LIMERICK GENERATING STATION

PROBABILISTIC RISK ASSESSMENT

REVISION 4 PAGE CHANGES

The attached Revision 4 pages, tables, and figures are considered part of a controlled copy of the Limerick Generating Station Probabilistic Risk Assessment. This material should be incorporated into Volume 1 and 2 of the PRA by following the collating instruction below:

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PROBABILISTIC RISK ASSESSMENT

LIMERICK GENERATING STATION

PHILADELPHIA ELECTRIC COMPANY

VOLUME 1

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largest inventory of radioactive material among all the sources in LGS.

Accidents which involve release of radioactivity from the refueling process, shipping casks, or the liquid radwaste would not result in public consequences nearly as serious as accidents involving melting of the fuel in the reactor core or in the SFSP. The melting of the fuel in the SFSP is assumed (as in WASH-1400) to be of sufficiently low probability as to not require detailed examination in the study.

Brookhaven National Laboratory (BNL) has determined, in a study for the Nuclear Regulatory Commission (NRC), that the highest risk to the public from operation of a nuclear power plant is presented by postulated accidents involving the reactor core (3-1). The other smaller sources of radioactivity at a nuclear power plant are found on a generic basis by BNL to make a negligible contribution to the public risk.

The Limerick Probabilistic Risk Assessment has focused on the potential for the release of radioactivity to the public by accidents involving the reactor core, and taking place during power operation.

3.2 ACCIDENT INITIATORS

Event trees provide a logical method for developing and displaying the sequences which may occur during postulated accidents. One of the most important aspects of the event tree technique is that it assists in ensuring that key accident

sequences following the identified initiators are evaluated. Therefore, the ability to establish a comprehensive list of initiators is a necessary part of the PRA. Items considered when identifying accident initiators include the following:

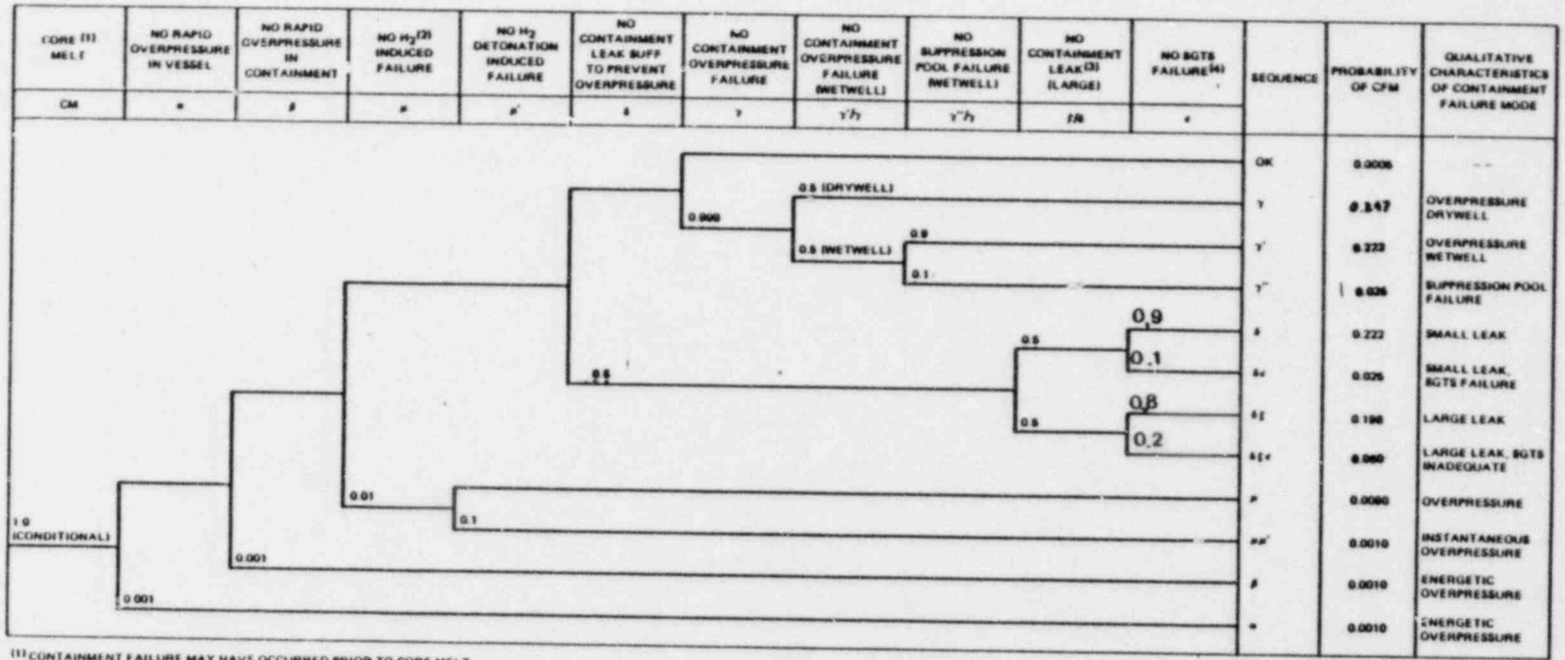
- Previous risk analyses (e.g., WASH-1400, CRBRP, IREP)
- Plant unique features leading to specific initiators
- Operating experience
- Licensing basis accidents.

See Appendix A for further discussion of accident initiators. The initiating events can be separated into three general groups:

1. Transients and manual shutdowns
2. Loss of Coolant Accidents (LOCAs)
3. Anticipated Transients Without Scram (ATWS).

Table 3.2.1 summarizes the transient frequencies used in the Limerick analysis. Evaluation of BWR operating experience data indicates that there are approximately 6.2 transients per year which result in a scram shutdown. As shown in the table, different transients are discriminated by type into five distinct groups:

1. Isolation events (MSIV Closure)
2. Turbine trips
3. Loss of offsite power
4. Inadvertent open relief valves (IORV)
5. Loss of feedwater.



[1] CONTAINMENT FAILURE MAY HAVE OCCURRED PRIOR TO CORE MELT. IN THOSE CASES (CLASS II AND CLASS IV), THE CONTAINMENT FAILURE MODES ARE ONLY USED AS MECHANISMS FOR RELEASE FRACTION DETERMINATION.

[2] ASSUMES THAT H_2 EXPLOSION IN CONTAINMENT CAUSES OVERPRESSURE FAILURE WITH DIRECT PATHWAY TO OUTSIDE ATMOSPHERE.

[3] LEAKAGE AT 2400 VOLUME PERCENT/DAY.

[4] FAILURE STANDBY GAS TREATMENT SYSTEM.

[5] VALUES SHOWN HERE ARE FOR CLASS I, II, AND III. SEE SECTION 3.5.4 FOR DISCUSSION AND VALUES FOR CLASS IV.

Figure 3.4.14 Containment Event Tree for the Mark II Containment

consequences than those calculated for γ , since no pool scrubbing is assumed for this failure mode. The probability of this occurrence is small.

<u>ζ/δ -- Large Leak</u>. The size of the leak from the primary containment is important in determining the radioactive releases to the environment. In this analysis any leak greater than 2400% vol/day is assumed to double the unavailability of SGTS.

For the purposes of the LGS study the assumption is made that for Class IV event sequences the containment pressurization is sufficiently rapid to result in some form of overpressure rupture; that is, leaks (i.e., low release fraction sequences) are precluded in the Class IV analysis. This assumption is the best estimate of the containment response under these conditions.

<u>ϵ -- Standby Gas Treatment System -- Secondary Containment</u>. This event represents the capability of the SGTS to process the effluent from the primary containment which enter the secondary containment. This event also includes sequences where containment leakage occurs. Success of this system is dependent on availability of electric power. Failure of the SGTS is dependent on the primary containment leakage rate.

Table 3.5.3

SUMMARY OF SEQUENCE FREQUENCIES (PER REACTOR YEAR) BY CONTAINMENT FAILURE MODE FOR DOMINANT SEQUENCES OF THE CLASS I VARIETY

DOMINANT SEQUENCES	CONTAINMENT FAILURE MODES						
	α 0.001	β,μ' 0.002	γ,μ 0.256	γ' 0.222	γ'' 0.025	$\zeta\epsilon,\delta\epsilon$ 0.078	ζ,δ 0.42
T_TQUV	$3.0x10^{-11}$	$5.9x10^{-11}$	$7.6x10^{-9}$	$6.6x10^{-9}$	$7.4x10^{-10}$	$2.3x10^{-9}$	$1.2x10^{-8}$
T_TQUX	$7.7x10^{-10}$	$1.5x10^{-9}$	$2.0x10^{-7}$	$1.7x10^{-7}$	$1.9x10^{-8}$	$6.0x10^{-8}$	$3.2x10^{-7}$
T_TPQUX	$7.8x10^{-12}$	$1.6x10^{-11}$	$2.0x10^{-9}$	$1.7x10^{-9}$	$1.9x10^{-10}$	$6.1x10^{-10}$	$3.3x10^{-9}$
T_MQUV	$8.4x10^{-12}$	$1.7x10^{-11}$	$2.2x10^{-9}$	$1.9x10^{-9}$	$2.1x10^{-10}$	$6.6x10^{-10}$	$3.5x10^{-9}$
T_MQUX	$2.2x10^{-10}$	$4.4x10^{-10}$	$5.6x10^{-8}$	$4.9x10^{-8}$	$5.5x10^{-9}$	$1.7x10^{-8}$	$9.2x10^{-8}$
T_FQUV	$1.4x10^{-10}$	$2.8x10^{-10}$	$3.6x10^{-8}$	$3.1x10^{-8}$	$3.5x10^{-9}$	$1.1x10^{-8}$	$5.8x10^{-8}$
T_FQUX	$3.6x10^{-9}$	$7.3x10^{-9}$	$9.3x10^{-7}$	$8.1x10^{-7}$	$9.1x10^{-8}$	$2.8x10^{-7}$	$1.5x10^{-6}$
T_FPQUV	$1.4x10^{-12}$	$2.8x10^{-12}$	$3.6x10^{-10}$	$3.1x10^{-10}$	$3.5x10^{-11}$	$1.1x10^{-10}$	$5.9x10^{-10}$
T_FPQUX	$3.7x10^{-11}$	$7.3x10^{-11}$	$9.4x10^{-9}$	$8.1x10^{-9}$	$9.1x10^{-10}$	$2.9x10^{-9}$	$1.5x10^{-8}$
T_EUV	$5.9x10^{-9}$	$1.2x10^{-8}$	$1.5x10^{-6}$	$1.3x10^{-6}$	$1.5x10^{-7}$	$4.6x10^{-7}$	$2.5x10^{-6}$
T_EUX	$6.9x10^{-10}$	$1.4x10^{-9}$	$1.8x10^{-7}$	$1.5x10^{-7}$	$1.7x10^{-8}$	$5.4x10^{-8}$	$2.9x10^{-7}$
T_EPUV	$5.8x10^{-11}$	$1.2x10^{-10}$	$1.5x10^{-8}$	$1.3x10^{-8}$	$1.4x10^{-9}$	$4.5x10^{-9}$	$2.4x10^{-8}$
T_EPUX	$5.8x10^{-12}$	$1.2x10^{-11}$	$1.5x10^{-9}$	$1.3x10^{-9}$	$1.4x10^{-10}$	$4.5x10^{-10}$	$2.4x10^{-9}$
T_IUV	$2.6x10^{-11}$	$5.2x10^{-11}$	$6.7x10^{-9}$	$5.8x10^{-9}$	$6.5x10^{-10}$	$2.0x10^{-9}$	$1.1x10^{-8}$
T_IUX	$6.8x10^{-10}$	$1.4x10^{-9}$	$1.7x10^{-7}$	$1.5x10^{-7}$	$1.7x10^{-8}$	$5.3x10^{-8}$	$2.9x10^{-7}$
$T_IC'UV$	$5.4x10^{-12}$	$1.1x10^{-11}$	$1.4x10^{-9}$	$1.2x10^{-9}$	$1.3x10^{-10}$	$4.2x10^{-10}$	$2.3x10^{-9}$
$T_IC'UX$	$1.4x10^{-10}$	$2.8x10^{-10}$	$3.6x10^{-8}$	$3.1x10^{-8}$	$3.5x10^{-9}$	$1.1x10^{-8}$	$5.9x10^{-8}$
S_1QUV	$1.1x10^{-11}$	$2.2x10^{-11}$	$2.8x10^{-9}$	$2.4x10^{-9}$	$2.7x10^{-10}$	$8.4x10^{-10}$	$4.5x10^{-9}$
S_1QUX	$2.1x10^{-12}$	$4.2x10^{-12}$	$5.4x10^{-10}$	$4.7x10^{-10}$	$5.3x10^{-11}$	$1.6x10^{-10}$	$8.8x10^{-10}$
S_2UX	$1.6x10^{-11}$	$3.2x10^{-11}$	$4.1x10^{-9}$	$3.6x10^{-9}$	$4.0x10^{-10}$	$1.2x10^{-9}$	$6.7x10^{-9}$
APPROXIMATE TOTAL PROBABILITY FOR CLASS I SEQUENCES	$1.2x10^{-8}$	$2.5x10^{-8}$	$3.2x10^{-6}$	$2.8x10^{-6}$	$3.1x10^{-7}$	$9.7x10^{-7}$	$5.2x10^{-6}$

The remainder of this section discusses the accident sequence calculations in various ways to provide a comparison with previous BWR probabilistic risk assessments. A summary chart of all the identified accident sequences within each class is given in Figure 3.5.2. This histogram provides a visual display of the claculated relative frequency of potential degraded core conditions for each of the classes of accident sequences. Also displayed on Figure 3.5.2 for comparison is the total frequency of postulated core melt taken from WASH-1400 (all values expressed as mean values).

Table 3.5.4

SUMMARY OF SEQUENCE FREQUENCIES (PER REACTOR YEAR) BY CONTAINMENT FAILURE MODE DOMINANT SEQUENCES OF THE CLASS II VARIETY

DOMINANT SEQUENCES	CONTAINMENT FAILURE MODES						
	α 0.001	β,μ 0.002	γ,μ 0.256	γ' 0.222	γ'' 0.025	$\zeta\epsilon,\delta\epsilon$ 0.078	ζ,δ 0.42
$T_TQW(Q)$	2.7×10^{-12}	5.5×10^{-12}	7.0×10^{-10}	6.1×10^{-10}	6.9×10^{-11}	2.1×10^{-10}	1.2×10^{-9}
$T_TPW(P)$	3.9×10^{-10}	7.7×10^{-10}	9.9×10^{-8}	8.6×10^{-8}	9.6×10^{-9}	3.0×10^{-8}	1.6×10^{-7}
$T_TPQW(P_Q)$	7.8×10^{-12}	1.6×10^{-11}	2.0×10^{-9}	1.7×10^{-9}	2.0×10^{-10}	6.1×10^{-10}	3.3×10^{-9}
$T_FQW(Q)$	1.6×10^{-10}	3.1×10^{-10}	4.0×10^{-8}	3.4×10^{-8}	3.9×10^{-9}	1.2×10^{-8}	6.5×10^{-8}
$T_FPW(P)$	1.4×10^{-10}	2.8×10^{-10}	3.6×10^{-8}	3.1×10^{-8}	3.5×10^{-9}	1.1×10^{-8}	5.8×10^{-8}
$T_FPQW(PQ)$	3.7×10^{-11}	7.4×10^{-11}	9.4×10^{-9}	8.2×10^{-9}	9.2×10^{-10}	2.9×10^{-9}	1.5×10^{-8}
T_EW	1.6×10^{-11}	3.1×10^{-11}	4.0×10^{-9}	3.5×10^{-9}	3.9×10^{-10}	1.2×10^{-9}	6.6×10^{-9}
$T_EPW(P)$	2.8×10^{-11}	5.7×10^{-11}	7.3×10^{-9}	6.3×10^{-9}	7.1×10^{-10}	2.2×10^{-9}	1.2×10^{-8}
T_IW	6.8×10^{-11}	1.4×10^{-10}	1.7×10^{-8}	1.5×10^{-8}	1.7×10^{-9}	5.3×10^{-9}	2.9×10^{-8}
$T_IC'W(C')$	5.0×10^{-11}	1.0×10^{-10}	1.3×10^{-8}	1.1×10^{-8}	1.3×10^{-9}	3.9×10^{-9}	2.1×10^{-8}
$T_IC'UW(C')$	5.6×10^{-12}	1.1×10^{-11}	1.4×10^{-9}	1.2×10^{-9}	1.4×10^{-10}	4.4×10^{-10}	2.3×10^{-9}
AJ	6.4×10^{-11}	1.3×10^{-10}	1.6×10^{-8}	1.4×10^{-8}	1.6×10^{-9}	5.0×10^{-9}	2.7×10^{-8}
$S_1QW(Q)$	1.0×10^{-12}	2.1×10^{-12}	2.7×10^{-10}	2.3×10^{-10}	2.6×10^{-11}	8.1×10^{-11}	4.4×10^{-10}
APPROXIMATE TOTAL PROBABILITY FOR CLASS II SEQUENCES	9.6×10^{-10}	1.9×10^{-9}	2.5×10^{-7}	2.1×10^{-7}	2.4×10^{-8}	7.5×10^{-8}	4.0×10^{-7}

WASH-1400 basically used one class of accident sequence and five containment failure modes to represent BWRs. Therefore, for the purposes of estimating the total calculated frequency of a potentially degraded core condition, the Limerick classes should be summed and compared with the value from WASH-1400. The Limerick evaluation produces a total estimate of degraded core conditions smaller than WASH-1400 (see also Figure 3.5.4). Figure 3.5.2 indicates that the events are all of relatively low probability.

Table 3.5.5

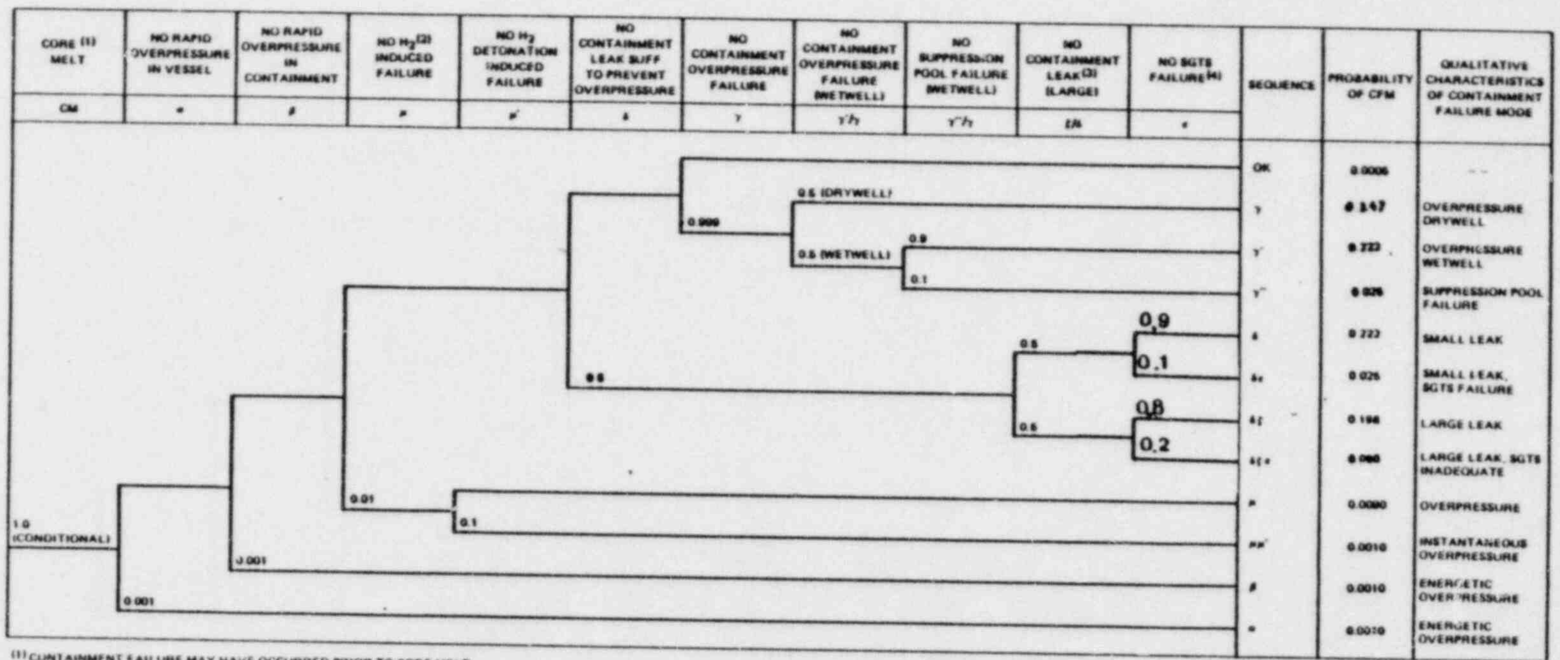
SUMMARY OF SEQUENCE FREQUENCIES (PER REACTOR YEAR) BY CONTAINMENT FAILURE MODE FOR DOMINANT SEQUENCES OF THE CLASS III VARIETY

