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MEMORANDUM FOR: Raymond F. Fraley, Executive Director
Advisory Committee on Reactor Safeguards

FROM: William J. Dircks
Executive Director for Operations

SUBJECT: 295TH ACRS MEETING (NOVEMBER 1-3, 1984)
FOLLOW-UP ITEMS

REFERENCE: Memo from R. Fraley to W. Dircks, subject as above, dated December 4, 1984

The following information is provided in response to those specific items in the referenced memorandum that pertain to NRR.

1. ACRS Report on Limerick Generating Station, Unit 1

The November 6, 1984 ACRS report stated that the NRC and industry should continue to work to develop methods which can be used to quantify seismic risk and to identify any seismic outliers which might exist. The NRC staff currently has established an Ad Hoc committee, composed of members from NRR and RES, to carry out a seismic design margin program. One of the purposes of this program is to develop more precise means to determine seismic margins. This effort is expected to be completed early in FY 86. The staff is tentatively scheduled to give the appropriate ACRS subcommittees a status report on this program on March 26, 1985. Additional reports will be given thereafter, as appropriate, and the ACRS will have an opportunity to comment prior to finalizing the resolution to this issue.

The ACRS, in its report, also recommended that Limerick receive special attention in the NRC staff's resolution of USI A-17 (Systems Interactions in Nuclear Power Plants). The staff has reviewed the Philadelphia Electric Company's effort to identify systems interaction problem areas and has concluded that additional interaction studies at Limerick would not be likely to yield improvements that would result in a significant improvement in risk. The staff has considered all systems interaction studies performed to date, including the one for Limerick, as part of developing the resolution of USI A-17. Should the generic resolution indicate the need for plant-specific actions, the staff will provide specific criteria and guidance as part of the resolution.

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5. Mitigation Studies Applicable to the Limerick Generating Station

The Research Development Association report on containment venting cited in your letter has been given to the appropriate ACRS members by the staff.

(Signed) William J. Dircks

William J. Dircks
Executive Director for Operations

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 22 1985

MEMORANDUM FOR: Hugh L. Thompson, Director
Division of Licensing

James P. Knight, Acting Director
Division of Engineering

FROM: Themis P. Speis, Director
Division of Safety Technology

SUBJECT: UNISOLATED LOCA OUTSIDE DRYWELL IN SHOREHAM

The enclosed draft report on unisolated LOCAs outside of the drywell in the Shoreham reactor building is a scoping study to identify high-energy line breaks that are important with respect to isolation requirements. It identified breaks in the RWCU, HPCI, and MSL drain lines as important.

This study used an upper bound assumption that the isolation valves in these lines do not work. The preliminary results of the analysis indicated that the estimate of core-damage frequency for unisolated LOCA outside the drywell at Shoreham assuming that the isolation valves failed to close upon demand is about $2x10^{-5}$/reactor-year. If the isolation valves were assumed to close upon demand, the estimate of the core-damage frequency would be about $4x10^{-7}$/reactor-year. These frequencies of core damage are predicated upon the assumption that the condensate system can be used to mitigate the consequences of an unisolable large LOCA, with an 80% success rate.

In order for RRAB to complete its review of this issue, it is necessary that DL obtain adequate information from the applicant to support the operability of the valves in the HPCI, RWCU and MSL drain lines under pipe break conditions. This information will need to be reviewed by DE to verify adequacy of the valves' operability. The operability of the isolation valves is important for putting these line breaks in the proper safety perspective.

For further information, contact E. Chow, RRAB, x24727.

Themis P. Speis, Director
Division of Safety Technology

Enclosure:
As stated

cc: T. Novak
[redacted]
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An Evaluation of Unisolated LOCA Outside Drywell in the Shoreham Nuclear Power Station

D. Ilberg
N. Hanan

Draft (for comments only)

Risk Evaluation Group
Department of Nuclear Energy
Brookhaven National Laboratory
Upton, New York 11973

February 1985

Prepared for
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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PREFACE

This work was prepared for the NRC which requested it within a one man month time frame. This dictated the use of all readily available information and refraining from physical analyses. Some of the phenomenological assumptions are approximate; hence, more accurate analysis may result in a somewhat different contribution to the core damage frequency for the medium LOCA outside the drywell (the major contribution to core damage frequency is this study). Nevertheless, the identification of the relative hierarchy of contributors is believed to be reasonable.

ACKNOWLEDGEMENT

The authors wish to thank Kenneth Perkins, Kelvin Shiu, and Robert Youngblood for their helpful comments. Cheryl Conrad is much appreciated for typing this document to meet a tightly imposed dealine.

1. INTRODUCTION

1.1 Background

The SNPS-PRA[1] considered LOCA outside the Drywell (LOCA in the Reactor Building) in two ways:

a) Interfacing System LOCAs: Appendix F of the SNPS-PRA estimates the initiator frequency and the core damage frequency for this case. The BNL review[2] of the Shoreham PRA re-evaluated the initiator frequency as well as the core damage frequency, and found an increase about an order of magnitude of the core damage frequency. This result is included in the present study, and for more details see Appendix C of Reference 2.

b) High energy line breaks inside the Reactor Building: The SNPS-PRA included in its analysis only pipes larger than 6 in. in diameter, on the premise that, if not automatically isolated ample time is available to isolate breaks in smaller lines before they cause adverse containment conditions. The frequency of unisolated line breaks downstream of the outboard isolation valve was calculated to be relatively small. The BNL review of this part agreed with the SNPS-PRA, as discussed in Appendix C of Reference 2. In the SNPS-PRA and the BNL review, all the isolation valves were assumed to be capable of operating under a postulated break and the resulting break-flow conditions; random failure to operate was used in both studies.

It is shown in Reference 2 that interfacing system LOCAs are the major contributor to LOCAs outside the drywell.

1.2 Objectives

This study is a special consideration of case (b) above stemming from the assumption of the failure of the corresponding isolation valves in the case of a line break outside drywell. NRC requested BNL to re-evaluate the core damage frequency from high energy line-breaks inside the Reactor Building (same as case (b) above) under the assumption that most of the isolation valves are not qualified to close under break-flow conditions, i.e., assuming the failure of most of the isolation valves. Under this assumption, there is

a need to examine the rupture of any pipe (regardless of diameter) opening a path that leads from the Reactor Pressure Vessel (RPV) to the Reactor Building, for potential adverse environment or flood effects.

This assumption obviously increases the contribution of the high energy line breaks to core damage frequency and requires consideration of other lines connected to the RPV of diameter < 6 in.

This study considers the following questions:

(a) What would be the increase in core damage frequency due to the assumption stated before, i.e., the failure of isolation valves to perform their function?

(b) What would be the contribution to core damage frequency from each pipe connecting the RPV and the Reactor Building?

(c) What isolation valves would be important for mitigating the outside drywell LOCAs?

(d) What is the characteristic time available for operator action?

1.3 Scope

The scope of the BNL study was defined to cover the following:

(a) To identify any significant(*) high energy lines leading from the RPV to the Reactor Building with a potential for affecting safety systems, if an unisolated break were postulated.

(b) To estimate the change in SNPS core damage frequency relative to the SNPS-PRA[1] and BNL review[2] due to the following assumptions on the operation of isolation valves following the occurrence of a line break:

(*) The contribution from downstream moderate energy lines of a system was neglected if it was smaller than the contribution of the lines upstream.

(1) The Main Steam Isolation Valves (Inboard and Outboard) on all four main steam lines will isolate in all the cases considered, having the failure rates shown in Table 2 (discussed in Appendix A).

(2) All check valves will close on reverse flow as designed with the failure rates shown in Table 2 (discussed in Appendix A).

(3) All other isolation valves will fail to close when receiving their signal to close. No partial closure is assumed for these valves.

(4) Manual valves are assumed to be available for isolation if accessible by the operator.

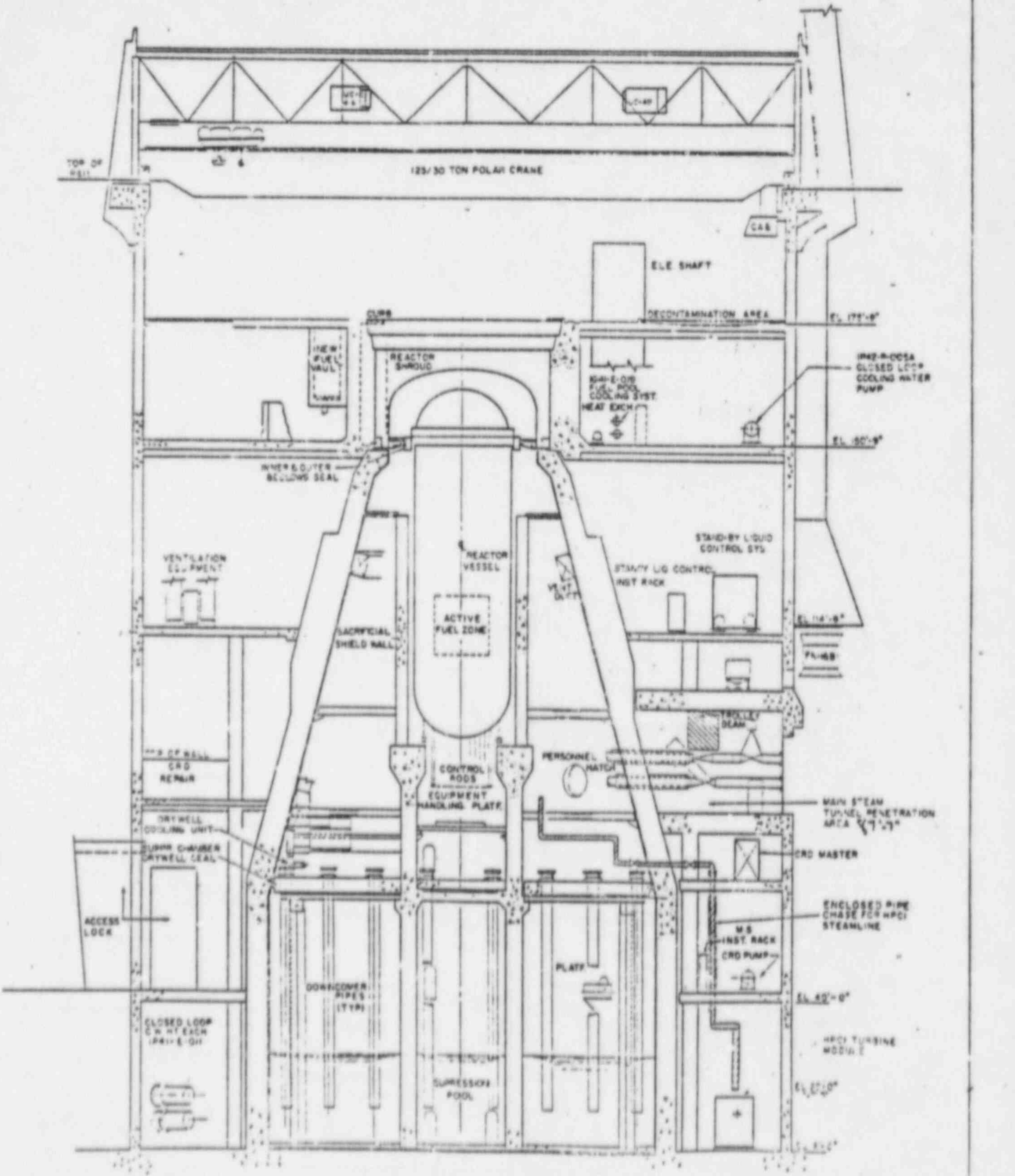
(5) Remote operated valves that do not receive automatic closure signals upon sensing break conditions are identified. However, no credit is given for them in this study.

(c) To provide the list of the more important isolation valves from the standpoint of reducing the core damage frequency.

(d) To provide some crude insights on the time available for the operator to respond to such accidents.

1.4 General Description of the Problem Evaluated

The Shoreham Reactor Building surrounds the MARK II containment structure (the drywell). At its lowest elevation (referred to here as Elevation 8), the building is an open cylindrical compartment, i.e., there are no barriers in Elevation 8 compartments. This open area presents the possibility that excessive water released into the compartment may adversely affect the ECCS equipment in Elevation 8. The SNPS Reactor Building has openings between its floors, and a line break at a high elevation will affect the entire reactor building (see section 3.1 for more details). Figure 1 provides a general description of the SNPS Reactor Building Elevations.



STEAM TO HPCI TURBINE

Fig. 1 General Description of SNPS Reactor Building Elevations (with Emphasis on HPCI Steam Line Routing)
From: Shoreham Nuclear Power Station - Unit 1 Final Safety Analysis Report

REVISION 14 - DECEMBER 1978

Figures 2a and 2b show lines that connect the RPV to the Reactor Building and provide a potential path from the RPV to the volume of the Reactor Building in the event of a break with a failure of the pertinent isolation valves to close. These figures do not show all isolation valves, but only those that are designated as containment isolation valves. In some cases, the most important being the RWCU, other valves are available to the operator for remote line isolation from the control room; these valves are not shown in Figures 2 and 3.

A list of the lines emerging from the RPV and some additional information associated with these lines (size, type of isolation valves, and process or standby line) is given in Table B.1 of Appendix B (reproduced from the SNPS-FSAR[3]).

