gamma$ $2x10^{-8}$ $S_1C-\gamma$ $3x10^{-9}$	$S_1GJ-\delta$ $2x10^{-10}$ $S_1GE-\delta$ $2x10^{-10}$ $S_1EI-\epsilon$ $1x10^{-10}$ $S_1GHI-\delta$ $2x10^{-10}$	
S_1 Probabilities	$1x10^{-8}$	$9x10^{-8}$	$2x10^{-7}$	$2x10^{-8}$	
SMALL LOCA DOMINANT ACCIDENT SEQUENCES (S_2)	$S_2J-\alpha$ $1x10^{-9}$ $S_2I-\alpha$ $1x10^{-9}$ $S_2HI-\alpha$ $1x10^{-9}$ $S_2E-\alpha$ $5x10^{-10}$	$S_2E-\gamma'$ $1x10^{-8}$ $S_2E-\delta$ $4x10^{-9}$ $S_2J-\gamma'$ $2x10^{-8}$ $S_2I-\gamma'$ $2x10^{-8}$ $S_2HI-\gamma'$ $2x10^{-9}$	$S_2E-\gamma$ $4x10^{-8}$ $S_2J-\gamma$ $8x10^{-8}$ $S_2I-\gamma$ $9x10^{-8}$ $S_2HI-\gamma$ $9x10^{-8}$ $S_2C-\gamma$ $8x10^{-9}$	$S_2CG-\delta$ $6x10^{-11}$ $S_2GHI-\delta$ $6x10^{-10}$ $S_2EG-\delta$ $3x10^{-10}$ $S_2GJ-\delta$ $6x10^{-10}$ $S_2GI-\delta$ $2x10^{-10}$	
S_2 Probabilities	$2x10^{-8}$	$1x10^{-7}$	$4x10^{-7}$	$4x10^{-8}$	
TRANSIENT DOMINANT ACCIDENT SEQUENCES (T)	$TW-\alpha$ $2x10^{-7}$ $TC-\alpha$ $1x10^{-7}$ $TQUV-\alpha$ $5x10^{-9}$	$TW-\gamma'$ $3x10^{-6}$ $TQUV-\gamma'$ $8x10^{-8}$	$TW-\gamma$ $1x10^{-5}$ $TC-\gamma$ $1x10^{-5}$ $TQUV-\gamma$ $4x10^{-7}$		
T Probabilities	$1x10^{-6}$	$6x10^{-6}$	$2x10^{-5}$	$2x10^{-6}$	
PRESSURE VESSEL RUPTURE ACCIDENTS (R)		P.V. RUPT. $1x10^{-8}$ Oxidizing Atmosphere	P.V. RUPT. $1x10^{-7}$ Non-oxidizing Atmosphere		
R Probabilities	$2x10^{-9}$	$1x10^{-8}$	$1x10^{-7}$	$1x10^{-8}$	
SUMMATION OF ALL ACCIDENT SEQUENCES PER RELEASE CATEGORIES					
MEDIAN (50% VALUE)	$1x10^{-6}$	$4x10^{-6}$	$2x10^{-5}$	$2x10^{-6}$	$1x10^{-4}$
LOWER BOUND (5% VALUE)	$1x10^{-7}$	$1x10^{-6}$	$5x10^{-6}$	$5x10^{-7}$	$1x10^{-5}$
UPPER BOUND (95% VALUE)	$5x10^{-6}$	$1x10^{-5}$	$4x10^{-5}$	$1x10^{-5}$	$1x10^{-3}$

NOTE: The probabilities for each release category for each event tree and the Σ for all accident sequences are the median values of the dominant accident sequences summed by Monte Carlo simulation plus a 10% contribution from the adjacent release category probability.

KEY TO TABLE 5.2 ON FOLLOWING PAGE

KEY TO BWR ACCIDENT SEQUENCE SYMBOLS[45]

A - Rupture of reactor coolant boundary with an equivalent diameter of greater than six inches.

B - Failure of electric power to ESFs.

C - Failure of the reactor protection system.

D - Failure of vapor suppression.

E - Failure of emergency core cooling injection.

F - Failure of emergency core cooling functionability.

