GENERAL ELECTRIC
PROPRIETARY INFORMATION

RESPONSE TO GESSAR SOURCE TERM
ISSUE 9

SUPPRESSION POOL BYPASS

IN BWR'S

4

CONTENTS

GENERAL ELECTRIC PROPRIETARY INFORMATION

TABLES

Table	Title	Page

ILLUSTRATIONS

Figure	Title	Page

GENERAL ELECTRIC PROPRIETARY INFORMATION

1. INTRODUCTION

Recent discussions between the General Electric Company and the US Nuclear Regulatory Commission and the National Laboratories have stressed the fission product retention capability of BWR suppression pools. For events which transport the fission products to the suppression pool, this retention capability has been shown to reduce the inventory of fission products available for release to the environment to levels far below levels prescribed by the US Regulatory Guides. As a conséquence, there are potentially significant impacts on BWR Emergency Planning, location of equipment, and Probabilistic Risk Assessments (PRA) - all supporting the extreme safety of the BWR and low consequence of potential release to the general public. Table 1-1 summarizes the results of a recently completed PRA showing the extremely low levels of public risk (0.265 manrem/reactor year).

A key concern following this assessment was the degree to which there may be pathways for fission products to be released to the environment which do not pass through (bypass) the suppression pool, although some bypass was treated by the PRA as shown on Table 1-1. The concern was that there may be pathways not addressed by the risk assessment which could have a large impact on exposure of the public.

This study was thus conducted to address these concerns and to show that the BWR design effectively removes potential bypass pathways from the standpoint of public risk to fission products. Although previous work was based on a BWR/6 with a Mark III containment, the study was extended to evaluate the impact of bypass on Mark I and II containments.

This study shows that small bypass lines are severely restricted so that an insignificant release of fission products occur. The design of the containment isolation system on larger lines is sufficiently reliable to make the risk of other bypass paths far less than the pathways which include the suppression pool. The study also shows that for all pathways there are natural fission product removal mechanisms which effectively eliminate these pathways from concern.

The conclusion of the study is that BWR suppression pool bypass pathways are not a concern from the standpoint of public risk from fission product exposure.

2. IDENTIFICATION OF BYPASS PATHS

Fission products released from the fuel in a severely degraded accident may be released to the environment by several pathways as indicated in Figure 2-1. Both liquid and gaseous release pathways are considered in the figure.

As shown, the dominant release pathways are to the suppression pool either through the safety relief valves (SRV) or through the drywell to wetwell vents. The dominance of these paths is assured by the plant design which isolates the significant release paths to direct the flow of RPV effluent to the suppression pool.

Other release paths are possible, however, through small leakage paths or paths where the isolation system has failed to function. The following subsections discuss these bypass paths.

2.1 LIQUID RELEASE PATHWAYS (FIGURE 1, PATH A)

Liquid leaks or condensate from steam leaks are normally collected by sumps in all areas which could potentially contain radioactive material. These sumps are discharged to the radwaste building where they are processed and recycled back to the plant. The radwaste building contains a basemat designed to withstand design basis seismic loadings so that even if tank failures occur in the Radwaste Building a release to the environment does not occur.

Once in a subcooled liquid form, liquid effluents are limited in reaching the general public by inadvertent releases, or vaporization. These methods are either unlikely or involve small release fractions.

For the above reasons, liquid releases to the environment are not considered a particular hazard. Furthermore if an inadvertent release to a river or lake were to occur the contamination could be quickly diluted and alternate drinking water supplies could be temporarily used to avoid an impact on the general public.

Potential releases from fluid systems have thus not been considered further in this study.

2.2 AIRBORNE RELEASE PATHS

The airborne releases during routine plant operation or following transients or accidents consist of noble gases, halogens and particulates. In the most severe accidents nearly all of the fuel inventory of fission products may be released from the fuel. However the halogen and particulate fission products are substantially retained within the vessel, associated piping, containment air or the suppression pool which limits the amount of activity which is available for release to the environment. As for noble gases, although there may be some holdup in plant buildings prior to release, most are expected to ultimately be released to the environment following a severe accident.

This study has concentrated on the halogen and particulate releases because these are the types of radioactive material which pose the greatest hazard to the general public if they are not retained. The release of even 100% of the core inventory noble gases has been shown to cause negligible health effects.

2.2.1 Potential Release Pathways Outside Secondary Containment (Figure 1, Path B)

The secondary containment boundary in BWRs contains all potentially radioactive systems except for systems supplied by the main steam system and its condensate. The lines which pass outside of this boundary and which communicate with the RPV or drywell are identified for a BWR/6 Mark III design in Table 2-1. Other product lines are similar in that the lines contain main steam and feedwater, RPV drains, HVAC exhaust, sump discharge, and cooling water supply and return lines.

Table 2-1 shows that all lines contain primary containment isolation valves which can be closed by remote manual control or which receive automatic closure signals in response to conditions representing a potential break outside containment. The lines in Table 2-1 can be viewed in three groups.

a. Lines which connect directly to the reactor pressure vessel are designed to General Design Criteria 55. These lines provide a potential release path for all severe accident events if the containment isolation system fails to function.

b. Lines which connect to the drywell atmosphere or a system (such as RWCU) not directly part of the coolant pressure boundary are designed to General Design Criteria 56. These lines represent a potential bypass pathway only for events which cause a break of the RPV pressure boundary inside the drywell. The RWCU lines represent a bypass path only for events where the RWCU system fails to isolate.

c. Lines which are closed inside the drywell are designed to General Design Criteria 57. These lines represent a bypass pathway only if there is a break in the closed system to cooling water interface combined with system failures such that a pressure difference favors release to the cooling water lines.

These potential bypass pathways occur only if the isolation system fails to function and, for groups b and c, if another system break has occured. For lines in group 1 a third remote manual isolation valve is also provided to provide an additional level of reliability beyond the primary containment isolation design.

For lines which pass outside secondary containment in the 238 Nuclear Island design, a positive leakage control system is also provided to reduce potential release due to isolation valve leakage. Other product lines contain MSIV leakage control systems which direct leakage to the standby gas treatment system.

The lines which are potential bypass pathways to areas outside the secondary containment are protected by a system design which provides an extremely high level of isolation reliability. These high reliabilities make the risk to the general public extremely small. Section 3 discusses the probability of release in more detail. Section 4 discusses the retention of fission products in these lines.

2.2.2 Potential Release Pathways Inside Secondary Containment (Figure 1, Path C)

Lines which connect either to the RPV or to the drywell atmosphere and which terminate inside the secondary containment on all product lines are identified in Table 2-2. Table 2-3 identifies other lines which terminate in secondary containment in Mark I and Mark II containment designs. On Mark III designs these lines terminate inside the primary containment, but outside of the drywell.

The release of radioactive material through small leaks in any of these lines to the environment is treated by the Standby Gas Treatment System (SGTS) which may be initiated in response to a process radiation system signal. Thus although these pathways may bypass the suppression pool the SGTS provides a degree of fission product removal for any particulates or halogens which might be released. Since the decontamination factor of the SGTS is about 1000, it provides a degree of removal nearly equivalent to that provided by suppression pool scrubbing.

Large breaks of the lines in Table 2-2 may cause overpressurization of their local compartments unless a protective feature such as blowout panels or equivalent are provided in this design. In any case, such large breaks present a potential release pathway to the environment which would not be treated by the Standby Gas Treatment System.

As shown in Tables 2-2 and 2-3 all lines which communicate with the reactor pressure boundary or drywell atmosphere also contain primary containment isolation valves which may be remote manually closed to limit the duration of any detected leak (instrument lines on Mark I and II plants contain excess flow check valves). In addition the non-essential lines automatically close on response to a detected system leak. These features limit the risk to the general public from releases through these pathways.

The suppression pool suction lines in general do not contain the same degree of isolation as do the other lines on Table 2-3. This is acceptable, however, because they are a liquid source (refer to Section 2.1) which does not pose as great a hazard to the general public as do the other sources.

The suppression pool suction lines, however, are also potential pathways which could lower the suppression pool water level and reduce the effectiveness of pool scrubbing. While breaks of these lines outside of the containment are a possibility, the break possibility is extremely remote ($<10^{-6}$ yr) because they do not contain high pressure fluid. Furthermore should a leak occur, (except at the containment boundary), it would be detected by the plant Leak Detection System and the appropriate line could be quickly isolated by the operator. On the Mark III design, an automatic suppression pool makeup system, actuated on low suppression pool water level, gives additional assurance that the suppression pool level is maintained for a period of time. Suppression pool suction lines are not considered further in this report.

Finally, there are release paths from the drywell through structural features such as hatches or penetrations. Periodic containment testing ensures that leakage through these paths is kept to a minimum. But regardless of the leak rate, except for catastrophic structural failure, they would be expected to provide a highly restricted flow path. Retention of fission products in such pathways is discussed in Sections 3 and 4.

2.2.3 Potential Suppression Pool Bypass Paths Inside Containment (Figure 1, Paths D and E)

As mentioned earlier, the BWR design directs the dominant flow in transients or in breaks inside the drywell to the suppression pool. The safety relief valves direct the decay heat steam to the suppression pool while any excessive pressure in the drywell (as a result of a break for instance) is relieved to the suppression pool through the vents. Bypass paths to the containment may be a concern if they provide a release path directly to the environment from breaks inside the drywell or wetwell or from drywell to wetwell leakage.

For Mark I and II containment designs, there are no components connected directly to the RPV in the wetwell above the suppression pool water level. Thus wetwell breaks are not possible. In the Mark III containment design there are several such systems which can provide a bypass path inside containment. These were identified in Table 2-3. In addition there are several high energy lines which pass through the wetwell in Mark III containments. All these lines contain guard pipes designed to direct any line break flow back to the drywell.