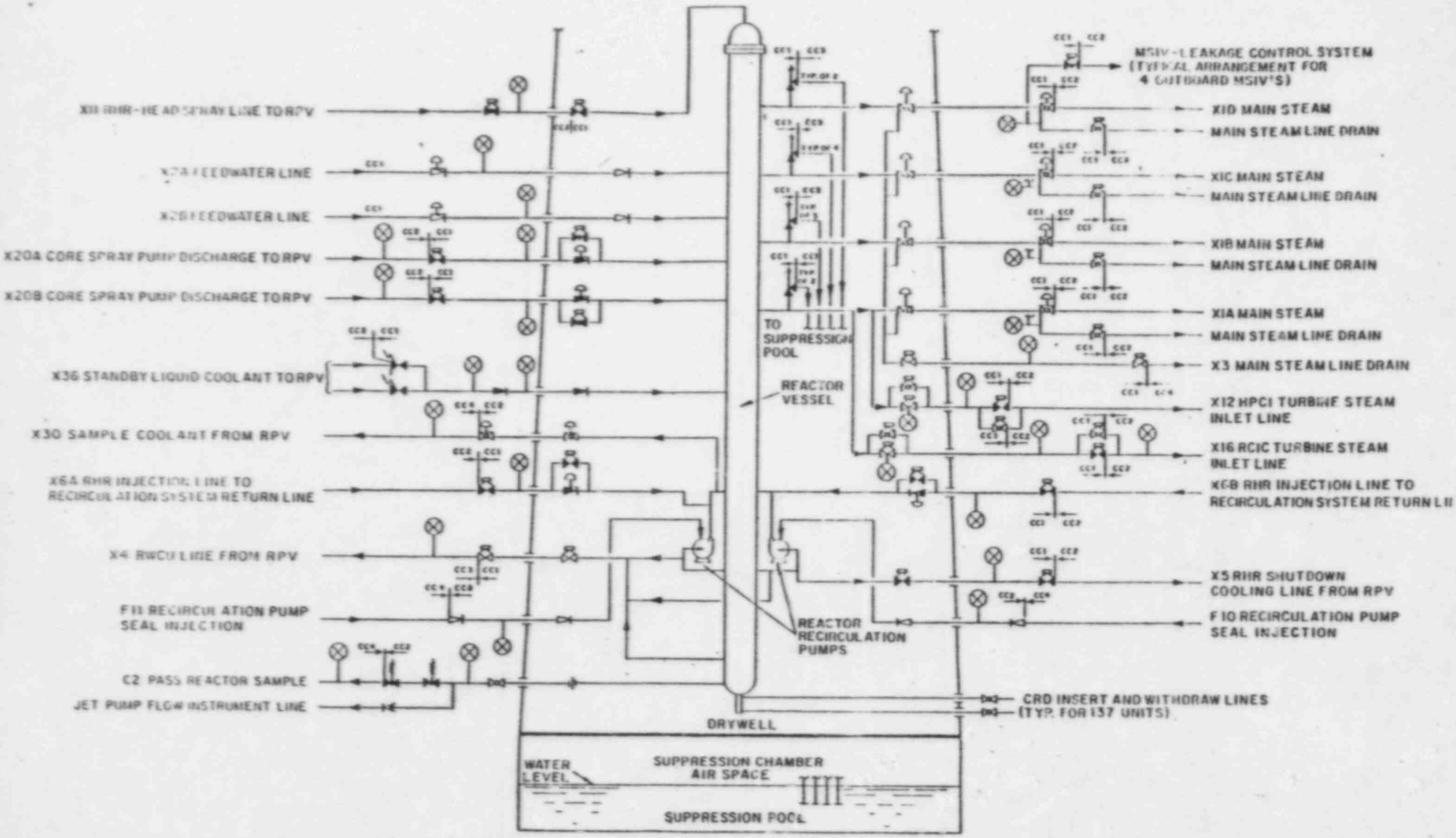


Fig. 2a Lines from Reactor Pressure Vessel to Reactor Building.

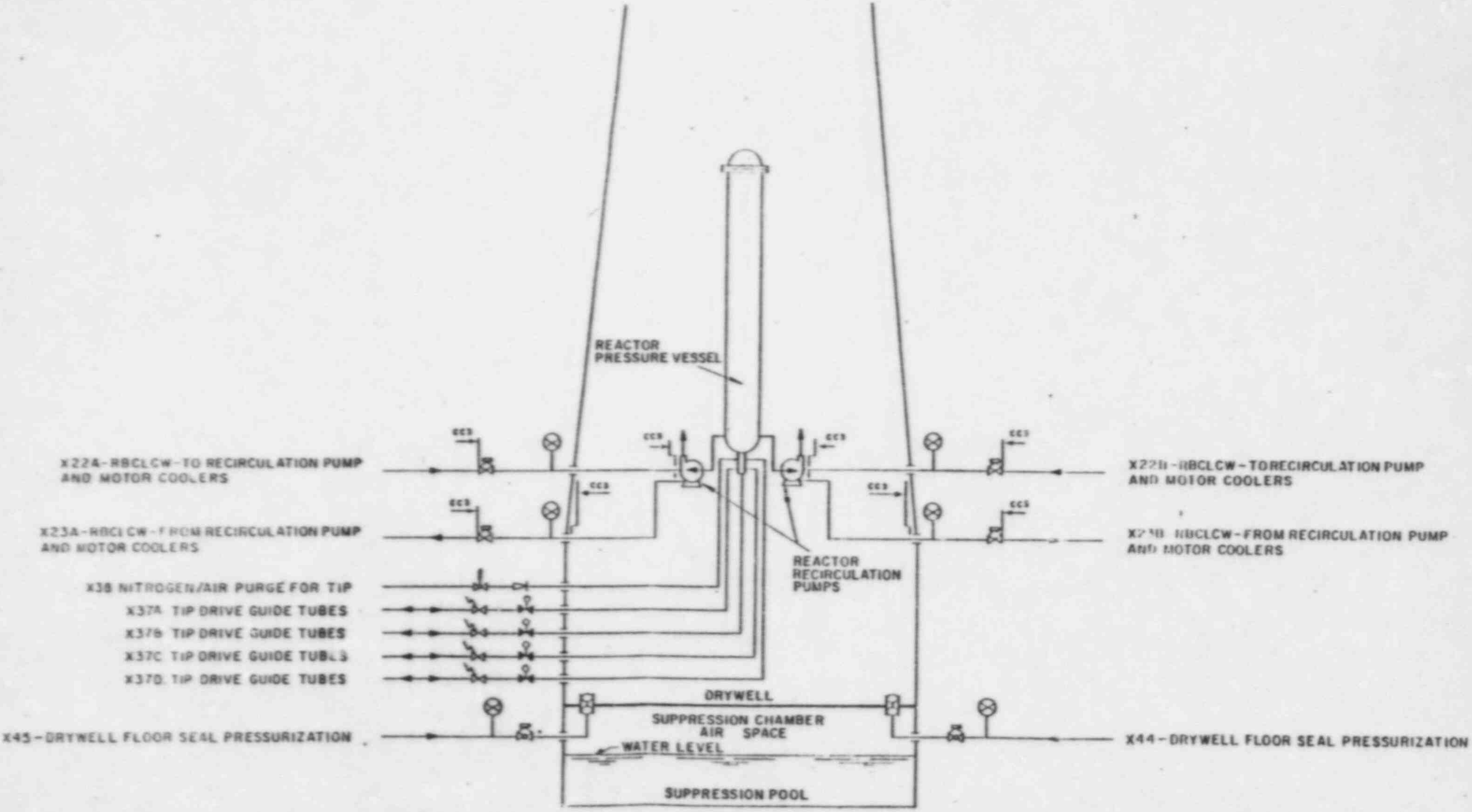


Fig. 2b Tip Drive Guide Tubes Connections to Reactor Pressure Vessel

2. EVALUATION OF PIPE BREAK FREQUENCIES

This section covers the evaluation of the frequencies of high and moderate energy pipe breaks excluding interfacing LOCAs. The interfacing LOCAS are addressed in Appendix C of Reference 2 and the results are included in Tables 2 and 3.

The pipes considered in this BNL study are listed in Appendix B. All lines which are associated with General Design Criterion (GDC) 55 are analyzed in this BNL study.(*) In addition, the Transversing Incore Probe (TIP) Drive Guide Tubes (GDC-57) are considered. All other lines referred to in Table B-1 as GDC-56 or 57 are not connected to the RPV; they are mainly connected to the Suppression Pool (the routing was rechecked).

The SNPS-FSAR(3) was the main source for determining the number of pipe sections and valves or other discontinuities on each line. The isometric drawings of pipe routing in the Reactor Building from Appendix 3C of the SNAPS-FSAR were used. They were compared with the system-specific drawings given in the other FSAR chapters. The summary of this task is presented in Appendix C of this report.

The evaluation of pipe break frequencies was made with the failure and unavailability data summarized in Table 1. The bases for the values shown in this table are further discussed in Appendix A. The failure and unavailability data were used with the number of sections and valves or discontinuities identified for each line, to compute the frequency of line breaks. The summary of this task is presented in Table 2. An example of this computation is shown in Appendix C.

The results of Table 2 were next grouped into seven different cases:

(a) Large Interfacing LOCAs (Liquid discharge through break)

(b) Large LOCAs outside Drywell: (1) steam and (2) liquid discharge

(c) Medium LOCAs outside Drywell: (1) steam and (2) liquid discharge

(d) Small LOCAs outside Drywell: (1) steam and (2) liquid discharge.

(*)In References 1 and 2 consideration was given to large break LOCA outside the drywell, i.e., lines which are 6 in. in diameter or more.

Table 1 Summary of Failure and Unavailability Data for Pipes and Valves

Component	Failure Mode	Failure Rate (Mean)	
		Break Exclusion	Non-Break Exclusion
Pipes > 3" (per section)	Rupture	8.6×10^{-11}/hr	8.6×10^{-10}/hr
Pipes $\leq$ 3" (per section)	Rupture	8.6×10^{-10}/hr	8.6×10^{-9}/hr
Check Valves	Severe Internal Leakage	--	3.3×10^{-3}/yr
	Rupture	1.5×10^{-10}/hr	1.5×10^{-9}/hr
Motor Operated Valves (MOV)	Failure to Operate (w/ command faults)	--	8×10^{-3}/d
	Failure to Operate (w/o command) faults)	--	6×10^{-3}/d
	Two MOVs (CMF)	--	2×10^{-3}/d
	Rupture	1.5×10^{-10}/hr	1.5×10^{-9}/hr

TABLE 2 Estimated Frequencies of Breaks Outside Containment

BREAK LOCATION	CASE	BREAK SIZE	NUMBER OF: LINES	SECTIONS	VALVES (*)	ISOLATION VALVES: VALVES DESIGNATORS	ASSUMED FAILURE PROBABILITY	INITIAL BREAK FLOW: STEAM OR LIQUID	ESTIMATED FREQUENCY OF BREAK OCCURRENCE	DESCRIPTION OF THE CASE ANALYZED
Main Steam Line	I	24"	4	1	1	IB21-AOV081 Inboard MSIV	6.0E-3	steam	5.0E-8	Break exclusion section and valve between Reactor Building penetration and the outboard MSIV. (Elevation 78).
	II	24"	4	1	0	Inboard and Outboard MSIV IB21-AOV082	2.0E-3	steam	6.0E-9	Break exclusion section from outboard MSIV up to the Jet-Impingement Barrier. (Elevation 78).
Main Feed-water Line	I	18"	2	1	1	Check Valve F002 A/B	3.3E-3	steam	1.4E-8	Break exclusion section and testable check valve between reactor building penetration and the testable checkvalve. (Elevation 78).
	II	18"	2	3	1	Testable C.V. IB21-AOV036 A/B and C.V. F002 A/B	$[3.3E-3]^2$	steam	7.8E-11	Break exclusion sections and IB21-MOV035A/B from testable check valve up to the Jet-Impingement Barrier (Elevation 78).
High Pressure Coolant Injection (HPCI) Steam Line	I	10"	1	1	1	IE41-MOV041	1.0	steam	2.1E-6	Break exclusion section and valve between Reactor Building penetration and the outboard isolation valve IE41-MOV042. (Elevation 66).
	II	10"	1	6	6	IE41-MOV041 and IE41-MOV042	1.0	steam	1.4E-6	Non break exclusion sections and valves from outboard isolation valve up to HPCI turbine. Four openings (24 hrs each) per year of valve MOV-042 are assumed. (Elevation 66 down to elevation 17).
	III	1"	1	17	17	IE41-MOV048 and IE41-MOV047	1.0	steam	1.0E-3	Non Break exclusion sections and valves from Reactor Building penetrations up to the 1-1/2" HPCI/RCIC drain line to condenser. Normally open path. (Elevation 66 down to elevation 11).

* This includes all discontinuities, i.e.: valves, pumps, reducers and heat exchangers (see Appendix A).

TABLE 2 Estimated Frequencies of Breaks Outside Containment Cont'd.

BREAK LOCATION	CASE	BREAK SIZE	NUMBER OF: LINES	SECTIONS	VALVES (*)	ISOLATION VALVES: VALVES DESIGNATORS	ASSUMED FAILURE PROBABILITY	INITIAL BREAK FLOW: STEAM OR LIQUID	ESTIMATED FREQUENCY OF BREAK OCCURRENCE	DESCRIPTION OF THE CASE ANALYZED
Reactor Core Isolation Cooling (RCIC) Steam Line	I	4"	1	1	1	IE51-MOV041	1.0	steam	2.1E-6	Break exclusion section and valve between Reactor Building penetration and the outboard MOV 042. (Elevation 87).
	II	3"	1	6	6	IE51OMOV041 and MO[illegible]	1.0	steam	5.8E-6	Non break exclusion sections and valves from outboard isolation valve up to RCIC turbine. Four openings per year of valve -042 are assumed (All elevation below elevation 87 down to elevation 8).
	III	1"	1	14	14	IE51-MOV048 and IE51-MOV047	1.0	steam	1.2E-3	Non break exclusion section and valves from Reactor Building Penetration up to the 1-1/2" HPCI/RCIC drain line to condenser. Normally open (Elevation 87 down to Elevation 8).
RCIC/HPCI Steam Drain Line	I	1-1/2	1	1	0	IE51/IE41 AOV-081 or AOV-082	1.0	steam	5.E-5	Section between HPCI and RCIC drain lines connection and the penetration to the main steam tunnel. (Between elevation 11 and 70).
Reactor Water Cleanup System (RWCU) Supply Line	I	6"	1	1	1	MOV033 (F001) and MOVF102 and MOVF100 F106	1.0	Liquid	2.1E-6	First section in Reactor Building. It is break exclusion and normal operating. (Elevation 112)
	II	6"	1	1	0	The above and IG33-MOV034 or(F004)	1.0	Liquid	7.5E-6	Section from outboard isolation vlave to the 6x3" reducer. Non break exclusion (Elevation 112).
	III	3"	2	3	3	same as above	1.0	Liquid	5.4E-4	Section and valves from reducer up to RWCU pumps. (Elevation 112)

TABLE 2 Estimated Frequencies of Breaks Outside Containment Cont'd.