G - Failure of containment isolation to limit leakage to less than 100 volume per cent per day.

H - Failure of core spray recirculation system.

I - Failure of low pressure recirculation system.

J - Failure of high pressure service water system.

M - Failure of safety/relief valves to open.

P - Failure of safety/relief valves to reclose after opening.

Q - Failure of normal feedwater system to provide core make-up water.

S_1 - Small pipe break with an equivalent diameter of about 2"-6".

S_2 - Small pipe break with an equivalent diameter of about 1/2"-2".

T - Transient event.

U - Failure of HPCI or RCIC to provide core make-up water.

V - Failure of low pressure ECCS to provide core make-up water.

W - Failure to remove residual core heat.

α - Containment failure due to steam explosion in vessel.

β - Containment failure due to steam explosion in containment.

γ - Containment failure due to overpressure - release through reactor building.

γ' - Containment failure due to overpressure - release direct to atmosphere.

δ - Containment isolation failure in drywell.

ϵ - Containment isolation failure in wetwell.

ζ - Containment leakage greater than 2400 volume per cent per day.

η - Reactor building isolation failure.

θ - Standby gas treatment system failure.

KEY TO TABLE 5.2

Table 5.3

DECAY HEAT REMOVAL SYSTEM PROBABILITY PER NUMBER OF REACTOR SHUTDOWNS IN A YEAR

NUMBER OF REACTOR SHUTDOWNS	PROBABILITY/REACTOR-YEAR DECAY HEAT REMOVAL SYSTEM
1	1.6×10^{-6}
2	3.2×10^{-6}
3	4.8×10^{-6}
4	6.4×10^{-6}
5	8.0×10^{-6}
6	9.6×10^{-6}
7	1.1×10^{-5}
8	1.3×10^{-5}
9	1.4×10^{-5}
10	1.6×10^{-5}

responsible for ~ 64.5% of the total risk. A recent study (EPRI, 1978) shows that the delay heat removal system is responsible for ~ 83% of the total risk.[48] On the other hand, Tony Buhl [46] and Robert Bernero [47] show that transient events and their consequences remain essentially unaffected by use of a non-inerted containment (Table 5.4 and 5.5). The model used in Buhl's study assumed that all core melts from LOCAs resulting in hydrogen explosions which rupture the containment demonstrate that the overall risk is insensitive to containment inerting. Bernero also indicates that with respect to the failure of the shutdown heat removal system, inerting has a negligible impact on the overall BWR risk.

TABLE 5.4

EFFECTS OF A NON-INERTED CONTAINMENT ON RISK[46]

EVENT TREE	CONTAINMENT ATMOSPHERE	RELEASE CATEGORIES			
LARGE LOCA	INERTED NON-INERTED	$8x10^{-9}$ $3x10^{-8}$	$6x10^{-8}$ $2x10^{-7}$	$2x10^{-7}$ $2x10^{-8}$	$2x10^{-8}$ $2x10^{-9}$
SMALL LOCA	INERTED NON-INERTED	$1x10^{-8}$ $3x10^{-8}$	$9x10^{-8}$ $3x10^{-7}$	$2x10^{-7}$ $3x10^{-8}$	$2x10^{-8}$ $3x10^{-9}$
SMALLEST LOCA	INERTED NON-INERTED	$2x10^{-8}$ $6x10^{-8}$	$1x10^{-7}$ $5x10^{-7}$	$4x10^{-7}$ $5x10^{-8}$	$4x10^{-8}$ $5x10^{-9}$
TRANSIENTS	INERTED NON-INERTED	$1x10^{-6}$ $1x10^{-6}$	$6x10^{-6}$ $6x10^{-6}$	$2x10^{-5}$ $2x10^{-5}$	$2x10^{-6}$ $2x10^{-6}$
REACTOR VESSEL RUPTURE	INERTED NON-INERTED	$2x10^{-9}$ $1x10^{-8}$	$2x10^{-8}$ $1x10^{-7}$	$1x10^{-7}$ $1x10^{-8}$	$1x10^{-8}$ $1x10^{-9}$
SUMMATION OF ALL SEQUENCES	INERTED NON-INERTED	$1x10^{-6}$ $1x10^{-6}$	$6x10^{-6}$ $7x10^{-6}$	$2x10^{-5}$ $2x10^{-5}$	$2x10^{-6}$ $2x10^{-6}$