Drywell to wetwell airspace leakage is possible whenever the differential pressure between the two zones favors it. It is limited, however, by the head created by the submergence depth of the vents. For all BWR designs this is only a few pounds of pressure (psi). Thus there is never a large driving force for drywell to wetwell air leakage.

For Mark III containments the drywell is completely surrounded by the wetwell and suppression pool. Only Mark I and II containment designs have the potential for direct drywell bypass leakage to the secondary containment buildings. For these designs only gross structural failure, such as from an external event* or hydrogen explosion or isolation valve failures can cause a significant pathway. Since Mark I and II containment designs are inerted during normal operation, only external events or isolation system failures are reasonable contributors to this type of pathway. Further holdup and retention of fission products in the secondary containment mitigates these releases as discussed in Section 4.

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*Seismic, tornado, tsunami, etc.

TABLE 2-1

PATHWAYS WHICH TERMINATE OUTSIDE SECONDARY CONTAINMENT

LINE	FLUID	LEAKAGE BARRIER TYPE (1)	ISOLATION DESIGN CRITERIA (2)	VALVE TYPES (3)
From RPV				
26" Main Steam	Steam	PC, 3IV, LCS, GP	55 (a)	AO Globe (I,0,0) (4 lines)
3" MSL Drain	Water/Steam	PC, SC, LCS, GP	55 (a)	MO Gate (I,0,0)
20" Feedwater	Water	PC, 3IV, LCS, GP	55 (a)	AO Check (I,0) (2 lines)
4" RWCU To Main Cond.	Water	PC, LCS, 2LCS	56 (b)	MO Gate (I,0)
3" Drywell Sumps Disch.*	Water	PC, LCS	56 (c)	MO Gate (I,0)
2" RWCU Backwash Drain*	Water	PC, 3IV, LCS	56 (c)	AO Globe (I,0,0)
Post Accident Liquid Sample	Water	PC, RO	55, 56	MO Gate (I,0)
From Drywell				
6" Chill. Water from Drywell	Water	PC, V, LCS	57 (c)	MO Gate (I,0)
Post Accident Gas Sample	Air	PC, RO	55, 56	MO Gate (I,0)

*Liquid drain path - see Section 2.1

NOTES TO TABLE 2-1

(1) Leakage Barrier Types

PC = Primary Containment Isolation Valves Provided
SC = Secondary Containment Isolation Valves Provided
WL = Water Leg Seal
V = Vented to Secondary Containment
LCS = Leakage Control System Provided
RO = Flow Restricting Orifice Restricts Bypass Flow
3IV = Third Isolation Valve Provided (Remote to Manual)
2LCS = Secondary Containment Leakage Control System Provided
GP = Guard Pipe Between Drywell and Secondary Containment (Mark III only)

(2) Isolation Signals (Remote Manual Plus)

(a) Low RPV Water Level (L1)
High Radiation
High Steam Flow
High Steam Tunnel Temp.
High Turbine Building Temperature
High Turbine Building Steamline Temp.
Low Condenser Vacuum
Low Main Steam Line Pressure (Run Only)
Interlocks with Pump or Valves
High Differential Flow

(b) Low RPV Water Level (L2)
High Drywell Pressure
High Steam Tunnel Temp.

(c) Low RPV Water Level (L2)
High Drywell Pressure

(3) Valve Actuator Types

MO = Motor Operated
AO = Air Operated

I = Inboard Primary Containment
O = Outboard Primary Containment

TABLE 2-2
PATHWAYS WHICH TERMINATE INSIDE SECONDARY CONTAINMENT

LINE	FLUID	LEAKAGE BARRIER TYPE (1)	ISOLATION DESIGN CRITERIA (2)	VALVE TYPES (3)
From RPV				
14" RHR LPCI Mode	Water	PC, 2LCS, CL	55 (a)	AO Stop Check (I) MO Gate (a) (3 or 4 lines)
20" RHR SD Cooling Line	Water	PC, GP	55 (b)	MO Gate (I,O)
10" RCIC Steam Line	Steam	PC, LCS, GP	55 (c)	MO Gate (I,O)
6" RCIC Pump Discharge	Water	PC, 3IV, CL, GP	55	AO Stop Check (I,O)
12" LPCS Pump Discharge	Water	PC, CL	55 (a)	MO Gate (I,O)
12" HPCS/HPCI Pump Discharge	Water	PC, CL	55	AO Stop Check (I) MO Gate (I,O)
6" RWCU Pump Suction	Water	PC, GP	55 (d)	MO Gate (I,O)
6" RWCU Return to FW	Water	PC, GP	56 (d)	MO Gate (I,O)
From Drywell				
2" Drywell Bleedoff Vent	Air	PC	56 (e)	MO Gate (I,O)(2 lines)
SUPPRESSION POOL SUCTION LINES				
24" RHR Pump Suction	Water	CL, 2LCS	56	MO Gate
8" RCIC Pump Suction	Water	CL	56 (c)	MO Gate
12" LPCS Pump Suction	Water	--	56	MO Gate
24" HPCS/HPCI Pump Suction	Water	CL, 2LCS	56	MO Gate
12" SPCU Pump Suction	Water	PC, CL	56 (e)	MO Gate
8" SPCU Return	Water	2 LCS	56 (e)	MO Gate

GENERAL ELECTRIC
PROPRIETARY INFORMATION

NOTES TO TABLE 2-2

(1) Leakage Barrier Types

PC = Primary Containment Isolation Valves
CL = Closed Loop Inside Secondary Containment
SC = Secondary Containment Isolation Valves
3IV = Third Isolation Valve Provided
2LCS = Secondary Containment Leakage Control System

(2) Isolation Signals (Remove Manual Plus)

(a)	Injection Valve Pressure (>450 psi)	
(b)	Low RPV Water Level (L1)	High RPV Pressure (>150 psi)
(c)	High RCIC Room Temperature	High Turbine Exhaust Press
	Low RCIC Steam Pressure	
(d)	Low RPV Water Level (L2)	High RWCU Room Temp.
	High Drywell Pressure	High RWCU Differential Flow
	High Steam Tunnel Temp.*	Interlocks*
(e)	Low RPV Water Level (L2)	High Drywell Pressure

(*Except RWCU Pump to Demin.)

(3) Valve Types

MO = Motor Operated
AO = Air Operated

I = Inboard Primary Containment
O = Outboard Primary Containment

GENERAL ELECTRIC
PROPRIETARY INFORMATION

TABLE 2-3

PATHWAYS WHICH TERMINATE INSIDE SECONDARY CONTAINMENT
(MARK I, II) AND INSIDE PRIMARY CONTAINMENT (Mark III)

LINE	FLUID	LEAKAGE BARRIER TYPE (1)	ISOLATION DESIGN CRITERIA (2)	VALVE TYPES (3)
From RPV				
1" CRD Insert/Withdraw (Incl. SDV)	Water	PC	56	AO Ball (177 lines)
1-1/2" SLC Supply	Water	PC	55	Check (I), MO Stopcheck (O) Expl. (O)
TIP Guide Tube	Air	PC, AP	56 (a)	SO Ball; XO Shear (6 lines)
3/4" Instrument Lines	Water, steam	XF, RO	55, 56	XF Check (Mark I, II only) (62 Lines)
Instrument Lines (SRV Tailpipe Pressure)	Air/Steam		56	(19 Lines)
3/4" Instrument Lines (DW Press & dp)	Air	PC	56	Manual Globe (4 Lines)
From Drywell				
18" Vacuum Relief or DW Purge	Air	PC	56 (b)	AO B'fly (vacuum relief) MO B'fly (H_2 Mixing)
1/2" Instrument Lines	Water	XF, RO	55, 56	XF Check (Mark I, II only) (4 Lines)
Drywell Air Lock	Air	T	N/A	N/A
Drywell Equipment Hatch	Air	T	N/A	N/A
Through Wall Leakage	Air	N/A	N/A	N/A

GENERAL ELECTRIC
PROPRIETARY INFORMATION

NOTES TO TABLE 2-3

(1) Leakage Barrier Type

CL = Closed System. These lines terminate inside primary containment (Mark III only).

PC = Primary Containment Isolation Valve (Mark I and II only).

AP = Air purge of lines limits the amount of drywell to containment bypass.

XF = Excess Flow Check Valve (Mark I and II only).

GP = Guard Pipe between Drywell and Secondary Containment (Mark III only).

RO = 1/4" reducing orifice limits amount of bypass leakage.

T = Periodic Leak Test.