BREAK LOCATION	CASE	BREAK SIZE	NUMBER OF: LINE	SECTIONS	VALVES (*)	ISOLATION VALVES: VALVES DESIGNATORS	ASSUMED FAILURE PROBABILITY	INITIAL BREAK FLOW: STEAM OR LIQUID	ESTIMATED FREQUENCY OF BREAK OCCURRENCE	DESCRIPTION OF THE CASE ANALYZED
Reactor Water Cleanup System (RWCU) Supply Line	IV		2	1	1	The above + 2 manual valves	1.0	Liquid	1.8E-4	Sections and valves from 2x3" reducers to 3x4" reducers. (Elevation 112).
	V	3"	2	2	2	Same as above	1.0	Liquid	3.5E-4	Sections and valves from 2x3" reducers to 3x4" reducer. (Elevation 112).
	VI	4"	1	5	2	Same as above + Manual valves	1.0	Liquid	4.0E-4	Sections and valves from 3x4" reducer up to the discharge of the non-regenerative Heat Exchanger and up to the normally closed IG33-MOV035. (Elevation 126).
Main Steam Line Drain (Inboard)	I	3"	1	1	1	IB21-MOV031	1.0	steam	8.8E-6	Normally open - break exclusion section and valve between reactor building penetration and the outboard valve IB21-MOV032. (Elevation 76).
	II	3"	1	1	1	IB21-MOV031 and MOV032	1.0	steam	8.8E-5	Normally open -non break exclusion section and valve from MOV032 up to the Jet-Impingement Barrier. (Elevation 76).
Main Steam Line Drain (Outboard) and MSIV leakage Control (inboard)	I	2"	4	2	4	Inboard MSIV IB21-AOV081	$6x10^{-3}$	steam	5.1E-7	Break exclusion sections and valves between main steam line connection and IE32-MOV021, IB21-MOV061 MOV062, MOV063, MOV064. (Elevation 76).
MSIV Leakage Control (out-board)	I	2-3"	3	1	2	Inboard and Outboard MSIV IB21-AOV082	$2x10^{-3}$	steam	6.0E-8	Break exclusions sections and valves between main steam line connection and IB21-MOV034, IE32-MOV024 IE32-MOV026. (Elevation 76).

TABLE 2 Estimated Frequencies of Breaks Outside Containment Cont'd.

BREAK LOCATION	CASE	BREAK SIZE	NUMBER OF: LINES	SECTIONS	VALVES (*)	ISOLATION VALVES: VALVES DESIGNATORS	ASSUMED FAILURE PROBABILITY	INITIAL BREAK FLOW: STEAM OR LIQUID	ESTIMATED FREQUENCY OF BREAK OCCURRENCE	DESCRIPTION OF THE CASE ANALYZED
Interfacing LOCA:										
- RHR Shutdown Cooling	I	20"	1	-	2	--	--	Liquid	2.0E-6	All four interfacing LOCA cases estimated on the basis of 0.02 for testable check valve unavailability times 10^{-3} for spurious MOV opening and 0.1 for probability of low pressure piping to fail before isolation. See detail in reference 2 (Elevation - 8 up to elevation - 87).
- RHR Head Spray Line	II	4"	1	-	2	--	--	Liquid		
- RHR/LPCI Injec. Line to Recirc. Lines	III	24"	2	-	2	--	--	Liquid		
- LPCS Injection	IV	10"	2	-	2	--	--	Liquid		
Standby Liquid Control (SLC)	I	1-1/2	1	1	1	LO-F008 and Inboard C.V. F007	3.3E-3	Liquid	1.5E-8	Break exclusion section of the SLC (Elevation 112)
	II	1-1/2	1	1	1	The above and Outboard C.V. F006	$[3.3E-3]^2$	Liquid	1.0E-9	Non break exclusion section of SLC (Elevation 112).
Control Rod Drive (CRD)	I	1-1-1/2					1.0	Liquid	1.0E-4	Scram Discharge Volume (SDV) header rupture. (Non break exclusion). The pipe break frequency is taken from NUREG-0803. (Elevation 78, 63 and 40).

TABLE 2 Estimated Frequencies of Breaks Outside Containment Cont'd.

BREAK LOCATION	CASE	BREAK SIZE	NUMBER OF: LINES	SECTIONS	VALVES (*)	ISOLATION VALVES: VALVES DESIGNATORS	ASSUMED FAILURE PROBABILITY	INITIAL BREAK FLOW: STEAM OR LIQUID	ESTIMATED FREQUENCY OF BREAK OCCURRENCE	DESCRIPTION OF THE CASE ANALYZED
Recirc. Pump Seal Injection	I	3/4	2	2	2		1.0	Liquid	2.0E-7	
Other 3/4" lines Branches from system shown in this table	I	3/4	1	20	20	Valves of the various system shown in this table	1.0	steam and Liquid	1.8E-3	(All elevation)
Sample Coolant from RPV	I	3/4	1	2	2		1.0	Liquid	1.8E-4	
Reactor Post Accident Sampling system (PASS)	I	3/4	1	2	2		1.0	Liquid	1.8E-4	
TIP Drive Guide Tubes	I	3/8	4	2	2	Ball valve and shear valve	1.0	Liquid	1.0E-5	(Elevation 60).

The combined frequency in each group is shown in Table 3. Note that the LOCA frequencies of the large and medium breaks groups are dominated by the line breaks of a single system. For the liquid breaks, it is the RWCU, and for the steam breaks, it is HPCI and MSL drain systems. In the latter case, the 10-in. HPCI line break has a frequency of 3.5×10^{-6}, while all other line breaks which contribute to the large LOCA steam line break have a frequency of 3% of that of HPCI. Similarly, in the case of the Main Steam Line (MSL) drain break, its frequency is 92% while the RCIC break frequency is only about 8%. Therefore, in the rest of this study, when discussing large or medium breaks, only the line breaks of the dominating systems are included; namely, the HPCI 10-in. line break, the RWCU 6-in. and 3-in. line breaks, and the MSL drain 3-in. line break.

The small steam line breaks are mainly due to HPCI and RCIC bypass line breaks (it is the case of a blowdown limited by the 1-in. bypass line). This will be referred to as the 1-in. line break even though the lines may be larger in diameter. The small liquid line breaks are represented in this BNL study by the RWCU 3/4-in. branches, and by the CRD SDV header piping rupture (reproduced from NUREG-0803[4]) which are about 1-1/2 in. equivalent diameter.

Table 3 also includes, for each of the LOCA-outside-Drywell groups, the liquid or steam break discharge flow rate at two different times:

(1) Initially, when the break occurs and flow rates are at their peak values, and

(2) At about 30 minutes later after coolant injection is established, depressurization of the RPV is completed and operator takes control of the injection according to procedures, keeping the core covered.

These flow rates values should be taken as crude estimates. They were obtained from NEDO-24708[5] for the purpose of providing some indication of the time available for operator diagnosis and response. The NEDO-24708 report provides this information for the entire spectrum of break size under consideration in this study.

3. ASSESSMENT OF MITIGATION CAPABILITY

In this section, the effects of LOCA outside the drywell are discussed according to the three different groups: small, medium, and large pipe breaks (see Table 3). Based on these effects, some insight on the time available for mitigation is presented. The first subsection provides general information on alarms available for diagnostics, containment sumps capacity and flooding data, and some crude information on the containment atmosphere temperature increase due to steam or saturated liquid discharges. The next subsections describe the mitigation conditions for small, large, and medium LOCAs outside the drywell.

3.1 Reactor Building Information

3.1.1 Instrumentation for Diagnostics

The following instrumentation and alarms are available to alert the operator in the case of a pipe break in the Reactor Building:

-- Reactor Building ventilation isolation alarm

-- Reactor Building equipment sump level alarm in the vicinity of the break

-- Reactor Building floor drain sump level alarm

-- Reactor Building flooding alarm at elevation 8 (see additional description below)

-- Area radiation monitor alarms

-- Reactor Building Standby Ventilation Exhaust high-radiation alarms

-- Area high-temperature alarms on elevation 8 and on the floor where the break occurs

-- Specific systems have their own break detection instrumentation such as the RWCU, MSL drain, HPCI, and RCIC.

-- Reactor Building low differential pressure alarms.

Most of these alarms are also sensitive to a small break LOCA of about 3/4-in. diameter but some set points will only be reached after about half an hour.

The Reactor Building (RB) water level at elevation 8 is detected by two RB level monitors installed on the RB floor. The flood alarms are activated by the monitors when the water level is more than 0.5 inch above the floor. The sump alarms will be activated when the water level reaches the sump alarm setpoints installed at a level just below the level that activates the RB flood alarms. Sump alarm sensors are installed at various locations in the RB.

The area high temperature alarms include the following:

-- RCIC and HPCI turbine steam line space high temperature (7 sensors each). Isolation signal setpoint at 155°F (elevation 8)

-- RHR space high temperature alarm (6 sensors) with setpoint at 175°F (elevation 8)

-- RWCU space high temperature (18 sensors) isolation signal at 155°F (elevation 112)

-- Main Steam line space high temperature (4 sensors per line) isolation signal at 200°F (elevation 78).

-- Main steam tunnel containment penetration area high temperature (4 sensors) located in the area of MSL drain lines. Isolation signal at 140°F.

3.1.2 Sump Pumps and Flooding Buildup Volumes

The open area of the elevation 8 floor is approximately 5,500 sq. ft. This area is the total floor area minus the area occupied by equipment foundations, columns, drain tanks, etc. Based on this area, flood buildup on elevation 8 is 3400 gal/in.

The drainage capabilities at SNPS are:

-- Reactor Building Floor Sumps - 2490-gal capacity

-- Reactor Building Equipment Sumps - 1660-gal capacity

-- Reactor Building Porous Concrete Sumps - 500-gal capacity.

Table 3

Summary of Frequencies of LOCA Outside Drywell

Initiator	Break Flow Conditions(*) Initial Stm/Liq	Initial lb/sec	After 30 Minutes Stm/Liq	After 30 Minutes lb/sec	Break Location (Main Contributor)	Initiator Frequency (Event/yr)
Large Size Breaks	Steam	1400	Liquid	1200	HPCI(**) (elevation 8')	3.6E-6
$\phi \geq 6"$	Liquid	1200	Liquid	700	RWCU (elevation 112')	9.6E-6
Total $\phi \geq 6$						1.3E-5
Large Interfacing LOCAs $\phi \geq 6"$	Liquid	1200	Liquid	700	LPCI/LPCS elevations 87' down to 8'	2.0E-6
Medium Size Breaks	Steam	120	Steam	60	MSL Drain	1.0E-4
$2" \leq \phi \leq 4.3"$	Liquid	400	Liquid	250	RWCU (elevations 112'-126')	1.5E-3
Total $2 \leq \phi \leq 4.3"$						1.6E-3
Small Size Breaks	Steam	10	Steam	5	HPCI/RCIC(**) (elevation 8')	~3.0E-3
$\phi < 2"$	Liquid	25	Liquid	12	RWCU Branches (elevations 112'-150')	~1.5E-3
Total $\phi < 2"$						~4.5E-3

(*) Approximate crude estimates of steam or liquid discharge through break from NEDO-24708.

(**) Break can occur between elevation 66 and 8, but the other break locations discharge through a pipe chase to elevation 8.

These systems have a total sump capacity of 4650 gallons. The total sump. pump capacity is 640 gpm, as follows:

-- Four 50 gpm equipment drain sump pumps (elevation(*) 9 ft)

-- Six 50 gpm floor drain sump pumps (elevation(*) 9 ft)

-- Two 20 gpm porous concrete sump pumps (elevation(*) 9 ft)

-- One 100 gpm leakage return pump (elevation(*) 12 ft).

The leakage return pump is designed to process radioactive water. If the floor drain sump pump indicators register radioactive material, all sump pumps will isolate. The leakage return pump can then be manually activated by the operator. In addition, only the leakage return pump is powered from onsite AC power.

It can be inferred that if flooding is not arrested before it reaches the 1 ft level above the elevation 8 floor (elevation 9), the sump pump capacity may drop from 600 gpm to 100 gpm. This corresponds to accumulation of about 42,000 gallons. Furthermore, since this study considers primary water release, it is assumed that only the leakage return pump would be operating (other sump pumps would be isolated).

RCIC, HPCI, LPCI/RHR, and LPCS are all located at elevation 8. It is assumed that they become disabled when water reaches 4 ft (about 160,000 gallons) as stated in SNPS-PRA.[1]

3.1.3 Containment Atmosphere

The SNPS-FSAR[3] includes in Appendix 3C a few calculations of Reactor Building temperatures for water and steam line breaks. Table 4 shows the results of one calculation for the discharge of 40,000 lb of saturated water at RPV normal power conditions out of a 4-in. line break at elevation 112 ft of

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(*)If water reaches this elevation, the pump is assumed to fail.

Table 4 Reactor Building Temperatures at Several Elevations Resulting from a 40,000 lb. Discharge

Reactor Building Elevation	Initial Temperature(*) [°F]	Equilibrium Maximum Temperatures [°F]	Comments
8'-0"	104	< 140	
40'-0"	"	148	
63'-0"	"	183	
78'-7"	"	194	
112'-9"	"	217	Break location at 112'. Outside the pump room temp is 177°F
150'-9"	"	148	
175'-9"	"	< 132	

(*) Reactor Building humidity changed from 50% initially to 100%.

the Reactor Building. In this deterministic analysis, the break was assumed to be isolated by an RWCU isolation signal at 40 sec after initiation of the break. This break results in less than 5,000 gal at elevation 8 or less than 1-1/2-in. water accumulation on that floor. It is seen from Table 4 that a break of this size is rapidly affecting Reactor Building atmosphere conditions.

The other calculations reported in Appendix 3C of SNPS-FSAR[3] are similar and lead to the assumption that conditions of 212°F in the Reactor Building elevation 8 will occur under the following circumstances:

(1) A RWCU line break discharging more than 500,000 lb. This is approximately the amount discharged from a RWCU 3-in. line break in 15 minutes (5 minutes for a 6-in. line).

(2) A MSL drain line discharging more than 100,000 lb of steam at RPV normal power conditions. For a 3-in. MSL drain line break this will occur in approximately 10 minutes.