Table 5.5

PERSPECTIVES ON RSS-BWR DESIGN (BWR 4, MARK I CONTAINMENT, INERTED)[47]

CONTAINMENT OVERPRESSURE FAILURE SCENARIOS				POTENTIAL RISK IMPACT OF SCENARIO	INERTING	CONTROLLED VENTING FILTER
Transient followed by failure to shutdown.	CONT. FAILS 77 MIN.	MELT STARTS 100 MIN.	MELT ENDS 144 MIN.	Large (Dominant) Sequence	Negligible	Small to Moderate
Transient followed by failure of shutdown heat removal system.	CONT. FAILS 2820 MIN.	MELT STARTS 3260 MIN.	MELT ENDS 3390 MIN.	Large (Dominant)	Negligible	Moderate to Large
Transient followed by failure to provide make-up water.	START MELT 160 MIN.	END MELT 200 MIN.	CONT. FAILS 232 MIN.	Medium to Small	Small	Moderate to Large
Small LOCA followed by failure to provide make-up water.	START MELT 57 MIN.	END MELT 102 MIN.	CONT. FAILS 117 MIN.	Medium to Small	Small	Small to Moderate

MARCH/CORRAL; BCL Accident Process Modeling

CHAPTER SIX

CONCLUSIONS AND RECOMMENDATIONS

The probabilistic safety analysis performed in this thesis shows that the inerting and air dilution systems have approximately the same overall probability of handling the hydrogen generated during an accident, preventing the hydrogen from reaching flammability limits that could lead to combustible mixtures. Inerting controls the combustible mixture over the entire range of accidents, from high probability-low consequence accidents, to low probability-high consequence accidents, while the air dilution system can handle the hydrogen only for low hydrogen generation accidents (high probability of occurrence). Depending on the assumptions made,(i.e.hydrogen produced in one hour or longer at a "medium" generation rate) the CAD system can handle the hydrogen with a higher (0.0899) or lower (0.0179) probability of success. For hydrogen generated in low probability-high consequence accidents, the CAD system is not useful. The fault tree analysis represents an organized source of information required to improve the designs of the hydrogen control systems (i.e., reducing operator dependence and including greater redundancy).

Accident initiating events can be reduced in non-inerted containments since drywell entries at power permit identification, evaluation, and repair of leaks. Entries cannot be done in inerted containments due to hazards due to lack of oxygen. The probability of leaks becoming breaks is so low that their tolerance does not affect the overall total BWR accident risk. However, entires can reduce the number of unscheduled shutdowns, affecting not only the capacity factor of the plant but also the probability of failure of the decay heat removal system because each time the reactor is shutdown (whether planned or not), the decay heat removal system has to be used to prevent core melt. This condition is included in the dominant transients vents in BWRs. These probabilities are affected directly by the number of reactor shutdowns.

The calculations in this thesis were performed using conservative values and assumptions in order to structure the methodology. The study was based on the four volume percent hydrogen flammability limit established by the NRC Regulatory Guide 1.7. Detailed further studies on hydrogen properties are recommended in order to establish more accurate data for designing better hydrogen control systems, and for calculating hydrogen control probabilities on the systems. It should be further be noted that inerting may have certain beneficial effects such as reduced corrosion that have not been accounted for in this analysis.

This thesis analyzed two hydrogen control systems operating in two existing plants, and provides a comparison between the two systems. Systems can vary from plant to plant, and the results can be different. However, when comparing an inerted plant with a non-inerted one, the number of unscheduled shutdowns will have a direct effect on the overall risk. Therefore, detailed analysis of shutdowns and their causes is recommended in order to extend the comparison between inerted and non-inerted containments.