(2) Isolation Signals (Mark I and II only; remote Manual Plus)

(a) RPV Water Level 2 — Drywell Pressure High
(b) RPV Water Level 2 — Drywell Pressure High
Exhaust Radiation High

(3) Valve Types

AO = Air Operated
MO = Motor Operated
XO = Explosive Operated
XF = Excess Flow

I = Inboard Primary Containment
O = Outboard Primary Containment

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

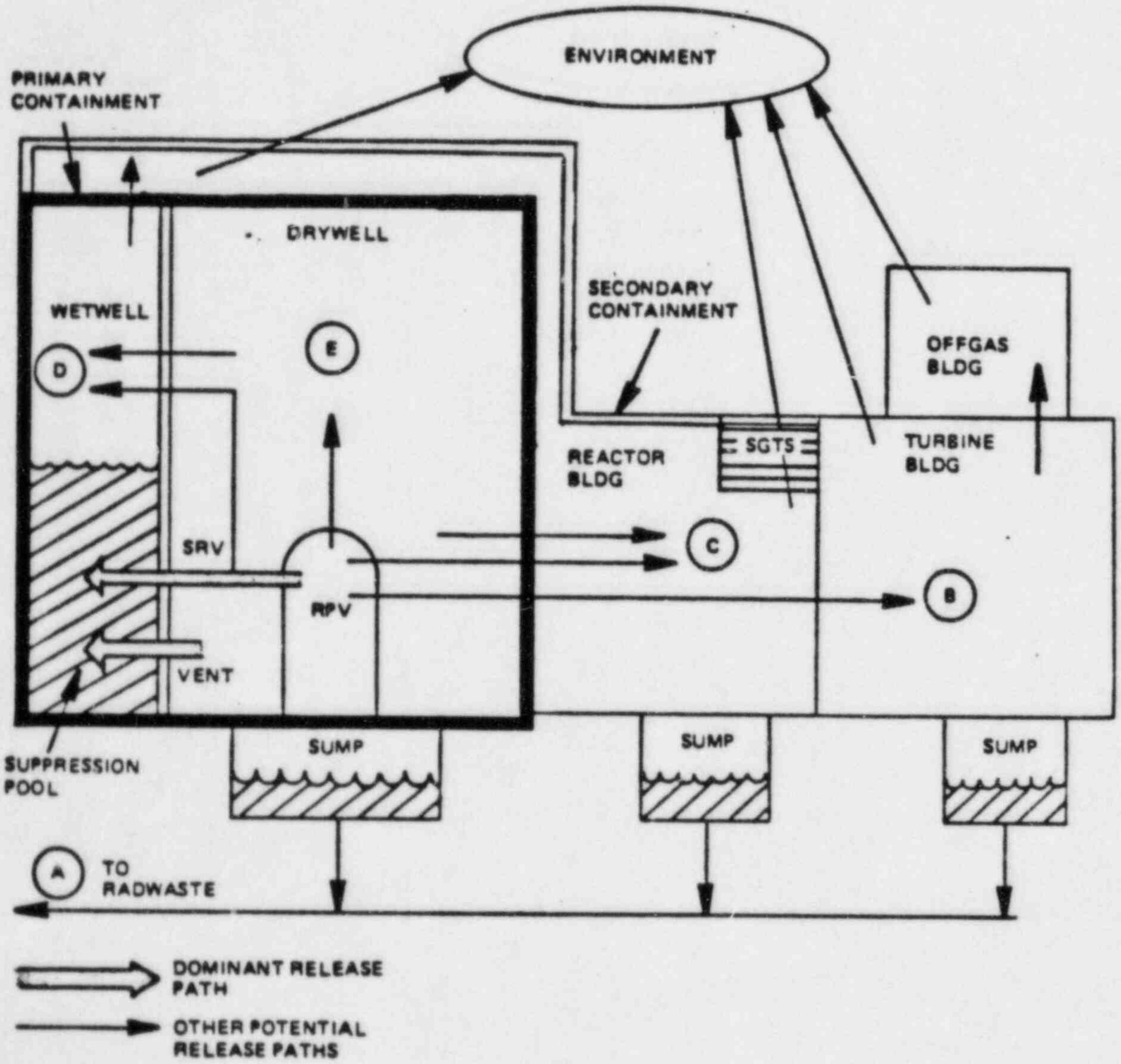


Figure 2-1. BWR Potential Bypass Pathways

3. SIGNIFICANCE OF SUPPRESSION POOL BYPASS

One way to assess the significance of the potential bypass paths identified in Section 2 is to evaluate their contribution to the annualized general public risk. The BWR/6 PRA evaluated the general public risk and, as shown in Table 1-1, found that it is dominated by transient events where no bypass paths occur. For these events, the offsite public exposure is largely dominated by the noble gas dose because the suppression pool effectively removes the particulate and the halogen fission products. The annualized general public risk is dominated by these events because of the relatively high event frequency for transients.

An assessment of the bypass paths identified in Section 2 has been made based on determining their contribution to annualized general public risk. Expressed mathematically, the annualized general public risk can be described as:

$$R = \sum_{\text{All events}} E \bullet F \qquad (1)$$

Where:

R = Annualized general public risk (manrem/year)
E = Exposure per event (manrem/event)
F = Frequency of event (events/year)

The exposure to the general public is composed of two parts: exposure to the noble gas cloud (E_N) and exposure to particulates and halogens (E_I). In simplified terms, the risk can thus be expressed as:

$$R = \sum_{\text{All events}} (E_N F + E_I F) \qquad (2)$$

GENERAL ELECTRIC
PROPRIETARY INFORMATION

Where the N and I subscripts refer to noble gas and iodine/particulate releases, respectively. Since pool bypass paths only affect the second of these expressions, the remaining discussion will focus on the $E_I F$ term.

The significance of each of these release pathways is evaluated by consideration of both the frequency of the event (F) taking any specific pathway and exposure terms (E_I). As shown on Figure 3-1, the pathways which bypass the suppression pool and those which pass through the suppression pool combine give a certain amount of activity available for release from the plant. As shown, there is a "resistance" to release of fission products through any pathway which is analogous to parallel electrical resistances. If the total "resistance" of the bypass pathways is much greater than the resistance of the suppression pool pathway, then the bypass paths are not significant contributors to overall risk.

For bypass pathways the "resistance" can be thought of in two parts: 1) the bypass probability (P_B) for a given pathway occurence concurrent with a core damage event (see Section 3.1) and 2) a factor (F_B) which represents the fraction of core damage release from the reactor pressure vessel which takes the bypass pathway. This factor (F_B) reflects the fact that small pathways are not capable of passing the full vessel release flow. Since the remaining flow is transferred to the suppression pool, this term is also referenced to on the flow split (see Section 3.2). When the bypass probability and flow split are combined they represent a release fraction which may be compared with the decontamination factor of the suppression pool to determine the significance of the pathway.

As discussed in Section 4, there are other retention mechanisms which are in effect for both the pathways which pass through the suppression pool and those which do not. These retention mechanisms function reduce the significance of core damage events on the general public. However, since they are separate phenomena and independent of the bypass or suppression pool pathway, they do not contribute to the significance of bypass pathways relative to the pool pathways.

3.1 PROBABILITY OF SUPPRESSION POOL BYPASS (P_B)

The probability of bypass for each line identified in Section 2 has been estimated. The probability values were obtained from the failure data used in the BWR/6 PRA (GESSAR 15D.3) and are summarized on Table 3-1. In general, these probabilities considered failures of the primary containment isolation valves and a line break probabilities as independent failures. The overall bypass probability also takes into account the types and number of valves, line size, and number of lines among similar or redundant groups of lines.

For certain lines, such as low pressure ECCS injection lines, which have a high pressure to low pressure interface, the failure of the control system logic (low pressure interlocks) was used in lieu of piping system failure. Such a failure could result in the possibility of over pressurization (and failure) of low pressure piping while the plant is still at high pressure.

The bypass probability represents the conditional probability, given a core damage event, that a certain bypass pathway may exist. The values obtained in this evaluation are summarized in Table 3-2.

3.2 BYPASS FLOW SPLITS (F_B)

To determine the flow split values, potential bypass flow rates were estimated from two sources and the more limiting was used on the evaluation: 1) formulae for flow of compressible fluids in pipes and orifices and 2) plugging of aerosols in small holes or cracks.

The results of the bypass flow split and plugging fraction evaluations are included on Table 3-2. It should be noted that the flow split evaluations are conservatively based on the full pipe size diameter. Restrictions due to valves and pipe crack exit effects would be expected to further restrict the potential effluent flow through bypass pathways.

3.3 EVALUATION RESULTS

The results of the bypass probability (P_B), flow split (F_B) evaluations and plugging fractions are shown on Table 3-2. In order to evaluate the importance of these bypass paths the product of the flow split (F_B) and bypass probabilities (P_B) should be compared against the fission product retention which occurs in the pathways which pass through the suppression pool as shown in Figure 3-1.

Figure 3-2 is a simplified event tree showing the overall resistance ($1/P_B F_B$) as compared with the decontamination factor expected in the pool due to pool scrubbing. The bypass lines listed on Table 3-2 havé been further grouped to show the relative significance of different release path types. The drywell pathway pool decontamination factor is based on the vent discharge to a saturated pool while the others are based on quencher discharge. It can be seen from Figure 3-2 that in all cases there is substantially greater resistance to release through potential bypass pathways than through the suppression pool. Consequently, source terms used to evaluate the consequences of severe accidents need only be concerned with the dominant release paths which are through the pool.

Reviewing the flow split and plugging fraction data on Table 3-2 shows that for the containment release pathways from the RPV, all lines contain significant restriction. By considering the conservatism in the methodology used, the conclusion can be reached that these pathways are not likely to realistically be a concern.

This evaluation also shows that the release pathways from the drywell are dominated by the TIP guide tubes and the guard pipe failure. The drywell vacuum breaker pathway, although it is potentially a large bypass pathway, does not significantly contribute to overall general public risk due to the low bypass probability.