DOMINANT SEQUENCES	CONTAINMENT FAILURE MODES						
	α 0.001	β,μ' 0.002	γ,μ 0.256	γ' 0.022	γ'' 0.025	$\zeta\epsilon,\delta\epsilon$ 0.078	ζ,δ 0.42
$T_T^1C_MPW_2$	2.7×10^{-11}	5.4×10^{-11}	6.9×10^{-9}	6.0×10^{-9}	6.8×10^{-10}	2.1×10^{-9}	1.1×10^{-8}
$T_T^1C_MPU$	2.7×10^{-10}	5.4×10^{-10}	6.9×10^{-8}	6.0×10^{-8}	6.8×10^{-9}	2.1×10^{-8}	1.1×10^{-7}
$T_T^1C_MC_{12}U$	1.4×10^{-12}	2.8×10^{-12}	3.6×10^{-10}	3.1×10^{-10}	3.5×10^{-11}	1.1×10^{-10}	5.9×10^{-10}
$T_T^1C_MC_2$	8.0×10^{-11}	1.6×10^{-10}	2.0×10^{-8}	1.8×10^{-8}	2.0×10^{-9}	6.2×10^{-9}	3.4×10^{-8}
$T_F{}^2C_ER$	4.4×10^{-12}	8.8×10^{-12}	1.1×10^{-9}	9.8×10^{-10}	1.1×10^{-10}	3.4×10^{-10}	1.8×10^{-9}
$T_F{}^2C_MR$	2.2×10^{-12}	4.4×10^{-12}	5.6×10^{-10}	4.9×10^{-10}	5.5×10^{-11}	1.7×10^{-10}	9.2×10^{-10}
$T_F{}^2C_MW_{12}$	1.4×10^{-10}	2.8×10^{-10}	3.6×10^{-8}	3.1×10^{-8}	3.5×10^{-9}	1.1×10^{-8}	5.9×10^{-8}
$T_F{}^2C_MUW_{12}$	9.6×10^{-12}	1.9×10^{-11}	2.5×10^{-9}	2.1×10^{-9}	2.4×10^{-10}	7.5×10^{-10}	4.0×10^{-9}
$T_F{}^2C_MUU_R$	2.3×10^{-10}	4.6×10^{-10}	5.9×10^{-8}	5.1×10^{-8}	5.7×10^{-9}	1.8×10^{-8}	9.7×10^{-8}
$T_F{}^2C_MPW_2$	1.7×10^{-11}	3.4×10^{-11}	4.4×10^{-9}	3.8×10^{-9}	4.3×10^{-10}	1.3×10^{-9}	7.1×10^{-9}
$T_F{}^2C_MPU$	1.6×10^{-10}	3.2×10^{-10}	4.1×10^{-8}	3.6×10^{-8}	4.0×10^{-9}	1.2×10^{-8}	6.7×10^{-8}
$T_F{}^2C_MC_2$	1.6×10^{-11}	3.2×10^{-11}	4.1×10^{-9}	3.6×10^{-9}	4.0×10^{-10}	1.2×10^{-9}	6.7×10^{-9}
$T_E{}^3C_MW_{12}$	1.1×10^{-11}	2.2×10^{-11}	2.8×10^{-9}	2.4×10^{-9}	2.7×10^{-10}	8.6×10^{-10}	4.6×10^{-9}
$T_E{}^3C_MUW_{12}$	1.9×10^{-12}	3.8×10^{-12}	4.9×10^{-10}	4.2×10^{-10}	4.7×10^{-11}	1.5×10^{-10}	8.0×10^{-10}
$T_E{}^3C_MUU_R$	2.4×10^{-11}	4.8×10^{-11}	6.1×10^{-9}	5.3×10^{-9}	6.0×10^{-10}	1.9×10^{-9}	1.0×10^{-8}
$T_E{}^3C_MPW_2$	5.4×10^{-12}	1.1×10^{-11}	1.4×10^{-9}	1.2×10^{-9}	1.3×10^{-10}	4.2×10^{-10}	2.3×10^{-9}
$T_E{}^3C_MPU$	5.4×10^{-12}	1.1×10^{-11}	1.4×10^{-9}	1.2×10^{-9}	1.3×10^{-10}	4.2×10^{-10}	2.3×10^{-9}
$T_I{}^4C_MW_{12}$	4.5×10^{-12}	9.0×10^{-12}	1.2×10^{-9}	1.0×10^{-9}	1.1×10^{-10}	3.5×10^{-10}	1.9×10^{-9}
$T_I{}^4C_MU$	6.4×10^{-11}	1.3×10^{-10}	1.6×10^{-8}	1.4×10^{-8}	1.6×10^{-9}	5.0×10^{-9}	2.7×10^{-8}
$T_I{}^4C_MPU$	7.1×10^{-12}	1.4×10^{-11}	1.8×10^{-9}	1.6×10^{-9}	1.8×10^{-10}	5.5×10^{-10}	3.0×10^{-9}
$T_I{}^4C_MC_2$	1.9×10^{-12}	3.8×10^{-12}	4.9×10^{-10}	4.2×10^{-10}	4.7×10^{-11}	1.5×10^{-10}	8.0×10^{-10}
AE/AI	1.6×10^{-12}	3.2×10^{-12}	4.1×10^{-10}	3.6×10^{-10}	4.0×10^{-11}	1.2×10^{-10}	6.7×10^{-10}
APPROXIMATE TOTAL PROBABILITY FOR CLASS III SEQUENCES	1.1×10^{-9}	2.2×10^{-9}	2.8×10^{-7}	2.4×10^{-7}	2.7×10^{-8}	8.5×10^{-8}	4.6×10^{-7}

Table 3.5.6

SUMMARY OF SEQUENCE FREQUENCIES (PER REACTOR YEAR) BY CONTAINMENT FAILURE MODE FOR DOMINANT SEQUENCES OF THE CLASS IV VARIETY

DOMINANT SEQUENCES	CONTAINMENT FAILURE MODES						
	α 0.001	β,μ' 0.002	γ,μ 0.503	γ' 0.443	γ'' 0.05	$\zeta\epsilon,\delta\epsilon$ 0.0002	ζ,δ 0.0002
$T_T^1C_MU_H$	6.6×10^{-12}	1.3×10^{-11}	3.3×10^{-9}	2.9×10^{-9}	3.3×10^{-10}	1.3×10^{-12}	1.3×10^{-12}
$T_T^1C_MD$	6.6×10^{-12}	1.3×10^{-11}	3.3×10^{-9}	2.9×10^{-9}	3.3×10^{-10}	1.3×10^{-12}	1.3×10^{-12}
$T_T^1C_MPW_2$	6.8×10^{-12}	1.4×10^{-11}	3.4×10^{-9}	3.0×10^{-9}	3.4×10^{-10}	1.4×10^{-12}	1.4×10^{-12}
$T_T^1C_MM$	3.7×10^{-12}	7.4×10^{-12}	1.9×10^{-9}	1.6×10^{-9}	1.9×10^{-10}	7.4×10^{-13}	7.4×10^{-13}
$T_T^1C_MC_2$	2.0×10^{-11}	4.0×10^{-11}	1.0×10^{-8}	8.9×10^{-9}	1.0×10^{-9}	4.0×10^{-12}	4.0×10^{-12}
$T_F^2C_MW_{12}$	3.6×10^{-11}	7.2×10^{-11}	1.8×10^{-8}	1.6×10^{-8}	1.8×10^{-9}	7.2×10^{-12}	7.2×10^{-12}
$T_F^2C_MU_H$	3.7×10^{-12}	7.4×10^{-12}	1.9×10^{09}	1.6×10^{-9}	1.9×10^{-10}	7.4×10^{-13}	7.4×10^{-13}
$T_F^2C_MD$	3.7×10^{-12}	7.4×10^{-12}	1.9×10^{-9}	1.6×10^{-9}	1.9×10^{-10}	7.4×10^{-13}	7.4×10^{-13}
$T_F^2C_MUW_{12}$	2.4×10^{-12}	4.8×10^{-12}	1.2×10^{-9}	1.1×10^{-9}	1.2×10^{-10}	4.8×10^{-13}	4.8×10^{-13}
$T_F^2C_MPW_2$	4.2×10^{-12}	8.4×10^{-12}	2.1×10^{-9}	1.9×10^{-9}	2.1×10^{-10}	8.4×10^{-13}	8.4×10^{-13}
$T_F^2C_MM$	2.2×10^{-12}	4.4×10^{-12}	1.1×10^{-9}	9.2×10^{-10}	1.1×10^{-10}	4.4×10^{-13}	4.4×10^{-13}
$T_F^2C_MC_2$	4.0×10^{-12}	8.0×10^{-12}	2.0×10^{-9}	1.8×10^{-9}	2.0×10^{-10}	8.0×10^{-13}	8.0×10^{-13}
$T_E^3C_MW_{12}$	2.8×10^{-12}	5.6×10^{-12}	1.4×10^{-9}	1.2×10^{-9}	1.4×10^{-10}	5.6×10^{-13}	5.6×10^{-13}
$T_E^3C_MPW_2$	1.4×10^{-12}	2.8×10^{-12}	7.0×10^{-10}	6.2×10^{-10}	7.0×10^{-11}	2.8×10^{-13}	2.8×10^{-13}
$T_I^4C_MW_{12}$	1.1×10^{-12}	2.2×10^{-12}	5.5×10^{-10}	4.9×10^{-10}	5.5×10^{-11}	2.2×10^{-13}	2.2×10^{-13}
$T_I^4C_MPW_2$	6.4×10^{-12}	1.3×10^{-11}	3.2×10^{-9}	2.8×10^{-9}	3.2×10^{-10}	1.3×10^{-12}	1.3×10^{-12}
AC	4.0×10^{-12}	8.0×10^{-12}	2.0×10^{-9}	1.8×10^{-9}	2.0×10^{-10}	8.0×10^{-13}	8.0×10^{-13}
$T_T^1C_ER$	7.2×10^{-12}	1.4×10^{-11}	3.6×10^{-9}	3.2×10^{-9}	3.6×10^{-10}	1.4×10^{-12}	1.4×10^{-12}
$T_T^1C_MR$	3.6×10^{-12}	7.2×10^{-12}	1.8×10^{-9}	1.6×10^{-9}	1.8×10^{-10}	7.2×10^{-13}	7.2×10^{-13}
APPROXIMATE TOTAL PROBABILITY FOR CLASS IV SEQUENCES	1.3×10^{-10}	2.5×10^{-10}	6.3×10^{-8}	5.6×10^{-8}	6.3×10^{-9}	2.5×10^{-11}	2.5×10^{-11}



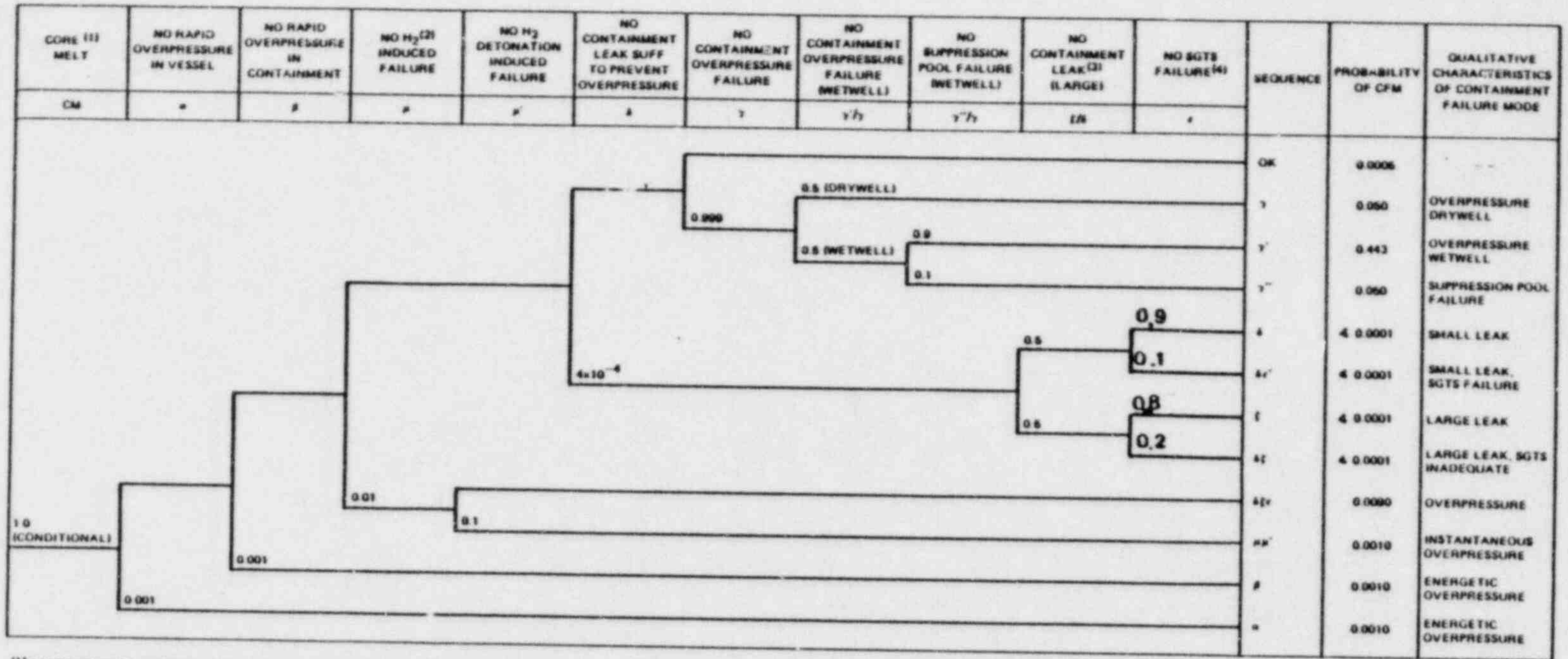
(1) CONTAINMENT FAILURE MAY HAVE OCCURRED PRIOR TO CORE MELT IN THOSE CASES (CLASS II AND CLASS IV), THE CONTAINMENT FAILURE MODES ARE ONLY USED AS MECHANISMS FOR RELEASE FRACTION DETERMINATION.

(2) ASSUMES THAT H_2 EXPLOSION IN CONTAINMENT CAUSES OVERPRESSURE FAILURE WITH DIRECT PATHWAY TO OUTSIDE ATMOSPHERE.

(3) LEAKAGE AT 2400 VOLUME PERCENT/DAY.

(4) FAILURE STANDBY GAS TREATMENT SYSTEM.

Figure 3.5.6a Containment Event Tree for the Mark II Containment for Class I, II, and III Event Sequences



(1) CONTAINMENT FAILURE MAY HAVE OCCURRED PRIOR TO CORE MELT. IN THOSE CASES (CLASS II AND CLASS IV), THE CONTAINMENT FAILURE MODES ARE ONLY USED AS MECHANISMS FOR RELEASE FRACTION DETERMINATION.

(2) ASSUMES THAT H_2 EXPLOSION IN CONTAINMENT CAUSES OVERPRESSURE FAILURE WITH DIRECT PATHWAY TO OUTSIDE ATMOSPHERE.

(3) LEAKAGE AT 2400 VOLUME PERCENT/DAY.

(4) FAILURE STANDBY GAS TREATMENT SYSTEM.

Figure 3.5.6b Containment Event Tree for the Mark II Containment for Class IV Event Sequences

μ, μ' -- Hydrogen Burn or Explosion in Containment. For the inerted Limerick containment, the possibility of a hydrogen detonation or burn appears quite remote; however, according to the tentative technical specification there may be short periods of time when the plant is operating at power and the containment is not fully inerted. This is anticipated to occur following reactor startups and prior to shutdowns. Based on past PECo experience and projected Limerick operating procedures, the probability of the plant not being inerted while operating at power considered to be 0.01. Relative to this 0.01 probability of not being inerted at power, if a core melt occurs during this time, then the probability of a burn or detonation sufficient to cause direct overpressure release, with a significant increase in the radioactive release fraction (i.e., comparable to a containment steam explosion) is no larger than 0.1[19]. This leads to a probability on the order of 10^{-3} for the μ' failure mode. However, the probability of some H_2 burn (μ) remains at 0.01. This may lead to a drywell overpressure release and is included in the γ containment failure mode.

γ -- Containment Overpressure In Drywell (No Leakage). Given that no containment leakage occurs, the possibility of containment overpressure without failure following a core melt is considered to be possible even though ultimate pressure is exceeded. Bechtel calculated the ultimate containment pressure capability to be 140 psig (approximately three times design pressure). For those core melt sequences where no leakage occurs, 140 psig and failure is reached with a high probability (0.999).

γ'/γ 20 -- Containment Overpressure (split between wetwell and drywell failure). Failure of containment due to overpressure has been divided into two types because of the potential difference in radioactive release terms. Failure in the drywell leads to direct release to the stack while a failure in the wetwell causes a release through the suppression pool. At present, evidence indicates failure at very high containment pressure may occur with equal likelihood in the wetwell or drywell. Therefore, $\gamma'/\gamma = 0.5$.

Table 3.5.13

RELEASE TERM CALCULATIONS REQUIREMENTS[a]

CONTAINMENT FAILURE MODES		RADIOACTIVE RELEASE FRACTIONS			
Designator	Description	Class I (C1)	Class II (C2)	Class III (C3)	Class IV (C4)
α	Steam explosion in vessel	Note f	Note f	Note f	Note f
β	Steam explosion in containment	Note f	Note f	Note f	Note f
μ'	H_2 explosion induced containment failure	Note e	Note e	Note e	Note e
μ	H_2 deflagration sufficient to cause containment overpressure failure	Note b	Note b	Note b	Note g
δ	Overpressure small leaks (A_R = 0.05 ft^2)	X	X	X	Note h
γ	Overpressure failure (A_R = 2.0 ft^2) Release through drywell	X	X	X	Note g
γ'	Overpressure failure (A_R = 2.0 ft^2) Release through wetwell break	Note b	Note b	Note b	Note h
γ''	Overpressure failure (A_R = 2.0 ft^2) Wetwell pool drained	X	X	X	Note h
ζ	Overpressure, large leak (A_R = 0.2 ft^2)	X	X	X	Note h
$\zeta\varepsilon$	Overpressure, large leak, SGTS failure (A_R = 0.2 ft^2)	Note c	Note c	Note C	Note c
$\delta\varepsilon$	Overpressure, small leak, SGTS failure (A_R = 0.05 ft^2)	Note d	Note d	Note d	Note d

(a) An "X" under the heading indicates that a calculation of release fraction must be made for the particular accident involving a BWR/4 with a Mark III containment; all other cases can either be extrapolated from the set of calculations or can be extracted directly from WASH-1400.

(b) Can be extrapolated from γ release by assuming a different decontamination factor for pool scrubbing. The principal difference between γ and γ' is that the γ' release occurs with much of the release passing through the suppression pool. The γ release occurs with much of the release occurring through the drywell.

(c) Can be extrapolated from equivalent ζ case by not using decontamination factor for SGTS (affects only portion of release flow).

(d) Can be extrapolated from equivalent δ case by not using decontamination factor for SGTS (affects all of release flow).

(e) Will be assumed to be equivalent to a β failure and same release fraction will be used.

(f) Release fractions will be extracted directly from WASH-1400 since the phenomenological nature of the accident does not change.

(g) Release fractions similar to those developed by the NRC using March-Corral are used in the characterization of Class IV radioactive release fractions for γ'.

(h) Extrapolated from the Class I, II, III results.

Table 3.5.14

SUMMARY -- GENERIC ACCIDENT SEQUENCE/RELEASE PATH COMBINATIONS

CONTAINMENT FAILURE MODE \ CLASS	CLASS I	CLASS II	CLASS III	CLASS IV	TOTAL PROBABILITY BY CONTAINMENT FAILURE MODE
α	1.2×10^{-8}	9.6×10^{-10}	1.1×10^{-9}	1.3×10^{-10}	1.5×10^{-8}
β,μ'	2.5×10^{-8}	1.9×10^{-9}	2.2×10^{-9}	2.5×10^{-10}	2.9×10^{-8}
γ,μ	3.2×10^{-6}	2.5×10^{-7}	2.8×10^{-7}	6.4×10^{-8}	3.8×10^{-6}
γ'	2.8×10^{-6}	2.1×10^{-7}	2.4×10^{-7}	5.6×10^{-8}	3.3×10^{-6}
γ''	3.1×10^{-7}	2.4×10^{-8}	2.7×10^{-8}	6.3×10^{-9}	3.7×10^{-7}
$\zeta\epsilon,\delta\epsilon$	9.7×10^{-7}	7.5×10^{-8}	8.5×10^{-8}	2.5×10^{-11}	1.1×10^{-6}
ζ,δ	5.2×10^{-6}	4.0×10^{-7}	$4\ 6\times10^{-7}$	2.5×10^{-11}	6.1×10^{-6}
TOTAL PROBABILITY BY CLASS	1.2×10^{-5}	9.6×10^{-7}	1.1×10^{-6}	1.3×10^{-7}	1.5×10^{-5}

Figure 3.5.7 indicates that the highest probability scenarios/are those involving a coupling of core melt accident sequences with postulated containment overpressure failures. The in-vessel steam explosion and containment steam explosion scenarios both have significantly lower probability than the others. However, the consequences for these scenarios tend to be larger than for overpressure failures. The postulated leaks are of relatively high probability, but they have smaller consequences than the containment overpressure failures.

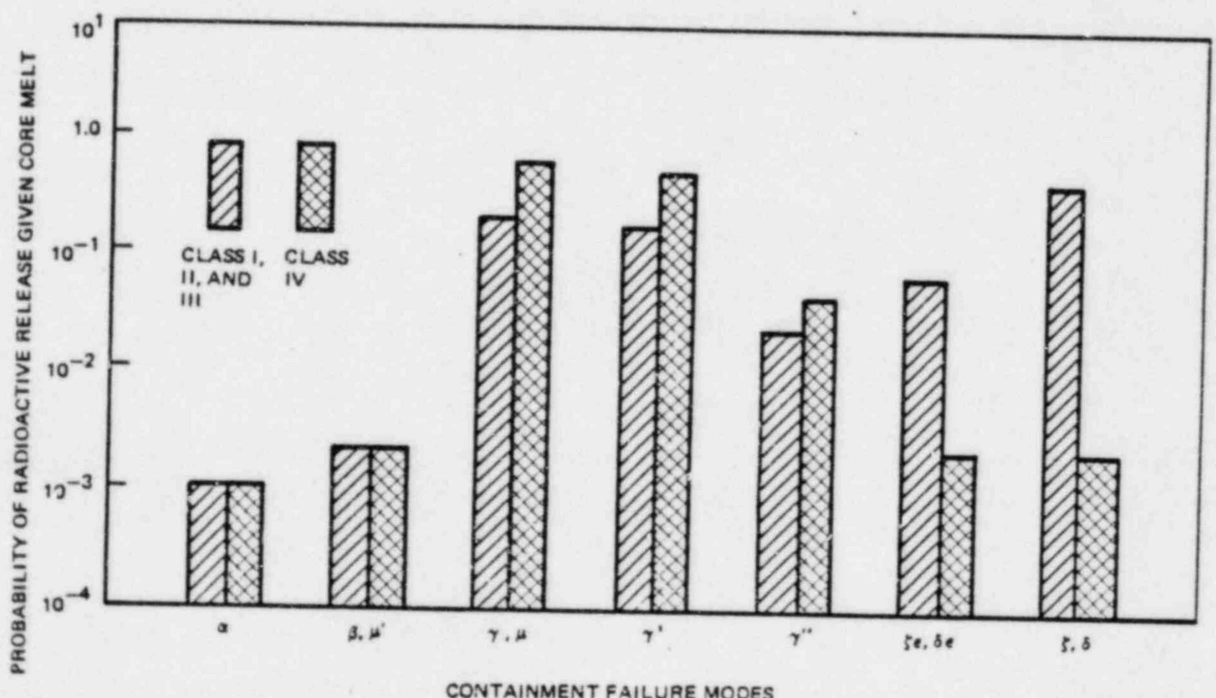


Figure 3.5.7 Probability of a Radioactive Release Given a Severe Degradation of Core Integrity -- Presented by Containment Failure Mode for All Classes.

Table 3.6.4

SUMMARY OF THE DECONTAMINATION FACTORS

Conditions	Meltdown Release	Vessel Failure and Vaporization
Containment Failure at End of Release	100	10*
Containment Failure Initiates Release	10*	10*

* Suppression pool considered saturated.

For each of the accident sequence classes there is a set of containment failure modes which will also affect the magnitude of the radionuclide releases. The principal ways the containment failure modes affect these releases are the following:

1. Size of Containment Breach: The size of postulated containment failures determine the usefulness of the reactor enclosure and the standby gas treatment system for providing additional decontamination.

2. Location of the Breach: The location of the postulated containment failure affects the degree of difficulty of the path for the radionuclide release. The most important aspect of the location is in relation to the suppression pool; that is, for some sequences which include drywell failure (i.e., γ) the radionuclide release during vaporization will bypass the suppression pool.