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Appendix

OTHER CONSIDERATIONS RELATED TO INERTING IN BWR CONTAINMENTS

A. Introduction

Two other considerations related to inerting were identified in conversations with utility engineers.* These include the potential for positive benefit from inerting in: (1) reducing the corrosion rates of the torus vessel and the termination boxes of the electrical outlets found in the torus, and (2) reducing the likelihood of fires inside the primary containment compartment. These considerations were not analyzed in detail in Lepervanche's engineers thesis since that analysis concerned itself primarily with the issue of hydrogen control and the differences in the impact on public health and safety. The issues of corrosion and fire are analyzed in more detail here.

B. Effects on Corrosion Rates in the Torus

Utility engineers contacted here observed some reduction in corrosion effects on the torus vessel at the Pilgrim I and Millstone BWR plants.[1,2] Although this effect has been attributed to the reduction in the oxygen content in the torus atmosphere due to inerting, a quantitative comparison of corrosion effects between non-inerted and inerted BWRs has not taken place. Even so, the corrosion effect would have limited impact on the overall BWR safety risk as the only major impact such an effect might have

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* The utility engineers contacted were those identified as having supported the concept of inerting from viewpoints other than (or including) the specific hydrogen control issue. These engineers included representatives of Northern States Power Co. (Musolf) and the Institute for Nuclear Power Operations (Rosen).

is in producing so much debris as to begin to clog up the screens on the ECCS system.[2] However, these screens have a grid size of 1/2" so that only very substantial corrosion effects could stand to produce the size of debris particle that might pose such a problem.

Other utilities contacted[3] have not observed such effects, and argue that the dissolved oxygen content in the torus water would not vary significantly between the inerted and non-inerted case to warrant a substantial effect on the torus corrosion rate. They further argue that the protective painted coating on the torus surface protects sufficiently against major corrosion problems, and therefore that the identified potential advantage of inerting due to corrosion is not a significant one in any case.

An additional effect observed by one utility[2] was a reduced corrosion effect on the termination boxes of the electrical outlets found in the torus. Again, the observed effect was not major, but as was pointed out in the conversation, might be of potential importance. As the termination boxes are fully insulated against water leakage and are designed to withstand high corrosion rates[4], the observed small reduction in box corrosion rate is not considered to have a significant effect on the failure rates of the electrical circuitry and instrumentation related to these boxes and therefore, has no significant impact on the overall risk calculation.

C. <u>The Impact of Inerting on the BWR Primary Containment Compartment Fire Hazard</u>

The issue of fires in the BWR primary containment compartment was identified as being a significant potential benefit of inerting.[2] In subsequent conversations with licensing engineers versed in the area of BWR fire risk,[4,5] it was discovered that there are two sources of combustible material inside the primary containment of a Mark I BWR: (1) the reactor recirculation pump fuel oil (50 gallons in each of the two pumps found on the primary coolant loop), and (2) the electrical cable, which is fire resistant but can ignite at higher temperatures.

There are three possible ways in which a fire inside the containment can be initiated: (1) oil leak from the recirculation pump during operation, (2) during shutdown and maintenance a welding related oil fire where welding catches the fuel oil on fire, and (3) electrical motor fire in the pump. All of these events relate to the recirculation pump; in cases (1) and (2), the event can lead potentially to a major fire in the containment defined to be where the electrical cabling would also be affected; in case (3), the fire would be confined to the pump casing itself but would result in pump failure. This would not be as significant a problem since adequate cooling can be maintained by either one of the recirculation pumps - even in the event of a simultaneous failure of both pumps, the BWR can be sufficiently cooled by natural recirculation.[4]

Impact of Primary Containment Fires on RHRS Availability

The worst possible scenario involving a fire in the primary containment would be a loss of both recirculation pumps as a result of an oil leak for one pump - igniting a fire spreading to the electrical cables, then igniting the second recirculation pump oil supply.* In this scenario, both pumps would thus be made inoperative, requiring that auxiliary cooling systems be employed to help ensure adequate cooling. Since a loss of primary coolant is not an expected result of even the worst fire scenario within the primary containment, failure of auxiliary cooling systems would not

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* This is a low probability scenario as the recirculation pump oil supplies are shielded against fire as a precautionary measure.