(3) A RCIC/HPCI 1-in. line discharging more than 15,000 lb of steam at RPV normal power conditions directly to elevation 8[*]. For a 1" line break, this will occur in more than 25 minutes, and therefore 212°F conditions at elevation 8 from these line breaks are not expected to occur[**].

Temperatures higher than 140°F in elevation 8 can result when steam is discharged directly to this elevation from a 1-in. RCIC or HPCI line continuously.

(*) RCIC and HPCI steam lines are enclosed in piping chase which protects higher elevation against a steam line break in these systems. However, for most steam line breaks in higher elevations, steam will exit at elevation 8.

(**) The 15,000 lb discharge would cause the saturation conditions only if discharged during a very short time, which is not the case here.

3.1.4 Procedures

Given a LOCA outside containment, the SNPS procedures dictate rapid manual depressurization of the RPV by the ADS. This action substantially reduces the flow rate through the break. If low pressure injection is provided at about 200 psi, break flow may become only about one-half of the initial break flow.

Given an RB flooding alarm, the operator is required to:

-- Monitor RB level to determine the approximate leak rate, and to ascertain the approximate location of the break (using additional sump alarms and high area temperature alarm)

-- Monitor parameters such as line pressures and flow rate of the safety systems, as a leak may affect these system parameters

-- If requireu and plant conditions permit, dispatch an operator to the RB floor to visually locate the source of leakage.

-- Isolate the break using the appropriate system procedure (HPCI, RCIC, RHR, others).

3.2 A Small LOCA Outside Drywell (< 1-1/2" Break Size)

3.2.1 Accident Conditions and Alarms

The description that follows is based on an analysis by NRC staff of a pipe break equivalent to a 1.2-in. line break. This is discussed in detail in NUREG-0803.[4] The description in this section applies to small line breaks, in general, and applies to the SNPS. It does not, in particular, apply to SDV header pipe breaks to which the original discussion refers.

The break described is a water line break discharging 550 gpm (~ 70 lb/s) initially. This is equivalent to a 1.2-in. line break discharging from the RPV at 1032 psi conditions.

Several alarms are available to the operator as described in section 3.1.1 above. The most expected early alarms are from the Reactor Building radiation monitors and from local area high temperature alarms.

NUREG-0803 cites a calculation for a typical BWR Reactor Building that shows a temperature rise to 110°F in 10 minutes and 140°F in 30 minutes for a discharge of 550 gpm at RPV conditions. (This amounts to about 130,000 lb over 30 minutes.) It may activate high temperature alarms if the set points is 120°F, but it will not isolate HPCI or RCIC systems.

The SNPS sumps and flooding setpoints are low (see section 3.1.1), i.e., at 1/2-in. above floor level which corresponds to 2000 or 4000 gallons of water accumulation. Therefore, the water accumulation at the 550 gpm flow rate will cause Reactor Building sump and flood alarms to actuate within 5 to 10 minutes (assuming 35% flashing into steam, travel time through stairwells and floors, and partial accumulation in equipment sumps (up to 2,000 gallons).

3.2.2 Reactor Building Environment

The water released from the break will exceed the local drain sump capacity, and some will flow to lower elevations through stairwells. Assuming that only the leakage return pump is available,(*) the accumulation of water at elevation 8 would be less than 0.13 in./min i.e., it would take six hours to reach the level that threatens ECCS equipment availability. Thus, ample time is available for the operator to recognize the need to depressurize the reactor and reduce break flow. Note that Appendix 3C in the SNPS-FSAR states that equipment along stairwells is protected against dripping of 212°F water.

During the initial blowdown, temperatures in the nearest area to the break can reach 212°F. The Reactor Building temperature is expected to rise significantly as shown in Table 4 for a discharge of 40,000 lbs of saturated water at elevation 112 ft. This is a 10 minute discharge from the 1.2-in. line break described here. While it may result in high Reactor Building temperatures when discharged over a short period of time, it results in 110°F in the Reactor Building if discharged during about 10 minutes (see section 3.2.1). However, the temperature in containment will continue to rise due to the continued discharge through the break and may reach the 155°F RCIC/HPCI isolation temperature after about one hour. The Reactor Building Standby Ventilation System (RBSVS) of SNPS has a heat removal capability of less than

(*)Radwaste system tanks capacity allows for about one day accumulations of untreated water at a 100 gpm pumping rate.

5% of the heat discharged by a 1.2-in. line break, before reactor is depressurized and the break flow is reduced.

3.2.3 Operator Response

At Shoreham, the operator will have a flooding alarm and high Reactor Building radiation alarm at about 10 minutes as discussed in the previous section.

For a small LOCA outside drywell, with feedwater still operating when the LOCA occurs, scram may not always occur immediately. Following the scram, the operator will try to keep the normal feedwater injection and therefore keep MSIV open. If the MSIV remains open (which is the more probable case), it may take a while before the operator will notice the abnormally high feedwater flow rate. It appears that the flooding and high reactor building radiation alarms will indicate that a small LOCA have occurred, and the increased feedwater injection flow may be used for verification.

Therefore, it is expected that the operator will recognize a small break LOCA in the reactor building within about 30 minutes after scram. Unless the operator perceives a LOCA, he will depressurize the reactor at a rate of only 100°F per hour. In such a case it will take 4 hours to depressurize the reactor to 100psi and reduce break flow by about a factor of 10. As seen in section 3.2.2, four hours are available at SNPS, without flooding to elevation 12. However, in this case, the temperature in Reactor Building may reach 155°F or higher(*) between 1 and 2 hours and trip HPCI and RCIC, and most probably require depressurization for low pressure injection. These events would lead the operator to recognize the small LOCA outside containment with very high probability, if he failed to recognize it during the first half hour.

It should be noted that unlike the generic analysis on NUREG-0803, the authors believe that recognition of a small break LOCA outside drywell at SNPS would be a high probability event mainly because of the improved arrangement for flooding detection at elevation 8 (relative to the arrangement assumed in NUREG-0803). High radiation and high temperature conditions in the reactor

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(*) A GE analysis estimates that the maximum bulk temperature in the reactor building would reach about 140°F (See NUREG-0803(4)).

building will enhance the probability of recognition. This BNL study assumed that it is most probable that manual depressurization of RPV to reduce flow and enthalpy discharge through the break would take place after about 30 minutes to 1 hour into the accident.

The depressurization of the RPV may reduce flow rate and enthalpy of the water discharged through the break to a level accommodated by the sump pumps, and may reverse the conditions in reactor building i.e., conditions may start to improve. It is indicated in NUREG-0803 that rupture of blowdown panels may be required to establish a path for leakage of hot humid air to outside containment (which is larger than the "natural" 100% per day leakage rate from reactor building), in order to improve the reactor building atmosphere conditions and to allow safe operator entry. As shown in NUREG-0803, depressurization reduces significantly the dose received by an operator entering the reactor building.

If an operator is required to enter the reactor building to isolate a break, it can be done for a 1.2-in. line break with early depressurization (and low primary water activity). It would be possible to stay for an hour, and this seems to be sufficient for isolation purposes. Appendix 3C of SNPS-FSAR considers 30 minutes to be sufficient time to walk through all SNPS elevations, locate a break, and isolate it.

3.2.4 Estimation of Core Damage Frequency

The description of the event and the reactor building conditions following a small break LOCA outside drywell were discussed in the previous sections. These are now summarized in the form of an event tree in Figure 3, and quantified. Feedwater and high pressure coolant injection are in general available under the circumstances of small LOCA. ADS, LPCI and LPCS have very low unavailabilities. The values for their quantification are taken from Reference 2. The events that are differently quantified are: (1) the probability that at 30-60 minutes the operators take actions and complete rapid manual depressurization, (X_H), and (2) the probability of controlling the condensate flow if required (V'''). The X_H=0.01 is taken basically from NUREG-0803 where 5×10^{-2} is used. The difference between NUREG-0803 and BNL values is due to the SNPS improved early flooding alarms which increase the probability that the operator recognizes the LOCA outside the drywell and follows the required depressurization procedure.

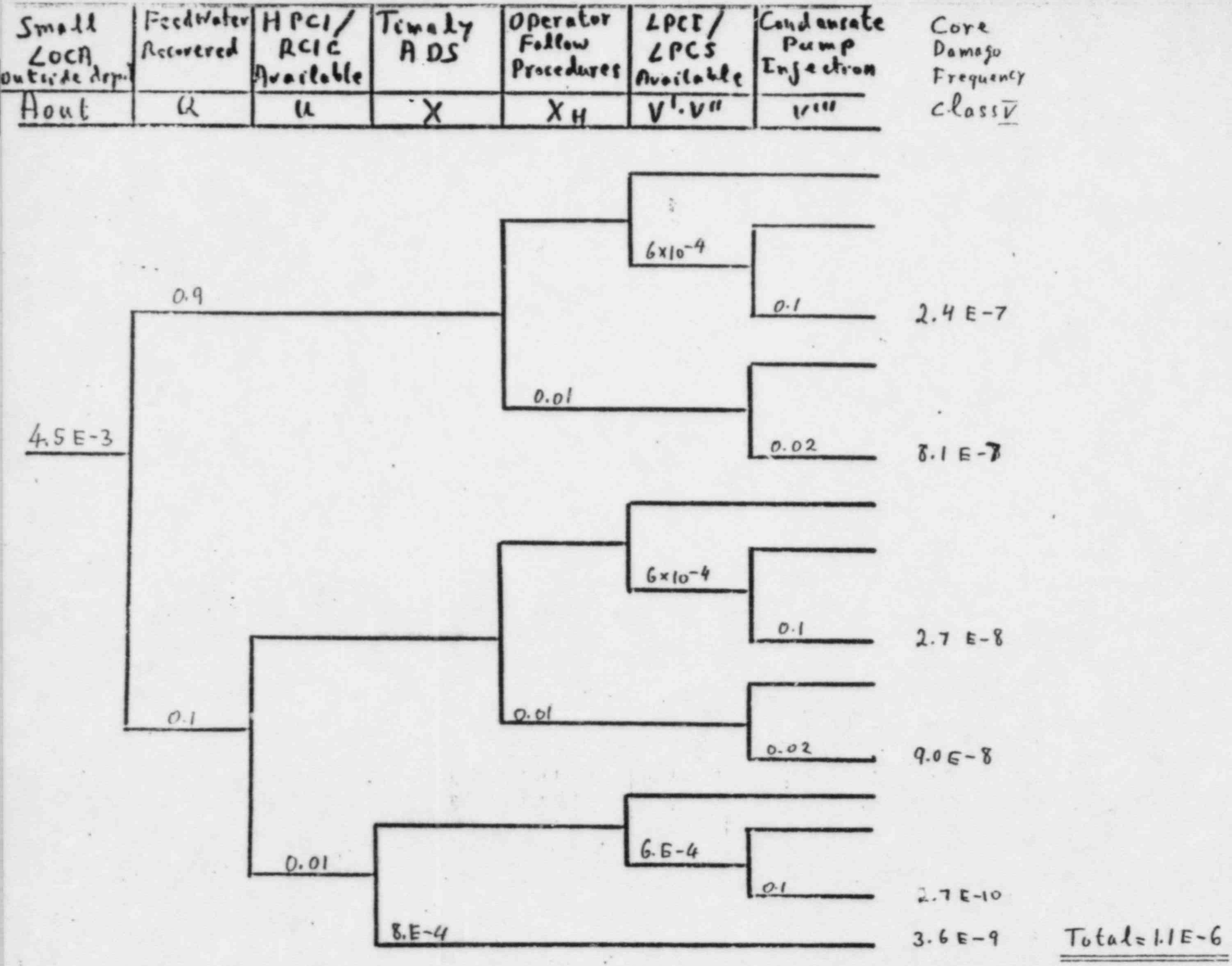


Figure - 3: Event Tree Diagram for Sequences Following Small LOCA Outside Drywell

The V'''=0.1 is the common value used by BNL in Reference 2 for controlling condensate injection if sufficient time is available to the operator (in our case 30 to 60 minutes). The V'''=0.02 includes a factor of 0.2 for the possibility that no damage to LPCI/LPCS will occur even under the circumstances that the operator does not depressurize the reactor early, but rather depressurizes it at 100°F per hour rate, for 4 hours or more. In such a case NUREG-0803 indicates that entry to the reactor building may be delayed for up to 20 hours. The LPCI/LPCS may survive the adverse environment in the reactor building for such a period, because they are qualified to sustain these conditions for at least several hours. A factor of 0.2 (compare V^{111} on Fig. 3) for the LPCI/LPCS availability apparently underestimate their availability.

The event tree quantification yields a core damage frequency of about 1.1×10^{-6} per year for small LOCA outside the drywell, when it is assumed that the motor operated isolation valves are failed.

Note that no distinction was made between steam and liquid breaks in the case of the small LOCA. The calculated core damage frequency would not change much if a distinction between liquid and steam break were made and apparently the flow out of a steam line break would be smaller after depressurization.

3.3 A large LOCA Outside Drywell (≥6" Break Size)

This case was treated in the BNL SNPS-review(2). However, the assumption in the present study is that HPCI and RWCU isolation valves would fail to close.

Only HPCI lines were treated in Reference 2, and a LOCA frequency of 2.7×10^{-8}/year was obtained. If we postulate that the isolation valves fail upon demand, a LOCA frequency of 3.5×10^{-6}/year is obtained for the 10-in. HPCI line break (see Table 2).

The 6-in. diameter RWCU line has three isolation valves inside the drywell. Only one of them close automatically on sensing line break conditions in the RWCU lines. In Table 3 when no credit is given to these valves a break frequency of 9.6×10^{-6}/yr is obtained as derived in Appendix C of this report. In Reference 2, the three valves were considered (having

different isolation signals and one of them is of a different design), and, it was estimated that their failure upon demand would be less than $2x10^{-4}$/d, and the frequency of the 6in. RWCU line break would be about 10^{-9}/year. Thus, in Ref. 2 it was not further considered because the frequency of interfacing system LOCAs, was calculated to be $2x10^{-6}$/year (see Tables 2 and 3 of this report for results and Ref. 2 for more details).

The interfacing LOCA frequency is also estimated in Reference 2. The results are reproduced in Tables 2 and 3. This LOCA frequency does not change under the specific assumptions of this report.