TABLE 3.6.5

EXAMPLES OF RADIONUCLIDE RELEASE PARAMETERS AND RELEASE FRACTIONS FOR DOMINANT ACCIDENT SEQUENCE CLASSES AND CONTAINMENT FAILURE MODES

PATHWAY SEQUENCE	TIME OF RELEASE (Hr)	DURATION OF RELEASE (Hr)	WARNING TIME FOR EVACUATION (Hr)	ELEVATION OF RELEASE (feet)	CONTAINMENT ENERGY RELEASE (10^6 BTU/Hr)	RELEASE FRACTIONS						
						Xe[(a)]	$I^{(b)}{}_2+CH_3I$	Cs[(c)]	Te[(d)]	Sr[(e)]	Ru[(f)]	La[(g)]
α												
C_1, C_3	2.0	0.5	1.0	82	40	1.0	0.40	0.40	0.50	0.05	0.50	3.0×10^{-3}
C_2	39.0	0.5	8.0	82	40	1.0	0.096	0.10	0.40	0.01	0.40	2.0×10^{-3}
C_4	2.0	0.5	1.5	82	40	1.0	0.096	0.10	0.40	0.01	0.40	2.0×10^{-3}
β												
C_1, C_2, C_3, C_4 ($\beta_1\beta'$)	4.0	0.5	3.0	82	36	1.0	0.20	0.06	0.50	0.007	0.40	1.0×10^{-5}
γ												
C_1	7.0	2.0	6.0	82	36	1.0	0.11	0.09	0.016	0.010	3.0×10^{-3}	3.0×10^{-4}
C_2	37.0	2.0	7.0	82	.3	1.0	0.06	0.023	0.4	6.3×10^{-3}	0.069	4.7×10^{-3}
C_3	7.0	2.0	6.0	82	36	1.0	0.04	0.024	0.073	2.7×10^{-3}	8.6×10^{-3}	9.1×10^{-4}
C_4	1.5	2.0	1.0	82	.3	1.0	0.261	0.202	0.434	0.029	0.952	5.23×10^{-3}
γ'												
C_4	1.6	2.0	1.0	82	.3	1.0	0.07	0.09	0.20	0.016	0.088	6.0×10^{-3}
γ''												
C_4	1.5	2.0	1.0	0	.3	1.0	0.73	0.70	0.65	0.09	0.12	7.0×10^{-3}

(a) Includes Xe, Kr
(b) Includes I (elemental), Br
(c) Includes Cs, Rb
(d) Includes Te, Se, Sb
(e) Includes Sr, Ba
(f) Includes Ru, Rh, Pd, Mo, Tc
(g) Includes La, Y, Zr, Nb, Ce, Pr, Nd, Np, Pm, Sm, Eu, Pu

The radionuclides suspended in containment following the oxidation release are assumed to be the same for each accident sequence. Effects due to the status of the suppression pool are considered to be negligible for this release.

3.6.3.3 μ' - Hydrogen Explosion

Hydrogen explosion is considered to be a low probability event for the limerick containment since it is usually inerted. However, there may be times when the plant is operating at power with the containment deine. .ed. Therefore, the possibility of hydrogen combustion is considered and the release fractions due to this type of failure are taken from WASH-1400.

The hydrogen combustion (μ') and containment steam explosion (α) are combined because of the similar manner in which they fail the containment and the assumption that they both have similar impacts on the radionuclide release fractions.

3.6.3.4 $\gamma, \gamma', \gamma''$ -- Relatively Slow Overpressure Failures During Postulated Core Melt Scenarios (Class I through IV)

The containment may fail due to a relatively slow pressure buildup due to core melt (assessed as the most likely type of failure). The various locations for such a failure are differentiated as follows:

- γ - Drywell Failure
- γ' - Wetwell Failure
- γ'' - Wetwell Failure below the suppression pool waterline.

These locations were chosen based upon a structural analysis of the LGS containment (see Appendix J).

The release fractions associated with γ', γ'' (wetwell) failures are nearly identical for all the classes of accident sequences used in the Limerick PRA quantification.

3.7 CONSEQUENCES ASSOCIATED WITH ACCIDENT SEQUENCES

This section summarizes the calculaton of offsite effects for the following:

- The calculational model used in the Limerick site-specific analysis (CRAC).
- The input data used in the CRAC evaluation.
- The results of the CRAC calculation.

3.7.1 Ex-Plant Consequence Model

CRAC (calculation of reactor accident consequences) is a computer code which was used in the Reactor Safety Study (WASH-1400) to assess the impact of reactor accidents on public risk. The CRAC evaluation in WASH-1400 was applied to specific sites but in the final assessment was applied to a composite site with population density derived in a manner to approximate an average site in the United States. This section focuses on the application of the CRAC model to the site-specific evaluation of the Limerick Generating Station. A discussion of the various aspects of the CRAC model is provided in Appendix E.

[16] See also the NRC BWR Rebaseline Case: E. J. Hanraham and L. Bickwit, Jr., "Report to the Commissioners, Subject: Report of the Task Force on Interim Operation of Indian Point", (Docket Nos. 50-247 and 50-286), June 12, 1980.

[17] Personal communication, Corradini (Sandia) to Burns and Parkinson (SAI).

[18] Personal communication between NRC (Taylor) and SAI (Burns).

[19] Any reduction of the hydrogen concentration by means of the hydrogen recombiners was not assumed due to the large amounts of hydrogen released during a core melt and the relatively small capacity of the recombiners.

[20] Y'/Y means Y' given Y.

[21] Either a failure below the elevation of the bottom of the downcomers or a containment wetwell failure which propagates to below the bottom elevation of the downcomers.

[22] SAI-REACT was also used to verify the CORRAL results.

[23] Both items are consistent with current NRC site review methods. See Appendix E for further discussion of radionuclide dispersion.

[24] Note mean values are used in all accident sequence calculations.

[25] WASH-1400 states that the error factor on LOCA initiators is 30. The actual implementaion of the data in accident sequences and the evaluation of their uncertainty do not reflect error bands of this magnitude. (A value of 7 appears to have been used.)

Table A.1.3

SUMMARY OF THE FREQUENCY OF TRANSIENT INITIATORS AND THE CATEGORIES INTO WHICH THEY HAVE BEEN CONSOLIDATED

ITEM	TRANSIENT	Frequency (Per Reactor Year) EPRI Survey of 12 BWRs			BWR OP. EXP.
		All Years	Exclude Year 1	Exclude Year 1 & > 25% Power	GE Assessment
1	MSIV Closure	1.34	.57	.35	1.08
	Closure of all MSIVs (5)**	0.67	0.19	0.13	1.00
	Turbine Trip Without Bypass (2,4)	0.00	0.00	0.00	0.01
	Loss of Condenser (8)	0.67	0.38	0.22	0.067
2	Turbine Trip	7.32	4.17	2.98	3.98
	Partial Closure of MSIVs (6,7)	0.12	0.14	0.12	0.20
	Turbine Trip with Bypass (3,13,30,32,33,34,35,36,37)	3.78	2.01	1.24	1.33
	Recirculation Problem (14,15,16,17,18,19)	0.38	0.09	0.09	0.25
	Pressure Regulator Failure (9,10)	0.43	0.35	0.31	0.67
	Inadvertent Opening of Bypass (12)	0.04	0.05	0.00	0.00
	Rod Withdrawal/Insertion (27,28,29)	0.14	0.14	0.06	0.10
	Disturbance of Feedwater (20,21,23,24,25,26)	1.39	0.70	0.53	0.68
	Electric Load Rejection (1)	1.04	0.70	0.63	0.75
3	Loss of Offsite Power (31)	.16	.11	0.09	.38
4	Inadvertent Open Relief Valve (11)	.20	.08	.03*	.06
5	Loss of Feedwater (22)	.27	.16	.06	.70
	TOTAL	9.29	5.09	3.51	6.2

* Modifies to 0.07 based upon NUREG-0626.

** Numbers in parentheses refer to transient numbers from Table A.1.2.

4. Inadvertent open relief valve which may lead to an initial heat up or pressurization of containment prior to any attempt to shutdown

5. Loss of Feedwater. The loss of feedwater initiator was separated out, but later combined with MSIV closure.

The consolidation of these transients into groups is defined in Table A.1.3.

Table B.5.4

SUMMARY OF SENSORS USED IN THE SAFETY SYSTEM INITIATION

		High Pressure		ADS		RHR and CS							
		HPCI	RCIC	ADS(A)	ADS(C)	CS(A)	RHR(A)	CS(B)	RHR(B)	CS(C)	RHR(C)	CS(D)	RHR(D)
INITIATOR	Level 2												
	692E		X										
	692F	X											
	692A		X										
	692B	X											
	692C	X											
	692G	X											
	697A		X										
	697E		X										
	Level 1												
	691A (1)			X		X	X						
	691B (2)							X	X				
	691F (2)							X	X				
	691C (3)				X					X	X		
	691G (3)				X					X	X		
	691D (4)											X	X
	691H (4)											X	X
	691E (1)			X		X	X						
PERMISSIVE	Level 3												
	695A			X									
	695C				X								
SHUTOFF	Level 8												
	693A		X										
	693B	X											
	693E		X										
	693F	X											

Table B.5.5

PRESSURE SENSORS USED BY SAFETY SYSTEMS

	SENSOR DESIGNATOR	SYSTEMS											
		HPCI	ADS(A)	ACS (C)	RHR(A)	CS(A)	RHR(B)	CS(B)	RHR(C)	CS(C)	RHR(D)	CS(D)	RCIC
INITIATION	Drywell												
	694A		X		X	X							X
	694E		X		X	X							X
	694B	X					X	X					
	694F	X					X	X					
	694C			X					X	X			X
	694G			X					X	X			X
	694D	X									X	X	
	694H	X									X	X	
LOW PRESSURE PERMISSIVE	Reactor												
	690A				X	X							
	690E				X	X							
	690B						X	X					
	690F						X	X					
	690C								X	X			
	690G								X	X			
	690D										X	X	
	690H										X	X	
	690J					X							
	690N					X							
	690K							X					
	690P							X					

REQUEST FOR ADDITIONAL INFORMATION AND CLARIFICATION ON LIMERICK PROBABILISTIC RISK ASSESSMENT (PRA)[1]

LIMERICK GENERATING STATION

Note: All items contained in this enclosure are grouped according to the Chapters and Appendices of the PRA.

Chapter 1

QUESTION 1.01

The text conveys the notion that no cross-ties between Unit 1 and Unit 2 were taken into account (p. 1-18). Cross-ties could be sources of redundancy as well as additionial failure causes. Are there cross-ties between units (e.g., RHRSW, RHRHX)? If yes, provide rationale for not considering them in the analysis.

RESPONSE

The referenced statement has been revised in Revision 3 to the PRA, as follows:

> "The system evaluation has been performed using design drawings from GE and Bechtel for Limerick Unit 1 only, and considers no cross-ties, benefits, or other effects between the two units with the exception of a cross-tie between Unit 1 and Unit 2 ESW and RHR service water pumps."

No other cross-ties, that would enter into the analysis are known to exist.

QUESTION 1.02

The ultimate containment capability is calculated to be in excess of 140 psig which is approximately 2.5 times the design pressure (p. 1-19). Given the fact that the containment exhibits leakage under design conditions, what is the increased leakage rate of the containment prior to reaching 140 psig? Provide a description of how the leakage between the primary and secondary containment was modeled.

RESPONSE

Leakage from the primary containment into the secondary containment is assumed to be totally pressure-driven through "cracks" modeled as an equivalent small containment break area. The estimated design leakage at the design pressure was used as the basis for calculating an equivalent hole size. The containment leakage calculations in INCOR prior to containment failure were modeled as flow through an orifice in the containment wall. The orifice area is an input variable. This area is calculated from equations for critical flow (Reference 1) based upon the estimated leakage rate from the containment evaluated at design pressure with the reactor building at atmospheric pressure. The flow rate through the orifice is sonic. Since the ratio of the reactor building pressure (atmospheric) to the containment pressure (design limit) is less than the critical pressure ratio, flow through the leakage paths is in the critical flow regime. Therefore, increasing the containment pressure beyond it's design limit would not result in an increase in containment leakage volumetric rate prior to reaching 140 psig.

This conclusion is reached considering two assumptions:

1. The potential for crack size to grow with increasing internal pressure is assumed to be small.

2. Steam and/or aerosols generated during scenario have been demonstrated to clog and plug small crack areas.

The effect of the containment elevated pressures on containment leakage is discussed in light of the core conditions during the pressurization period prior to reaching 140 psig. First, for the cases where the containment is intact during core melt (i.e., Class I and Class III), the aerosols generated during core meltdown would tend to plug the containment "cracks" (Reference 2). This plugging effect has been observed in various situations where aerosols,

(e.g., smoke) have plugged leaks. Therefore, the containment leakage rate should not increase prior to reaching 140 psig.

Secondly, for the accident sequences where the containment is failed prior to core melt (i.e., Class II and Class IV), the containment leakage prior to core melt would not result in "increased" radionuclide release to the environment. Nevertheless, in the Containment Systems Experiments (Reference 3) it was observed that leaks were also plugged by condensed steam.

It should be noted that leakage of steam from the containment into the Reactor Building could result in some environmental degradation in the Reactor Building, however, due to the compartmentalization and system of room cooling units in the Limerick Reactor Building this leakage is expected to be easily treated and would not adversely affect the operation of ECCS equipment. This question only affects the timing of operator action. Since Class IV scenarios have containment pressure rises to the ultimate pressure capacity of the containment in less than one hour, operator action for repair is minimal to begin with and would therefore, be minimally affected by increased leakage. Class II sequences are potentially affected by adverse environment since operator action for equipment repair has been included in the Class II evaluation.

Based on the above discussion, there should not be an increased leakage rate of the containment prior to reaching 140 psig. By considering the design leakage of the containment in the analysis, the Limerick PRA is judged to be realistic.

(1) Wheat, L. L., et. al., CONTEMPT-LT, ANCR-1219, Idaho, June 1975.

(2) Morowitz, H. A., "leakage of Aerosols from Containment Buildings", to be published in Health Physics.

(3) Witherspoon, M. W., and Postma, A. K., Leakage of Fission Products From Artificial Leaks in the Containment Systems Experiments, BNWL 1582, Battelle Northwest Laboratories, Richmond, Washington.

QUESTION 1.03

Based on the design leakage of the containment (p. 1-20), it is expected that some amount of containment environment constituents will escape into the reactor building; this may become more pronounced when the containment is at an elevated pressure. Given the long-time nature of some transients, what is the probability of hydrogen combustion inside the reactor building? In the event that there is containment failure prior to core melt, what is the likelihood of hydrogen combustion in the reactor building? Does hydrogen combustion inside the reactor building further aggravate radioactive releases, and if so, in what way?

RESPONSE

Assuming perfect mixture of hydrogen with the air in the secondary containment, ignition quantities for hydrogen combustion have been calculated for the Limerick secondary containment. These are shown in Table 1 as a function of temperature. This indicates that at the design leakage rates (see answer to PRA 1.02), given the composition of the vapor region of the primary containment, the amount of H2 gas which could escape to the reactor building will be far below the amount shown in the table*. Therefore, H2 combustion inside the reactor building would not be possible at all for Class I and Class III sequences where the containment is intact during core meltdown and hydrogen is produced from metal-water reactions.

Table 1

IGNITION QUANTITIES OF HYDROGEN IN THE REACTOR BUILDING*

Imperative	62°F	210°F	290°F
Lower Limit Upper Limit	490 lb moles 3730 lb moles	360 lb moles 2990 lb moles	250 lb moles 2440 lb moles

* Assumes mixing and dry air at the operating pressure of 14.6 psia in the RB.

In the event that the containment were failed prior to core melt (Class II and IV), the reactor building (RB) would be steam laden and steam inerting would occur with the oxygen being displaced from the RB. Hydrogen combustion would be unlikely.

If hydrogen combustion in the reactor building should occur it would result in a pressure rise in the compartment which might increase radioactive release rate at that time. The pressure increase at ignition could result in a puff release of radionuclides reducing the potential for radionuclide removal in the reactor building. This possibility is extremely remote and has not been explicitly modeled in the Limerick radionuclide release calculations since the possibility of hydrogen combustion inside the reactor building is quite remote.

---

* The relative volumes and leak rates of the reactor building and an intact containment provide a dilution ratio of 2000:1. Even if the hydrogen concentration in the containment were 10-20%, the resulting concentration in the reactor building would remain far below the ignition level.

QUESTION 1.04

The sources of data used for the Limerick study were summarized into four categories. Please provide criteria for the selection of one data source over the other? What was the rationale for combining several data sources in some instances and not others? What are the guidelines used to determine whether or not the data base should be integrated (p. 1-23)?

RESPONSE

Sources of data used for the Limerick event tree/fault tree models in order of priority are:

I. Plant or component-specific

II. NRC

III. General Electric, EPRI, SAI (BWR generic data)

IV. WASH-1400 and WASH-1270

Example of plant specific data are:

- Diesel failure rates at Peach Bottom
- PJM data for Loss of Offsite Power frequency
- Susquehanna Technical Specification which should be similar to the future Limerick Technical Specification
- Peach Bottom Limiting Conditions of Operation which should be similar to the future Limerick LCO

An Example of a component specific datum is:

- Failure probability of Target Rock safety/relief valves

Example of NRC compiled data are:

- Pump LERs - NUREG/CR-1205
- Diesel generator LERs - NUREG/CR-1362
- Valve LERs - NUREG/CR-1362
- Human reliability - NUREG/CR-1278
- BWR transient initiators - NUREG/CR-0460
- Diesel generator unavailabilities - NUREG/CR-1362

Examples of General Electric, EPRI, and SAI compiled data are:

- BWR transient initiators frequencies - unpublished GE evaluation, EPRI-NP-801 (SAI), EPRI-NP-438 (SAI), SAI-154-79-PA
- Equipment and component failure rate data - unpublished GE data on specific SRV valves, level sensors, containment pressure sensors, valves, and pumps
- Human reliability - SAI-010-76-PA, <u>Transations of ANS</u> June 1977 (SAI)
- System unavailability due to maintenance - unpublished GE data
- Diesel generator reliability - McLagan et. al., SAI/AMES 1980
- Diesel generator failure rates for Cook, Zion, and Peach Bottom power plant

Examples of WASH-1400 and WASH-1270 failure rate data used in the analyses or for comparison are:

- BWR accident initiators - WASH-1400 and WASH-1270

- Component failure rate data - WASH-1400

- Human failure rate data - WASH-1400

- System unavailability due to maintenance - WASH-1400

The scope of this PRA did not generally include integration and re-evaluation of different data sources. In such infrequent instances, data were combined and re-evaluated when it appeared that the data were compatible, uncertainty was reduced, and the integrated results was at least as realistic as the results from either of the individual data bases. Applicable data evaluations were collected from each of the four categories without priority discrimination. Subsequently, comparisons of the applicable results were used to determine a failure rate probability based on relevancy of the data base to the Limerick failure rate assessment.

For example, the LGS evaluation of diesels generator failure rates was divided into three probability areas: (1) Failure to "start and run" probability for a single diesel is based on the PECo operating experience at the Peach Bottom Station which has similar diesel generators; (2) The conditional probability of miltiple diesel failures given a single diesel failure is based on combining the 23 LWR (McLagan et. al., "Preliminary Assessment of Diesel Generator Reliability at Light Water Reactors, "SAI/AMES, 1980) with the NUREG/CR-1362 data to use the most information available; (3) Recovery of a diesel is based upon the NRC evaluation and is consistent with the Peach Bottom data. This combined use of the available data was delieved to provide a valid and the most applicable approach for the LGS PRA.

QUESTION 1.05

It is stated that for the purposes of the analysis, the Technical Specifications of the Peach Bottom Station and the test frequencies from the Susquehanna Station were used (p. 1-24, 1-25). Provide rationale for this combination as representative of LGS.

RESPONSE

The Technical Specifications of Peach Bottom Atomic Power Station were used since they are representative of Philadelphia Electric Company's operating and maintenance practices. However, the test frequencies are more reflective of design. For this reason, the test frequencies of Susquehanna, a BWR 4/Mark II, were selected as being more representative.

QUESTION 1.06

(a) What is the rationale for Guideline No. 11 (p. 1-32)?

(b) Provide the reference for the "improved chronic-health-effects model" referred to (p. 1-10).

RESPONSE

(a) Guideline No. 11 of the LGS PRA assumes that the failure of display information to the operator is not dependent on the accident sequences.

This guideline was adopted in the detailed engineering evaluation which accompanied the LGS PRA, no identified link could be made between the accident sequences investigated and common cause adverse effects on all the information available to the operator. It is recognized that there may be a possibility for very low frequency accident sequence initiators which could cause a loss of display information*. However, these initiators were not quantified explicitly in the PRA, since an engineering investigation determined that such circumstances were highly unlikely at Limerick. Therefore it was judged that the overall probability of such sequences would be much less than other identified sequences and this initiator was not quantified.

(b) The referenced statement was inaccurate. In Revision 3 to the Limerick PRA, the statement has been corrected to read:

"These improvements include corrected calculational routines and improved output routines providing for better analysis of results, including sensitivity studies where applicable."

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* Cases of loss of display information have occurred at B&W reactors. However, the Limerick design is significantly different that the B&W design and the display instrumentation is not subject to the same relatively high frequency of disabling events.

QUESTION 1.07

"Table 1.2 Summary Of Success Criteria For The Mitigating Systems Tabulated As A Function Of Accident Initiators (p. 1-26)." For each initiator, including all 5 transients, reference the section of the FSAR that describes the adequacy of the selected success criteria. For those initiators, including all 5 transients with success criteria not described in the FSAR, provide the reference that justifies the adequacy of the selected success criteria.

RESPONSE

The content of Table 1.2 defines the minimum system requirements to successfully terminate a transient or LOCA initiating event (with scram). All success criteria in this Table were developed from the analyses given in NEDO -24708, "Additional Information Required for NRC STaff Generic Report on Boiling Water Reactors," December, 1980. The analyses in that report are based on realistic conditions of core heatup, instead of the conservative licensing basis analysis performed for the FSAR.

The analysis in the FSAR demonstrates the performance of the safety systems under the "single active failure" criteria, and is not applicable to the PRA ground rules or conditions (multiple failures with realistic conditions).

QUESTION 3.01

For the event trees shown in Chapter III, provide the reference for the probabilities assigned, to each system success or failure and/or frequency of initiators. Provide rationale and method used whenever different probability values are used for the same event.

Identify values obtained from fault trees and provide cross reference to corresponding fault tree figure.

RESPONSE

- The initiator frequencies used in the event tree quantification are presented in Appendix A.1.
- The conditional probabilities used in the event tree quantification are in general developed from detailed system level fault trees. In certain cases, multiple systems are quantified through combinations of system fault trees. These combinations are explicitly defined in a separate proprietary document entitled, "Quantification of The Limerick Generating Station Event Tree Functions". Rational and methodology are discussed for each of the functional fault trees provided in the report.

QUESTION 3.02

The Limerick FSAR reported that the vapor suppression system reliability and effectiveness varies as a function of the LOCA size. However, in the Limerick PRA study, it does not appear that this particular aspect of the system has been incorporated into the containment event trees. If it was neglected, what is the justification? If it was included, provide additional details on how the system was modeled.

RESPONSE

1. Quantification of the vapor suppression failure probability and affect is subjective since there is only limited test experience to indicate potential bounds.

2. A large LOCA coupled with vapor suppression failure leading to core melt was found in WASH-1400 not to be a contributor to risk, therefore the decision was made early in the development of the LOCA event trees to eliminate this failure mode from LOCA sequences since the Limerick reactor and containment design would tend to suppress its impact and not enhance the contribution to potential unacceptable conditions. This was reviewed by study participants to ensure that a dominant contributor was not being overlooked. This failure mechanism could be incorporated into the system level LOCA event trees for completeness, however, scoping studies still demonstrate that inclusion of vapor suppression failure would not contribute to the calculated level of risk at Limerick.