result in a serious problem since the BWR is designed such that adequate cooling is provided for natural circulation in the reactor core. Further, the auxiliary cooling system major components are located exterior to the primary containment (see Figure A.1). A fire in the primary containment cannot likely lead, therefore, to the initiation of a loss of coolant accident; although the fire would result in an additional reactor shutdown, it would not likely lead to an effect on the RHRS (residual heat removal system). With a fire occurrence frequency in the drywell of between $1.6 \times 10^{-2} - 10^{-3}$ per reactor year* it can be shown that the impact of a fire on the number of shutdowns per year is negligible as the average number of unplanned shutdowns currently rests between 2-6 per reactor-year (see Table A.I). Thus, the impact of fires in the primary containment on the dominant BWR accident sequence is relatively negligible.

From another standpoint, it is also possible to show that the potential benefit from inerting with respect to fires is overshadowed by the disbenefit of inerting with respect to early maintenance and inspection (this disbenefit leads to a possible order-of-magnitude increase in the small-to-medium size LOCA initiation rate (see p. 110, Lepervanche)). Figure A.2 shows the fire event tree for the inerted vs non-inerted cases. With the addition of an oil leak collection system on each recirculation pump, the significant fire initiation rate drops from 1.16×10^{-6}/hr to 1.01×10^{-7}/hr in the non-inerted case, compared with a range of 2.6×10^{-7}/hr to 1.09×10^{-7}/hr in the inerted case (see Table A.II).

* The value of 1.6×10^{-2}/r-year comes from a paper by Apostolakis and Kazarian,[6] which reports this value for a fire in an LWR containment. Although not specifically applicable to the BWR drywell, this number is assumed to provide an upper estimate on the frequency of a drywell fire.