The total frequency of large LOCA outside the drywell assuming isolation failure, and including interfacing LOCA becomes $1.5x10^{-5}$/year. When this is used with the event tree of Ref. 2 (see Fig. 4), a core damage frequency of $3.0x10^{-6}$/yr is found. The 0.2 factor is the probability of operator failure to control the condensate system pumps' flow to the RPV in the short time available (about 10-15 minutes).

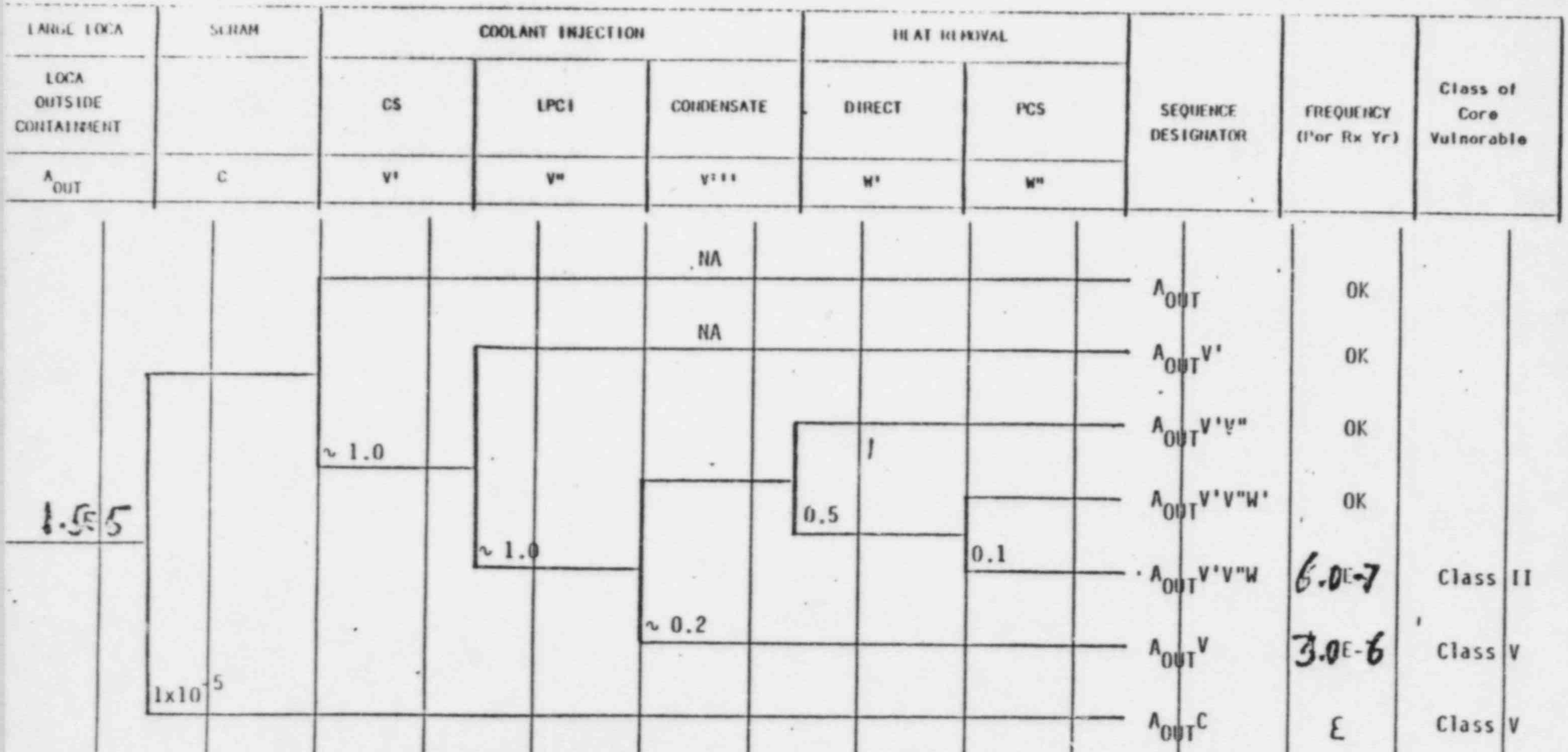
In the case of a large LOCA outside drywell, the discharge to containment is about 1200 lb/s and saturation conditions in the bulk atmosphere of the reactor building are reached within 5 to 10 minutes. The ECCS equipment at elevation 8 would be flooded in about 15 to 25 minutes (the latter number corresponds to 35% flashing). Thus, it is obvious that no isolation is possible, as it was also assumed in the SNPS-PRA and the BNL review for large LOCA discharging saturated water or steam into the reactor building.

This core damage frequency of $3x10^{-6}$/yr is 7 times larger than that given in Reference 2. This is because in Reference 2 the interfacing LOCAs were the dominant contributors. They are dominant when credit to isolation valve closure is considered.

3.4 A Medium LOCA Outside Drywell (2"< Ø < 4")

3.4.1 Accident Conditions Alarms and Operator Response

The most dominant case of the medium LOCA is the 3-in. RWCU line break as shown in Table 2. The frequency of a RCIC 4-in. line break is small compared to the total medium LOCA frequency of $1.6x10^{-3}$/yr; the RWCU 4-in. line break



~~Table 6B-1~~ Figure-4: Event Tree Diagram for Sequences Following Large LOCA Outside ~~Containment~~ Drywell

frequency is significant but the sections considered are relatively downstream and estimated to be 1/4 of the total RWCU break frequency, whereas the other 3/4 are for 3-in. line break or less. Thus, our discussion in this section refers to a 3-in. RWCU line break.

The RWCU is located at elevation 112 ft to 150 ft. At 150 ft the demineralizers are located, which process water at low pressure and at about 125°F and, therefore, not considered. Thus, the break location of significance can occur at the 112 ft or 126 ft elevations. On these elevations, the line are enclosed within concrete shields providing physical separation from all safety related equipment (see App. 3C of the SNPS-FSAR).

Table 4 present the approximate temperatures in the reactor building following a RWCU 4-in. line break at elevation 112 ft in the RWCU pumps room. It is estimated that about 10 times the amount discharged in that case, i.e. 500,000 lb, would result in saturation conditions in the reactor building. This will take about 20 minutes if the flow rate of Table 3 (400 lb/s) is assumed. It apparently will take longer because of the decrease expected in the break flow due to depressurization after a few minutes (up to 10 minutes).

It is expected that the blowdown from the break will cause immediate MSIV closure and loss of the feedwater system. In about 10 minutes or less, the temperature at elevation 8 will reach 155°F and trip the RCIC and HPCI, which started a few minutes before that on low level (L2). Therefore, in this case, it is immaterial whether the operator depressurizes the RPV, because early automatic ADS actuation is expected for this case.

The water discharged during the first 10 minutes would flash (~35%) and the remainder (about 20,000 gallons) will cascade through the stairwells to elevation 8. Appendix 3C of the SNPS-FSAR considers this effects and states that no safety system would be affected. This accumulation is equivalent to 0.5 ft and will result in flooding alarm in the control room.

The radiation and temperature alarms are expected to be on in many areas of the reactor building. Therefore, it is believed that the situation of LOCA outside drywell and the reactor building adverse conditions and flooding would be recognized with high probability within the first 10 minutes. Earlier recognition of the LOCA and depressurization of the RPV would not change

much the progress of this accident sequence. However, if operators fail to recognize the event and follow the procedures (which call for keeping RPV at low pressure and controlling the injection flow), then the reactor building conditions may severely deteriorate.

The depressurization would apparently happen at about 10 minutes. Then the LPCI, LPCS and condensate pumps, may all inject water to the PRV, and discharge a large amount of hot water through the break. While this hot water would have less enthalpy than the saturated water discharged during the first 10 minutes, it has flooding potential because of its high flow rate. Flooding may occur in an additional 30 minutes if the flow rate to the RPV is not reduced by keeping it at the lowest possible pressure without uncovering the core. This is the operator action specifically required for the case of medium LOCA outside the drywell. In such a case LPCI/LPCS may maintain core cooling for long period and condensate would not be needed until several hours into the accident.

3.4.2 Estimation of Core Damage Frequencies

The estimation of core damage frequency for the case of a medium LOCA outside drywell is shown in the event tree in Figure 5.

The initiating event does not distinguish between water or steam line breaks. They are considered similar because even though the steam discharge through the break is smaller, the impact on containment atmosphere temperature and pressure is about 5 times higher for a steam line break than for the case of a similar size water line break.

In the long run, after the RPV is depressurized, the flow out of a steam break may be significantly smaller if the core is not flooded so that water is discharged through the break. If the water level is kept below level 8 (L8), then the steam flow out of the break is expected to be relatively small. Thus, it may not be sufficient to create flooding sufficient to damage the ECCS equipment.

The liquid line break is therefore the dominating case. Thus, the event tree starts with the medium LOCA frequency from Table 3. The feedwater and RCIC/HPCI are assumed to be unavailable. Depressurization by ADS is

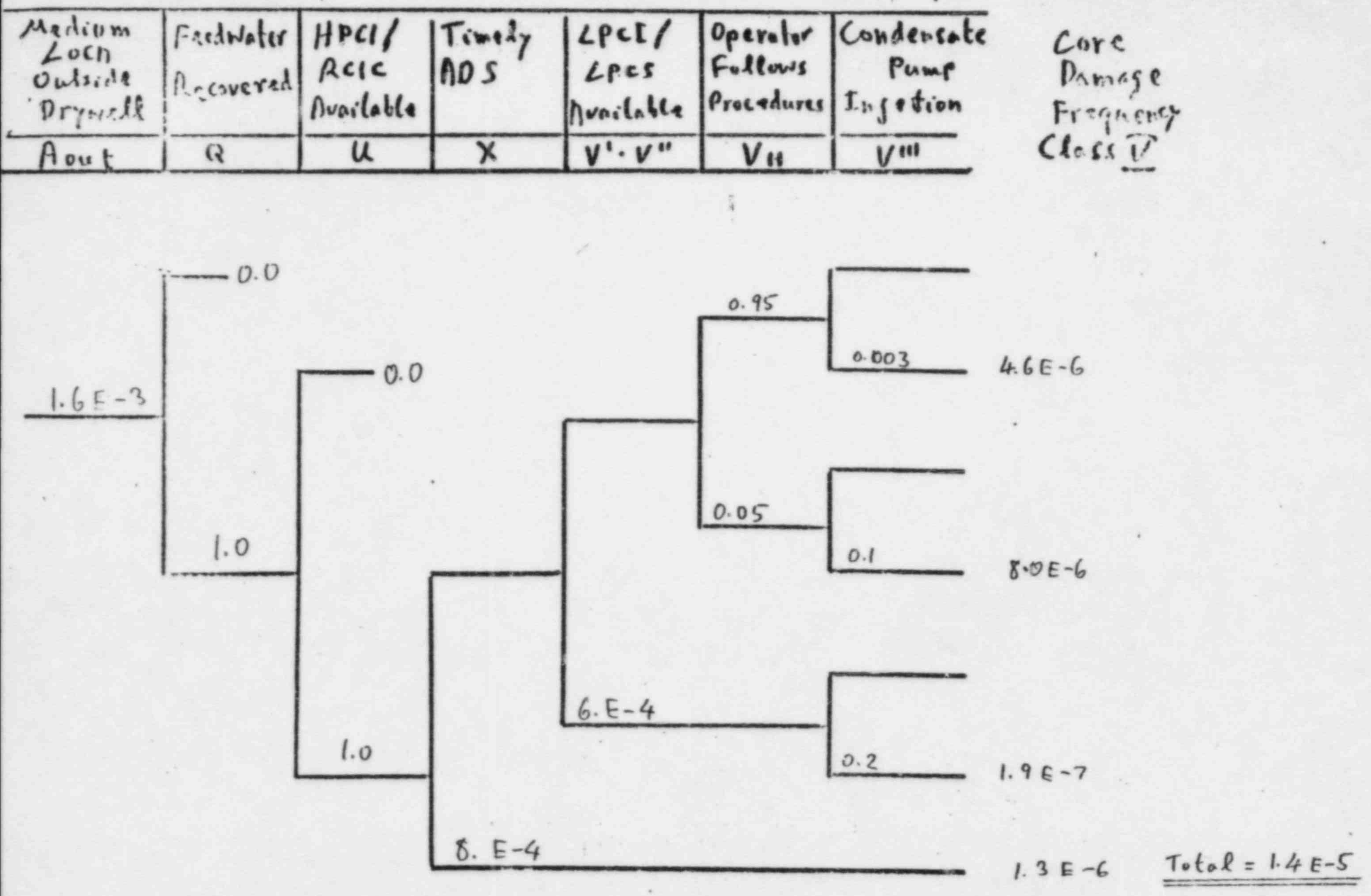


Figure-5: Event Tree Diagram for Sequences Following Medium LOCA Outside Drywell

considered to occur at about 10 minutes into the sequence. The low pressure injection systems will start to flood the core. Therefore, operator action to control the injection flow rate is needed to reduce the impact on the reactor building and gain time before the condensate system would be required. If the operator recognizes the need to control the injection, then the condensate system pumps will also be controlled at a later time with a higher reliability. If the operator fails to control the injection, less time will be available to control the condensate pumps injection because they may be needed at about 10 minutes into the accident.

The values used for the probability of successful operator action are thought to be on the conservative side given the time estimated to be available. Therefore, the core damage frequency for medium LOCA outside drywell may be smaller than $1.4x10^{-5}$ for the case that no credit is given for RWCU isolation valves. On the other hand, the phenomenological assumptions used may not be realistic and may underestimate the break-flow and Reactor Building conditions, so that less time will be available for operator corrective action than assumed above.

4. SUMMARY

The BNL review[2] of SNPS-PRA estimated a core damage frequency of 4.2×10^{-7} for LOCA outside the drywell in the SNPS; this is mainly due to interfacing system LOCAs. In this study, an additional assumption was introduced at NRC request: namely, that isolation valves would be treated as failing to close upon demand. The only exceptions to this assumption are the MSIVs and check valves. The effect of this assumption is shown in Table 5. It is seen that the core damage frequency increased by a factor of almost 50. The leading contribution comes from medium LOCA outside the drywell; in particular, the RWCU 3-in. line break is seen to be the most important (see Table 3).