3. The assessment of sequences involving potential degraded core conditions using the containment event trees does take into account the potential for premature containment failure due to vapor suppression failure during challenges to the containment integrity in the assignment of the conditional failure probability for drywell failure (γ).

QUESTION 3.03

In addressing manual shutdown as an initiating event, there are situations in which the reactor operator is required to shutdown the reactor in order to be in compliance with technical specifications due to the unavailability of certain safety systems. Provide a summary of how these types of manual shutdowns were included in the event tree depicted in Figure 3.4.2?

RESPONSE

A previous review of operating experience data regarding manual shutdown [1] ranked the causes for BWR manual shutdowns as follows:

(1) Refueling outages

(2) Turbine/generator problems

(3) Relief valves, BOP valves

(4) Recirculation Pumps

(5) Drywell leakage

(6) Gaseous radwaste system

(7) Planned outages for maintenance

(8) Ventilation system

(9) Operator training/exams/inspections

For the quantification of the LGS manual shutdown event tree some simplifications were made. For example, violations of technical specification leading to a manual shutdown were not explicitly included, since based upon the limited operating experience data available this did not appear to be a noticeable contributor to manual shutdowns.

In order to ensure that a possible dominant sequence was not being overlooked, the potential contribution to core melt frequency due to the possibility of RHR outages leading to a manual shutdown challenge was scoped. The comparison between the frequencies of PCS containment heat removal challenges as evaluated in the event tree versus those which could result if one RHR is disabled initially is as follows:

The potential frequency of PCS challenges following manual shutdowns initiated by RHR failures outside technical specification requirements can be estimated as the product:

Manual shutdown freq. due to RHR tech. spec. violation per Rx Yr.	x	Probability of losing feedwater and PCS during shutdown	x	Conditional failure probability of the PCS and remaining RHR given a manual shutdown

5×10^{-3}/RxYr** x 7×10^{-3}/d x 1.6×10^{-5}/d =

5.6×10^{-10}/Rx Yr

This frequency is negligible compared to the dominant sequence for manual shutdowns. (2.2×10^{-7})

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* Excluding the few cases which affected the condensate pumps.

** Frequency is calculated based upon the data from reference [1] and the assumption that 1/2 of these failures would be recoverable in the 20 hour period subsequent to shutdown prior to containment overpressure.

REFERENCES

[1] Component Failures that Lead to Manual Shutdowns, SAI-180-PA, Prepared by Science Applications, Inc. for Sandia Laboratories (1980)

QUESTION 3.04

Plateout and settling is assumed to "remove" radioactivity. Can the radioactivity be released back to the environment by some physical means, for instance water flash (p. 3-125)?

RESPONSE

Containment failure at elevated temperature and pressure will in most cases result in the "flashing" of suppression pool water to vapor until equilibrium saturation conditions are reached.

Radioactivity dissolved in the suppression pool was conservatively assumed to be re-released back into the containment air space upon containment failure for Class I and Class III accident sequences. During the sudden depressurization it was conservatively assumed that the dissolved radionuclides in the water which boil off would be released back in direct proportion to the amount of water flashed. The vaporized water would then leave the radionuclide salts in the air space to be released to the environment through the postulated containment break. Radionuclides (i.e., elemental I) can also be partitioned between the aqueous phase and the vapor phase resulting in another mechanism for possible release. This is treated in CORRAL in the equilibration of I in the containment spray water.

QUESTION 3.05

Isn't the δ sequence a drywell overpressure and not a wetwell overpressure as labeled in the far right column of the containment event trees (see for example p. 3-82)? Explain the difference between δ , δ' , and δ'' . The definition of δ is confusing. In the containment event trees it means containment overpressure either drywell or wetwell. The definiation on the top of page 3-133 indicates it is a drywell failure.

RESPONSE

(The Greek letter used to identify relatively slow overpressure failures of the containment postulated to occur due to slow containment pressure increases is γ , not δ .)

The location of potential overpressure failures is described by the following conditional probability nomenclature:

γ - containment overpressure failure occurring in the drywell

γ' - containment overpressure failure occurring in the suppression pool air chamber space above the waterline

γ'' - containment overpressure failure occurring below the water line of the suppression pool

There were several places where the symbols were incorrect in the original PRA. Corrections were made in Revision 3; however, several places still remain incorrect. Revision 4 includes additional corrections.

QUESTION 3.06

In Section 3.2, the text states that one of the most important aspects of the event tree technique is that it ensures that all of the key accident initiators are identified. How does the event tree technique identify key initiators, and how does it identify all key initiators?

RESPONSE

Appendix A.1 describes the origin of the initiators chosen for the LGS PRA. The event tree methodology requires the identification of initiators but it does not of itself assure completeness in the initiator identification. Revision 4 changes the sentence in Section 3.2 to read "One of the most important aspects of the event tree technique is that it assists in ensuring that key accident sequences following the identified initiators are evaluated. Therefore, the ability to establish a comprehensive list of initiators is a necessary part of the PRA".

The method used in the LGS PRA to establish the initiator list is to build upon past evaluations (probabilistic and deterministic) and operating experience data.

The selection of possible event tree initiators was made after the examination of the following available information:

- site specific operating experience (loss of offsite power from the PJM data)
- operating experience of BWRs (especially Peach Bottom)
- operating experience of all LWRs (including LER data)
- previous industry evaluations (WASH-1400, RSSMAP, Big Rock Point, Clinch River PRA)
- published reliability evaluations
- plant operator insight and other expert input

The initiators have been grouped together in two categories for discussion purposes:

- Transient challenges (includes ATWS sequences)

- LOCAs

This is similar to the WASH-1400 approach.

Identification of the transients and quantification of their frequency of occurrence is based upon operating experience data. A list of 37 anticipated transients is summarized in Table A.1.2. Consolidation of these 27 possible initiators into five major transient categories is summarized in Table A.1.3.

The frequency and type of loss of coolant accidents is also characterized by operating experience data, however because of the lower frequency of occurrence the results have larger uncertainty bands associated with them. The small, medium, and large LOCA (WASH-1400 precedent events) are presented in Table A.1.6.

In addition, the anticipated transients also have detailed event trees constructed which are used to assess the frequency of sequences which may occur if a failure to scram followed a transient.

The area where there is the highest uncertainty in potentially missing important initiators are those very low frequency initiators which have not yet been identified because of the limited amount of operating experience data.

Some scoping evaluations were performed on low frequency accident initiators such as loss of DC power and loss of instrument air which indicated that sequences following these initiators were less than 10% contributors to core melt frequency and that they were in the Class I category, that is, were sequences for which the lowest consequences were calculated.

QUESTION 3.07

Further justify the statement in Section 3.4.3.2 that potential failures of the reactor pressure vessel as an initiating event have a very low probability of expected occurrence. What is the effect on the overall consequences by omission of this event?

RESPONSE

The probability of a reactor pressure vessel failure was found to be too low to be treated explicitly in the PRA. A separate event tree with reactor pressure vessel failure as the initiating event is, therefore, not included in the PRA based on the low event frequency. The basis for this treatment is discussed in detail below.

A survey of existing literature, data and reactor pressure vessel expert opinion was employed. The following are the pertinent points from this survey:

1. Section 8 of the Appendix XI of the Reactor Safety Study (Reference 1) cites the conclusion of WASH-1318, that the upper limit (99% confidence) probability of a disruptive RPV failure event in any one nuclear reactor during any service year falls within the range of 10^{-7} to 10^{-6} and the actual value of this probability would be expected to be even smaller. This conclusion was based on 725,000 vessel-years of service in U.S. fossil-fueled power plants without a disruptive failure.

2. The "leak before break" phenomenon is another key consideration which justifies a failure rate less than 10^{-7} per vessel year (Reference 2). Reference 2 further states that the "pressure vessel will have a considerable margin to failure by (a) brittle fracture, even with large postulated initial flaws and (b) that leak-before-break capability is maintained even after a LOCA." This means that long before a crack could propagate to the point that a disruptive failure could occur the crack would propagate through the vessel wall and be detected due to significant leakage. The leak detection system would detect the existence of leaks and allow shutdown and depressurization of the reactor to avoid propagation of the crack and vessel failure.

3. Reactor vessels are subjected to periodic inspections, in accordance with Section XI of the ASME Code. This inspection is generally more intensive than that for non-nuclear vessels, and

consists of an ultrasonic inspection of weld joints, and surface inspections (visual, liquid penetrant test and magnetic particle test) before the vessel goes into service and inspections every 10 years thereafter.

4. Reactor vessels are designed and operated with a higher degree of protection from pressure transients and temperature events than are non-nuclear vessels. This higher degree of protection is assured by virtue of design measures, including over-pressure relief devices and operational control procedures.

5. Due to low neutron flux, BWR vessels are not significantly subjected to nil-ductility phenomena during the course of their expected operating lifetime.

6. Reactor vessels are designed and constructed in accordance with Section III of the ASME Code. These rules are more restrictive than the rules of Section I and VIII, which are used for non-nuclear vessels.

7. Reactor vessels are operated in accordance with the limitations specified in NRC License Technical Specifications, wheras no such requirements are imposed on non-nuclear vessels.

Based on the above considerations it was concluded that, the probability of RPV disruptive failure is so low that its explicit inclusion in the analysis would not significantly impact the PRA results. The RPV failure modes that are mechanistically plausible would produce consequences similar to the higher-probability LOCA events because of the "leak-before-break" phenomenon. These latter events are analyzed in detail and reported in the PRA.

An indication of the significance of the reactor vessel failure on the LGS risk can be obtained by considering the treatment given this subject in WASH-1400. In WASH-1400 a pressure vessel rupture accident was included in release category BWR-3 at a frequency of 10^{-7}/year while an accident at a frequency of 10^{-8}/year was included in release category BWR2.

Release category BWR2 contains accident sequences which have early containment failure and limited fission product removal and is similar to the LGS type IV accident class. This class has a total frequency of $1.3x10^{-7}$/year in the Limerick analysis. The vessel failure as quantified in WASH-1400 would increase the LGS frequency to $1.4x10^{-7}$ forthis class.

The BWR3 category has smaller fission product release with greater credit for removal in the containment and/or the

reactor building, and is similar to the LGS type III accident class which has a frequency of $1.1x10^{-6}$/year. The WASH-1400 assessed vessel failure frequency for this category would increase the Class III frequency to $1.2x10^{-6}$.

From the above it is again concluded that reactor vessel failure would add insignificantly to the risk from other accident sequences and is considered negligible. The same conclusion wasreached in WASH-1400:

> "5.3.4.2 reactor vessel rupture (R). As with the PWR, reactor vessel rupture is defined as a vessel failure large enough to negate successful operation of the ESF's required to prevent core melt. Again, the probabilities of such events are small and make no significant contribution to the release histogram in Figure 5.2."

REFERENCES

1. Report, "Reactor Safety Study - Assessment of Accident Risks in U.S. Commercial Power Plants," 1975, WASH-1400.

2. Technical Report, "Analyses of Pressure Vessel Statistics From Fossil-Fueled Power Plant Service and Assessment of Reactor Vessel Reliability in Nuclear Power Plant Service," USAEC Regulatory Staff; May, 1974, WASH-1318.

QUESTION 3.08

According to p. 1-14, Section 3.2 will discuss the subject of completeness. Further details as to why those events noted in the section satisfy the completeness requirements are needed. Have events like, RCP seal failure, loss of instrument and control air, loss of DC power, etc. been examined in the Limerick study?

RESPONSE

(a) As noted in previous reviews of PRAs [1], the ability to ensure completeness may not be possible to demonstrate. PRAs do not attempt to assess all possible accident scenarios; they attempt to assess all significant accident sequences (the risk from the identified sequences includes more than 50% of the risk from all possible accident sequences [1]). The LGS PRA attempted to include all major and significant accident sequences by:

(1) incorporating all contributing sequences which have previously been identified,

(2) incorporating new initiating events and functional responses which are judged to present unique contributions to risk in either higher frequency of occurrence or in possible consequences,

(3) inclusion of more detailed event trees in certain cases which have previously been judged to be potential contributors to BWR risk.

The event tree methodology affords a technique to assemble together in a single format an array of postulated sequences so that the engineering details of the systems can be integrated with the minimum success requirement and known problem areas to quantify accident scenarios. This allows a convenient ranking mechanism to ascertain potential dominant contributors based upon the plant specific design considerations.

This appraoch also attempts to ensure completeness by presenting a thorough engineering evaluation in the framework of past PRAs.

(b) The following events have also been considered in scoping analyses to determine if these low frequency initiators would produce sequences which might significantly contribute to the potential risk of the Limerick plant operation:

- Recirculation Pump Seal Failure: This particular failure is included in the small LOCA initiator. Recirculation pump seal failure during accident scenarios is judged to not significantly alter the progression or quantification of the accident sequences. Therefore, no further event tree development was performed.

- Loss of instrument air: While instrument air has a pervasive influence on many balance of plant functions the evaluation of its contribution to risk has been adequately assessed by inclusion in the MSIV closure initiated event tree. The impact of MSIV closure on the key balance of plant system at Limerick is similar for both loss of instrument air and MSIV closure, i.e.:

  - all MSIVs close

  - condenser becomes unavailable as a heat sink

  - feedwater becomes unavailable

  - ECCS equipment remains unaffected to perform safety function (i.e., air operated valves in safety systems fail safe)

  - Recovery of instrument air and reopening the MSIVs are both given relatively little credit during the initial 30 minutes and some additional credit during the subsequent 20 hours.

- Loss of DC Power: SAI has performed a PRA evaluation of the contribution of Loss of DC Power initiator to the risk spectrum of a BWR/4 plant which has two emergency DC buses. The results of this non-Limerick evaluation indicated that the loss of DC power initiators represented less than 10% of the core melt frequency and that the resulting accident sequences would produce

consequences in the lower release categories, i.e., lower risk to the public.

The calculated contribution to early fatalities from the analyses was negligible. Since this evaluation was performed on a plant with 2 emergency D.C. buses (one battery each), and the Limerick plant has four emergency DC buses (one battery each) it is judged that the LGS plant would have an event lower contribution to risk from loss of DC power.

References

[1] Risk Assessment Review Group Report to the U.S. Nuclear Regulatory Commission, U.S. Nuclear Regulatory Commission NUREG/CR-0400.

QUESTION 3.09

In order to successfully operate the ADS, it must be manually initiated in a timely fashion (p. 3-17).

Provide the basis for the time limit on how soon the depressurization should begin.

Is there a time limit beyond which depressurization is not possible?

Is there any requirement on the rate of depressurization?

It is stated (p. 3-18) that the alternate methods of depressurization are given low probability for success, since they involve "creative operator actions under potential stressful conditions:. Will there be approved procedures delineating steps required to implement these alternate methods? Are these methods included in the quantification of the sequence?

RESPONSE

The operator will follow procedures. Procedures for Limerick will follow the EPG [1], which was used as a basis for the Limerick PRA analysis.

The referenced statements (p. 3-17, 3-18) are now on p. 3-22, 3-23, 3-24, and 3-25 of Revision 3 to the PRA. The statement that ADS must be manually initiated carries the underlying assumption that automatic ADS has not occurred. The referenced statements are mode in reference to the turbine trip event tree (Figure 3.4.1). For this event, automatic ADS usually will not occur. On the event tree, ADS is shown to be required in the event of failure of high pressure injection (FW/cond, HPCI, and RCIC).

Failure of high pressure injection would result in decreasing reactor water level. At this time, the operator should initiate ADS. The operator would have at least 30 minutes (see p. 3-23 of the PRA) to accomplish ADS. The basis for this time limit is the boil-off time. The water level in the core would reach top of active fuel (TAF) in 25 minutes and would be slightly below TAF in 30 minutes [2]. The core would be adequately steam cooled to remain undamaged.

The time limit beyond which depressurization is not possible is dependent on various conditions of the event, but is always limited by the effect on suppression pool temperature and/or suppression pool water level. The attached page from the EPG defines these limits.

The operator will be instructed to limit the rate of depressurization to maintain the RPV within a cooldown rate of 100 F/hr; however, he may exceed this rate if necessary.

In Revision 3 to the PRA, the statement on p. 3-24, 25 now reads as follows:

> "The probability of successful completion of such actions is difficult to quantify. Therefore, they are given low probability of success." (in the PRA)

In Revision 3 to the PRA, these alternate means of depressurization were given no credit. The reason is not only that they are difficult to quantify (as stated above), but that they would have little effect on the results, since ADS success is limited by the operator. However, these are viable means of depressurization, they are in the EPG (see attached page), and they will be written into the plant procedures.

REFERENCES

(1) Emergency Procedure Guidelines (DRAFT), Revision 1E, BWR 1 through 6, March 1, 1982 (General Electric Co. for the TMI BWR Owners Group)

(2) Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors, NEDO 24708, Revision A, December, 1980. See Figure 3.1.1.1 - 58.2

RC/P Monitor and control RPV pressure.

> If while executing the following steps Emergency RPV Depressurization is required, enter [procedure developed from CONTINGENCY #2].

RC/P-1 If any SRV is cycling, initiate IC and manually open SRVs until RPV pressure drops to [935 psig (RPV pressure at which all turbine bypass valves are fully open)].

RC/P-2 Stabilize RPV pressure below [1090 psig (lowest SRV lifting setpoint)] with the main turbine bypass valves.

RPV pressure control may be augmented by one or more of the following systems:

- IC
- SRVs #12
- HPCI #11
- RCIC
- Other steam driven equipment
- RWCU (recirculation mode) if boron has not been injected into the RPV.
- Main steam line drains
- RWCU (blowdown mode) if boron has not been injected into the RPV. Refer to [sampling procedures] prior to initiating blowdown.

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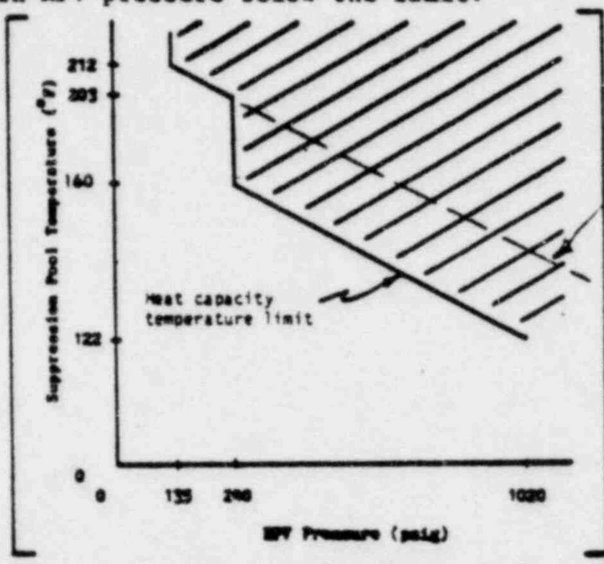
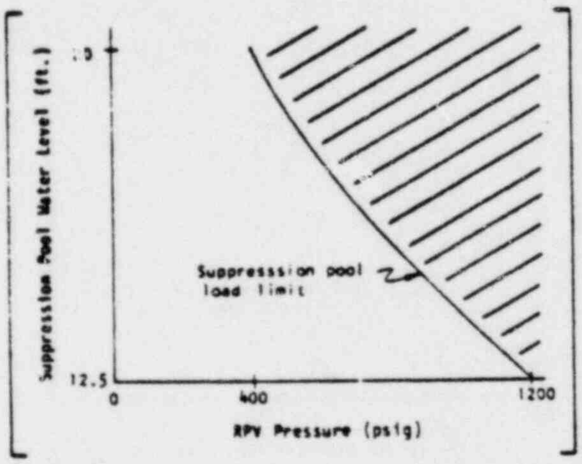
If while executing the following steps:

- Suppression pool temperature cannot be maintained below the heat capacity temperature limit, maintain RPV pressure below the limit. | #13 #14 #15 |



- Suppression pool water level cannot be maintained below the suppression pool load limit, maintain RPV pressure below the limit. | #13 #14 #15 |



- Steam Cooling is required, enter [procedure developed from CONTINGENCY #3].

(RC-5) Rev 1E

TABLE I

ABBREVIATIONS

ADS	-	Automatic Depressurization System
APRM	-	Average Power Range Monitor
CRD	-	Control Rod Drive
ECCS	-	Emergency Core Cooling System
HCU	-	Hydraulic Control Unit
HPCI	-	High Pressure Coolant Injection
HPCS	-	High Pressure Core Spray
IC	-	Isolation Condenser
LOCA	-	Loss of Coolant Accident
LPCI	-	Low Pressure Coolant Injection
LPCS	-	Low Pressure Core Spray
MSIV	-	Main Steamline Isolation Valves
NDTT	-	Nil-Ductility Transition Temperature
NPSH	-	Net Positive Suction Head
RCIC	-	Reactor Core Isolation Cooling
RHR	-	Residual Heat Removal
RPS	-	Reactor Protection System
RPV	-	Reactor Pressure Vessel
RSCS	-	Rod Sequence Control System
RWCU	-	Reactor Water Cleanup
SBGT	-	Standby Gas Treatment
SLC	-	Standby Liquid Control
SORV	-	Stuck Open Relief Valve
SPMS	-	Suppression Pool Makeup System
SRV	-	Safety Relief Valve

QUESTION 3.10

Why does a "controlled" manual shutdown require SRV actuation? Isn't the plant scrammed from a power level below the bypass valve capacity? Explain the major sequence of events which are expected for a normal reactor shutdown.

RESPONSE

Manual shutdown does not require SRV actuation. The incidence of inadvertent SRV actuation during manual shutdown is so low as to be considered negligible. In Revision 3 to the Limerick PRA, SRV actuation has been deleted from the manual shutdown event tree (Figure 3.4.2).

The sequence of events for a manual shutdown will be described in the plant operating procedures. The major sequence of events maybe summarized generically as follows. (Plan-specific procedures may vary somewhat from these guidelines):

1. Pre-shutdown preparation and system checks.
2. Reduce reactor recirculation flow to 80% power.
3. Perform system checks (PSC)*
4. Reduce reactor recirculation flow to 65% power.
5. PSC
6. Shut down one reactor feed pump
7. PSC
8. Reduce reactor recirculation flow to 55% power.
9. Shut down one condensate pump.
10. Insert control rods to 30% power using Rod Sequence Control System (RSCS).
11. PSC
12. Transfer feedwater control to MANUAL and shut down feedwater heaters.
13. Insert control rods to 10% power using RSCS.
14. PSC
15. At ~ 5% power, transfer reactor level control to MANUAL and place mode switch in STARTUP.

16. PSC

17. Begin scamming individual rods.

18. Reduce generator load to 10MW or less (Bypass valves will open).

19. PSC

20. Trip turbine and shut down generator.

21. PSC

22. Shut down second reactor feed pump and second condensate pump.

23. Continue scramming indivdual rods. Hold reactor pressure at 900 psig for two hours.

24. PSC

25. Manually open bypass valves further to reduce pressure (limit reactor cooldown rate to 100°F/hr)

26. PSC

27. At ~200 psig, shut down remaining reactor feed pump. Remove steam jet air ejector and start mechanical vacuum pump.

28. PSC

**29. At ~75 psig, place RHR in service in Shutdown Cooling Mode. Hold reactor temprature at 125°F.

30. At ~50 psig, close bypass valves and MSIV's.

31. Scram all remaining control rods, and place mode switch in SHUTDOWN.

* Numerous checks of system conditions and process parameters are conducted throughout the shut down procedure.