GENERAL ELECTRIC
PROPRIETARY INFORMATION

Table 3-2

BYPASS LINE PROBABILITIES AND FLOW SPLITS

LINE	# OF LINES	ISOLATION BARRIERS	BYPASS PROBABILITIES	FLOW SPLIT	PLUGGING FRACTION	NOTES
LINES TO OUTSIDE SECONDARY CONTAINMENT						
From RPV						
26" Main Steam	4	4	4×10^{-10}	1.0		A
20" Feedwater	2	4	$<10^{-12}$	1.0		
3" Main Steam Drain	1	3	1×10^{-12}	2×10^{-1}		A
4" RWCU To Main Condenser	1	6	$<10^{-12}$	5.8×10^{-1}		A
Post Accident Liquid Sample	2	4	1.6×10^{-4}	5×10^{-6}	1×10^{-3}	B
From Drywell						
6" Drywell Cooling Water	1	3	$<10^{-12}$	4.4×10^{-2}		C,K
Post Accident Gas Sample	2	4	1.6×10^{-4}	4.5×10^{-3}	2×10^{-4}	B,K
LINES TO SECONDARY CONTAINMENT						
From RPV						
20" RHR Shutdown Cooling	2	4	$<10^{-12}$	1.0		D,E
10" RCIC Steam Lines	1	3	1×10^{-12}	1.0		
6" RCIC Pump Discharge	1	3	$<10^{-12}$	1.0		
12" HPCS Pump Discharge	1	3	1×10^{-12}	1.0		
14" LPCI/LPCS Discharge	4	4	4×10^{-9}	1.0		D
6" RWCU Lines	1	5	$<10^{-12}$	1.0		F
From Drywell						
2" Drywell Bleedoff	2	2	2×10^{-7}	4.4×10^{-2}	1.8×10^{-2}	K

Table 3-2 (Continued)

LINE	# OF LINES	ISOLATION BARRIERS	BYPASS PROBA-BILITIES	FLOW SPLIT	PLUGGING FRACTION	NOTES
LINES TO CONTAINMENT						
From RPV						
1" Instrument Lines	80	1	1.8×10^{-2}	2.2×10^{-3}	1×10^{-4}	I,J
1-1/2" SLC Line	1	3	2.3×10^{-9}	3×10^{-2}	2.6×10^{-2}	J
1" CRD Lines	177	3	4.1×10^{-2}	2×10^{-5}	1×10^{-3}	G,J
3/8" Sample Line	1	3	2.3×10^{-10}	5.0×10^{-3}	4×10^{-4}	J
From Drywell						
10" Vacuum Relief + H_2 Mixing	4	2	4×10^{-8}	1.0		J,K
TIP Guide Tubes	5	0	1.0	1×10^{-4}	1×10^{-6}	H,J,K
Airlock/Equipment Hatch	2	1	2×10^{-4}	2×10^{-3}	2×10^{-2}	J,K,L,M
Guard Pipe Failure	1	1	1×10^{-3}	4.4×10^{-2}	118×10^{-2}	J,K,N
Unidentified Drywell Leakage	-	0	1.0	2×10^{-3}	3×10^{-5}	J,K,L

GENERAL ELECTRIC
PROPRIETARY INFORMATION

NOTES TO TABLE 3-2

A. Release path through main condenser failure; condenser vacuum loss assumed. Drains presume discharge above condenser water level.

B. Flow split assumes flow restricting orifice; liquid sample line assumes scrubbing in line (10^{-3}).

C. Flow split assumes small (<2") opening in heat exchanger tubes.

D. Assumes check valve and low pressure interlock failure at high pressure causes break of low pressure RHR or LPCS piping.

E. Operator error to inadvertently open valve while at high pressure is assumed.

F. Probability is based on RWCU pump suction line. Other lines are less likely to be bypass paths.

G. Flow split assumes in-vessel scrubbing (10^{-3}) prior to vessel failure likely due to bottom entry; Not a likely pathway after RPV failure.

H. No isolation provided in Mark III design. Flow restriction severe due to probe left in guide tube. Mark I and II designs contain dual barrier protection.

I. Mark I and II designs have excess flow check valves; bypass probability is about 10^{+3} lower.

J. These pathways discharge to primary containment in Mark III designs; containment failure also required to provide bypass path.

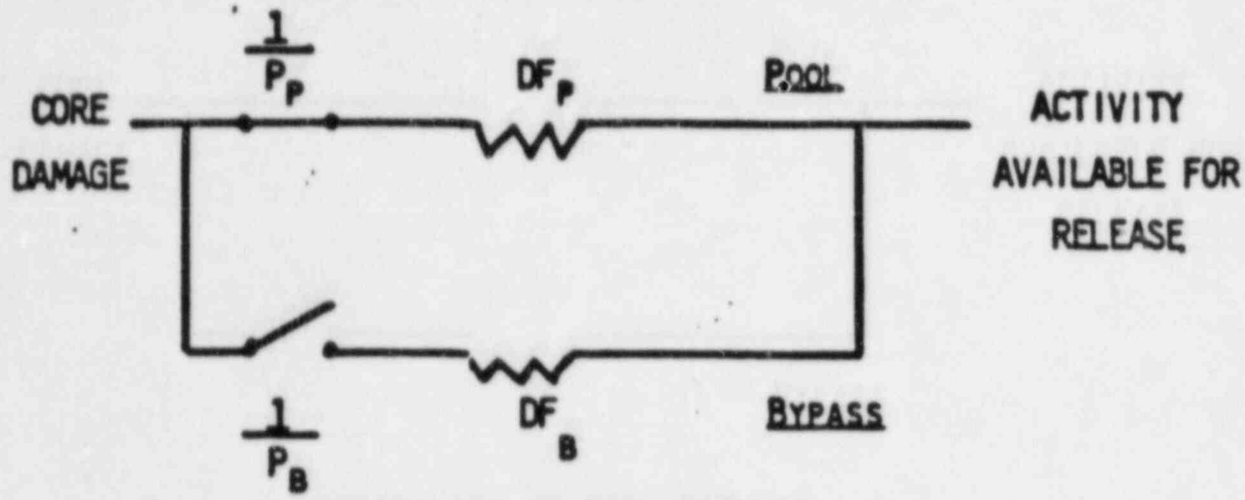
K. Pathway from drywell air is most likely after vessel failure. Bypass probability assumes probability of release in drywell at 1.0.

L. Pathway consists of numerous small paths. Equivalent path less than 125 in. assumed (conservative).

M. Bypass probability presumes a core damage event which generates sufficient drywell pressure to cause excessive leakage.

N. Bypass probability assumes failure of the guard pipe due to hydrogen burning inside the Mark III containment.

GENERAL ELECTRIC
PROPRIETARY INFORMATION



P_P = PROBABILITY OF POOL PATHWAY

P_B = PROBABILITY OF BYPASS

DF_P = SUPPRESSION POOL SCRUBBING DF

DF_B = 1/FRACTION OF FLOW BYPASSING POOL (F_B)

OVERALL "RESISTANCE":

$$\frac{1}{R} = \frac{1}{R_P} + \frac{1}{R_B} = \frac{P_P}{DF_P} + P_B F_B$$

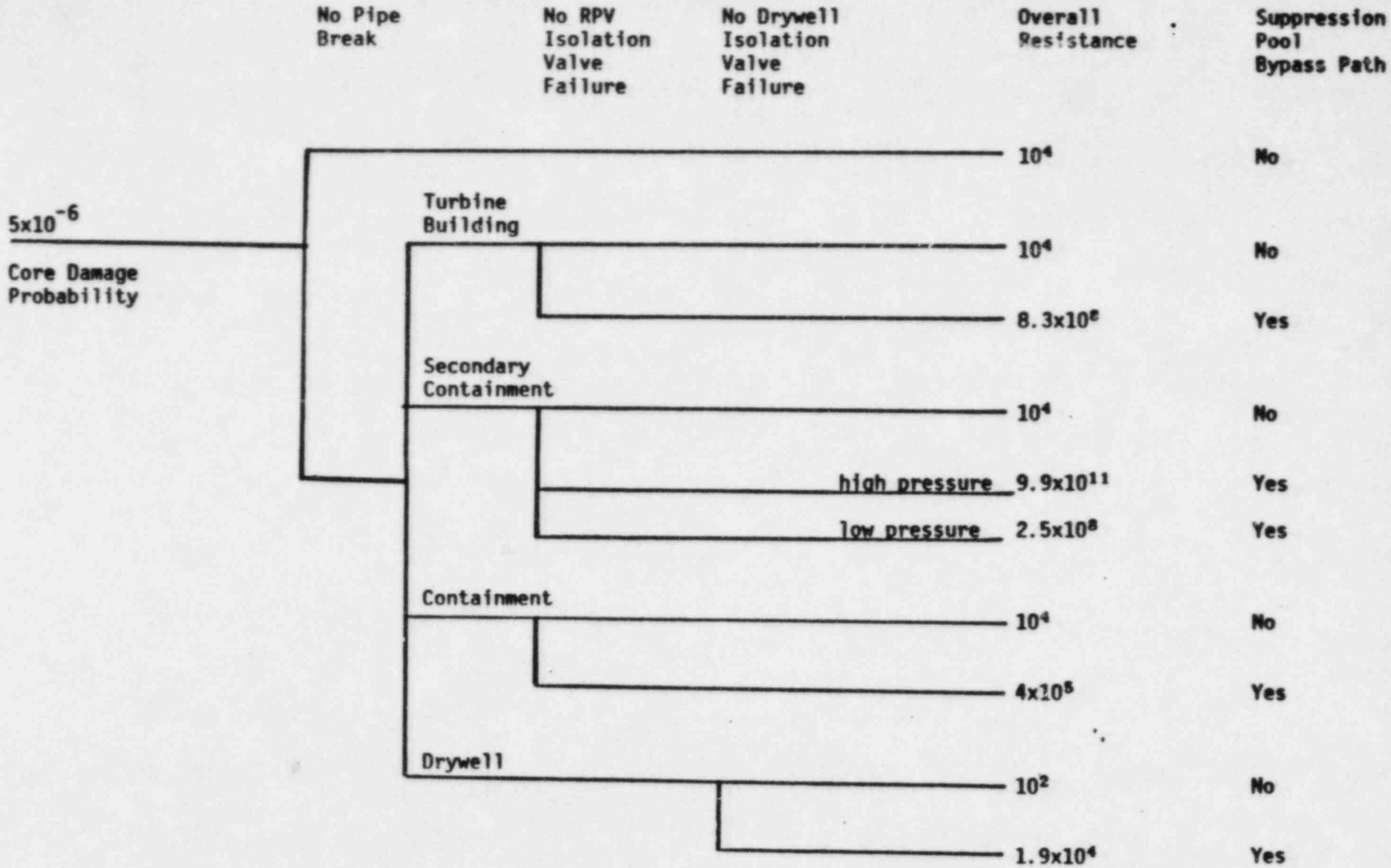
BYPASS IS NEGLIGIBLE IF:

$$P_B F_B << \frac{P_P}{DF_P}$$

Figure 3-1

FIGURE 3-2

RELEASE PATHWAY EVENT TREE



GENERAL ELECTRIC
PROPRIETARY INFORMATION

4. NATURAL FISSION PRODUCT RETENTION ON BYPASS PATHWAYS

There are other natural removal mechanisms which prevent release of particulates and halogens inside the reactor pressure vessel and along the release pathway. These mechanisms are in effect in all severe accident related paths and further reduce the source term of radioactive material to which the general public may be potentially exposed.