Table 5 Core Damage Frequencies for Unisolated LOCA Outside Drywell

Initiator	Class V Core Damage Frequency	
	Isolation Valves Assumed to Close on Demand (from BNL Reference 2)	Isolation Valves Assumed to Fail to Close on Demand (from this analysis)
Interfacing LOCA	4.0 E-7	4.0 E-7
Large LOCA Outside Drywell	2.0 E-8	2.5 E-6
Medium LOCA Outside Drywell	--	1.4 E-5
Small Loca Outside Drywell	--	1.1 E-6
Total	4.2 E-7	1.8 E-5

Table 2 provides the information on the most important isolation valves whose failures contribute to the results of Table 5. RWCU isolation valves are the most important. Next, but by far less important, are HPCI and MSL drain isolation valves.

Tables 3 and 5 show that under the assumptions used in this study, the core damage frequency from LOCA outside drywell is dominated by the RWCU medium LOCA breaks. Also, the large LOCA contribution comes mainly from the RWCU system. Therefore, it should be noted that beside the inboard and outboard containment isolation valves, the RWCU also has two additional isolation valves that do not receive an automatic signal to close when a line break occurs and are available for timely remote closure. This action can be taken half an hour after initiation of the accident when the reactor is depressurized, and before the loss of low pressure injection.

5. References

1. "Probabilistic Risk Assessment Shoreham Nuclear Power Station Long Island Lighting Company, Final Report," Science Application, Inc., June 24, 1983.

2. D. Ilberg, K. Shiu, N. Hanan, and E. Anavim, "A Review of the Shoreham Nuclear Power Station Probabilistic Risk Assessment," NUREG/CR-4050, BNL-NUREG-51836, December 1984.

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4. "Generic Safety Evaluation Report Regarding Integrity of BWR Scram System Piping," NUREG-0803, August, 1981.

5. "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors," GE Report NEDO-24708, December 1980.

6. Reactor Safety Study--"An Assessment of Accident Risks in U.S. Commerical Nuclear Power Plants," WASH-1400, NUREG/74-014, October 1975.

7. S.L. Basin and E. T. Burns, "Characteristics of Pipe System Failures in Light Water Reactors," EPRI-NP-438, August 1977.

8. W. H. Hubble and C. F. Miller, "Data Summaries of LERs on Valves at U.S. Commercial Nuclear Power Plants," NUREG/CR-1363, EGG-EA-5125, May 1980.

9. Oconee PRA, A Probabilistic Risk Assessment of Oconee Unit 3, NSAC/60, June 1984.

Appendix A

Pipes and Valves Failure Rates

A.1 Pipe Rupture

The main data sources used for probability of pipe ruptures were the Reactor Safety Study[6] (RSS) and the EPRI-NP-438 report[7]. In the Reactor Safety Study, pipe rupture rates are based on the large amount of data prior to 1973. The EPRI report includes data for an additional two years. Even though it does not change the RSS results on pipe break rates, it provides more insights on the failure mechanisms leading to pipe breaks, mainly vibrations and pressure surges. It also points out that expansion joints and reducers may be at locations more susceptible to breaks. In the BNL study, reducers and valves were considered as rupture locations, in addition to pipe sections.

The SNPS-PRA[1] uses the RSS data for pipe breaks. However, it distinguishes between pipe sections which are "Break Exclusion," i.e., are designed to criteria provided in Appendix 3C of SNPS-FSAR[3], which basically allow for larger design margins and higher quality control of these sections. These increased margins are assumed by SNPS to reduce the failure rate of these sections by a factor of 10. BNL accepted this assumption, and the basic values used in the study are similar to the SNPS-PRA and are summarized in Table A.1 below.

Table A.1

Pipe Rupture Rates

Component	Assessed Range (non break-exclusion pipes)	Computational Median	Computational Mean	
			Break Exclusion	Non-Break Exclusion
Pipes > 3" dia. per section	$3x10^{-12}$ - $3x10^{-9}$/hr	$1x10^{-10}$/hr	$8.6x10^{-11}$/hr	$8.6x10^{-10}$/hr
Pipes ≤ 3" dia. per section	$3x10^{-11}$ - $3x10^{-8}$/hr	$1x10^{-9}$/hr	$8.6x10^{-10}$/hr	$8.6x10^{-9}$/hr

The pipe rupture data of the RSS is applied section by section, where a section is defined (RSS, page III-41) as follows:

> A section is an average length between major discontinuities such as valves, pumps, etc. (approximately 10 to 100 ft). Each section can include several welds, elbows, and flanges.

In this study, piping was also divided into sections where discontinuities were considered to be:

-- Valves
-- Reducers
-- Pumps
-- Heat Exchangers

Appendix C presents the details of the pipings and their division into sections.

A.2 Valve Failure Rates

The main sources used for valve rupture or excessive leakage failure rates were the Reactor Safety Study(6) and NUREG/CR-1363 report(8). The values of the NUREG/CR-1363 evaluation are about a factor of three higher than those in the RSS (see Table A.2 for comparison). However, the NUREG evaluation includes also small leakages such as from packing failure. Similarly, the internal leakage rate of check valves given in the NUREG evaluation includes many small leakages which are just violations of the Technical Specifications limits, and too small to be considered in this study.

The NUREG/CR-1363 evaluation reports about 130 LERs under the title of "External Leakage/Rupture." However, no case of valve external rupture has occurred. SNPS-PRA(1) estimated from this list that a value of 1/18 may be used to modify the RSS rupture rate to better represent severe rupture of valves. This value of 1/18 is also used in this study.

Based on the above, the BNL study essentially adopted the SNPS-PRA approach, i.e.:

Table A.2 Valve Rupture or Excessive Leakage Rates

Component	Source	Failure Mode	Assessed Range $[hr^{-1}]$	Computational Mean	
				Break Exclusion $[hr^{-1}]$	Non-Break Exclusion $[hr^{-1}]$
Check Valves	RSS	Internal Leakage (Severe)	10^{-7} - 10^{-6}	---	$3.8x10^{-7}$
	NUREG/CR-1363	Internal Leakage (all sizes)	---	---	$1x10^{-6}$
Check Valves and Motor Operated Valves	RSS	Rupture	10^{-9} - 10^{-7}	$2.7x10^{-9}$	$2.7x10^{-8}$
	NUREG/CR-1363	External Leakage/ Rupture	---	$7x10^{-9}$	$7x10^{-8}$

Table A.3 Motor Operated Valves Failure Rates

Component	Source	Failure Mode	Assessed Range	Mean Value	Value Used in BNL Study
Motor Operated Valves (MOV)	RSS	Failure to operate (include command)	$3x10^{-4}$ - $3x10^{-3}$/d	$1.3x10^{-3}$/d	---
	NUREG/CR-1363 (for BWRs)	Failure to operate (include command)	---	$8x10^{-3}$/d	$8x10^{-3}$/d
	NUREG/CR-1363 (for BWRs)	Failure to operate (w/o command)	---	$6x10^{-3}$/d	$6x10^{-3}$/d
	Command Failure of Both MOVs (Inboard and Outboard)	Failure of Inboard and Outboard MOVs	---	$2x10^{-3}$/d	$2x10^{-3}$/d
	Oconee PRA(9)	MOV Spurious Opening	---		10^{-3}/y

Table A-4 A Comparison of Frequencies of Loss of Coolant Accidents

Pipe Break Diameter (Inch)	RSS		EPRI-NP-438		SNPS-PRA	This Study: LOCA Outside Drywell
	90% Range	Mean LOCA Frequencies	All Pipes (Mean)	LOCA Sensitive Pipes(*) Mean	Mean LOCA Frequencies	
Small LOCA 1/2" - 2"	$1x10^{-}$[illegible] - $1x10^{-2}$	$2.7x10^{-3}$	$\sim 10^{-2}$	$8x10^{-3}$	$8x10^{-3}$	$5x10^{-3}$
Medium LOCA 2" - 6"	$3x10^{-5}$ - $3x10^{-3}$	$8x10^{-4}$	---	$3x10^{-3}$	$3x10^{-3}$	$1.6x10^{-3}$
Large LOCA $\geq$ 6"	$1x10^{-5}$ - $1x10^{-3}$	$2.7x10^{-4}$	$\sim 1x10^{-3}$	$7x10^{-4}$	$7x10^{-4}$	$3.5x10^{-5}$(**)

(*)It is assumed that 10% of plant piping are LOCA sensitive pipes.(Ref.1)

(**)The large diameter pipes are "break-exclusion" and are assumed to have 1/10 of the RSS rupture rate.

(1) Use of RSS failure rates for valves.

(2) Apply a modifying factor of 1/18 to the RSS valve rupture data.

(3) Distinguish between valves which are in the break exclusion zone and those which are not. A factor of 1/10 is applied to the rupture rates of the break-exclusion valves, similarly to the factor applied to the pipe section they are located on.

To summarize, the value used for valve failure rates were:

check valve internal leakage: $3.8\times10^{-7} \times 8760 = 3.3\times10^{-3}$/year

valve rupture (break exclusion): $2.7\times10^{-9} \times 8760 \times 1/18 = 1.3\times10^{-6}$/year

(non-break exclusion): $2.7\times10^{-8} \times 8760 \times 1/18 = 1.3\times10^{-5}$/year.

For simplification of the analysis, the valve rupture rates were also used with other discontinuities between pipe sections, such as reducers or pumps; this may be a conservative assumption.

In addition to valve rupture and internal leakage, other failure modes of motor-operated valves were needed in this study. The additional failure modes and failure rates used are summarized in Table A.3.

A.3 Comparison with LOCA Frequencies

The analysis in the main part of this report involves a large number of pipe sections and valves. In general, more pipe sections and valves are located outside the drywell. Thus, the frequency of LOCA outside containment should be a large fraction of the plant's LOCA frequencies. Table A-4 compares the results of the LOCA frequencies from this BNL study with the RSS results (table III-6-9 of RSS), the EPRI-NP-438[7] results, and those of the SNPS-PRA.

Table B-1

PROCESS PIPELINES PENETRATING PRIMARY CONTAINMENT

(Numbers in parentheses are keyed to notes on pages B-6 and B-7; signal codes are listed on page B-8).

PRIMARY CONTAINMENT PENETRATIONS	LINES ISOLATED (22)	GDC	NUMBER OF LINES	VALVES PER LINE	NOMINAL PIPE SIZE (IN.)	VALVE LOCATION RELATIVE TO PRIMARY CONTAINMENT	VALVE AND/OR OPERATOR TYPE (6,22)	POWER TO OPEN (5,6)	POWER TO CLOSE (5,6)	ISOLATION SIGNAL	CLOSING TIME (SEC) (10)	NORMAL STATUS (8,9)	REMARKS
1A,B,C,D	Main Steam	55	4	1	24	Inside	AO Globe	Air/AC/DC	Air/Spring	B,C,D,E,P,R,T,RM	3-5	Open	(1)
				1	24	Outside	AO Globe	Air/AC/DC	Air/Spring	B,C,D,E,P,R,T,RM	3-5	Open	(1)
	Main Steam Line Drain and MSIV-Leakage Control System	55	4	1	2	Outside	MO Globe	AC	AC	B,C,D,E,P,R,T,RM,O	4	Open	
		55	4	1	2 1/2	Outside	MO Globe	AC	AC		6	Closed	(19)
2A	Feedwater	55	1	1	18	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
				1	18	Outside	VTC	Flow	Reverse Flow/ Air/Spring	Reverse Flow/F, G,RM	N/A	Open	(11)
2B	Feedwater	55	1	1	18	Inside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
				1	18	Outside	VTC	Flow	Reverse Flow/ Air/Spring	Reverse Flow/F,G, RM	N/A	Open	(11)
3	Main Steam Line Drain	55	1	1	3	Inside	MO Gate	AC	AC	B,C,D,E,P,R,T,RM,O	16	Open	
				1	3	Outside	MO Gate	DC	DC	B,C,D,E,P,R,T,RM,O	16	Open	
4	RWCU Line from RPV	55	1	1	6	Inside	MO Gate	AC	AC	A,J,RM,O	30	Open	
				1	6	Outside	MO Gate	DC	DC	A,J,W,T,RM,O	30	Open	
5	RHR Shutdown Cooling from RPV	55	1	1	20	Inside	MO Gate	AC	AC	A,F,U,RM	23	Closed	
				1	20	Outside	MO Gate	DC	DC	A,F,U,RM	23	Closed	
6A,B	RHR Injection Line to Recirculation System Return	55	2	1	24	Inside	VTC	Flow	Reverse Flow	Reverse Flow	N/A	Closed	(3)
				1	2	Inside	MO Gate	AC	AC	A,RM	19	Closed	
				1	24	Outside	MO Gate	AC	AC	RM	24	Closed	(12)
7A,B	RHR - Containment Spray Drywell	56	2	1	10	Outside	MO Gate	AC	AC	F,G,RM	51	Closed	(2)
				1	10	Outside	MO Angle	AC	AC	F,G,RM	10	Closed	(2)
8A,B	RHR - Containment Spray Suppression Chamber	56	2	1	6	Outside	MO Globe	AC	AC	F,G,RM	71	Closed	(2)
				1	16	Outside	MO Gate	AC	AC	F,G,RM	71	Closed	(2)
				1	16	Outside	MO Globe	AC	AC	F,G,RM	79	Closed	(2)
9A,B,C,D	RHR Pump Suction	56	4	1	20	Outside	MO Gate	AC	AC	RM	106	Open	(13)

Table B-1 (continued)