** RHR required only if going to cold shut down.

QUESTION 3.11

Explain why a value of $1.1x10^{-4}$ was used for the unavailability of RHR/RHRSW or PCS, given a failure of the SRV's to reclose, in Figure 3.4.3. This is the same as for turbine trip or manual scram event trees. The additional problem of recovering feedwater (the initiating event) should increase the unavailability as stated on p. 3-27 under event W description.

RESPONSE

For MSIV closure initiated sequences with multiple relief valves stuck open it is conservatively assumed that the RHR heat exchangers are eventually required to remove decay heat from the suppression pool. Consequently, the failure rate of the containment heat removal path is determined without including credit for the ability to use the power conversion system as an adequate heat sink for these sequences.

Similarly, in turbine trip or manual scram sequences*, if there are multiple stuck open relief valves, also no credit is taken for the PCS system. Therefore, the containment heat removal path failure rate is the conditional probability of failure of containment heat removal throught the RHR heat exchangers. ($9.9x10^{-6}$ in Revision 3)

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* For Revision 3 of LGS PRA, the manual shutdown case Figure 3.4.2 is realistically modeled without SORV since relief valves seldom open on manual shutdown.

QUESTION 3.12

Provide supporting documentation and/or calculations showing that a feedwater pump can add water to the reactor vessel following a scram and a subsequent stuck open SRV. The event trees for turbine trip and MSIV closure show the feedwater availability to be the same, independent of the condition of the SRV. It is realized that, should the feedwater pump not be able to continue running, due to low steam pressure, the condensate pump would take over at approximately 600-700 psig pressure. However, operator actions and additional valve operation would seem to reduce the probability of successful operation. Have these items been considered?

RESPONSE

Refer to Feedwater P&ID, Bechtel Drawing No. 8031-M06. No operator action is required to initiate condensate pumps. When main feedwater pumps trip, condensate pumps continue to run and deliver water to the RPV through the centrifugal feedwater pumps. If RPV pressure is too high, the condensate pumps continue to run with minimum flow bypass until RPV pressure decreases to the point where injection begins. No valving is needed.

Condensate pumps will continue to deliver water to the RPV and will eventually overfill the RPV unless the operator takes manual control by diverting flow rods to the main condenser hotwell by means of flow control valves.

QUESTION 3.13

(a) The event tree for manual shutdown has different feedwater system unavailabilities depending upon the condition of the SRV's. Why does the difference exist in this case?

(b) The statement at the top of p. 3-20 discusses overriding of the low vacuum interlocks for the turbine bypass valves. Have the operator actions required to bypass the MSIV low vacuum interlocks been considered in calculating the unavailability of the power conversion system?

RESPONSE

(a) See response to Question 3.10. Manual shutdowns are judged to be slow, controlled event for which the feedwater/condensate system can provide short term coolant makeup with a high reliability. In the manual shutdown event tree there is considered to be a negligible probability that the safety relief valves may be required to operate; therefore, the operation of safety relief valves has been deleted from the manual shutdown event tree in Revision 3 to the Limerick PRA.

(b) The statement at the top of p. 3-20, in more camplete context, reads as follows:

> "The main steam isolation valves (2) in one of the four main steam lines must either remain open or be reopened. A turbine bypass valve must open to control reactor pressure during reactor depressurization. If the condenser vacuum cannot be maintained below seven inches of Hg, the low vacuum interlocks on the bypass valves must be overriden."

The assumption that the MSIV low vacuum interlock must be overridden is correct. For completeness, the referenced sentence should so state. Both interlocks (turbine bypass and MSIV) can be overridden by the operator from the control room.

The referenced statement is in the discussion of the turbine trip event tree. As explained on p. 3-18, this event tree is for turbine trips with bypass. For this event, the frequency of loss of condenser vacuum is very low, and the availability of feedwater and the PCS is high.

Loss of condenser vacuum as an initiating event is included in the MSIV closure event tree (Figure 3.4.3). For this event, the probability of recovering feedwater and the PCS is lower and does require the operator to override both high vacuum interlocks (bypass valve and MSIV) for those cases where vacuum is lost. This is accounted for in the analysis.

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* The incidence of relief valves opening during manual shutdown is so low as to be negligible and has been deleted from the manaul shutdown event tree for Revision 3 to the PRA.

QUESTION 3.14

For the MSIV transient, the report indicated that the RCIC steam condensing mode was not evaluated (p. 3-28). On p. 3-20 it is stated that the RCIC steam/condensing mode was included in the turbine trip event tree. Why is this decay heat removal method not consistently included in the analysis?

RESPONSE

In Revision 3 of the Limerick PRA, credit for the steam condensing mode of RHR was deleted. A definitive statement is given as Item 3 on pages 3-26 and 3-27 of the PRA, and is repeated here:

"3. Heat removal via the RHR steam condensing mode is viewed as an additional design feaure which allows the operator flexibility in maintaining a safe reactor condition in the face of unusual plant occurrences. The RHR steam condensing mode utilizes the HPCI steam lines, the RCIC turbine and pump, RHR exchangers, and RHR service water to transfer reactor decay heat to the ultimate heat sink. The steam condensing mode will be available for plant operation. However, it is not included in the system fault trees. A scoping analysis has shown that a small net benefit would be derived from the use of steam condensing, but no credit was taken in the analysis."

QUESTION 3.15

In order to establish natural ventilation in the HPCI and RCIC rooms, operator action is required. Is this going to be part of the emergency procedures?

RESPONSE

Yes.

QUESTION 3.16

Further elaboration on the removal of the emergency core cooling functionability from the event tree is required (p. 3-40).

RESPONSE

The issue was not addressed explicitly in the Limerick PRA for the following reasons which were included in WASH-1400 Appendix XI, Section 7.

1. The question of the success of failure of ECCS -- as a matter of functionability, as opposed to operability -- does not readily lend itself to analysis by the methods used in WASH-1400. Thus, the study decided to examine what level of failure probability would cause ECF to contribute to potential accident risks. As noted in Appendix V, Section 4.2, sensitivity studies reveal that "... even if values as high as 10^{-1} for ECF failure (probability) were to be used, any contribution made would be within the accuracy of the overall calculations."

2. Thus, although there appears to be no current basis for making a rigorous quantitative assessment of the probability of ECF failure, the analysis referenced showed that even if ECF failure probability were as high as 10^{-1}, it would not change the results of the study significantly. It is the view of the study that the probability thay ECCS will fail to cool the core adequately is less than 10^{-1}.

In addition, there has been further efforts within the nuclear research community to verify that these assertions are true. Based upon the assumption used in WASH-1400 and the additional verification of these assumptions by efforts such as LOFT, it was judged that no new information has become available since WASH-1400 which would change the sensitivity evaluation indicating that even significantly higher failure probabilities of ECF would not change the results of the study.

There is nothing unique in the LGS design which would change the conclusion presented in Appendix XI of WASH-1400.

QUESTION 3.17

Are there any erroneous actions expected upon a plant scram condition, i.e., containment isolation due to level shrink or turbine trips (main and RFPT) due to actual or sensed level swell? If so, how has this been taken into account in the accident sequences? Is MSIV closure trip point at Level 2 or Level 1?

RESPONSE

Approximately 95% of scrams at BWR's result in turbine trips (65% with bypass and feedwater available; 30% with feedwater pump trip). These statistics are in the data base used for the Limerick PRA. They constitute the transient initiating events leading to the accident sequences described in the transient event trees (Figures 3.4.1-3.4.5). The root causes of the scrams and trips are discussed in Appendix A and listed in Table A.1.2. Deviations from the normal sequence of events following a scram are analyzed in the various accident sequences identified in the event trees. Functional and equipment failure modes leading to the off-normal accident sequences are identified and analyzed in the system and functional fault trees.

The MSIV closure trip point at Limerick will be Level 1.

QUESTION 3.18

Please explain the basis for assigning a reactor scram failure of $1x10^{-5}$ for a large LOCS (p. 3-42) and $3x10^{-5}$ for medium and small LOCAs (p. 3-45 & 3-47).

RESPONSE

The treatment of LOCA coupled with a failure to scram can be explained through the following key facts:

1. The failure to scram conditional probability has been extracted from the NRC document NUREG-0460 and is estimated at $3x10^{-5}$/demand. Based upon the BWR precursors which have occurred, the ratio of common-mode mechanical to electrical scram system failure without ARI is 1/2 as discussed in Appendix B. Therefore, the conditional probability of common-mode mechanical failure of the scram system is estimated to be $1x10^{-5}$/demand and electrical to be $2x10^{-5}$/demand.

2. Following a large LOCA, the SLC is assumed to be ineffective as a means of inserting negative reactivity into the core for shutdown, and scram system failure is taken to lead directly to core melt of the Class IV type. ARI has the effect of reducing the common-mode electrical failures in the scram system by approximately a factor of 100.

3. Medium and small LOCA's coupled with a failure to scram are judged to be capable of being effectively mitigated through the use of the SLC system and ARI. Transfers to the IORV event trees is used to model both ARI and SLC mitigation capability.

4. As noted in the text, the evaluation of failure to scram for a large LOCA has been simplified for the purposes of the quantification. Since the remote probability of a failure to scram has been assumed to be independent of the low frequency of a large LOCA (i.e., initiator and blow down forces), the calculated frequency of a large LOCA coupled with an ATWS is extremely low.

A simplification has been made in the treatment of failure to scram for each of the LOCA initiators, as follows:

- For large LOCA initiators, a specialized ATWS event tree is not drawn. The following two simplifications reduce the problem:

    - electrical common-mode failures plus failures in ARI are of significantly lower frequency than mechanical common-mode failures.

- SLC is assumed to be ineffective

Therefore, $1 \times 10^{-5}/d$ is used as the conditional probability that the large LOCA would be followed by a common-mode failure to scram.

- A similar simplification could be done for small and medium LOCA's; however, these are treated in the IORV ATWS event tree (Figure 3.4.11) to which transfer occurs. Therefore, the total conditional probability of failure to scram is used to assess the transfer.

An additional ATWS tree could be drawn for large LOCA initiators; however, the quantified results would not be changed. In particular, the simplification for large LOCA does not affect the sequence quantification used in the calculation of risk due to Class IV sequences.

QUESTION 3.19

The text indicates (p. 3-43) that the success criteria and calculated probability of long term coolant recirculation and short term coolant injection are similar. Why are the success criteria for long term and short term demands the same? What is the difference in system configuration between coolant injection and coolant recirculation?

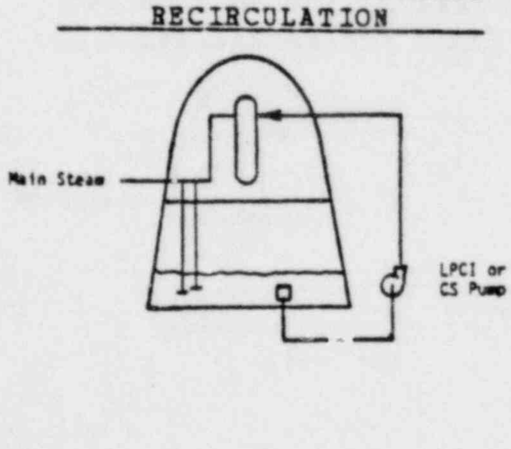
Given the long time nature of some of the accident scenarios-in the order of twenty to thirty hours, was failure subsequent to successful system actuation addressed in the Limerick study (failure to run)? If yes, were the degraded environmental conditions under which the systems must operate taken into consideration?

RESPONSE

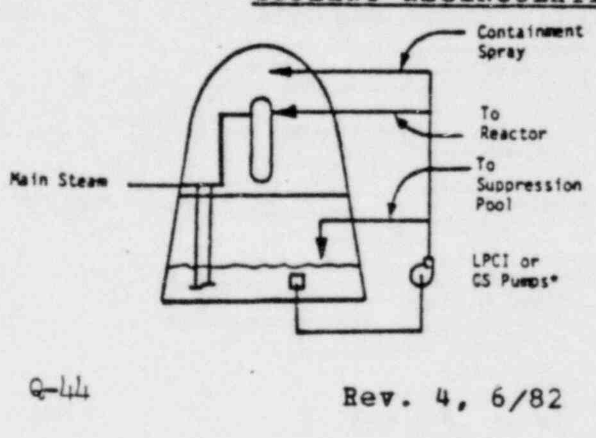
(a) The success criteria indicate the minimum complement of systems necessary to successfully fulfill a function. The success criteria derived by GE indicate that all accident sequences requiring low pressure coolant injection considered in the LGS PRA could be adequately mitigated by one leg of any of the low pressure systems, i.e., any one of the four LPCI pumps or either of the two core spray subsystems. The success criteria for coolant recirculation requires the same set of minimum components.

(b) The following two simplified schematics are provided to identify the configuration differences between coolant injection and coolant recirculation. Coolant recirculation has the potential for a wide variety of return paths to the containment which increase the success of the recirculation function. However, the dominant contributors to failure of coolant recircu1cation are included in either the coolant injection function or containment heat removal function (not shown here).

COOLANT INJECTION AND ONE MODE OF THE COOLANT RECIRCULATION



COOLANT RECIRCULATION



It should be noted that the coolant recirculation function is a carryover from the WASH-1400 PWR terminology and is not a BWR system or function.

(c) Failure of coolant injection/recirculation functions over the period 0-20 hours is included in the evaluation of the conditional probabilities of component failure. In this way the failure of components to run are included in the quantification. It must be noted that the dominant contributors to the calculated conditional probability of system failure are demand failures.

(d) The most likely scenarios following reactor shutdown are that the required ECCS or PCS equipment are available to safely cool the core and containment and that no significant degradation of containment or reactor building environmental conditions exist. The PRA does however consider the possibility that an initial failure to start or a subsequent failure to run may occur in the long term containment heat removal process (includes recirculation as a function). If this occurs, a conditional probability for successful repair within 20 hours is also incorporated. The degraded conditions which might exist were taken into account in so far as they may affect repair. The degraded conditions, i.e., high suppression pool temperatures and potentially high Reactor Building temperatures, are within the envelope of ECCS equipment operating capability. High containment pressure conditions were modeled to include an appropriate high exhaust pressure trip of HPCI and RCIC turbines.

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* Core spray can recirculate through the reactor vessel only.

QUESTION 3.20

What are the set points for the high radiation interlock for the COR? Page 3-46 states that the COR is assumed available for a medium LOCA. What would the containment radiation level be from this event and would the COR actually be available?

RESPONSE

COR is not included in the LGS PRA plant model. See Revision 3 submitted April 30, 1982

QUESTION 3.21

Some sequences on the Turbine Trips ATWS event trees (p. 3-53) are designated as negligible. The $T_T\ C_M\ C_{12}\ U$ sequence is 8.3×10^{-8} which is not negligible when compared to other sequences on the same event tree which are assigned probability values such as 6.4×10^{-8} for $T_T\ C_M\ D$. This discrepancy is present in other sequences and for other events. What is your criterion for assigning sequence path probability as negligible?

RESPONSE

The $T_T C_M C_{12}\ U$ frequency is included as a contributor to the Class III sequences. It has a frequency of 1.4×10^{-9}/reactor-year per revision 3 of Figure 3.4.8b. Originally, sequences with frequencies that were less than two orders of magnitude lower than the maximum sequence frequency in the event were treated as negligible. In Revision 3 to the PRA, all sequences with frequencies below 1.0×10^{-9} were treated as negligible. All others were evaluated.

QUESTION 3.22

What is the probability of failure for the secondary containment (p. 3-50)?

Given the unity probability for a number of the branches with MSIV not open, what do TW, TWE, TA, TAE, TQ, and TQE signify?

RESPONSE

(a) The presence of the function, "Secondary Containment", on the event tree is superfluous. It was originally intended to distinguish between cases with relatively small leakage even though the rapid pressurization of the containment during an ATWS condition without mitigation generally precludes obtaining significant benefit from the secondary containment. However, in the most likely scenario in which the intent of Alternate 3A is accomplished, the following will occur:

- feedwater will be successfully runback and the recirculation pump will be tripped,
- the turbine bypass will open and accommodate the steam flow,
- the condenser will be available as a heat sink,
- the MSIV's will remain open, and
- SLC will eventually be used to bring the reactor to a shutdown condition.

For this scenario, the secondary containment remains intact.

(b) Again the secondary containment branches are superfluous and are not used in the quantification of the event tree.

QUESTION 3.23

The report states (p. 3-56) that with multiple relief valves failed open, the RHR is required to operate successfully.

Is there a time limit on how long multiple relief valves could stay open before exceeding the capability of the RHR system? Has this been accounted for in the PRA?

RESPONSE

The referenced statement has been revised in Revision 3 to the PRA and is now on p. 3-72. The statement now reads as follows:

> "For those cases where multiple relief valves fail open, the analysis conservatively requires the RHR to operate successfully on the assumption that the MSIV's will close."

The above statement is in reference to a turbine trip ATWS event. For cases where the MSIV's do not close (or are reopened), there would be no definitive time limit (at least 5-6 hours), since the reactor would depressurize with most of the generated steam going to the main condenser. Credit for this case was not taken in the analysis (with relief valves open).

The cases analyzed assumed an isolation and a requirement for both RHR's to operate when multiple relief valves are open. This a somewhat conservative treatment as discussed in Footnote 11 of Section 3 of the PRA. With one relief valve open, the suppression pool heatup rate would be 2-3°F/minute for the conditions of this event. No analysis was performed for mulitiple valves stuck open.

The probability used in the event tree for operator initiation of RHR was 0.99, based on Table 21-1 in Swain and Guttmann [1]. The referenced table can be applied in several ways, i.e., with or without a dedicated operator, and with or without shift operator backup. In an ATWS event, RHR initiation is a vital function which must be performed manually. It is expected that this will be stated clearly in the Emergency Operator Procedures and understood by the operator and other control room personnel, so that necessary actions can be expected to be taken within 5-10 minutes resulting in peak suppression pool temperatures below saturation. At 15 minutes, with multiple relief valves open, it may be assumed that pool temperatures would peak at a temperature above the pool saturation point and containment pressure would rise (at 750°F, containment pressure would be ~15 psig)

REFERENCES:

[1] Handbook of Human Reliability Analysis With Emphasis on Nuclear Power Applications, NUREG/CR-1278, A. D. Swain, H. E. Guttman, Sandia Laboratories, October, 1980.

QUESTION 3.24

The $T_T C_m C_L$ sequence (p. 3-57) does not use COR due to "high radiation associated with incipient fuel failure". Why is there no incipient fuel failure with the $T_T C_m R$ sequence on that same page? A related question is to give the basis of the 90% MSIV isolation assumption for the $T_T C_m C_L$ sequence.

RESPONSE

The referenced statement has been revised in Revision 3 to the PRA and now appears on p. 3-73. The $T_T C_m R$ sequence is treated as a Class IV sequence; i.e., containment fails prior to core melt (due to the high rate of steam flow to the suppression pool). This treatment also applies to the $T_T C_E R$ sequence.

The statement regarding 90% MSIV isolation is not applicable to Revision 3 of the PRA.

(Note also that COR is not included of the analysis.

QUESTION 3.25

The $T'_T C_M R$ sequence (p. 3-57) states that it is "assumed" that RPT and FW runback are tripped from the same set of logic and sensors. Are they in fact tripped from the same logic and sensors? What flow rate does the FW run back to? Has the case been investigated in which the FW runback does occur, but the RPT does not? This would seem to be a more limiting case, since vessel inventory would be rapidly decreasing.

RESPONSE

The referenced statement is on p. 3-73 of Revision 3 to the PRA and has been revised as follows:

> "Since RPT and FW runback are tripped from the same set of logic and sensors it was conservatively assumed that RPT failure would also result in failure of feedwater runback and recriticality due to dilution of the boron."

RPT and feedwater are tripped from the same set of logic and sensors, but could still fail independently. Independent failure was modeled for feedwater runback given successful RPT, but RPT failure was treated as always resulting in a Class IV core melt.

Feedwater runback is to zero flow.

The case where feedwater runback occurs, but RPT does not, has been investigated and is found to be the more limiting case in regard to the effect on the core. However, the common failure of both feedwater runback and RPT has the greater affect on risk since it results in a Class IV event, whereas feedwater runback with RPT failure would result in a Class III event.

QUESTION 3.26

Page 3-69 states that ARI is successful if, and only if RPT is successful. Provide detailed information on ARI.

RESPONSE

(The statement is on p. 3-86 of Revision 3 to the PRA.) The referenced statement is the following:

"b) ARI is effective if and only if RPT is successful."

The reason for this is that ARI requires 25 seconds (maximum) to insert control rods. Therefore, in the absence of RPT, the reactor would remain at full power for 25 seconds. For cases where the MSIV's are closed, the RPV pressure would rise to 1400 psi with all safety relief valves open. In the Limerick PRA, this was conservatively assumed to result in a LOCA and Class III core melt.

ARI is a diverse means of providing a scram signal to the control rod drives. It uses different sensors, logic, and valves than the reactor protection system. The ARI signal is generated by the same sensors and logic as the RPT signal. Both are initiated by either reactor water level 2 or RPV high dome pressure.

QUESTION 3.27

Page 3-65 shows ARI either working or failing independent of a success or failure of RPT. Is this consistent with the requirement on page 3-69 that ARI is successful if, and only if RPT is successful?

RESPONSE

The referenced figure (3.4.11a) is on p. 3-82 of Revision 3 to the PRA. The reference to p. 3-69 is on p. 3-86 of Revision 3.

The statement that "ARI is effective if and only if RPT is successful" is correct as explained in the answer to Question 3.26. The statement applies to event sequences requiring prompt scram Figure 3.4.11 is for an IORV initiator, in which prompt scram is not required. There is adaquate time for ARI to be affective. For this event, there would be no automatic signals to initiate RPT or ARI. The event is modelled as if the operator initiated a manual scram, received no response, then initiated ARI. The operator would probably also initiate RPT, but this is conservatively not modelled.

QUESTION 3.28

(Top Paragraph, p. 3-86) The statement is made that the diaphragm floor is drained into a sump and the downcomer pipes. This drainage capability eliminates the possibility of a molten core dropping in one large mass from the vessel directly into a pool of water. How does this statement apply if containment spray is used? The downcomers are approximately one foot above the floor level so a large amount of water can accumulate on the floor prior to the molten core dropping. It is realized that no credit for containment spray has been assumed, but have negative effects, such as the above or excessive steam production, been accounted for?

RESPONSE

There are three possible scenarios which could lead to an accumulation of a large amount of water on the drywell floor. These include:

1) A degraded core accident initiated by a loss of coolant from a pipe break in the primary containment which results in the primary coolant discharging into the drywell region.

2) A core melt accident where the containment sprays are initiated but fail prior to the RPV bottom head failure, and

3) A core melt accident where the sprays work and remain functional throughout the degraded core accident progression.

For these postulated core melt accident scenarios, the core debris could potentially fall into a large pool of water. For steam explosions, the important parameter to consider is the maximum quantity of melt which could interact with the coolant and efficiently mix before the melt solidifies or an interaction occurs.

Some Sandia National Laboratory experiments indicate that the relative volume fraction of the melt was small compared to that of steam and water at the time of the spontaneous explosion [1]. Based on the total amount of water that could accumulate on the drywell floor, it seems likely for steam explosion to occur for the three secarios stated above. However, it is important to note that the maximum amount of melt available for mixing would be that portion of the core melt mass which is molten at the time the interaction occurs, since this would determine the potential for containment failure.