Several principle mechanisms have been identified which are in effect. First, plateout and/or deposition of the material inside the reactor vessel, its components such as separators, dryers, and channels, and also plateout along the vessel piping pathways provide a mechanism for retention prior to release to the environment. Secondly, once released to the building outside the primary containment (or within the primary containment for breaks) a significant retention in the highly humid environment and relatively cold surfaces would also be expected due to condensation or deposition. These naturally occurring mechanisms are currently being studied by various National Laboratories. The following sections summarize the current estimates of the potential retention factors which may ultimately be demonstrated by these programs.

4.1 PRIMARY SYSTEM PLATEOUT

A computer code (TRAP-MELT) developed by Battelle-Columbus for the NRC has been used to estimate the amount of fission produce retention in the primary system. Overall invessel retention ranging from DF of 1 to 10 are expected for paths which include vessel separators and dryers and steam lines. TRAP-MELT verification testing is being conducted by the Oakridge National Laboratory. Early results support the models included in the TRAP-MELT Code.

An internationally sponsored testing program is being conducted at the Marvikin facility in Sweden which is attempting to provide test data to support the level of in-vessel and ex-vessel piping plateout assumed. This work is a multi-year program which was initiated in 1983.

An NRC sponsored program with INEL is conducting "in-pile" tests with release of hot aerosols through piping to show the degree of removal which occurs. Significant amounts of piping deposition are being observed.

Finally, the retention of fission products is highly restricted pathways such as cracks, leaking hatches, or leaking valves is expected due to agglomeration of the solid fission products. Several NRC sponsored research projects are developing models and testing aerosol behavior to verify these models.

The result of the above studies are providing technical justification for assuming a high level of primary system plateout. Mechanisms of deposition in piping bends, gravity settling in low flow regions and condensation on cold pipes are expected to be in effect and provide a removal fraction of 10^{-1} to 10^{-2} in addition to the suppression pool scrubbing values on the majority of paths.

4.2 BUILDING PLATEOUT (RAIN FOREST)

Stone and Webster is including a treatment of rain forests in BWRs in its source term paper for the ANS. They have published a paper* based on PWR studies which shows decontamination factors greater than 30 for a break outside of containment.

*Assessment of the radiological consequence of particulate reactor accidents, CSA Warman, November 1982, Presented at Second Internals and Conference in Nuclear Technology Transfer, Buenos Aires, Argentina.

A similar assessment for BWR line breaks outside containment would be expected to show substantial fission product retention inside building/rooms outside containment due to deposition on wet building surfaces and gravity settling of larger particles. Another removal fraction of 10^{-1} to 10^{-2} for buildings is expected.

4.3 CONCLUSION

The ongoing studies identified above are expected to show that substantial removal mechanisms exist in BWRs independent of the suppression pool. The judgements used in PRAs, the Stone and Webster paper, and the ongoing NRC research programs, lend confidence that a removal fraction on the order of 10^{-2} can be justified for large bypass pathways due to inherent retention mechanisms other than suppression pool scrubbing.

5. SUMMARY/CONCLUSIONS

The potential pathways for fission products to be released to the environment which bypass the suppression pool have been identified. These pathways are of concern because they do not benefit from the fission product retention capability of the suppression pool.

In examining each potential line it was found that the general public risk is not greatly affected by bypass lines either due to the relatively low amount of flow which would pass through the small pathways or due to the high reliability of the containment isolation design for the larger lines.

As a consequence source terms based solely on suppression pool pathways with credit for suppression pool scrubbing are justified.

Several studies are in progress to show that the risk is also not significant because of natural removal mechanisms which are in effect independent of the containment isolation system design.

Based on these conclusions and the likelihood that the continuing studies on natural removal mechanisms will confirm the presence of significant retention in bypass pathways, the suppression pool bypass pathways are not considered a source of concern for BWR's.

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H.4 TYPICAL BWR CONFIGURATIONS

Boiling water reactors (Figure H.4-1) have a core support configuration in which the control rod drive (CRD) guide tubes support the core from below. There is essentially one CRD tube for each group of four fuel assemblies such that the support is not only from below but it is also localized.

Given this core support configuration, ~~which is illustrated in Figure H.4-2~~, it is virtually impossible to conceive of a sequence whereby a degraded core would catastrophically collapse into water. In addition, with the extensive CRD guide tube structure, it is equally difficult to envision any process whereby rapid and intimate mixing could occur. The specific details of this reasoning process are given below.

1. Under normal operating conditions, the guide tube structure is designed to support the entire core. The major change in the material properties occurs when substantial overheating takes place, but this can only occur in the absence of water. If water is absent steam explosions are not possible.

2. In addition, each group of assemblies is, in effect, individually supported and if a degraded core condition is assumed, the most likely way in which molten core material would migrate to the lower plenum is through the assembly orifice located within the support tube. This would undoubtedly be an incoherent process and the molten core material would flow into the interstitial spaces between the CRD guide tubes and perhaps contact the steel wall and freeze. However, thermal attack of the tube itself would not begin until the water had been boiled away inside of the tube. Consequently, not only would

H.4 TYPICAL BWR CONFIGURATION (Continued)

the melt progression be incoherent, but the core material could not participate in a global interaction until the water was vaporized. This eliminates the potential for any steam explosion.

3. If all the above physical restraints are completely disregarded and one assumes that coherent core collapse occurs in any event, then one must consider the forest of support tubes, control rod thimbles, and instrument tubes which exist below the core. This massive, cold structure, which could freeze the core debris on contact, would prevent any large scale, intimate mixing of the molten debris and coolant.

These three points, all dealing with the below-core structure, show that catastrophic collapse in the presence of water cannot occur, the downward progression of any postulated scenario would be incoherent and occur within the support tubes (and only in the absence of water), and large scale, intimate mixing could not be achieved. Therefore, large scale steam explosions involving substantial masses of core material can be ruled out on geometric considerations alone. In addition, these can be considered remote in light of the massive, coherent interaction *required* in WASH-1400 before vessel failure was calculated. The below-core structure was ignored for the WASH-1400 BWR analyses.

One can be equally critical of the slug formation, displacement, and impact model from WASH-1400 as it relates to the actual design.

1. With the below-core structure segmenting the water with the core support tubes, the formation of a continuous, overlying liquid slug can also be discarded.

H.4 TYPICAL BWR CONFIGURATIONS (Continued)

2. If such a slug is postulated the core grid at the top of the fuel assemblies and the upper plenum dome would destroy the coherence as the material travels upward through the vessel.

3. Steam separators, located above the core as ~~shown in Figure H.?-1,~~ are large structural components which do not provide straight-through flow paths. This would also prevent the upward transmission of a coherent liquid slug.

4. Steam dryers are positioned above the steam separators. These components, like the steam separators, also have a tortuous flow path, and thus, provide another barrier to the postulated coherent behavior.

5. In addition to destroying the coherency of a liquid slug, the mentioned structures will also attenuate the energy of dispersed material.

These arguments have been formulated on the basis of specific components available in the reactor vessel but ignored in the Reactor Safety Study. As discussed, these differences are indeed extensive and the discussion of each shows that their neglect in WASH-1400 grossly overestimated 1) the likelihood of an event, 2) the amounts of material involved, and 3) the damage potential represented by an event. Considerations of the structural components allows one to individually rule out 1) catastrophic collapse, 2) rapid and intimate mixing, 3) coherent slug formation, 4) coherent slug transmission, and 5) coherent slug impact. As summarized in Table H.2-1, all of these are required for the WASH-1400 analysis to predict steam explosions. However, there

H.4 TYPICAL BWR CONFIGURATIONS (Continued)

is even a more fundamental misrepresentation in WASH-1400 and that is the characterization of steam explosion themselves. This is addressed in the next section.

H.5.2 Low System Pressures (Large Break Sequence) (Continued)

Since the major concern of this evaluation is the damage potential represented by in-vessel steam explosions, one must evaluate the amounts of material which can come into contact and mix on an intimate scale prior to the onset of an explosive interaction. To make this assessment, necessary criteria for achieving a significant interaction in the RPV must be defined and each evaluated with respect to governing physical principles.

If a steam explosion is conceived to be a physical process whereby the reactor pressure vessel integrity can be violated and as a result also violate the containment integrity, several specific criteria must be satisfied for the physical processes to achieve such a magnitude. These are listed below in essentially their chronological sequence and each is discussed individually. As will be shown in this discussion, each physical process represents a highly improbable if not impossible condition, and as such this lists provides a description of why steam explosions are of no practical importance in the containment assessment.

H.5.2.1 Prerequisites for Loss of RPV Integrity by In-Vessel Explosion

To develop a steam explosion of sufficient energy to violate a reactor pressure vessel requires a) sufficient molten corium poured into the lower plenum, b) a sufficient molten condition at the time of initiation of the explosion, c) insufficient pressurization to permit inter penetration of the melt and water, d) coarse intermixing to a sufficiently small scale prior to the explosion, e) a sufficient trigger to mix these materials on an explosive time scale (∿10 msec), f) either sufficiently high shock pressures to rupture the lower head or g) a slug formation and h) transmission upward through the RPV with a coherent impact on the vessel head. Each of these is discussed in more detail below.