PROCESS PIPELINES PENETRATING PRIMARY CONTAINMENT

PRIMARY CONTAINMENT PENETRATIONS	LINES ISOLATED (22)	GDC	NUMBER OF LINES	VALVES PER LINE	NOMINAL PIPE SIZE (IN.)	VALVE LOCATION RELATIVE TO PRIMARY CONTAINMENT	VALVE AND/OR OPERATOR TYPE (6,22)	POWER TO OPEN (5,6)	POWER TO CLOSE (5,6)	ISOLATION SIGNAL	CLOSING TIME (SEC) (10)	NORMAL STATUS (8,9)	REMARKS
X-10A	RHR Test Line Return to Suppression Chamber, Suppression Pool Cleanup Return, RHR Steam Condensing Discharge, RHR Minimum Flow, Core Spray Test Line, and Core Spray Minimum Flow	56	1	1	16	Outside	MO Globe	AC	AC	F,G,RM	79	Closed	(2)
			1	2	6	Outside	MO Gate	AC	AC	A,F,RM	31	Closed	
			1	1	4	Outside	MO Gate	AC	AC	F,G,RM	20	Closed	
			1	1	4	Outside	MO Gate	AC	AC	RM	20	Open	(16)
			1	1	10	Outside	MO Globe	AC	AC	F,G,RM	67	Closed	
			1	1	3	Outside	MO Gate	AC	AC	RM	16	Open	(16)
X-10B	RHR Test Line Return to Suppression Chamber, RCIC Minimum Flow, HPCI Minimum Flow, RHR Steam Condensing Discharge, RHR Minimum Flow, Core Spray Test Line, Core Spray Minimum Flow, and Relief Valve Discharge from RHR Supply to RCIC Pump Suction	56	1	1	16	Outside	MO Globe	AC	AC	F,G,RM	79	Closed	(2)
			1	1	2	Outside	MO Globe	DC	DC	RM	18	Closed	(16)
			1	1	4	Outside	MO Globe	DC	DC	RM	29	Closed	(16)
			1	1	4	Outside	MO Gate	AC	AC	F,G,RM	20	Closed	
			1	1	4	Outside	MO Gate	AC	AC	RM	20	Open	(16)
			1	1	10	Outside	MO Globe	AC	AC	F,G,RM	67	Closed	
			1	1	3	Outside	MO Gate	AC	AC	RM	16	Open	(16)
			1	1	2	Outside	Relief Valve	High Differential Pressure	Spring	N/A	N/A	Closed	
X-11	RHR - Head Spray Line to RPV	55	1	1	4	Inside	MO Gate	AC	AC	A,F,U,RM	20	Closed	
				1	4	Outside	MO Globe	DC	DC	A,F,U,RM	13	Closed	
X-12	HPCI Turbine Steam Inlet Line	55	1	1	10	Inside	MO Gate	AC	AC	K,RM	11	Open	(7)
				1	1	Inside	MO Globe	AC	AC	K,RM	12	Open	(7)
				1	10	Outside	MO Gate	DC	DC	K,RM	43	Closed	(7)
				1	1	Outside	MO Globe	DC	DC	K,RM	12	Open	(7)
X-13	HPCI Turbine Exhaust	56	1	1	18	Outside	MO Gate	DC	DC	RM	102	Open	
				2	18	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Closed	
X-14	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
X-15	HPCI Pump Suction	56	1	1	16	Outside	MO Gate	DC	DC	K,RM	71	Closed	
X-16	RCIC Turbine Steam Inlet Line	55	1	1	3	Inside	MO Gate	AC	AC	K,RM	16	Open	(7)
				1	1	Inside	MO Globe	AC	AC	K,RM	12	Open	(7)
				1	3	Outside	MO Gate	DC	DC	K,RM	16	Closed	(7)
				1	1	Outside	MO Globe	DC	DC	K,RM	12	Open	(7)

Table B-1 (continued)

PROCESS PIPELINES PENETRATING PRIMARY CONTAINMENT

PRIMARY CONTAINMENT PENETRATIONS	LINES ISOLATED (22)	GDC	NUMBER OF LINES	VALVES PER LINE	NOMINAL PIPE SIZE (IN.)	VALVE LOCATION RELATIVE TO PRIMARY CONTAINMENT	VALVE AND/OR OPERATOR TYPE (6,22)	POWER TO OPEN (5,6)	POWER TO CLOSE (5,6)	ISOLATION SIGNAL	CLOSING TIME (SEC) (10)	NORMAL STATUS (8,9)	REMARKS
X-17	RCIC Turbine Exhaust	56	1	1 2	8 8	Outside Outside	MO Gate Check	DC Flow	DC Reverse Flow	RM Reverse Flow	38 N/A	Open Closed	(13)
X-18	RCIC Vacuum Pump Discharge	56	1	1 1	2 2	Outside Outside	MO Stop Check Check	Flow/DC Flow	Rev. Flow/DC Reverse Flow	Rev. Flow/RM Reverse Flow	13 N/A	Closed Closed	(13,21)
X-19	RCIC Pump Suction	56	1	1	6	Outside	MO Gate	DC	DC	RM	31	Closed	
X-20A,B	Core Spray Pump Discharge to RPV	55	2	1 1 1	10 2 10	Inside Inside Outside	VTC MO Globe MO Gate	Flow AC AC	Reverse Flow AC AC	Reverse Flow RM RM	N/A 18 43	Closed Closed Closed	(3) (18)
X-21A,B	Core Spray Pump Suction	56	2	1	14	Outside	MO Gate	AC	AC	RM	76	Open	
X-22A,B	RBCLCW to Recirc. Pump and Motor Coolers	57	2	1	4	Outside	MO Gate	AC	AC	RM	23	Open	
X-23A,B	RBCLCW from Recirc. Pump and Motor Coolers	57	2	1	4	Outside	MO Gate	AC	AC	RM	23	Open	
X-24A to H	RBCLCW to Drywell Unit Coolers	56	8	1 1	3 3	Inside Outside	Check MO Gate	Flow AC	Reverse Flow AC	Reverse Flow F,G,Z,RM	N/A 16	Open Open	
X-25A,B	RBCLCW from Drywell Unit Coolers	56	2	1 1	4 4	Inside Outside	MO Gate MO Gate	AC AC	AC AC	F,G,Z,RM F,G,Z,RM	20 20	Open Open	
X-26	Purge Air to Drywell	56	1	1 1	18 18	Inside Outside	AO Butterfly AO Butterfly	AC/Air AC/Air	Spring Spring	F,RM F,RM	5 5	Closed Closed	(17) (17)
X-27	Purge Air from Drywell	56	1	1 1	18 18	Inside Outside	AO Butterfly AO Butterfly	AC/Air AC/Air	Spring Spring	F,RM F,RM	5 5	Closed Closed	(17) (17)

Table B-1 (continued)

PROCESS PIPELINES PENETRATING PRIMARY CONTAINMENT

PRIMARY CONTAINMENT PENETRATIONS	LINES ISOLATED (22)	GDC	NUMBER OF LINES	VALVES PER LINE	NOMINAL PIPE SIZE (IN.)	VALVE LOCATION RELATIVE TO PRIMARY CONTAINMENT	VALVE AND/OR OPERATOR TYPE (6,22)	POWER TO OPEN (5,6)	POWER TO CLOSE (5,6)	ISOLATION SIGNAL	CLOSING TIME (SEC) (10)	NORMAL STATUS (8,9)	REMARKS
X-28	Purge Air to Suppression Chamber	56	1	2	18	Outside	AO Butterfly	AC/Air	Spring	L,RM	5	Closed	(17)
X-29	Purge Air From Suppression Chamber	56	1	2	18	Outside	AO Butterfly	AC/Air	Spring	L,RM	5	Closed	(17)
X-30	Sample Coolant from RPV	55	1	1 1	3/4 3/4	Inside Outside	AO Globe AO Globe	AC/Air AC/Air	Spring Spring	B,C,RM B,C,RM	15 15	Open Open	
X-31	Equipment Drains from Drywell	56	1	2	3	Outside	MO Gate	AC	AC	A,F,RM	16	Open	
X-32	Floor Drains from Drywell	56	1	2	4	Outside	MO Gate	AC	AC	A,F,RM	16	Open	
X-33	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
X-34	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
X-35	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
X-36	Standby Liquid Coolant to RPV	55	1	1 1 2	1 1/2 1 1/2 1 1/2	Inside Outside Outside	Check Check Explosive	Flow Flow AC	Reverse Flow Reverse Flow N/A	Reverse Flow Reverse Flow RM	N/A N/A Instantly	Closed Closed Closed	
X-37A	Nitrogen/Air Purge for TIP	57	1	1	3/8	Outside	Check	Flow	Reverse Flow	Reverse Flow	N/A	Open	
X-37B,C,D	TIP Drive Guide Tubes	57	3	1 1	3/8 3/8	Outside Outside	Ball Explosive Shear	AC N/A	Spring DC	- RM	0.5 Instantly	Closed Open	(14) (14)
X-38	TIP Drive Guide Tubes	57	1	1 1	3/8 3/8	Outside Outside	Ball Explosive Shear	AC N/A	Spring DC	- RM	0.5 Instantly	Closed Open	(14) (14)
X-39A,B	Instrument Air to Suppression Chamber	56	2	1 1	1 1	Outside Outside	Check MO Globe	Flow AC	Reverse Flow AC	Reverse Flow F,G,RM	N/A 5	Open Closed	

Table B-1 (continued)

PROCESS PIPELINES PENETRATING PRIMARY CONTAINMENT

PRIMARY CONTAINMENT PENETRATIONS	LINES ISOLATED (22)	GDC	NUMBER OF LINES	VALVES PER LINE	NOMINAL PIPE SIZE (IN.)	VALVE LOCATION RELATIVE TO PRIMARY CONTAINMENT	VALVE AND/OR OPERATOR TYPE (6,22)	POWER TO OPEN (5,6)	POWER TO CLOSE (5,6)	ISOLATION SIGNAL	CLOSING TIME (SEC) (10)	NORMAL STATUS (8,9)	REMARKS
X-40	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
X-41	HPCI Vacuum Breaker	56	1	1 2	2 18	Outside Outside	MO Globe Check	DC Flow	DC Reverse Flow	F and X, RM Reverse Flow	13 N/A	Open Closed	
X-42	RCIC Vacuum Breaker	56	1	1 2	1 1/2 8	Outside Outside	MO Globe Check	DC Flow	DC Reverse Flow	F and X, RM Reverse Flow	16 N/A	Open Closed	
X-43	RHR Relief Valve Discharge Vacuum Breaker, RHR Heat Exchanger Vent, RHR Heat Exchanger, and HPCI Steam Supply to RHR Heat Exchanger	56	1 2 2 2	N/A 2 1 1	N/A 1 1 6	N/A Outside Outside Outside	N/A MO Globe Relief Valve Relief Valve	N/A AC High Pressure High Pressure	N/A AC Spring Spring	N/A RM N/A N/A	N/A 10 N/A N/A	N/A Closed Closed Closed	
X-44	Containment Atmospheric Control from Suppression Chamber, and Drywell Floor Seal Pressurization	56 57	1 1	1 1 1	6 4 1/2	Outside Outside Outside	MO Gate MO Gate MO Globe	AC AC AC	AC AC AC	RM RM RM	31 20 6	Closed Closed Open	
X-45	Containment Atmospheric Control from Suppression Chamber, and Drywell Floor Seal Pressurization	56 57	1 1	1 1 1	6 4 1/2	Outside Outside Outside	MO Gate MO Gate MO Globe	AC AC AC	AC AC AC	RM RM RM	31 20 6	Closed Closed Open	
X-46	Containment Atmospheric Control from Drywell	56	1	1 1	6 6	Inside Outside	MO Gate MO Gate	AC AC	AC AC	RM RM	31 16	Closed Closed	
X-47	Containment Atmospheric Control from Drywell	56	1	1 1	6 6	Inside Outside	MO Gate MO Gate	AC AC	AC AC	RM RM	31 16	Closed Closed	
	CRD Insert and Withdraw Lines	55	137 137	1 1	3/4 1	Outside Outside	Globe Globe	Manual Manual	Manual Manual	N/A N/A	N/A N/A	Open Open	(20) (20)
XS-1	Spare	-	-	-	-	-		-	-	-	-	-	(15)
XS-2	Spare	-	-	-	-	-		-	-	-	-	-	(15)

Table B-1 (continued)

PROCESS PIPELINES PENETRATING PRIMARY CONTAINMENT

PRIMARY CONTAINMENT PENETRATIONS	LINES ISOLATED (22)	GDC	NUMBER OF LINES	VALVES PER LINE	NOMINAL PIPE SIZE (IN.)	VALVE LOCATION RELATIVE TO PRIMARY CONTAINMENT	VALVE AND/OR OPERATOR TYPE (6,22)	POWER TO OPEN (5,6)	POWER TO CLOSE (5,6)	ISOLATION SIGNAL	CLOSING TIME (SEC) (10)	NORMAL STATUS (8,9)	REMARKS
X5-3	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
X5-4	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
X5-5	HPCI Steam Supply to RHR Heat Exchanger, RHR Heat Exchanger Vent, and RHR Heat Exchanger	56	2 2 2	1 2 1	6 1 1	Outside Outside Outside	Relief Valve MO Globe Relief Valve	High Pressure AC High Pressure	Spring AC Spring	N/A RM N/A	N/A 10 N/A	Closed Closed Closed	
X5-6	Suppression Pool Cleanup/ Pump Down	56	1	2	10	Outside	MO Gate	AC	AC	A,F,RM	51	Closed	
X5-7	Containment Atmospheric Control to Suppression Chamber	56	1	2	6	Outside	MO Gate	AC	AC	RM	32	Closed	
X5-8	Containment Atmospheric Control to Suppression Chamber	56	1	2	6	Outside	MO Gate	AC	AC	RM	32	Closed	
X5-9	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
X5-10	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
X5-11	Drywell Service Air	56	1	1 1	1 1/2 1 1/2	Inside Outside	Check Gate	Flow Manual	Reverse Flow Manual	Reverse Flow N/A	N/A N/A	Closed Locked Closed	(4) (4)
X5-12	Containment Drywell Radiation Monitoring Subsystem	56	1	1 1	1 1/2 1 1/2	Inside Outside	MO Globe MO Globe	AC AC	AC AC	F,G,RM F,G,RM	14 14	Open Open	
X5-13	Containment Drywell Radiation Monitoring Subsystem	56	1	1 1	1 1	Inside Outside	MO Globe MO Globe	AC AC	AC AC	F,G,RM F,G,RM	14 14	Open Open	
X5-14	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
X5-15	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)

Table B-1 (continued)