Considering the BWR Reactor pedestal geometry, the CRDM and its associated support structures represent a large heat sink. The thermal capacity of these structures should allow some of the core melt mass to solidify prior to fuel coolant mixing.

In addition, the geometry of the drywell floor* is such that an efficient core-coolant mixing and fragmentation may not be likely in a shallow pool. Based on the above, a steam explosion that could generate enough energy which can create a missile that could fail containment is judged to be unlikely.

It may be possible, however, that the steaming rates for the first two scenarios may be sufficient to fail the containment by overpressure, depending on the accident class. That is, the required steaming rate to fail the containment is reduced if the accident class is such that the containment pressure is already elevated at the time of RPV bottom head failure. For the third scenario, where the sprays are functional at the time of the interaction, the steaming rate required to fail containment must be much greater than the high condensation rate on the spray droplets, in order to fail the containment by overpressurization.

These types of scenarios are implicit in the ex-vessel steam explosion failure probability used in the PRA, based on the analysis given in Appendix H, Section IV.A., pages 59-90.

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* The downcomers lip extends approximately 1 1/2 feet above the floor which limits the amound of debris and water mass that can remain in this area.

(1) Sandia National Laboratories, Light Water Reactor Safety Research Program Quarterly Reports for the following periods: July-September, 1979, October-December, 1979, April-June, 1980, July-September, 1980.

QUESTION 3.29

Limerick takes a 10^{-2} failure rate per demand for COR (see bottom of p. 3-102). A Sandia Report on "Risk Assessment of Filtered-Vented Containment Options for a Mark-I Containment" shows a value looks to be optimistic. (See also the bottom of p. 3-21).

Explain why Limerick COR has given an unavailability of 10^{-2} per demand.

RESPONSE

Same as for Question 3.20

QUESTION 3.30

What are the bases for the selection of the probabilities on the containment event tree? Address each containment failure mode in detail.

RESPONSE

General Response

The containment event tree for Class I, II, and III is Figure 3.5.6a on p. 3-114 of Revision 3 to the PRA. The Class IV CET is Figure 3.5.6b on p. 3-115. These figures contain numerical errors as submitted in Revision 3. Corrected figures are included. Conditional probabilities as utilized in the Limerick containment event trees (CET) were developed utilizing:

- analysis and data extrapolated from WASH-1400
- deterministic and probabilistic analysis from the literature or performed since WASH-1400 by a variety of contractors and national laboratories, including the Limerick GS architectural engineering firm, Bechtel; and PECo consultant, Fauske Associates
- engineering judgment and expert opinion

Two sets of CET conditional probabilities were developed, each containing a spectrum of potential containment failure modes extending from small leaks within the capability of the SGTS to large energetic failures of containment. The following distinction can be made between the treatment of the static overpressure failures:

- for Classes I, II, and III, accident process sequences were predicted upon the fact that pressure would build slowly inside containment and that the eventual failure of containment due to static overpressure could involve any of a spectrum of potential leakage path sizes.
- for Class IV it was predicted that even a "static" overpressure failure would occur sufficiently fast to preclude the likelihood of leakage before failure and; therefore, the failure would most probably correspond to a large containment failure size.

In the Limerick PRA, it was conservatively modelled that essentially all (99.9%) core melts would lead to containment failure by one of the postulated failure modes.

Conditional Probabilities for Specific Containment Failure Modes (CFM)

α - in-vessel steam explosion - this CFM refers to failure of the primary containment as a result of a large scale molten core-water interaction in the vessel which produces a blast wave and/or energetic projectiles. In WASH-1400 this event was given a frequency of 0.01 given a core melt. Since that time, significant efforts have been made to identify conditions under which such a release of energy would occur. At the time the LGS PRA was performed, no probabilistic data was available other than WASH-1400. Reference deterministic data from Sandia indicated that such reactions were extremely unlikely under conditions of high in-vessel pressure and high coolant temperatures. Work by Fauske & Associates (Appendix H) further indicated that the configuration of BWR internals and the additional amound of channeling in a BWR core made the likelihood of a "coherent" oxidation reaction very unlikely. Based upon conversation with Corradini (Sandia), a conditional probability of in vessel steam explosion sufficiently energetic to simultaneously fail containment was assessed to be 10^{-3}/core melt. This probability is afactor of 10 less than that used in WASH-1400 to characterize this event.

β - ex-vessel steam explosion - this CFM corresponds to failure of the primary containment as a result of a large scale core/water reaction taking place in the containment after failure of the reactor pressure vessel. In WASH-1400, values for the frequency of this CFM were assessed only for a limited number of sequences, principally sequences initiated by large LOCA's with a significant amount of water in the drywell. Non-zero values estimated ranged from 0.01 to 0.18. For the LGS MK II over-under containment design, the potential for in-containment steam explosions was assumed to exist for all sequences resulting in core melt. Direct access of a molten core to the wetwell pool in the LGS continament would have to occur through the downcomer vents or through small drains to penetrations in the floor of the CRD room (vessel pedestal region). Access is limited since the risers on downcomers would have to be failed or the drain cover melted through. Fauske & Associates (Appendix H) work indicated that there would not be sufficient core/water interface in the drywell to allow a major reaction. Furthermore, reactions in the suppression pool would be limited in scope and the large amound of steel and concrete structural components would tend to cause a non-coherent reaction to occur. In light of the above qualitative observations a value of 1×10^{-3}/core melt was used to reflect the judgment that the event of an ex-vessel steam explosion is considered highly unlikely.

μ - H_2 burn failure - for this CFM, failure of the primary containment is assumed to result from increased containment pressure due to a reaction of hydrogen in the primary containment. The TMI-2 accident indicated that potentially large releases of hydrogen are possible given a core overheating event. Normally the LGS containment is inerted, and even in a core overheat scenario insufficient oxygen would exist in primary containment for hydrogen combustion. It was assumed that the containment may not be inerted during reactor-operation for up to 70 hours per year. Therefore, for the amount of zirconium available in the LGS core and the size of the primary containment, it was determined that a hydrogen burn, when the containment is not inerted, could be generated with sufficient pressure increase to fail the containment. Since it can be construed that a core melt during this potentially uninerted time would likely be accompanied by hydrogen combustion, a conditional probability of 0.01/core melt (70 hours/7000 hours) was conservatively* chosen for containment overpressure failure due to hydrogen combustion.

μ' - H explosion - This CFM is instantaneous overpressure due to a pressure spike caused by a hydrogen explosion. The probability of an explosion and the size of the resulting pressure spike are both controlled by the degree of H concentration. It was estimated that no more than 10% of the time would conditions exist to provide sufficient hydrogen to produce a pressure spike large enough th fail the containment.

δ, - containment leak sufficient to prevent overpressure - In the event that no steam explosion or H combustion induced failure occurs, the containment may fail by overpressure. These CRM's refer to a failure of the containment with an equivalent cross sectional area geater than 2 ft² in a variety of locations including (in the drywell), (in the wetwell and (in the wetwell containment wall below the water line such that the suppression pool inventory may be drained into the Reactor Building). In lieu of any deterministic or probabilistic information, an overall probability of 0.5 was assigned for large failures in Class I through Class III CET's, and a value of 0.9996 was used for Class IV for the reason discussed above in the general comments.

γ, γ', γ'' - containment overpressure failure - Given the fact that the two most likely containment failure locations were identified as:

- at the interface between the diaphragm and the primary conatinment wall, or

- mid height in the wetwell

the and failures were estimated to be equally likely, with the failure mode having a probability reduction factor of 10.

S / δ - large leak - The δ and S CPM's refer conditions in which significant leakage may occur to prevent energetic containment failure. These are failures of the containment for which secondary containment decontamination may be significant; refers to the smallest size break equivalent to less than a 3" diameter hole and to a larger break equivalent to a hole 0.5 ft² in cross-sectional area. Without further information on the leak before failure, these two modes were evaluated as equally likely.

δ_ϵ, S_ϵ - large and small leaks with SGTS failure - These CPM's represent large and small primary conatinment breaks in which the standby gas treatment system failed to operate resulting in structural failure of the secondary containment and direct leakage to the outside. Reliability of the SGTS is estimated on the basis that it is similar to systems designed to minimum single failure criteria operating in an unusually hostile environment.

For the small (S) break a conditional probability of 0.1 that the system will not be effective is estimated. For the larger (ζ) break some further degradation of system effectiveness would be expected and the probability of SGTS failure is doubled.

OK - Containment failure does not occur - the possibility exists that if no significant leakages occur, some sequences would not result in containment failure because of the passive containment capability or active damage control measures. Since no credit was taken for active recovery measures such as recovery of coolant makeup or containment sprays, the probability of this CPM was estimated to be quite small (0.0005 for Class I, II and III, and 0.001 for Class IV.)

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* It is unlikely that the reactor core will be at full power during times when the containment is not inerted.

QUESTION 3.31

Why was an average value of 10^{-3} per event used for a coherent in-vessel steam explosion when more detailed values of 10^{-2} for a steam explosion during a LOCA event and 10^{-4} for a steam explosion during non-LOCA events were stated on p. 3-114?

RESPONSE

The value of 10^{-3} percent used in the Limerick analysis is based in the Sandia Laboratories analyses and small scale experiments. The Sandia evaluation concluded that steam explosions could occur but with insufficient energy to fail containment. Therefore, the WASH-1400 value of 10^{-2} with a reduction factor of 10 was used in the Limerick PRA. Lower values (e.g., 10^{-4}/challenge) have been identified for certain sequences at high reactor pressure. However, a value of 10^{-4} for non-LOCA events was judged not applicable since for BWR's the emergency procedures guide calls for the operator to depressurize the primary system during all non-ATWS transients.

QUESTION 3.32

Provide supporting analysis and/or calculations to show that RCIC (as stated on the bottom of p. 3-104) alone or HPCI alone is adequate for coolant inventory makeup during an ATWS condition and does not result in core meltdown.

RESPONSE

The basis for the success criteria for HPCI/RCIC during an ATWS (Table 1.3) was extrapolation of licensing design basis transient analysis to realistic conditions. Subsequent to the issue of the PRA report, an analysis was performed for the Susquehanna BWR/4. (1)

The NSSS for Limerick is very similar to Susquehanna. The only major differences are that Limerick has greater HPCI capacity, and HPCI is split between core spray and feedwater sparger in Limerick; whereas HPCI enters entirely through the feedwater sparger in Susquehanna. These differences have no effect on a case in which HPCI is failed, so the results of the Susquehanna study are directly applicable to Limerick for the RCIC-only case. The REDY computer code was used to simulate an MSIV ATWS with HPCI failure. Power values from this REDY run were then input to the SAFE-06 computer code to calculate transient water level. It was found that the minimum water level was 0.8 ft above the top of the active fuel. The conclusion based on the Susquehanna study is that RCIC alone is capable of maintaining water level above the active fuel.

Since HPCI capacity is nearly an order of magnitude greater than RCIC capacity, HPCI alone is also capable of maintaining adequate coolant inventory. The differences in HPCI injection method do not materially change the success criteria evaluation.

Reference 2 discusses fuel clad analysis for the worst case ATWS MSIV closure event with HPCI/S failure. This event was for a BWR/6 with 86 GPM SLCS. Less than 1 ft fuel uncovery was experienced. The peak clad temperature calculated for the covered portion of the fuel was 1784°F, well below the 2200°F limit for fuel integrity. The peak clad temperature in the uncovered portion of the core was even lower, since the power level was lower. Since the fuel remains covered in the BWR/4 (Susquehanna) analysis, the fuel conditions are much better, and meltdown will not occur.

REFERENCES:

1. Letter from Greg C. Storey to E. C. Eckert, January 15, 1982, "SAFE Comparisons for BWR/4 Hendrie Rule Analysis"

2. Letter from J. V. Hice to E. C. Eckert, February 18, 1982, "ATWS Fuel/Clad Analysis for Hendrie Rule Events"

QUESTION 3.33

Why are transients in which the SRV's fail to open transferred to only the large LOCA event tree?

RESPONSE

For most of the identified transients, there is a pressure surge following the MSIV closure or turbine stop valve closure. This pressure surge is limited by the opening of the SRV's. Failure of a sufficient number of SRV's to open is assumed to result in sufficiently high peak pressures to lead to a break of the primary system. While it is true that the entire spectrum of potential break sizes could be modeled, it was judged that the large LOCA represented the most severe test to the ECCS response. Transients for which the SRV's fail to open were only transferred to the large LOCA event tree to bound the problem.

QUESTION 3.34

The Emergency Operator Guidelines (Rev. 0) state in Steps LC-2.5 and SD-2.2 that if the SRV's are cycling, the operator is supposed to manually open one SRV to reduce pressure to 150 psi below the SRV set points. Page C-29 shows the reactor pressure to be cycling about the SRV set points indicating that this action has not been included in the analysis. Has this operator action been included in the event tree and fault tree quantification and if not, why hasn't it?

RESPONSE

The thermal hydraulic calculations for Class I model the case in which high pressure coolant injection is unavailable and reactor depressurization is also unavailable (i.e., TQUX sequences). As such, the operator action to reduce pressure is precluded by these occurrences in the sequences being modeled. The benefit of the operator action to perform this function has not been included in the fault tree/event tree evaluation since it is judged to be conservative to neglect this action. The action will reduce the frequency of challenges to the safety relief valves and therefore reduce the frequency of a stuck open relief valve. The negative impact of this action, i.e., inadvertent depressurization to less than 100 psi and consequential loss of HPCI AND RCIC has been evaluated and is considered a negligible contributor to HPCI/RCIC unavailability.

QUESTION 3.35

The containment event tree assumes an equal probability of containment failure occurring in the wetwell or drywell. Interpretation of the information in Appendix J, and the information presented at the February meeting results in our assumption that the containment failure always starts at the midwall of the wetwell and rapidly progresses upward to the drywell. Thus, both the wetwell and drywell will be failed simultaneously. This would remove the distinction of whether or not the failure was in the drywell or wetwell and whether or not the suppression pool has been lost, i.e., all failures rapidly lead to drywell failure and loss of the suppression pool scrubbing is irrelevant. What is the rationale for using the containment event tree sequences as presented in Figure 3.4.14?

RESPONSE

The referenced Appendix J containment analysis provides the results of a deterministic calculation of containment integrity for very specialized assumptions concerning stated dynamic pressure and temperature loadings of containment. The conclusion of the analysis indicated that one of the most likely locations of failure due to static overpressure was at the wetwell mid height.

The following items must be considered in assessing the possible containment failure modes:

- There are other postulated failure mechanisms for containment other than static overpressure. One such example is the structural failure of the concrete diaphragm floor due to core melt interaction while the containment is at elevated pressures.

- Since there is a probability that the containment may fail at secondary locations identified by Bechtel, or other locations not identified by Bechtel these incipient crack locations should be accounted for. This is important, since other containment failure modes may have a higher potential for radionuclide releases for certain accident sequences, than the wetwell failure above the suppression pool which has the greatest potential for pool decontamination.

- The analysis does not indicate that a crack initiated in the wetwell will necessarily grow to the drywell. Therefore, this assertion, while

possibly true, is covered by the postulated split in containment failure modes between wetwell and drywell.

Appendix J indicates on page J.7 "The predicted failure above 140 psig is a split along a meridianal (vertical) crack at the wetwell wall midheight. The verticle crack failure is contained to the midheight of the containment by the restraint of the base slab and the diaphragm slab. However, at the failure of the diaphragm slab connection (across liner), the wall loses its restraint at the diaphragm slab and the vertical crack will propagate very rapidly "towards the top of the wetwell wall."

It was judged that a 50-50 split was a reasonable conservatism despite the best estimate wetwell volume space crack. This will not reduce risk much (at best a factor of 2) over the case of only wetwell failures, but would increase risk by the ratio of the actual drywell failure probability to the wetwell failure probability (as much as from 10 to 1 or 100 to 1).

On page J -6 it says "An evaluation of the finite element analysis concludes that the ultimate strength capacity can be increased due to the influence of the base slab and diaphragm slab" (The increase was chosen to be from 120 to 140 psig.) "However at approximately 120 psig internal pressure, the diaphragm slab containment wall connection becomes overstressed and a general yield state occurs in the midheight of the containment wall."

It was concluded from this discussion that the increasing internal pressure would result in the formation of a liner tear along a meridianal crack at the wetwell wall midheight. This would occur at approximately 140 psig. However, a significant pressure increase would still have to occur prior to failure of the diaphragm slab. Hence the 140 psig criteria and the statement, "However, at the failure of the diaphragm slab. . . ." Indications are that such a liner tear would not propagate after pressure relief occurred due to the tear. It is stated in a WASH-1400 analysis of a reinforced concrete structure that "When this [a concrete failure extending to the liner] happens over a large enough area, the combined tension and bending will cause a blowout with the possibility of the crack propagating several feet before the sudden release of the internal pressure will cause it to stop." Appendix J also states that the crack will propagate "<u>towards</u> the top of the wetwell wall." The distance between the midheight and the drywell volume is nearly 30 feet. Since this is much larger than "several feet" and since a tear is likely to occur prior to the pressure reaching a value sufficient to fail the diaphragm slab (first quotation), it was felt with reasonable confidence that the crack would remain within the wetwell volume due to pressure relief.

QUESTION 3.36

Where in the report is the propagation of uncertainties for the dominant sequences T QUX, ATWS and LOCA documented?

RESPONSE

The characterization of uncertainties in the Limerick PRA is addressed in Section 3.8. The progagation of uncertainties for the dominant sequences is described briefly in Section 3.8.1.

QUESTION 3.37

In the table in page 3-122, Iodine released to the environment is typed as vapor. Clarify whether its deposition to the ground was considered in the subsequent CRAC analysis or not.

RESPONSE

The ground deposition of iodine vapor was considered in the CRAC analysis. The deposition velocity for iodine vapor was 0.01 meters per second. This is the same value as was used in WASH-1400. Additionally, washout ground deposition by rain was assumed to be the same as WASH-1400.

QUESTION 3.38

Provide discussions as to how the following parameters in Table 3.6.5, which were part of the inputs to CRAC, were determined.

- Time of release
- Duration of release
- Warning time
- Elevation of release, and
- Energy release

What were the values of the above mentioned parameters for the sequences $C_2\Upsilon'$ and $C_3\Upsilon'$ in the same table?

RESPONSE

Data on release parameters as given in Table 3.6.5 is extracted form several sources including LGS design WASH-1400, and the LGS in-plant consequence analysis. Reference should also be made to Table 3.6.3 of the LGS PRA Volume 1. Generally, release parameter data definitions are those utilized in WASH-1400:

A. time of release - the time at which a hypothesized release from the plant occurs relative to shutdown. This is the time at which release from the plant can begin, which is assumed to be either of the following two cases:

   1. At the initiation of gap release and core melt, for those cases in which containment may have failed prior to placing the core in jeopardy, i.e., Classes II and IV

   2. Following containment failure, for sequences in which the containment is challenged substantially after core melt initiation.

B. Duration of release - the time over which a release actually occurs. This is the time over which release of radionuclides would be dispersed and it is influenced by the rate of nuclide release as well as the time and energetics of containment failure. Although only puff releases are utilized in the ex-plant consequence model, this parameter is used to establish an effective plume width allowing for plume expansion as a result of variations in wind direction and speed over a specific period of time. For energetic and fast reactions the release duration is short since nuclide release occurs quickly following an abrupt containment failure.

For release in which containment fails after core melt occurs, containment failure initiates the release. The duration of the release is approximately the time necessary for complete discharge of the containment.

For releases in which containment failure precedes core melt, the duration of the release will be approximately the time from gap release to the end of the fuel solids vaporization.

C. Warning time - the time available for initiation of an evacuation. This is the time available between the determination of an iminent release and the time when the release occurs. An evacuation is initiated at the time the determination of an imminent release is made and the effectiveness of the evacuation is influenced by the length of time available. There is some imprecision as to the time when an evacuation would be initiated. In the LGS PRA, for those sequences in which the containment is postulated to fail first (i.e., Classes II and IV), the warning time is the time available from the decision that the containment could be in danger until the time that the release occurs. For sequences in which core melt precedes containment failure, evacuation was be presumed to be initiated at the time core melt becomes imminent.

D. Elevation of Release = the height at which a release of materials from the plant is assumed to occur. The height of release is of significance since the wind direction and speed are often quite different at an elevated location beyond the ground level wind shear. The two types of likely release location at the Limerick site include the following: a) blowout panels in the secondary containment and; b) through the HVAC system which discharges through a vent from the release treatment system, but which is not assumed to be effective for massive releases, the ducting and vent provide an exit path from secondary containment. For the LGS PRA, two elevations of wind data were available: a) at essentially ground level and; b) at 25 meters in height. These releases were divided into those which were presumed to fail a low blowout panel and those which would likely initiate failures of an upper level panel or be channeled out the vent.

E. Containment Energy of Release - the amount of energy in terms of thermal energy (sensible heat) which is available as a driving force for release of nuclides from a failed containment. Release energetics are a significant influence since a

highly energetic release tends to elevate the radionuclide plume and reduce fatalities near the plant for some accident sequences. For energetic and fast releases, the release energy is expected to be high. For releases which are the result of overpressure after the completion of core melt and substantial vaporization, the initial energy of release would be high. For releases which occur after failure of the primary containment, the available driving force of release from containment is essentially only that of the core melt process, which is small.

SPECIFIC PARAMETER ESTIMATES 3.38.2

Specific estimates of release parameters for the LGS PRA are defined as follows using the preceding general criteria and Table 3.6.3 of the LGS PRA.

A. Time of release

- in-vessel steam explosions - the time of release is estimated as the at time which the steam explosion occurs. In keeping with WASH-1400, the steam explosion for sequences where the containment is intact (such as TQUV), is assumed to occur when core melt is half completed. For cases in which containment fails first (such as Class II and IV) it is assumed that core melt reaches completion and that the steam explosion occurs in the bottom head. Estimates from RACAP for expected core melt initiation times are used to estimate the time of the steam explosion.

- ex-vessel steam explosion - again the time of release is estiamted as the time of the explosion. It is assumed that the explosion immediately follows vessel failure and the times of vessel failure are taken from RACAP.

- overpressure failure of the drywell - this data is taken directly from RACAP data. For sequences in which containment failure occurs first, the start time is based on the assumption that all makeup to the reactor core becomes unavailable at the time containment failure.

B. Duration of Release - only two values were utilized, depending on whether or not release and containment failure were simultaneous or not. Thus, all steam explosion events were assumed to be distributed over a half an hour, with overpressure events taking approximately two hours.

C. Warning time - Warning time is estimated separately for each accident class. For Classes II and IV, where containment failure initiates melt down, the warning time is the period following containment failure through the start of substantial melt down and is taken from RACAP data. For Classes I and III, the warning time, as illustrated in Table 3.6.3, is taken from RACAP as the time between the initiation of substantial melting and the time of containment failure.

D. Elevation of Release - Only two elevations of release are considered. All β, α, γ and γ' releases are assumed to fail secondary containment blowout panels high up or be released through the vent. For the γ'' failure mode, release is assumed to be at ground level.

E. Containment Release Energy - A value of 40×10^6 BTU/hr for steam explosion was extracted from WASH-1400. Containment overpressure failures preceded by core melt are assumed to be almost as energetic as early failures of containment followed by core melt. Data are again taken from WASH-1400.