H.5.2.1 Prerequisites for Loss of RPV Integrity by In-Vessel Explosion (Continued)

a) The principal consideration is not whether steam explosions can occur inside the RPV but whether they can be of such magnitude as to fail the RPV and thereafter violate the containment integrity. Therefore the primary question is whether sufficient molten material is available to provide the necessary work for violation of the reactor pressure vessel. Typically this would require a minimum of 1.2 tons of molten core material for a theoretically perfect thermal interaction and realistically considering the ability to vaporize the liquid in close proximity to the fuel, one should consider several times this amount of material, i.e. perhaps 12 tons. This material must be available to the water on the time interval sufficiently short such that pressurization resulting from steam generation during film boiling as the material enters the water does not provide sufficient forces to prevent penetration of the material. This will be addressed later, but it is closely interwoven with this issue of sufficient corium availability.

b) To initiate an explosive interaction, the corium must be in a molten condition when it contacts the water and it must maintain this molten state during the pre-explosion stage. As a result, it requires that this coarse fragmentation and mixing take place in a sufficiently short time that the surface does not solidify. In fact, they should not even approach freezing since corium mixtures become viscous as they approach the liquidus point. This limitation on surface temperature is even more restrictive since the material must be molten and have

H.5.2.1 Prerequisites for Loss of RPV Integrity by In-Vessel Explosion (Continued)

a sufficiently low viscosity so that the pre-explosion, coarse fragmentation process can proceed with the limiting hydrodynamic forces associated with liquid-liquid film boiling. The rate at which the surface cools is dependent upon the particle size, but in all cases is typically on the order of magnitude 1 sec. As a result, this is coupled with the sufficient corium requirement discussed in item (a) and these combined establish the rate at which the molten material must be added to the water in the lower plenum to provide a material state for a sufficient explosive interaction.

c) As the material enters the water in a film boiling state, the energy transfer in film boiling tends to pressurize the water which in turn attempts to separate the water and core debris. In assessing the needs for a sufficient explosive interaction, the conditions must be such that the pouring or dropping process is of sufficient character that pressurization of the water and corium does not occur at the interface and thus preclude the interpenetration of the overheated material into the water. This is related with issues (a) and (b), but is also separate since it represents the ability for continued interpenetration of the corium debris into the water regardless of the failure mechanism which provided the material pour.

d) After sufficient material has been generated and released to the water under conditions providing for its global penetration into the water on a short time scale, the material must have sufficient time to undergo fragmentation (in the film boiling state) to the level dictated

H.5.2.1 Prerequisites for Loss of RPV Integrity by In-Vessel Explosion (Continued)

by the hydrodynamic stability of the water and core debris. In this regard the film boiling fragmentation model provides for a descriptive formulation of the level of such pre-fragmentation that can occur. Another means of assessing the sufficient fragmentation size in the liquid-liquid film boiling state is to identify an available trigger (and its energy level) in the reactor system to initiate such an explosive interaction and equate this to the mixing energy, which will be discussed below. With this energy level, the size required for establishing a triggerable system can then be determined which sets the scale of the pre-explosion fragmentation. Comparisons can then be made between the scale of this necessary fragmentation and that achievable in liquid-liquid film boiling.

e) Given that such large quantities of molten materials can be available and added to the water over a sufficiently short time interval and premixed to sufficient level while still in a molten state, a sufficient trigger must then be available at the appropriate time to provide the necessary mixing energy to carry out the explosive interaction. This can be addressed in terms of the available pre-explosion fragmentation size, the amount of water that must be interacted, and the effective drag coefficient for rapidly intermixing materials on the size and time scales necessary for such large thermal energy transfers. This is a particularly crucial question since it can be principally based on the availability of a sufficiently large trigger as opposed to the statistical question of whether a trigger is delivered at the appropriate time.

H.5.2.1 Prerequisites for Loss of RPV Integrity by In-Vessel Explosion (Continued)

f) Given the completion of all the above necessary steps to achieve a sufficient explosive interaction for reactor pressure vessel failure, the first structural question is the integrity of the RPV lower head. Failure of this part of the reactor vessel is independent of slug acceleration and impact which will be discussed later and would only be the result of a very strong pressure wave. A typical operating pressure for a BWR system is 7 MPa (1015 psia) as compared to an upper bound pressure of a steam explosion during the expansion phase of approximately 10 MPa (1450 psia). As a result, the failure of the lower head would require explosive pressures on a sustained level essentially equal to the maximum values observed to date. If such a failure is feasible it would still have to be considered highly unlikely.

g) If an explosion is to be considered and the lower head of the reactor vessel remains intact, the other vessel failure mechanism considered as potentially leading to the loss of containment integrity is the acceleration of a continuous overlying liquid slug upward through the reactor pressure vessel and impingement of the slug on the RPV upper head. This requires both the formation of such a continuous slug and the transmission of this slug in a sufficient coherent fashion to impact and fail the upper head. The question which would be asked at this stage is whether such a continuous overlying slug could indeed be formed. Imbedded in such an evaluation is the effect of rapid steam formation during the pre-explosion fragmentation interval. The steam formed in this time

H.5.2.1 Prerequisites for Loss of RPV Integrity by In-Vessel Explosion (Continued)

interval must either be transmitted through this slug region, transmitted up the by-pass region or stored within the water. If the steam is transmitted up the downcomer, the water removal is far greater than has been suggested by the above questions, and the availability of water for sustained interaction of the fuel is considerably less than was discussed above. If this steam remains within the water, the pressurization as a result of corium entering into the coolant is far greater than was alluded to above. If the steam is transmitted upward through the overlying slug, the slug cannot remain continuous and an evaluation of the steaming rates and the transmission of this steam through the slug require that this material would have a considerable void fraction, i.e. one not identifiable with a continuous overlying slug. As a result, the formation of such a slug would be highly questionable.

h) If all the above restraints were assumed to be violated so that a sufficient explosion was conceived with a continuous overlying slug formed, this slug must then be transmitted upward through the remainder of the original core configuration, through the upper core support plate and through the upper internals before it could coherently impact upon the upper head of the vessel. This transmission must be such that sufficient energy is retained to cause failure of the head. This transmission must be sustained through the remnants of the core, the upper core plate, the steam separators and the steam dryers, all of which represent sizable energy absorption capacity. As a result, the transmission of

H.5.2.1 Prerequisites for Loss of RPV Integrity by In-Vessel Explosion (Continued)

the slug would be such that it could be broken up, dispersed, and deliver incoherent impact forces with a total energy less than that provided by the explosion itself.

Each of these above points must be satisfied before an explosive interaction can be sufficiently energetic to result in failure of the reactor pressure vessel and eventually the loss of containment integrity. In the following subsections, these individual behaviors are quantified to provide the basis for an engineering evaluation of the likelihood for such an event.

H.5.2.2 Quantitative Evaluation of the Prerequisites for the BWR/6 RPV Loss of Integrity by In-Vessel Steam Explosion

H.5.2.2.1 Sufficient Molten Material

The slug impact energy required to fail a BWR reactor vessel head has been estimated to be 500 MJ (References H.5-13, H.5-14). To accomplish the work by a steam formation process requires the vaporization of a water mass which is dependent upon the actual path involved.

The calculated mass of 1200 kg (2650 lbs) is a conservative estimate (by an order of magnitude) of the material mass required to initiate an explosive interaction which could threaten the RPV. Considering the theoretical density to be 7000 kg/m^3 (437 lbm/ft^3), H.5.2.2.1 Sufficient Molten Material (Continued)

this would be a spherical accumulation of a molten debris 0.7 m (27.5 in.) in diameter. This dimension is much greater than either the CRD tubes or their pitch, and as a result, the downward movement would occupy several CRD channels and their interstitial spaces. Coherent downward migration would be extremely unlikely in such a loosely coupled system.

H.5.2.2.2 Sufficiently Molten and Coarsely Fragmented

These are considered together since the two behaviors are intimately coupled. As the material fragments, the cooling rate increases and as the corium approaches the liquidus point the fragmentation process in a liquid-liquid film boiling state becomes more difficult. However, as will be shown, the corium decreasing temperature enables the coarse fragmentation to continue to a smaller scale.

H.5.2.2.2 Sufficiently Molten and Coarsely Fragmented (Continued)

If this analysis is applied to a reactor accident scenario in which 1200 kg (2650 lbm) of core material is assumed to fall into the lower plenum~~, the fragmentation limit for a BWR geometry with the CRD guide tubes is given in Table H.5-4. For illustration purposes calculations are presented in Table H.5-4 for the same geometry in the absence of CRD guide tubes.~~ As shown the particle sizes are very large for such a large amount of very hot material in a small area. Obviously such large particles would not exist, but the calculation demonstrates (by orders of magnitude) that in a reactor system water cannot remain in the presence of fine particulation. As a result, finely dispersed configurations in intimate contact with water are physically unattainable.

Another feature of the BWR system noted early in this report which also is relevant in assessing the potential for intermixing of molten corium and water is the extensive below-core structure. In the analyzed BWR plant the core is supported from below by 177 control rod guide tubes.

The CRD flow inside these tubes is separated from the inlet plenum water outside the forest of tubes, and except for minor leakage at the inlet to the fuel assembly, these two sources of water do not mix below the top of the core. As a result, the only cross-sectional area available for the intermixing process is that restricted area between the CRD tubes making fragmentation even more difficult ~~as illustrated in Table H.5-4~~. In the hypothetical case, the water would be displaced by the downward moving corium and any initiation of fragmentation would only drive the water away faster. It should be noted that only gravity retains the water and that it can readily be displaced into colder (outer) regions of the core, backwards through the jet pumps, etc.