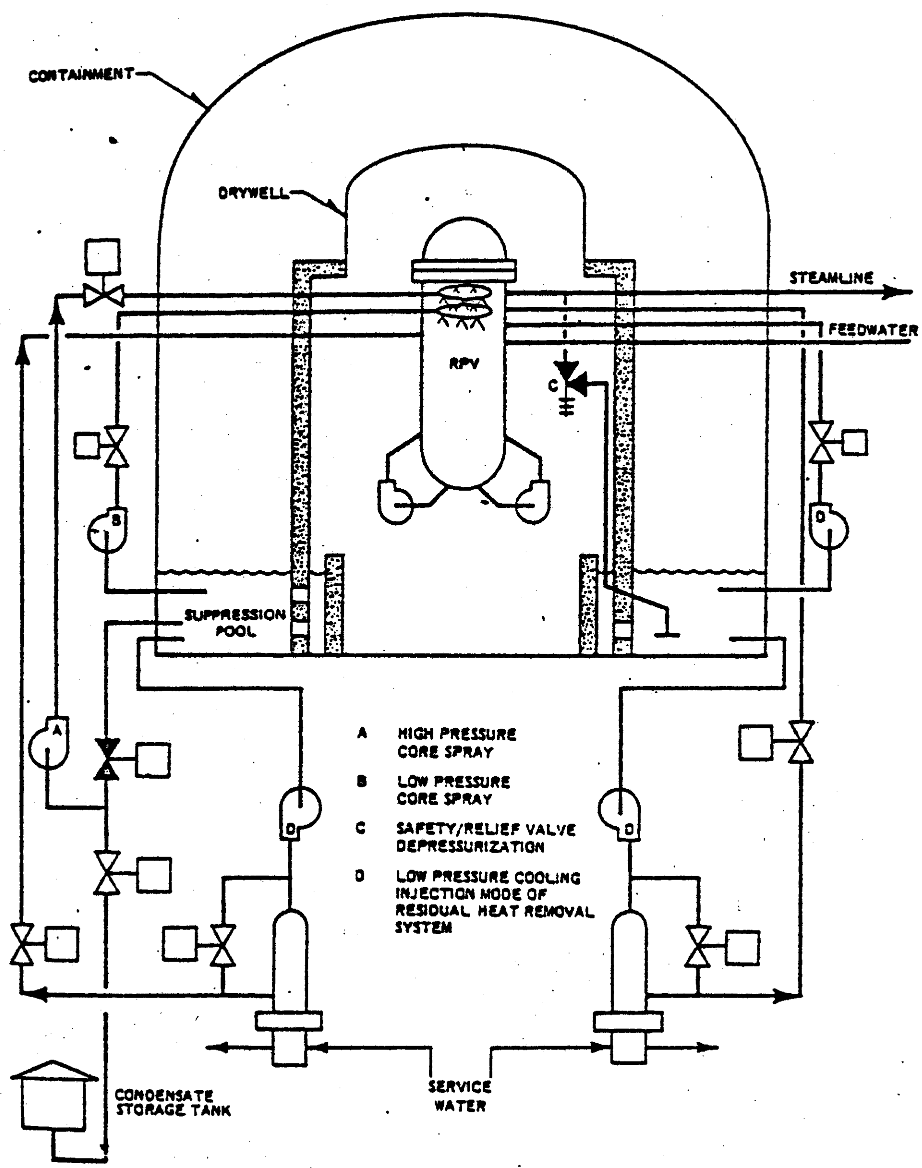


Figure A.1 ECCS SYSTEM INCLUSIVE OF LOW PRESSURE COOLING INJECTION MODE OF RESIDUAL HEAT REMOVAL SYSTEM

Table A.I

COMPARISON OF EXPERIENCE: INERTED BWRs VS. NON-INERTED BWRs*

Plant Name	Avg. No. of unscheduled entries/yr	% of entries resulting in plant shutdown for repair	# shutdowns/yr	Plants normally operated with small leakage	Entries normally performed with plant inerted
Hatch, Unit 1	5	64	3.2	Yes	No
Cooper	1	100	1	Yes	No
Nine Mile Point, Unit 1	3	92	2.8	Yes	No
Brunswick, Unit 1	6	70	4.2	Yes	No
FitzPatrick	2	100	2	Yes	No
Quad Cities, Unit 1	4	54	2	Yes	No
Quad Cities, Unit 2	2	43	1	Yes	No
Peach Bottom, Unit 2	3	?	3	Yes	No
Peach Bottom, Unit 3	4	?	4	Yes	No
Monticello	2	100	2	Yes	No
Pilgrim	3	100	3	Yes	No
Dresden, Unit 2	3	90	2.7	Yes	No
Dresden, Unit 3	2	90	1.8	Yes	No
Duane Arnold	2	100	2	Yes	No
Browns Ferry, Unit 1	3	?	3	Yes	No
Browns Ferry, Unit 2	1	?	1	Yes	No
Browns Ferry, Unit 3	4	?	4	Yes	No
Vermont Yankee	4	20	.8	Yes	N/A
Hatch, Unit 2	9	100	9	Yes	N/A

* From NRC staff position (Butler, 1980).

Table A.II

Serious Fire Initiation Rate Per Hour in a BWR Drywell
(With/Without Oil Leak Collection System on Recirc Pumps)

Fire Initiation Event	Inerted		Non-Inerted	
	With	Without	With	Without
Welding	1.09×10^{-7}		9.89×10^{-8}	
Oil Leak	1.5×10^{-10}	1.5×10^{-7}	1.06×10^{-9}	1.06×10^{-6}
Total	1.09×10^{-7}	2.59×10^{-7}	1.01×10^{-7}	1.16×10^{-6}

From Figure A.2, it is evident that when no oil spill collection system is installed in the non-inerted case, the fire initiation rate is ~5 times greater than for the inerted case. The installation of such a system causes the welding initiated fires to dominate the overall yearly fire risk in the drywell such that the difference between the inerted and non-inerted cases is quite small. Also, since inerting may result in producing more unscheduled reactor shutdowns per year, the fire initiation rate may be less for the non-inerted case (by a factor of ~1.08 given oil collection systems are installed in both cases and assuming twice as many unscheduled shutdowns per year for the inerted case).* Thus, it is concluded that as long as an oil leak collection system is installed on each recirculation pump, the difference in fire hazard between the inerted and non-inerted cases is relatively negligible since then the welding initiated fires during shutdowns dominate the total yearly drywell fire risk.

* This factor may be a negligible one in that the dominant contribution to the yearly shutdown time is the 6 week period assumed for annual refueling.