Values for $C_2\gamma'$ and $C_3\gamma'$ have been added to Table 3.6.5 in Revision 3 to the PRA.

QUESTION 3.39

Provide the description of those failures that were identified by the study that could disable more than one ADS valve.

RESPONSE

Common-mode failure of the ADS valves considers causes such as:

- Loss of DC electric power (two busses)
- Maintenance or testing errors which may disable valves when demanded
- Degradation of solenoids from adverse environmental conditions during an accident.
- Loss of pneumatic supplies (two supplies plus individual accumulators)

QUESTION 4.01

Explain in detail how Figure 4.3 was generated. What exactly was used to compute the risk of Limerick at WASH-1400 composite site with WASH-1400 date methods?

How was Figure 4.2 "WASH-1400 BWR with updated methods and data" obtained?

RESPONSE

The answer to this qustions was presented at the review meeting in King of Prussia on May 14, 1982, and is repeated herein as a step-by-step procedure.

STEP 1: Figure 1 (attached) was constructed. This Figure 4.4 in Revision 3 to the PRA.

In this figure, the upper curve is the early fatality CCDF for the WASH-1400 BWR, and is copied directly from WASH-1400, as published. This curve was not reproduced by the Limerick analysis.

The lower curve is the result of the Limerick PRA analysis and represents the Limerick early fatality CCDF. This curve was generated from the Limerick Final CRAC run.

The difference between the two curves represents the difference between the Limerick PRA and the WASH-1400 PRA and is due to:

1) Differences between the Limerick site and the WASH-1400 composite site (site differences),

2) Differences between the Limerick plant and the WASH-1400 BWR (design differences), and

3) Differences between the Limerick analysis and the WASH-1400 analysis (data methods).

STEP 2: Figure 2 (attached) was constructed. This is Figure 4.1 in Revision 3 of the PRA.

In this figure, the lower curve is repeated from Figure 1, and again represents the WASH-1400 BWR, as published.

The upper curve represents the WASH-1400 BWR at the Limerick site, and was generated by running CRAC with WASH-1400 accident sequence inputs and Limerick site

data inputs (population and meteorology). Evacuation data was the same for both analyses.

The difference between the two curves represents the effects of the Limerick site to a very close approximation. The site difference is actually slightly greater (<5%), since the upper curve was generated using the Limerick CRAC code and thus includes the difference between the Limerick CRAC code and the WASH-1400 CRAC code. The code difference is small, with the Limerick code producing slighly lower results than the WASH-1400.

STEP 3: Figure 3 (attached) was constructed. This figure is not included in the Limerick PRA report.

In this figure, the lower curve is the Limerick CCDF repeated from Figure 1.

The upper curve represents Limerick without design differences, and was generated by running CRAC with the accident sequence probability inputs revised to delete the effects of the most significant differences between the Limerick plant and the WASH-1400 plant.

The same release fractions were used as in the basic Limerick run. For purposes of comparison, the upper curve is also equivalent to the WASH-1400 BWR (Limerick with design differences deleted) at the Limerick site, and using Limerick data and methods.

The difference between the two curves approximates the design differences between the Limerick plant and the WASH-1400 BWR. The exact difference would actually be somewhat greater, since many small design improvements were not evaluated and the release fractions were not changed to reflect the effect of differences between the MK I and MK II containments.

STEP 4: Figure 4 (attached) was constructed. This is Figure 4.3 in Revision 3 to the PRA.

This figure was constructed only to allow direct comparison to the WASH-1400 CCDF. In the figure, the upper curve is the WASH-1400 CCDF repeated from Figure 1.

The lower curve was not generated directly from a CRAC run. It was generated by applying the difference between the two curves on Figure 3 to the WASH-1400 CCDF, by ratio. Points on the lower curve of Figure 4 have the same ratio to the upper curve as the ratio between the lower and upper curves on Figure 3.

The difference between the two curves represents the same design differences as shown on Figure 3, and that is the only difference between the two curves. Thus,

for comparitive purposes, the lower curve represents the Limerick plant at the WASH-1400 site with WASH-1400 data and methods.

STEP 5: Figure 5 (attached) was constructed. This figure is not included in the PRA report.

This figure was constructed to provide a comparison between the effect of the data and methods used in the Limerick analysis and the data and methods used in the WASH-1400 analysis. In this figure, the curve that is lower on the left represents the WASH-1400 BWR at the Limerick site using WASH-1400 data and methods. This curve is repeated from Figure 2.

The curve that is uppermost on the left represents the WASH-1400 BWR at the Limerick site with Limerick data and methods. This curve is repeated from Figure 3.

The only difference between these two curves is the difference between data and methods. The data and methods used in the Limerick analysis produced a high CCDF at the lower consequence part of the curve, and a lower CCDF at the higher consequence end.

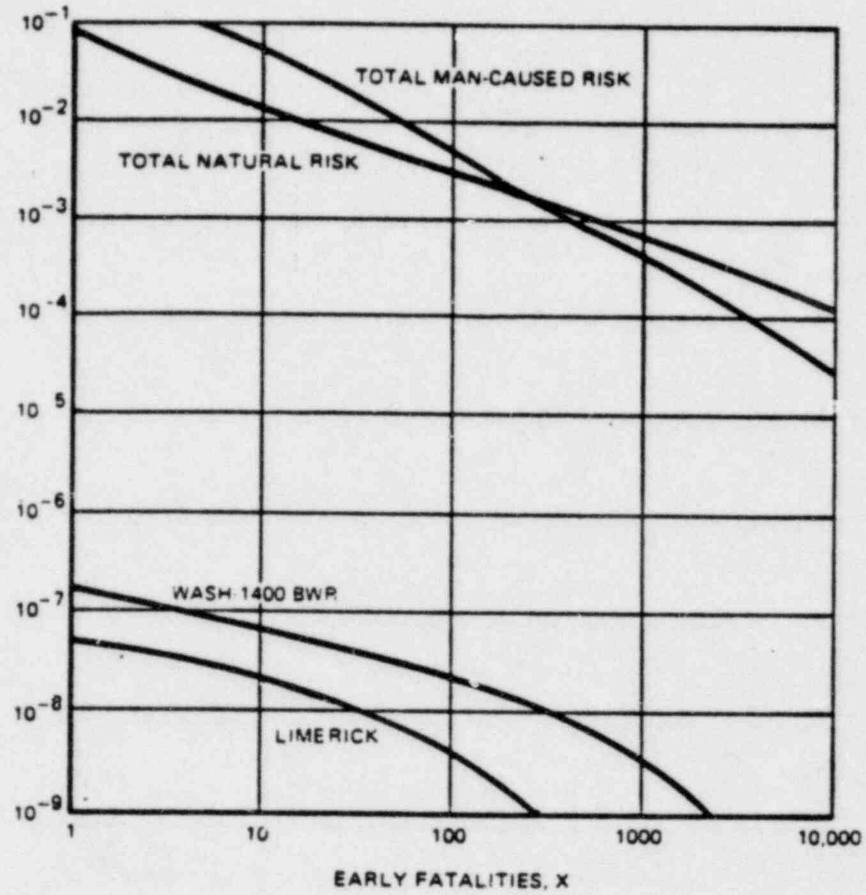
STEP 6: Figure 6 (attached) was constructed. This is Figure 4.2 in Revision 3 to the PRA.

This figure was constructed only to allow direct comparison to the WASH-1400 CCDF. The curve that is lower on the left is the WASH-1400 CCDF repeated from Figure 1.

The curve that is uppermost on the left represents the WASH-1400 BWR at the WASH-1400 site with the data and methods used in the Limerick analysis. This curve was not generated directly from a CRAC run. It was generated by applying the difference between the two curves of Figure 5 to the WASH-1400 CCDF, by ratio. Points on this curve have the same ratio to the WASH-1400 CCDF as the ratio between the two curves on Figure 5.

The only difference between the two curves is the effect of differences in data and methods. As shown on Figure 5, the methods and data used in the Limerick analysis produce a somewhat higher CCDF at lower consequences and a somewhat lower CCDF at high consequences.

FIGURE 1

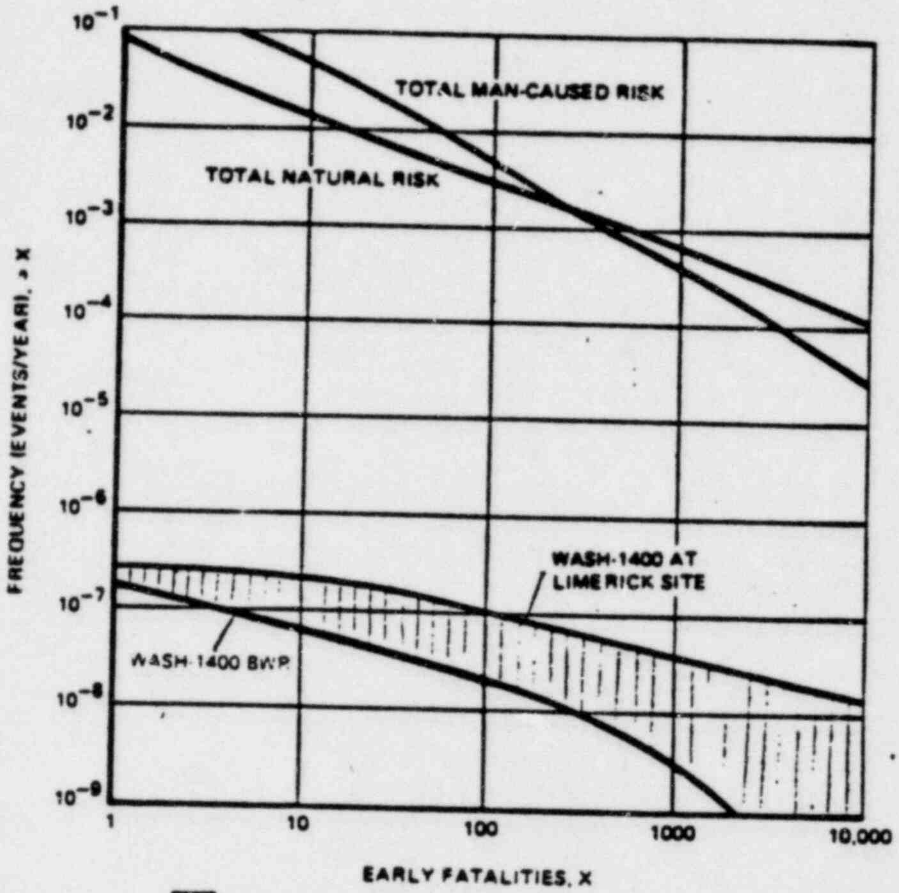


Comparison of WASH-1400 and Limerick

Rev. 4, 6/82

FIGURE 2

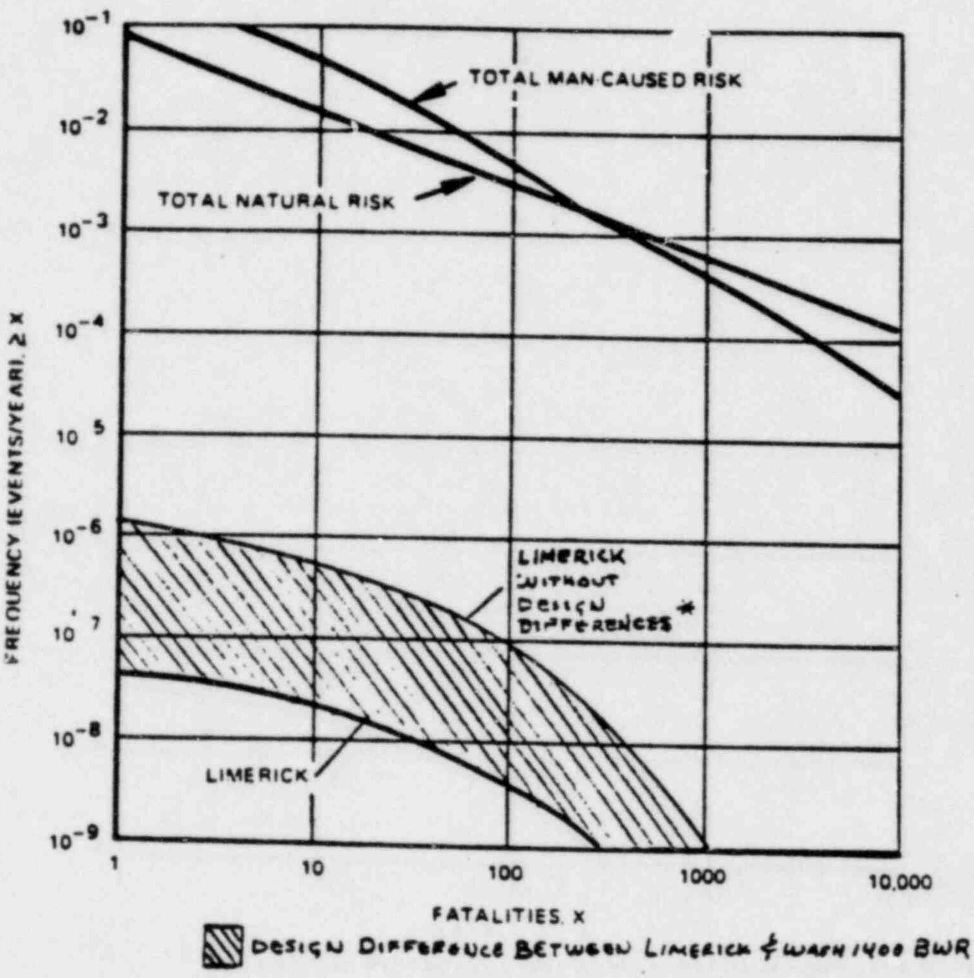
EFFECT OF SITE DIFFERENCES



SITE DIFFERENCES BETWEEN LIMERICK SITE & WASH 1400 COMPOSITE SITE.

FIGURE 3

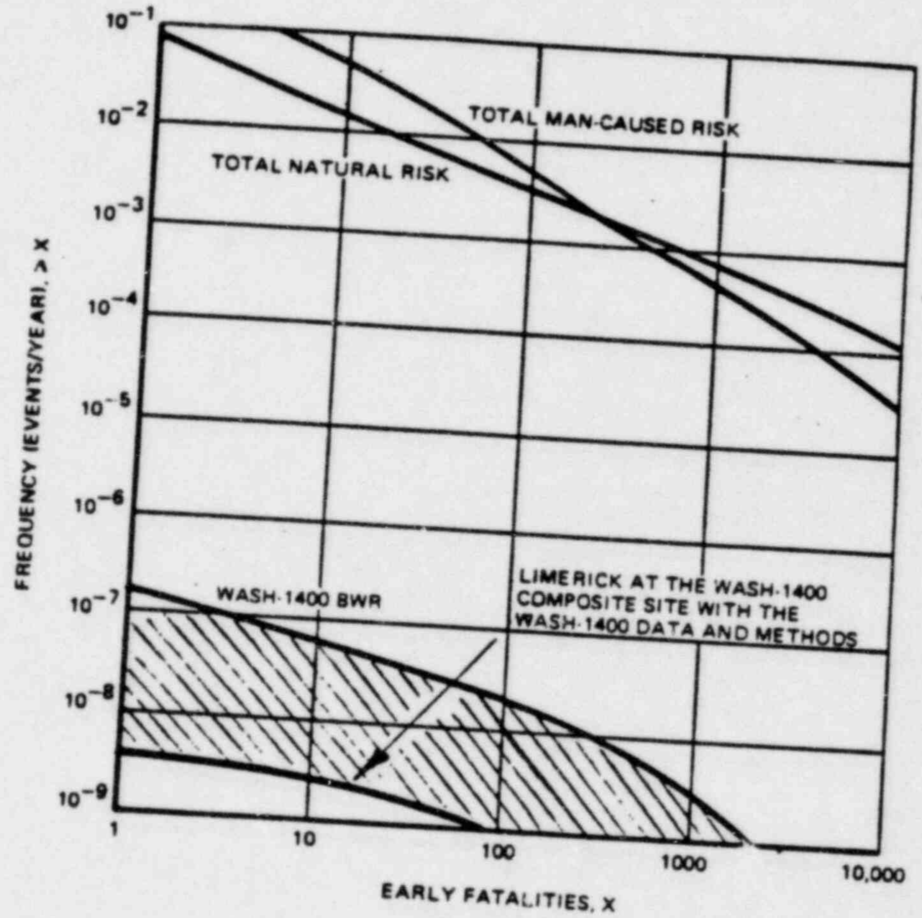
DEVELOPMENT OF DESIGN DIFFERENCES



* EQUIVALENT TO WASH 1400 PLANT AT LIMERICK SITE WITH DATA & METHODS OF LIMERICK PRA.

FIGURE 4

EFFECT OF DESIGN DIFFERENCES



DESIGN DIFFERENCES BETWEEN LIMERICK PLANT AND WASH 1400 BWR

FIGURE 5

DEVELOPMENT OF DATA & METHODS DIFFERENCES

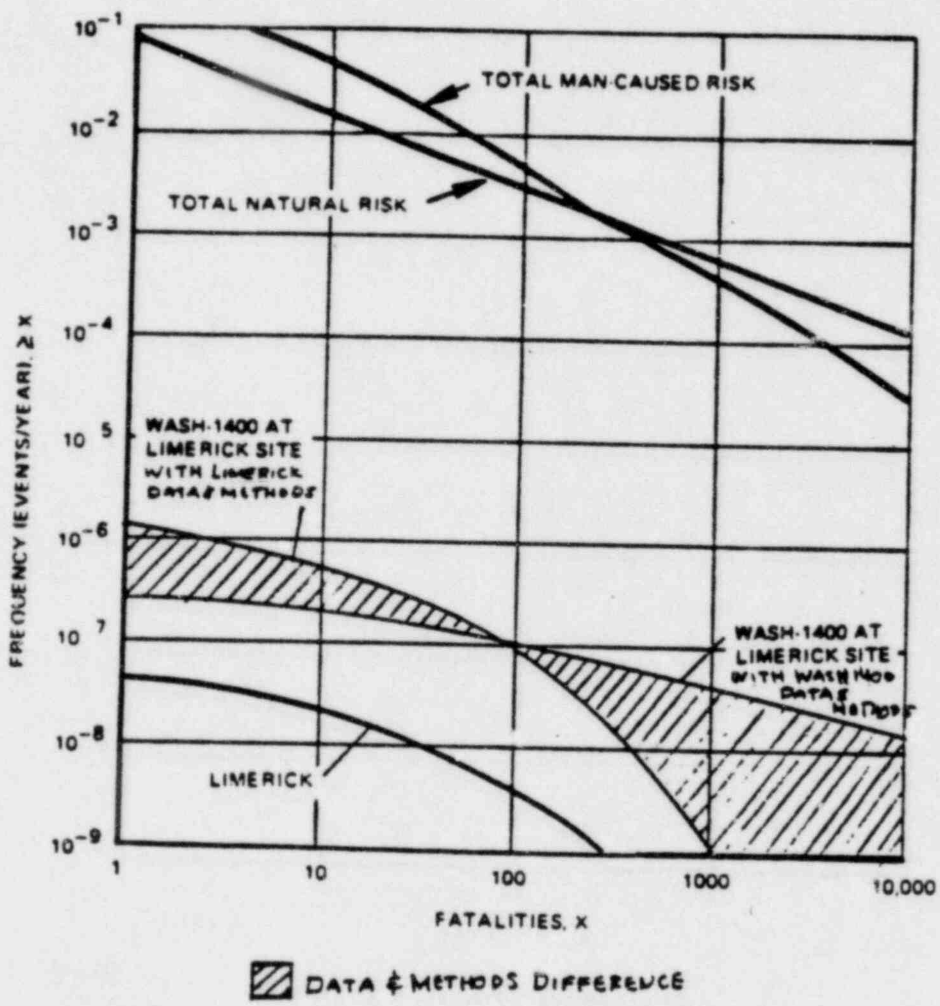
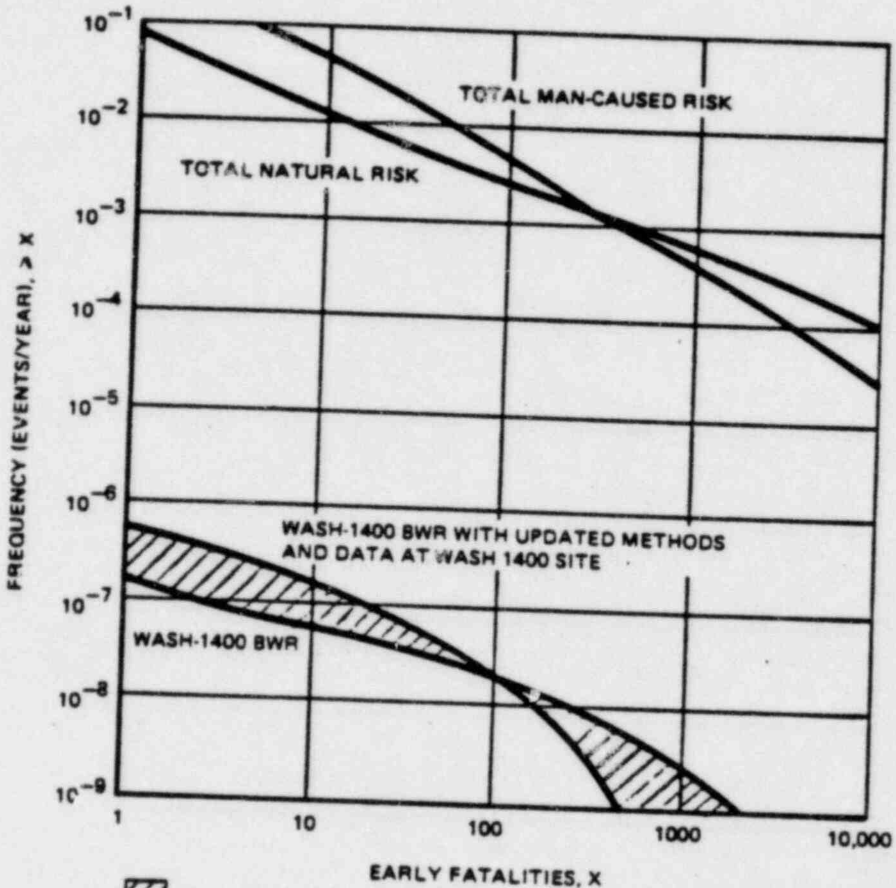


FIGURE 6

EFFECT OF DATA & METHODS DIFFERENCES



DATA & METHODS DIFFERENCE BETWEEN LIMERICK PRA (REVISED) & WASH 1400 ANALYSIS

Appendix A

QUESTION A.01

Tables A.1.2 and A.1.3 list the anticipated transients considered in the EPRI-SAI study and GE assessment. Provide clarification as to why transients #14-19 and #22 of Table A.1.2 do not appear in Table A.1.3

In Table A.1.3, under Turbine trip with bypass, transients #36 and #37 are indicated; there are no corresponding numbers 36 and 37 in Table A.1.2

In view of the EPRI survey and the GE assessment, discuss the major differences noted in Table A.1.3, e.g., loss of condenser, inadvertent opening of bypass, turbine trip with bypass. . . etc.

RESPONSE

(a) Revision 3 of the LGS PRA has provided revised tables which correct the typographical errors in the original PRA. These changes have no effect on the identified frequencies.