H.5.2.2.3 Slug Dispersal and Pressurization

While somewhat out-of-phase, these two phenomena are considered together since they both involve the effect of steam generated in the film boiling, coarse dispersal and intermixing process. In one instance, sufficient time is available to allow the steam to escape upwards through the overlying pool and the necessary conditions to allow this escape are evaluated. For the second case the steam is assumed to be retained within the pool, thus pressurizing the system.

As the mixing and inner dispersion progresses, the hot and cold liquids, are in liquid-liquid film boiling. Since the molten core debris is at a temperature of 2500°K or greater, the principle mode of energy transfer would be via radiation from the hot particles to the water. This energy transfer can be expressed as

H.5.2.2.3 Slug Dispersal and Pressurization (Continued)

Therefore, in a slowly developing dispersion (time scale of 1 sec or longer) the vapor throughput would be substantial and preclude the formation of a continuous overlying liquid slug. If the vapor is assumed to be retained in the pool, the pressurization would disperse the pool, hence no slug formation. Without the continuous slug formation, the only pressure imposed on the vessel is that due to the explosion itself, which experiments have shown to be a few MPa typically, and could conceivably be as high as 10 MPa. However, such pressure levels do not even threaten the integrity of the vessel.

H.5.2.2.4 Rapid Liquid-Liquid Mixing (Continued)

To summarize, the amount of material required to rupture a BWR vessel would be a minimum of 1200 kg (2650 lbm) of molten debris. This high temperature material must then coarsely fragment in the water contained within the lower plenum. Because of both the high material temperature and liquidus point, the fragmentation must occur in film boiling and can only proceed as long as the water can remain in place, i.e. the hydrodynamic stability limit of the water cannot be exceeded. At typical BWR accident conditions, the spherical fragment sizes are essentially the same as the CRD tube pitch, and therefore, greater than the characteristic dimension of the interstitial space between the tubes where the water is located. Rapid liquid-liquid mixing from this coarsely fragmented state down to a size capable of rapid thermal response can require substantial mechanical work depending primarily on the initial material size and the length over which the mixing occurs. Though the proposed analytical models for the pre-explosive fragmentation and mixing energy are based on mechanistic considerations, their application in the case of a reactor requires verification against available large scale experiments. This is done in the next section.

H.5.3 References

H.5-1 R. E. Henry and H. K. Fauske, "Nucleation Processes in Large Scale Vapor Explosions," Trans. of ASME, Journal of Heat Transfer, Vol. 101, pp. 280-287, May 1979.

H.5-2 D. J. Buchanan, Journal of Physics D; Applied Physics, Vol. 7, pp. 1441-1457, 1974.

H.5-3 R. E. Henry and L. M. McUmber, "Vapor Explosion Experiments with an External Trigger," Second CSNI Experts Meeting on the Science of Vapor Explosions, Grenoble, France, September 1978.

H.8 STEAM EXPLOSION - EX-VESSEL

H.8.1 Explosion Scenario

If, in a defined accident sequence, water cannot be supplied to the reactor vessel to establish in-vessel removal of the decay power from the damaged core, then eventually the core will melt along with the fuel channels, the core support plate, and the core support tubes. This mass of molten material will accumulate in the lower head of the vessel and will thermally attack the vessel wall and vessel penetrations and result in the corium penetration of the vessel head. This would lead to the discharge of the molten material collected in the lower plenum of the PRV into the pedestal cavity below.

For accident sequences such as a large break LOCA, the pedestal cavity may be covered with up to 5 feet of water from the blow-down of the vessel. In this case, as the molten material is released from the reactor vessel it will encounter water and the potential for a steam explosion would exist. Such a steam explosion could be triggered when the molten material contacts the wetted floor of the pedestal cavity. In this section, the maximum work potential of such an explosion and its effect on the drywell boundary are estimated. The physical processes and the specific criteria associated with ex-vessel explosions are the same as in the case of the in-vessel explosion discussed earlier.

H.8.2 Molten Corium/Water Interaction

The structure below the vessel of a Boiling Water Reactor assembly (Figure H.8-1) consists of the bottom head insulation support beams, the control rod drive housing support beams (Figure H.8-2) in-core flux monitors, and the forest of control rod drives and

H.8.2 Molten Corium/Water Interaction (Continued)

the associated hydraulic drive lines (Figure H.8-3). Thus even if one were to assume the RPV bottom head failed and consequently spilled the molten material into the space below the downward progression of the molten material would not be in one coherent mass. Molten metal discharged from the vessel will first encounter thermal insulation which will not provide any significant resistance to continued penetration of the high temperature discharge. The structures mentioned earlier are below the insulation and would temporarily break up and disperse the debris causing some material to be discharged through the pedestal windows, and in addition it would distribute the debris in a fairly uniform manner.

The size of the globules reaching the water in the pedestal cavity after a freefall below the control rod drives, will exceed 9 inches in an extreme case of bottom head collapse.

It was shown in Appendix H.2, in connection with the interaction of molten corium and water inside the vessel, that the surfaces of such large particles would freeze rapidly as they attempt to mix coarsely with water due to intense radiation heat transfer. Furthermore the downward penetration would be limited by the pressurization of the pedestal water as the corium enters and the steam formation from the film boiling in the coarsely mixed state would disperse the incoming corium and the generated steam through the CRD door and the hydraulic line tunnels. Once again, as in the case of the in-vessel interaction, the energy required to rapidly mix the coarsely fragmented debris far exceeds that of a realistic trigger and also exceeds the mechanical work delivered by the explosion itself.

H.8.2 Molten Corium/Water Interaction (Continued)

On the other hand, molten corium could penetrate through the RPV bottom in a less dramatic manner. Boiling Water Reactors have a forest of penetrations in the lower head because the control rods are driven from the bottom and the incore instrumentation also enters the vessel from the bottom. For the BWR/6-238 plant there are about 177 control rods, each with its own penetration, 55 penetrations for in-core neutron flux monitors and a reactor vessel drain. The weld area around these penetrations would be subject to a three-dimensional thermal attack in the presence of a significant accumulation of degraded core material. Because of the large number of penetrations and the three-dimensional type of melting attack that these would experience, as opposed to the essentially one-dimensional melting at the vessel wall, one would expect these penetrations to be the first element of the primary system pressure boundary to fail and admit molten corium into the pedestal cavity.

In the event that a control rod drive support is melted through and the mechanism is ejected the resulting vessel breach would be approximately 7.5 cm (3.0 inches) in diameter. Thus for an assumed failure of one CRD penetration the total breach area would be around 44 cm^2 (7 square inches). Consequently, the amount of degraded core material in contact with water in the pedestal, at the time its front contacts the pedestal floor, would essentially be the breach cross-sectional area times the water depth (5 ft), i.e., about .0067 m^3 and 47 kg (0.24 ft^3 and 104 lbs). If this is at a temperature of 2200°C and the water is at 100°C, the thermal energy contained within the melt is 60 MJ. Using the experimental data reported in Reference H.5-24 the upper bound on the efficiency of such interactions, which were conducted with an iron-aluminum oxide thermite and melt quantities of this

H.8.2 Molten Corium/Water Interaction (Continued)

magnitude, was 1% of the thermal energy of the melt. This would yield 0.6 MJ of mechanical work which is a negligible level compared to that required for failure of the containment boundary and its major effect would be to displace the water from the pedestal cavity.

The immediate reaction of an ex-vessel steam explosion would be to disperse the water and degraded core material through the drywell. This would enhance the contact between the two media and result in rapid steam production. The remainder of the material released from the vessel at this point in time, while not participating in the explosion, could be rapidly quenching as a result of this dispersion process.

In summary, ex-vessel steam explosions could occur for those defined sequences where water is available in the pedestal cavity, but the amount of material would be very limited. In fact, the major effect would be a rapid quenching of that material which had been released from the vessel at the time of the event. Consequently, it is concluded that a loss of containment integrity will not occur as a result of ex-vessel steam explosions.