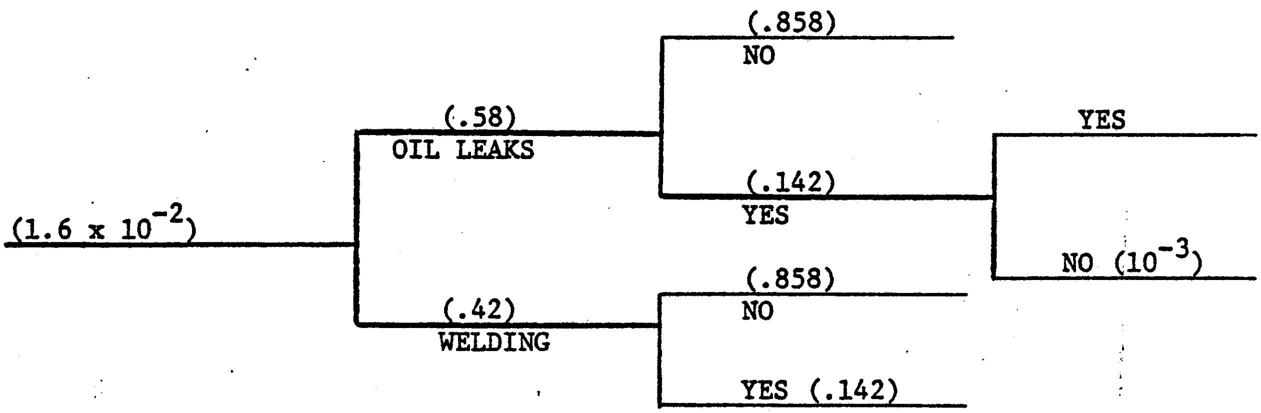
Figure A.2

EVENT TREE FOR FIRES INITIATED INSIDE BWR
INERTED (OR NON-INERTED) DRYWELL CONTAINMENT STRUCTURES

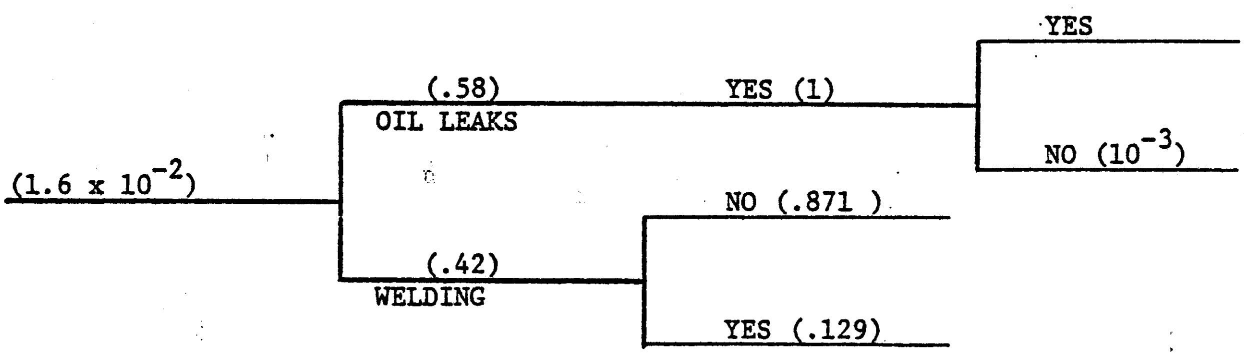
Event Catgories

Yearly Fire Initiation Rates in LWR Containments (from Apostolakis et al.)[6]	Percentage of Recorded Fires Due to Oil Leaks and Welding (from French Study)[7]	Percentage of Year Containment Susceptible to Indicated Fire Type*	Oil Collection System Installed on Recirc Pumps (Failure Probability of System Assumed 10^{-3} per year) Fire Resistant

Inerted Case



Non-Inerted Case



* For the inerted case, the containment is only susceptible to fires of any type during periods of shutdown; likewise, for the non-inerted case in the welding event sequence. The percentages were calculated assuming 6 weeks for annual refueling added to the number of unscheduled shutdowns per year (assumed 4 per year for inerting, 2 per year for non-inerting).

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