PROCESS PIPELINES PENETRATING PRIMARY CONTAINMENT

PRIMARY CONTAINMENT PENETRATIONS	LINES ISOLATED (22)	GDC	NUMBER OF LINES	VALVES PER LINE	NOMINAL PIPE SIZE (IN.)	VALVE LOCATION RELATIVE TO PRIMARY CONTAINMENT	VALVE AND/OR OPERATOR TYPE (6,22)	POWER TO OPEN (5,6)	POWER TO CLOSE (5,6)	ISOLATION SIGNAL	CLOSING TIME (SEC) (10)	NORMAL STATUS (8,9)	REMARKS
XS-16	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
XS-17	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
XS-18	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
XS-19	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
XS-20	Containment Atmospheric Control to Drywell	56	1	1 1	6 6	Inside Outside	MO Gate MO Gate	AC AC	AC AC	RM RM	32 32	Closed Closed	
XS-21	Containment Atmospheric Control to Drywell	56	1	1 1	6 6	Inside Outside	MO Gate MO Gate	AC AC	AC AC	RM RM	32 32	Closed Closed	
XS-22	Containment Vent to RBHVS	56	1	1 1	6 6	Inside Outside	AO Butterfly AO Butterfly	AC/Air AC/Air	Spring Spring	L,RM L,RM	5 5	Closed Closed	(17) (17)
XS-23	Spare (Reserved for RPV Internal Inspection)	-	-	-	-	-	-	-	-	-	-	-	(15)
XS-24	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
XS-25	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
XS-26	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
XS-27	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
XS-28	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
XS-29	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
XS-30	Spare	-	-	-	-	-	-	-	-	-	-	-	(15)
H-7	Instrument Air to Drywell	56	1	1 1	1 1/2 1 1/2	Inside Outside	Check MO Globe	Flow AC	Reverse Flow AC	Reverse Flow RM	N/A 7.5	Open Open	
H-5	Instrument Air to Drywell	56	1	1 1	1 1/2 1 1/2	Inside Outside	Check MO Globe	Flow AC	Reverse Flow AC	Reverse Flow RM	N/A 7.5	Open Open	
I-10	Recirc. Pump Seal Injection	55	1	1 1	3/4 3/4	Inside Outside	Check Check	Flow Flow	Reverse Flow Reverse Flow	Reverse Flow Reverse Flow	N/A N/A	Open Open	
I-11	Recirc. Pump Seal Injection	55	1	1 1	3/4 3/4	Inside Outside	Check Check	Flow Flow	Reverse Flow Reverse Flow	Reverse Flow Reverse Flow	N/A N/A	Open Open	

Table B-1 (continued)

PROCESS PIPELINES PENETRATING PRIMARY CONTAINMENT

These notes are keyed by number to correspond to numbers in parentheses.

1. Main steam isolation valves require that both solenoid pilots be deenergized to close valves. Accumulator air pressure plus spring set together close valves when both pilots are deenergized. Voltage failure at only one pilot will not cause valve closure. The valves are set to fully close in less than 5 seconds.

2. Containment spray to drywell and suppression chamber and RHR test line return to suppression chamber isolation valves will have the capability to be manually reopened after automatic closure. This setup will permit containment spray for high drywell pressure conditions and/or suppression water cooling. When automatic signals are not present, these valves may be opened for test or operating convenience.

3. Testable check valves are designed for remote opening with zero differential pressure across the valve seat. The valves will close on reverse flow even though the test switches may be positioned for open. The valves will open when pump discharge pressure exceeds reactor pressure even though the test switch may be positioned for close.

4. This line is only needed during maintenance. Service air supply is disconnected during plant operation by administrative control.

5. AC motor operated valves required for isolation functions are powered from the emergency AC power buses. DC operated isolation valves are powered from the station batteries.

6. All motor operated isolation valves will remain in the last position upon failure of valve power. All air-operated isolation valves will close upon air failure.

7. Signal B opens, signal K overrides to close.

8. Power operated valve can be opened or closed by remote manual switch for operating convenience during any mode of reactor operation except when automatic signal is present (see Note 2).

9. Normal status position of valve (open or closed) is the position during normal power operation of the reactor.

10. The specified closure rates are as required for containment isolation only.

11. Special air testable check valves with a positive closing feature are designed for remote testing during normal operation to assure mechanical operability of the valve disc. The remote testing feature will cause only a partial movement of the disc into the flow stream, with only a minor effect on flow. Upon receipt of an isolation signal, the actuator spring force will either cause a slight reduction in flow when the feedwater system is available or cause the valve to close, providing a positive closure differential pressure on the seated disc, when the feedwater flow is not available.

12. This valve will open when both a low reactor pressure vessel pressure and an accident signal are present.

13. The motor operator of this valve is key locked open during normal operating conditions.

14. Traversing In-Core Probe (TIP) Systems:

    When the TIP system cable is inserted, the ball valve of the selected tube opens automatically so that the probe and cable may advance. A maximum of four valves may be opened at any one time to conduct the calibration, and any one guide tube is used, at most, a few hours per year.

    If closure of the line is required during calibration, as indicated by a containment isolation signal, the cable is automatically retracted and the ball valve closes automatically after completion of cable withdrawal. To ensure isolation capability, if a TIP cable fails to withdraw or a ball valve fails to close, an explosive, shear valve is installed in each line. Upon receipt of a remote manual signal, this explosive valve will shear the TIP cable and seal the guide tube.

15. All unused penetrations (designated "Spare") are capped and seal welded.

16. Valve will close on system high flow.

17. Isolation signals A or F will initiate the reactor building standby ventilation system which in turn isolates the purge air isolation valves.

18. This valve will open when both a low differential pressure across the valve and an accident signal are present.

19. Pressure sensors and sensing steam line pressure are used for interlock control to prevent inadvertent valve opening at high steam line pressures (above 35 psig).

Table B-1 (continued)

PROCESS PIPELINES PENETRATING PRIMARY CONTAINMENT

Notes (Continued)

20. Control Rod Drive (CRD) Insert and Withdraw Lines:

    Criteria 55 concerns those lines of the reactor coolant pressure boundary penetrating the primary reactor containment. The CRD insert and withdraw lines are not part of the reactor coolant pressure boundary. The classification of the insert and withdraw lines is Quality Group B, and therefore designed in accordance with ASME Section III, Class 2. The basis to which the CRD lines are designed is commensurate with the safety importance of isolating these lines. Since these lines are vital to the scram function, their operability is of upmost concern.

    In the design of this system, it has been accepted practice to omit automatic valves for isolation purposes as this introduces a possible failure mechanism. As a means of providing positive actuation, manual shutoff valves are used. In the event of a break on these lines, the manual valves may be closed to ensure isolation. In addition, a ball check valve located in the insert line inside the CRD is designed to automatically seal this line in the event of a break.

21. This MO stop check valve is normally in a closed position due to its check valve feature, but its MO is in the open position. The MO provides a backup to close the valve to provide additional high leak tight integrity.

22. Abbreviations used in table:

    AO - Air Operated

    MO - Motor Operated

    VTC - Pneumatic Testable Check Valve

    RHR - Residual Heat Removal System

    RPV - Reactor Pressure Vessel

    RCIC - Reactor Core Isolation Cooling System

    RWCU - Reactor Water Cleanup

    HPCI - High Pressure Coolant Injection

    GDC - General Design Criterion

    RBCLCW - Reactor Building Closed Loop Cooling Water

    TIP - Transversing Incore Probe

    CRD - Control Rod Drive

    MSIV Main Steam Isolation Valve

Table B-1 (continued)

PROCESS PIPELINES PENETRATING PRIMARY CONTAINMENT

ISOLATION SIGNAL NOTES

SIGNAL	DESCRIPTION
A*	Reactor vessel low water level 3 - (A scram will occur at this level)
B*	Reactor vessel low water level 2 - (The reactor core isolation cooling system and the high pressure coolant injection system will be initiated at this level, and recirculation pumps are tripped)
C*	High radiation - main steam line
D*	Line break - main steam line (high steam flow)
E*	Line break - main steam line (steam line tunnel high temperature)
F*	High drywell pressure
G	Reactor vessel low water level 1 - (The core spray systems and the low pressure core injection mode of RHR systems will be initiated at this level)
J*	Line break in reactor water cleanup system - high space temperature, high differential flow, high differential temperature
K*	Line break in steam line to/from turbine (high steam line space temperature, high steam flow, low steam line pressure or high turbine exhaust diaphragm pressure)
L	Reactor building standby ventilation system initiation
M	High radiation signal downstream of primary containment purge filter train
O	High ambient temperature in main steam tunnel penetration area (MSTPA)
P*	Low main steam line pressure at inlet to turbine (RUN mode only)
R	Low condenser vacuum
T	High temperature in Turbine Building
U	High reactor vessel pressure
W*	High temperature at outlet of cleanup system nonregenerative heat exchanger
X	Low steam pressure
Y	Standby liquid control system actuated
Z	Low level in RBCLCW head tank
RM*	Remote manual switch from main control room

* These are the isolation functions of the primary containment and reactor vessel isolation control system; other functions are given for information only.

Appendix C

Identification of Pipe Sections and Discontinuities for Break Frequency Estimation

Main Steam Lines

All sections of the four lines in the Reactor Building are break exclusion. Two sections are considered: one to the outboard MSIV; one from the outboard MSIV.

Main Feedwater Lines

All sections of the two lines in the Reactor Building are break exclusion. They include check valve inboard and testable check valve outboard. Their failure rate is assumed to be similar.

High Pressure Coolant Injection (HPCI)

Reference: FSAR and LILCO drawings no. M10121-17 and M10122-14.

Description: 10 in.: one section and valve to the outboard valve (MOV-041). Break exclusion. Under normal RPV pressure conditions because inboard valve is open.

10 in.: six nonbreak exclusion sections (4 challenges per year of 24 hrs each are assumed in these sections):

- To reducer
- Branch SHP-171 + valve MOV-049
- Reducer/valve F001
- To steam turbine stop valve
- To turbine admission valve
- The turbine assumed to be equivalent to one section

1 in.: two bypass sections and a valve. Six sections downstream to the RCIC/HPCI drain line. Two branches. All nonbreak exclusion. Normally open.

RCIC

Reference: FSAR and LILCO Drawings No. M10116-16 and M10117-13

Description: 4 in.: open MOV inside drywell to the outboard MOV. It has a bypass line of 1 in., normally open. Break-exclusion. six sections and discontinuities:

3 in.:

- to 3x6 reducer
- to drain pot and 3x6 reducer
- to steam turbine stop-valve
- to steam admission valve
- to steam turbine governing valve
- the turbine treated as one section.

Following the turbine, low energy assumed.

1 in.:

- Bypass is 2 sections
- Drain lines from drain pot to RCIC/HPCI drain line are considered six sections.

Branches: two or more 3/4 in. branches.

Quantification:

4 in.: [8.6(-11) + 1.5(-10)] * 8760 = 2.1(-6)

3 in.: [8.6(-9) + 1.5(-9)] * 6 * 4 (times per year) x 24 (hrs) = 5.8(-6)

1 in.: [6 + 6 + 2] * [8.6(-9) + 1.5(-9)] * 8760 = 1.2(-3).

Reactor Water Cleanup System (RWCU) Supply Line

Reference: FSAR Figures 3C-4-15A,B,C and Figure 5.5.8-1,2,3

Description: 6 in.: One break exclusion section and valve

6 in.: One section nonbreak exclusion to reducer

3 in.: Two lines (having three sections each), two valves each and one pump each.

2 in.: Two lines with section and reducer/check valve.

3 in.: Two line with section, valve, section, reducer

4 in.: One section and two valves. One of these valves is normally closed. Another line with section, HX, section HX. The heat exchanger (HX) considered as one section in our approximation.

Beyond the second heat exchanger, temperature is less than 125°F and not considered to be high energy, and will not result in a large environmental effect. The high energy part of the RWCU on the return line from the regenerative HX to the feedwater line is not considered a significant additional contributor, compared to the part already included.

Standby Liquid Control (SLC)

Reference: Figure 4.2.3-11 of FSAR and LILCO Drawing M10115-16

Description: 1-1/2 in.-line; 2 check vlaves one inside and the othe routside drywell designated F006 and F007 respectively.

Sections: up to CV-F006 is break exclusion section; from F006 to the two normally closed explosive valves is nonbreak exclusion section.

Branches: four 3/4 in.-branches from the main 1-1/2 in.-line.

Quantification: [8.6(-10) + 1.5(-10)] * (8760/2) * 3.3(-3) = 1.5(-8)
$[8.6(-9) + 2 \times 1.5(-9)] * (8760/2) * [3.3(-3)]^2 = 1.0(-9)$

Control Rod Drive

Reference: NUREG-0803

The contribution comes from the Scram Discharge Header rupture as explained in NUREG-0803. The value of the rupture frequency of 10^{-4} is derived from that report.

Recirculation Pump Seal Injection

Reference: FSAR

Description: Two 3/4 in.-lines; 2 check valves one inside and the other outside drywell. Apparently, it is not break exclusion pipe.

Quantification: Similar to SLC but not break exclusion -- 2.0(-7)

Sample Coolant From RPV

Reference: FSAR

Description: 3/4 in.-line; one normally open inboard air-operated globe valve. One normally open outboard air operated globe valve. Assumed to have one line, two sections, and two valves in reactor building. Nonbreak exclusion.

Quantification: 2 * [8.6(-9) + 1.5(-9)] * 8760 = 1.8(-4)

Reactor Post Accident Sampling System (PASS)

Reference: FSAR

Description: 3/4-in. line. One manually operated globe valve outboard, normally open. Two solenoid operated globe valves, normally closed, downstream.

Quantification: same as above.

TIP Drive Guide Tubes

Reference: FSAR

Description: four lines of 3/8 in. The tubes are normally with nitrogen. In order to cause LOCA, all the following must occur:

- One tube rupture inside RPV
- Nitrogen system alarm fail to alert the operator
- Operator error in using the system, failing to operate the shear valve. (The TIP is assumed to be used 4 times per year.)

Quantification: $4 * 4x10^{-2} \times 10^{-1} \times 2.5 \times 10^{-3} * 10^{-1} - 4 \times 10^{-6}$

Other 3/4-in. Lines

It is estimated that there are about 20 sections of 3/4 in., test lines, and other lines branching from the systems listed in this table. Many of them are in the RWCU and are potential "liquid" break location. Other branch out of HPCI, RCIC, and other steam lines, and are potential "steam" break location.