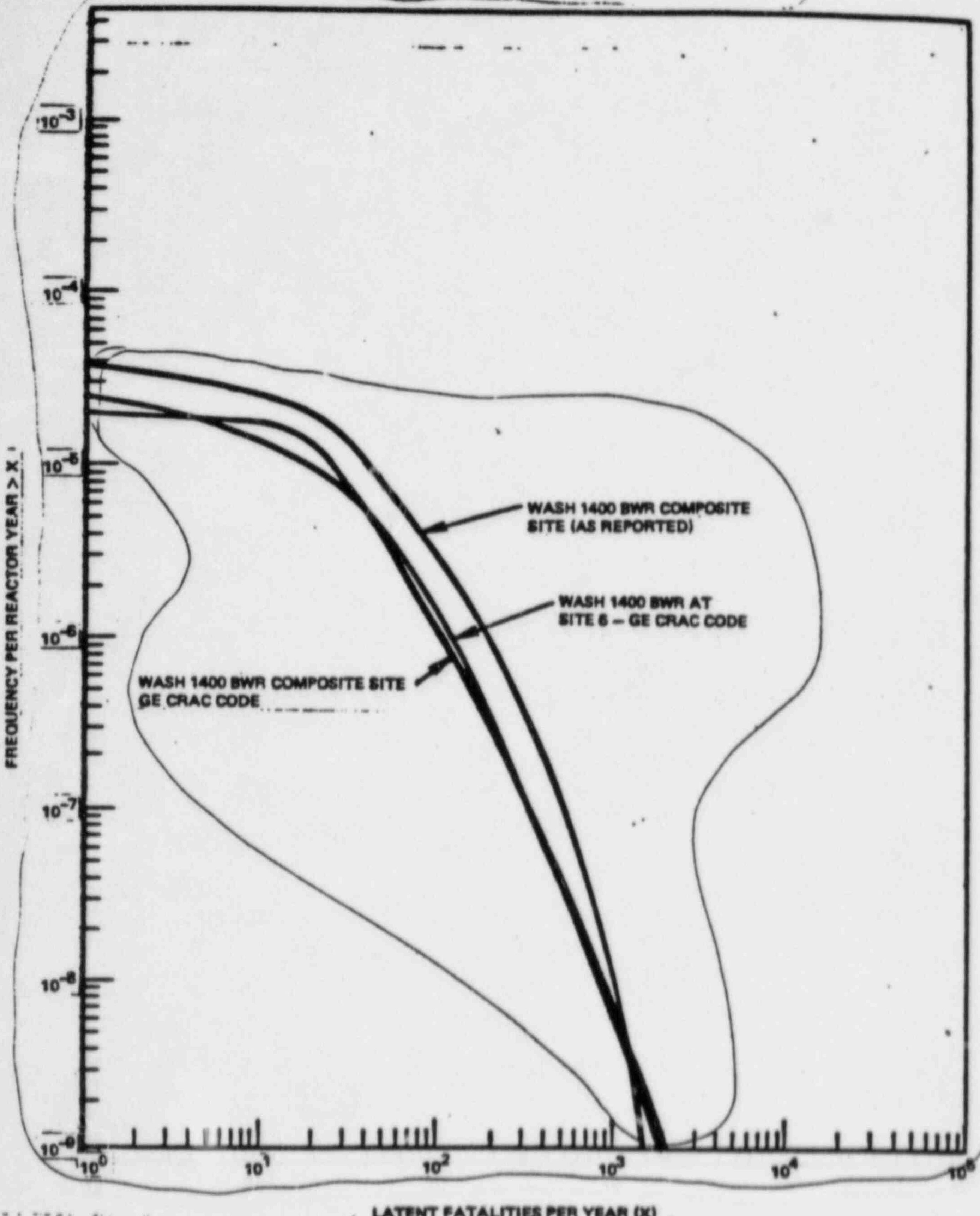

Figure F.4-2. Site Comparison of Latent Fatality Curves for WASH-1400 BWR

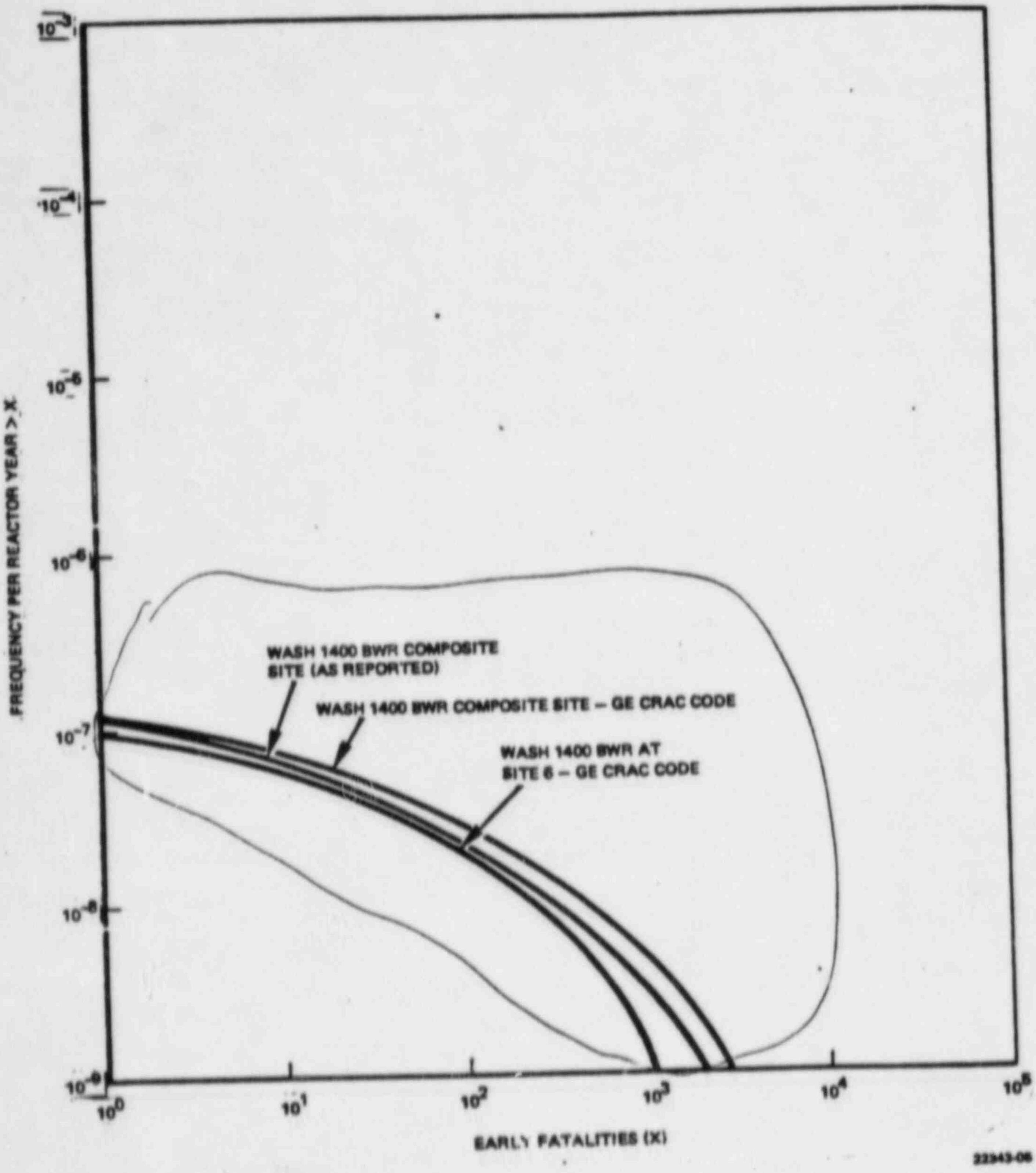

Figure F.4-1. Site Comparison of Early Fatality Curves for WASH-1400 BWR

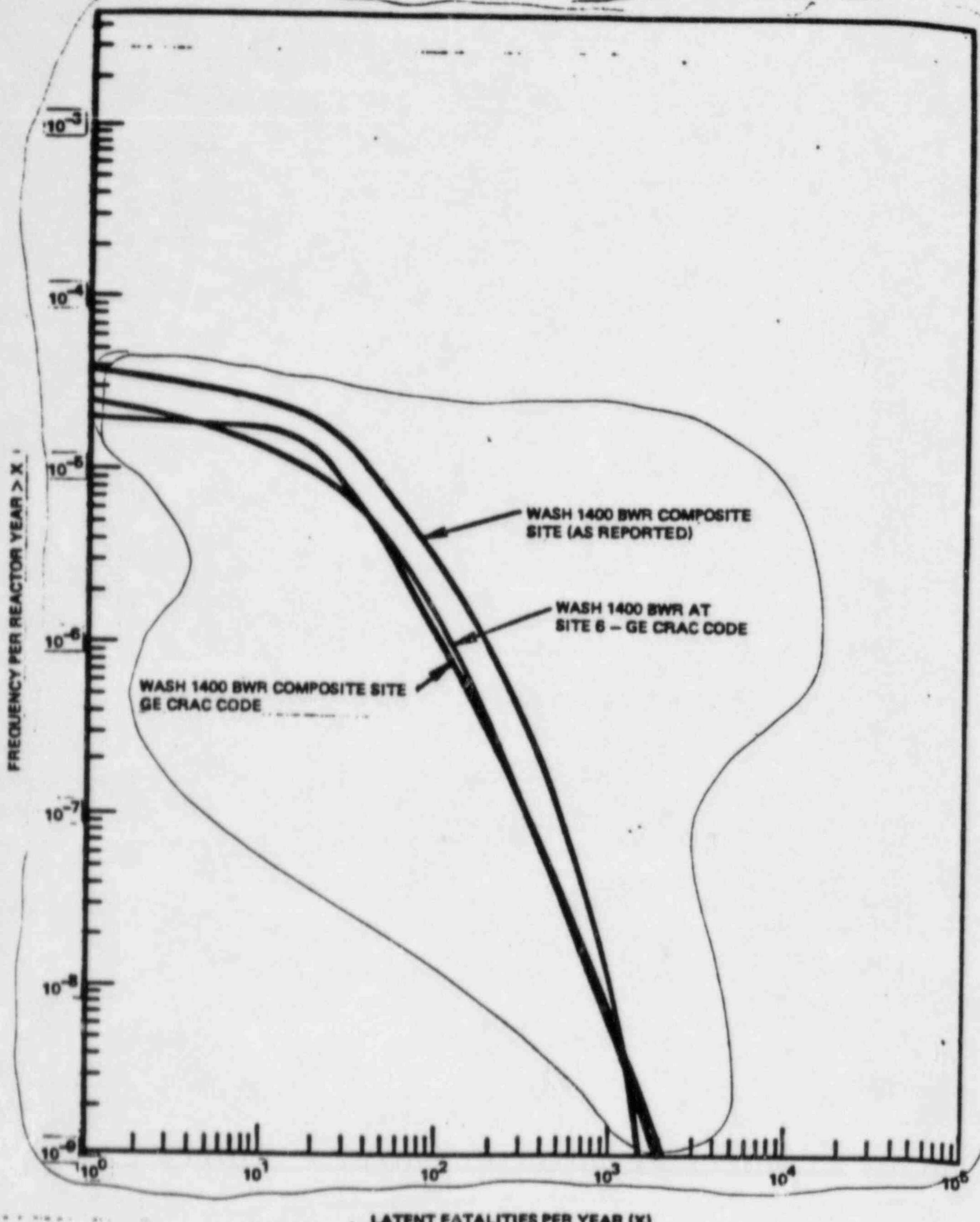

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Figure F.4-2. Site Comparison of Latent Fatality Curves for WASH-1400 BWR