(b) The EPRI survey from EPRI NP-801 (July 1978) included transient initiators for 12 BWR's for operating experience prior to that time. The GE evaluation includes additional data beyond that which is included in EPRI NP-801. This additional data includes other BWR's, more recent years of operation, and the initial years of operation is homogenized in the data base.

QUESTION A.02

The electric load rejection with bypass valve failure in Table A.1.3 is listed under turbine trip. Shouldn't this be listed under MSIV closure so as to be consistent with the statement on p. 3-15? This would change the transient initiator frequencies used in the PRA for MSIV closure and turbine trip.

RESPONSE

Cases in which the turbine bypass fails closed are included in the LGS PRA as events similar to MSIV closure (conservative assumption). The error pointed out in Table A.1.3 is a typographical mistake. Table A.1.3 has been revised to accurately reflect this conservatism.

Since there are no recorded cases in the data base for electric load rejection with turbine bypass failure, there is no change in the initiator frequencies used for event tree quantification.

QUESTION A.03

In the footnote on page A.12, a statement is made to the effect that due to the controlled nature of manual shutdown, there is an increased reliability of feedwater to maintain reactor inventory. What is the qualitative and quantitative basis for such a statement?

Provide clarification for the statement: "However, coolant injection functions, ATWS, and LOCA sequences are not affected by these initiators when they are quantified (p. A-12)."

RESPONSE

(a) Transients (i.e., scrams) lead to the abrupt interruption or diversion of normal power conversion operation. Specifically, MSIV closure and loss of offsite power initiators lead to the immediate loss of the main condenser as a heat sink and feedwater as a coolant makeup source. Relative to these initiators a manual shutdown has a relatively benign effect on these functions, and the reliability of FW and PCS for manual shutdown are significantly higher than for these isolation initiators.

(b) The discussion in Appendix A.1 is provided only to give the reader a qualitative assessment for the impact of including a unique event tree for manual shutdown initiated events. The quoted statement is an interpretation of the impact of the manual shutdown event tree which is:

- ATWS is judged not to be a contributor to risk during manual shutdowns since a scram challenge is not induced. Scram challenges induced by improper manual shutdown procedures are included in the transient frequencies quoted in Appendix A.1.

- A LOCA is not considered as a potential outcome of sufficiently high probability following a manual shutdown to affect the LOCA event trees. The likelihood of a pressure or temperature transient sufficient to induce a LOCA is quite small.

- Coolant injection is explicitly quantified in the manual shutdown event tree. The impact on core melt frequency is approximately two orders of magnitude less than the total core melt frequency.

QUESTION A.04

On the discussion of reported failures for all sizes of piping, the sum total of all the failure percentages comes to 64.4% (p. A.13); furnish information on the remaining 35.6%. Given the magnitude of the balance in failures (35.6%) versus the largest failure category (25.1%), how does one justify the accuracy of the data if the 35.6% is not included in the data base?

RESPONSE

The question refers to the compilation of the causes of pipe failures. This comilation is provided as reader information and is not explicitly used in establishing the frequency of pipe failure. The crucial information used in establishing the LOCA initiating frequencies is the number of failures recorded in the industry by pipe size. By way of further explanation concerning the lack of descriptive information on the cause of 35.6% of the piping failures it must be realized that this information is provided through LER's. The lack of descriptive information is due to either a failure on the part of the reporter to cite the cause or a lack of ability to identify the cause. However, since the cause is not used directly in the calculation of initiating frequency, the uncertainty introduced because of a lack of information on the cause of some failures is minimal.

QUESTION A.05

In addition to pipe rupture, there are other causes which could lead to LOCA, for instance, valves failed open, failure of recirculation pump seals. Were they addressed and properly included in the analysis as LOCA initiators?

RESPONSE

The LOCA initiating frequencies used in the LGS PRA are derived from operating experience data. The data examined included all failures which could lead to a loss of coolant accident inside containment. Therefore, based upon the data source used for these initiators they are considered to characterize the spectrum of possible failures which could result in the release of primary coolant to the containment. For example:

Small LOCA's	• leaks in large diameter pipes • instrument line break
	• recirculation pump seal failure
	• CRD hydraulic line breaks
	• Leaks from valves or other equipment
Medium/Large	• large flow from pipe leaks
LOCA	• large diameter piping breaks
	• Pump casing leaks/breaks

In general the failure of a valve to close (i.e., failed open) does not lead to a LOCA condition. There must be another failure in addition to the failed open valve. Stuck open relief valves are considered in the analysis, however. Since they are piped directly to the suppression pool, they are treated differently than LOCA's which release directly to the drywell.

In addition to LOCA's developed as a class of initiators, pipe rupture is also included in the fault tree development. In the fault tree quantification the conditional probability of this input takes into account not only the pipe and weld failures but also external ruptures of major component i.e., pumps and valves. These failures in ECCS equipment do not lead to LOCA's but rather to leaks in the ECCS equipment.

QUESTION A.06

The use of a 10% reduction of the probability of pipe rupture for the probability of LOCA seems to be a rough estimate (p. A-16). Are pipe failrue probability data given per unit length? Are there any data on primary piping ruptures? If yes, have they been compared to the 10% estimate?

RESPONSE

(a) The estimate that 10% of the large diameter piping is inside containment and subject to a large LOCA (i.e., LOCA sensitive) was the assumption made in WASH-1400.

The probabilities of a large and medium LOCA have been calculated by assuming that only 10% of the pipe in a nuclear plant is potentially sensitive to producing a LOCA.

A comparison with other PRA's shows:

	Calculated Mean per Reactor Year
LGS	4 E-4
WASH-1400	3 to 9E-4 (estimated here for comparison)*
RSSMAP	1E-4 (point estimate)
Zion	9.4E-4 (Section 1.5.2)

(b) The initiator frequencies for LOCA event trees are not given a per unit length basis. Pipe failure probabilities for fault tree inputs are calculated using failure rates (per hour-ft) as taken from Reference 3 of Question A.07.

(c) The data on primary system piping ruptures is limited to small LOCA's.

(d) No comparison has been made.

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* Function of the assessed error factor

QUESTION A.07

Table A.1.6 gives the probabilities of a LOCA for various cases. Discuss the method, analysis and criteria used in the selection of the Limerick values.

RESPONSE

The following facts must be considered in the assessment of LOCA probabilities: 1) Based on estimates from other sources (e.g., RSS, Bush), a pipe rupture leading to a large LOCA is expected to occur once every 10,000 plant year. With this estimated frequency and the fact that U.S. cumulative reactor experience used was only approximately 270 reactor years, it is difficult to assess the probability of a LOCA accurately by only considering LOCA sensitive pipes because sufficient operating experience has not accrued; 2) there have been pipe ruptures occurring in high integrity piping in secondary systems of nuclear plants (see Table A.1.10). Based upon these rupture failures, a probability of a pipe rupture failure in LOCA sensitive piping can be estimated using the fact that LOCA sensitive piping represents approximately 10% of the piping in a reactor plant.

The following discussion specifies how each of the estimates of LOCA initiator were derived from the observed pipe failure in nuclear operating experience.

Large LOCA

There has not been a single large pipe break (LPB) event in about 150 reactor-years of BWR operating experience. Estimates of failure probability based on chi-square distribution are as follows:

At 50% confidence level,

$$\lambda_{LPB} = \frac{1.386}{2\times150} = 4.6\times10^{-3}/\text{Rx Yr}$$

$$\lambda_{LPB}\ (\text{LOCA}) = 4.6\times10^{-3} \times 0.1^{*} = 4.6\times10^{-4}/\text{Rx Yr}$$

At 90% confidence level,

$$\lambda_{LPB} = \frac{4.605}{2x150} = 1.5x10^{-2} \text{ Rx Yr}$$

$$\lambda_{LPB} \text{ (LOCA)} = 1.5x10^{-2} \text{ x } 0.1^{*} = 1.5 \text{ x } 10{-}3 \text{ Rx Yr}$$

The 50% confidence level value used as the mean value estimate of the large LOCA initiating frequency.

Medium LOCA:

The evaluated frequency of a medium LOCA is derived from the medium size pipe break frequency. There is one instance of a 4" diameter pipe break in 125 reactor years of domestic BWR operating experience in this data base. A Bayesian approach is adopted to evaluate medium size pipe break frequency per reactor year.

Assuming an exponential pipe break model, we make use of the natural conjugate prior, gamma distribution for the pipe break frequency .

Let $\pi(\cdot)$ be the gamma prior probability density of , i.e.,

$$\pi(\lambda) = \frac{b^{a} \lambda^{a-1} e^{-b}}{\Gamma(a)}$$

It is shown in [1] that the posterior probability density (./D) is also gamma, and

$$\pi(\lambda/D) = \frac{(b+T)^{a+k} \lambda^{a+k-1} e^{-(b+T)}}{\Gamma(a+k)}$$

where k is the number of observed failures and T is the total time on test.

---

* 0.1 = LOCA sensitive factor.

It is noted that k and T together are sufficient.

The posterior mean of $\bar{\lambda}$ is

$$\int_0^\infty \lambda \pi(\lambda/D)\, d \quad = \frac{a+k}{b+T} \quad .$$

We have assumed a uniform improper prior, i.e., a = 1, b = 0; then the posterior mean for $\bar{\lambda}$ is $\frac{k+1}{+T}$.

Applying the result above to our data, we have the posterior mean of the medium size pipe break frequency for the entire plant as

$$\frac{1+1}{125} = 2x10^{-2}$$

However, as assumed in WASH-1400 [2], only 10% of the pipes are LOCA sensitive and the medium LOCA frequency is evaluated to have a mean of $2x10^{-3}$/reactor-year.

Small LOCA:

The initiating frequency of a small LOCA has been conservatively estimated using the available data from all LWR primary systems. Specifically the three incidents cited in the table below are used along with the corresponding accumulated reactor operating experience to calculate a mean estimate of the initiating frequency. By use of the Bayesian analysis presented above, we can derive similarly the posterior mean of the initiating frequency of a small leak/rupture in the primary system as follows:

$$\frac{3+1}{270 \text{ Rx Yr}} = 1x10^{-2}\text{/Reactor Year}$$

The probabilities of a LOCA in a BWR are summarized in Table A.1.6.

REFERENCES

[1] Barlow, R. E. and Proschan, F. "Inference for the Exponential Lift Distribution" ORC79-16, Operations Research Center, University of California, Berkeley, December 1979.

[2] WASH-1400 "Reactor Safety Study" U.S. NRC, October 1975.

[3] Bush, S. H., "Reliability of Piping in Light-Water Reactors," Nuclear Safety, Vol. 17, No. 5, September-October 1976, pp. 568-579.

INCIDENTS OF PRIMARY SYSTEM PIPE FAILURES IN THE U.S.

Surry 1, November 1972 - During a reactor startup a resistance temperature detector (RTD) in a hot leg became dislodged causing 30,000 gallons of primary coolant to be discharged to the containment. Safety injection was initiated manually (all three pumps operated).
CAUSE: Swagelock failure
(Referred to as a small LOCA)

Beaver Valley 1, June 1976 - 28% power

A failed RCS pressure transmitter sensing fitting resulting in the discharge of 5,300 gallons of primary coolant. The leakage rate was within the makeup capacity of the discharge pump.
CAUSE: Swagelock fitting failure

Brunswick, November 1976 - Reactor Coolant Pump seal failure

Table A.1.6

PROBABILITY OF A LOCA

PIPE SIZE	WASH-1400 (MEDIAN)	VALUE USED IN THE LIMERICK PRA
Large Pipe $>4"\phi$	1.0×10^{-4}*	4.0×10^{-4}
Medium Pipe $\leq 4"\phi$	1.0×10^{-3}*	2.0×10^{-3}
Small Pipe $\leq 1"\phi$	1.0×10^{-3}*	1.0×10^{-2}

* Mean values are approximately three times larger.

Table A.1.10

REPORTED LARGE DIAMETER PIPE BREAKS IN U.S.
NUCLEAR POWER PLANTS

Date of Occurrence	Approx. No. of Operating Years	Reactor	Type	Failure Location	Pipe Size	Probable Cause	Reactor Status	Characterization Of the Event by Utility
FEEDWATER SYSTEM								
11/73	1	Indian Pt. 2	PWR	Pipe-HAZ	10"	Pressure Surge	Auto Scram	"180° Circumferential Break"
8/75	3½	Quad Cities 2	BWR	4" x 6" Reducer	4"	Vibration	Manual Scram	"Break"
STEAM PIPING								
10/73	½	Ft. Calhoun	PWR	Expansion Joint	10"	Vibration	No Shutdown Required	"Rupture"
6/71	1	Robinson 2	PWR	Pipe Nozzle	6"	--	During Shutdown	"Complete Failure"
6/72	½	Turkey Pt. 3	PWR	Pipe	10"	Surge	During Shutdown	"A Failure"
8/72	½	Surry	PWR	Pipe Nozzle	4"	Surge	During Shutdown	"Nozzle Separated from Pipe"

QUESTION A.08

Table A.2.1 compares median and mean values. It is further stated on p. A-22 that "mean values of failure rates used in WASH-1400 appear lower than mean values reported in other sources". Where is this shown?

RESPONSE

Table A.2.1 is an attempt to present comparable values used as point estimates of component failure rate. The WASH-1400 values have been converted from the medians presented in WASH-1400 to means*, so that they can be consistently compared with the values from the GE and NRC LER data which were interpreted as mean values.

P. A-22 contains the quote "Much of the data from the three sources ** are similar; however, the mean values of failure rates used in WASH-1400 appears lower than mean values reported in the other two data sources mentioned above." This conclusion is based upon a comparison of the mean values presented in Table A.2.1.

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* In order to convert from medians to means, the formulas presented in Appendix A.2 and the error factors given in WASH-1400 are utilized.

** NRC, WASH-1400, GE as noted in Table A.2.1.

QUESTION A.09

Provide more detail on the modeling of how a component could fail to run for the duration of the accident. Also, explain the basis or justification on how 20 hours was selected (p. A-29).

RESPONSE

(a) The principal contribution to the component failure probability as provided in the available data sources is attributed to failure to start and run initially. Since the failure probability is established on challenges and tests which occur over short test times or run times, we must ensure that potential failures which may occur due to extended run times are accounted for. Therefore, the failure rates per hour times an assumed operation time of 20 hours is used to estimate this contribution to the component failure probability.

(b) The 20 hour time period is chosen based upon two considerations:

(1) Within this time, it is judged that sufficient expertise and equipment can be brought to bear to augment the operating piece of equipment.

(2) The PRA analysis terminated at hot stable shutdown. An estimate judged to be conservative of 20 hours was used to encompass all sequences.

QUESTION A.10

The fact that values in Table A.2.4 agree does not necessarily mean that the 4 cases can always be indiscriminately used. What criteria was used in the selection of cases in the LGS/PRA?

RESPONSE

The LGS PRA was conducted using generic data sources to quantify the failure probability of componenets as input to the fault trees. Using these generic sources of data there are several methods which may be used to calcualte a component failure rate. The examples in Table A.2.4 are used only as an illustration to indicate that the application of Bayesian updating will not in general drastically alter the numerical point estimates of component failure rates. The frequentist approach was used in the application of the generic data sources for the LGS PRA.

The results of several methods of combining existing data or "states of knowledge" were provided in Table A.2.4. As shown, the mean value for the particular component chosen (pumps) is relatively insensitive to the method of combining existing generic data.

The above example demonstrates that data from individual plants can be characterized by a common distribution model, and the total population of plants combined in a Bayesian fashion, to determine a posterior distribution representing the current state of knowledge. However, this method, which may be rather time consuming, produces point value results which are similar to the classical statistical approach. While the establishment of a specific method of combining component data from various sources is desirable, there are a number of variations in the currently available basic data used in the quantification of the accident sequences. These variations may tend to obscure any usefulness which could be gained by establishing a rigorous method of combining existing data. These potential variations in the data are due to such items as:

1. Lack of specificity as to the function/type of component (e.g., main circulating water pump or RHR pump). All types of pumps are treated together because of the very small population available.

2. Age of the components is generally not considered.

3. Variations which occur among different manufacturers are not included in the LER reporting scheme.

4. Local plant test and maintenance procedures, training programs, and management/personnel factors may vary.

Compared with variations arising from the above listed items which may be encountered at a specific plant, the calculated "expected values" from Table A.2.4 show very small differences which do not warrant an extensive Bayesian analysis. See Appendix F for further discussion of the statistical treatment used in the analysis.

(p. A-62, top paragraph, last sentence). This sentence states that "components involved in the room cooling and ventilation are not included in the estimate of maintenance unavailability". Page B-5 (bottom) states that "cooling must be available to maintain acceptable temperatures in the HPCI compartment: for long term operation. Is there an inconsistency in ignoring the cooling system? The same comment applies to RCIC (see p. B-8).

RESPONSE

The method chosen for the quantification of safety system unavailability and even tree sequences is through the use of Peach Bottom operating experience. The technique used in the calculation of maintenance unavailability is judged to adequately incorporate any potential unavailability due to maintenance outage involving room cooling equipment.

The maintenance unavailability calculated on a per system basis in WASH-1400 was found to be too high based upon PECo. experience at Peach Bottom which is typical of operating experience at BWR's. Therefore, in the quantification of system level fault trees, the latest Peach Bottom operating experience information was used to determine the maintenance unavailability for each system.

Room cooling systems may be required for operation of ECCS equipment over a long period of time and therefore the data upon which the analysis was performed would include any repair incidents due to these components. The analysis has included additional system unavailability in the calculated values of system unavailability to account for possibly unreported items.

In summary, it is estimated that the calculated maintenance unavailability adequately incorporates any unavailability due to the room cooling support systems. It should also be noted that even without room cooling, ECCS operation can be successfully performed for a period of time greater than 2 hours.

QUESTION A.12

In the fault tree model of the diesels (p. A-72), not all the dependencies are shown, for instance, based on Table A.4.1 if one diesel is out of service, the LPCI, both core sprays, remaining diesel generators and the containment cooling systems have to be operational. How are these dependencies accounted for in the fault tree model?

RESPONSE

System dependencies specified in the Limiting Conditions of Operation (LCO) as shown in Table A.4.1 are included in the individual system fault trees and combined using Boolean alegebra. For example, see Sheet 13 of Figure 7, Core Spray Fault Tree.

In the Limerick PRA fault trees, the dependency among diesel generators is modeled in the Electric Power System whereas the dependency of diesel generators on other safety systems is modeled in the fault trees of the other safety systems. The maintenance unavailability models of the LPCI, both core sprays and the containment cooling systems all incorporate diesel generator maintenance. For example, diesel generator maintenance is incorporated in the LPCI fault tree on page 19 through the transfer ECUM4.

QUESTION A.13

The average demand of 65.4/diesel-year seems to be low, compared to the data given for Zion and Cook. We could not verify this number because of the lack of necessary data regarding Plant X. Provide additional information (p. A-91).

RESPONSE

The average diesel generator demand of 65.4/diesel-year was not used in the LGS PRA. Even though this value is not used, for completeness of the discussion of A.5, the following data used for plant "X" is provided:

No. of demands/diesel year = 30.2

No. of diesel years = 21.8

The average of 65.4 demands/diesel year is calculated as a weighted average of the five plants (Zion 1 & 2, Cook 1 & 2, and Plant X).

QUESTION B.01

A statement is made (top paragraph, p. B-91) about the large uncertainty of bringing the reactor from hot to cold shutdown. Is this consistent with the statement at the bottom of page 1-17 which states that the operation is of a routine nature?

RESPONSE

The actions during the period between hot and cold shutdown are very flexible. These actions taken during this period may vary widely depending on the purpose of the shutdown, the condition of the plant, the load demand, the time of day, and many other factors. In many cases, the plant is restarted without going to cold shutdown.

Even though there is wide variability of actions, the operations are "routine" in the sense that the plant is in a stable condition, and there is much time available for whatever actions are indicated. There may indeed be "uncertainty" as to what actions are indicated because of the variability of the factors mentioned above.

QUESTION B.02

Explain why ADS pressure sensor is not included in Table B.5.5 (p. B-53).

RESPONSE

The ADS drywell pressure sensors were not included in the Table B.5.5 due to an error of omission in the typing of the table. The corrected table is included in Revision 4.

QUESTION B.03

Figure B.9.2 depicts a generic fault tree of a MOV. How are redundant demands on a MOV modeled? If one assumes a situation in which the first demand is to close the valve and the second demand is to open the valve, how is this modeled in the study?

RESPONSE

The MOV's modeled in the LGS PRA for operation in the short term, e.g., for coolant injection, are such that the MOV's are demanded to remain as is or to change state. In several identified cases multiple demands on MOV's to change state several times are included (e.g., HPCI and RCIC operation). These multiple demands are modeled such that the failure probability of the valves to answer the subsequent demand challenges is less than the initial demand. Since there is no data available which can be used to characterize a subsequent demand failure probability during and accident sequence, an engineering judgment was made that the failure probability for the second demand was 1/2 that for the initial demand, given that it operated successfully during the first demand.

QUESTION B.04

Have DC failures been considered and if so, did they include operator and maintenance contributions?

RESPONSE

"Loss of DC Bus" (EDC125A, B, C, D) is presented on Sheet 7 of Figure 8, Electrical Power System Fault Tree, and is applied to all system fault trees as appropriate.

Battery maintenance is included in Battery Set Unavailability (EBV1A1HWI, B, C, D), also on Sheet 7 of Figure 8. No other operator action is modelled for DC power.

QUESTION B.05

There are more than 600 penetrations in the containment. Has their ability to withstand pressure up to 145 psi been evaluated? Would some of these penetrations, for instance the electrical penetrations, yield to excessive leakage under elevated temperature and pressure environments?

RESPONSE

The containment penetrations have been evaluated and found to withstand elevated temperature and pressure environments.

The manufacturer of the containment electrical penetration assemblies has documentation for penetration assemblies similar to those used for Limerick that have passed testing in environmental conditions including a simulated LOCA exposure of 340°F temperature with a 180 psi peak pressure.

Sections 4 and 5 of Appendix J to the PRA discuss mechanical penetration and their ability to withstand extreme environmental conditions.

Appendix C

QUESTION C.01

How was the metal water reaction of the fuel bundle zirconium channels considered?

Which core melt model and metal-water reaction model is assumed?

RESPONSE

The model chosen is the core meltdown model A, which assumes that the heat in the molten pool is transfered downwards. The core material slumps downwards once predicted to be molten (Reference 1). The melt fron progression downwards is maximized, as was down in WASH-1400.

The cladding-water reaction is calculated from either the Baker rate law or a gas phase diffussional model, whichever predicts the limiting rate. The model is the same as that used in NURLOC (Reference 2). The reaction is stopped: a) once the fuel rod node is predicted to be molten, b) if the steam is totally consumed, or c) if the zirconium cladding is totally oxidized.

(1) WASH-1400, Appendix VII

(2) Walters, E. T., and Genco, J. M. "NURLOC 1.0 a Digital Computer Program for Thermal Analysis of a Nuclear Reactor Loss of Coolant Accident" BHI 1807, July 1967.

QUESTION C.02

The statement is made (top paragraph of p. C-16) that HPCI is allowed to stay on even after high exhaust pressure trip point is reached (i.e., the operator overrides the interlock). Does the operator have enough time to do this since the containment fails in less than 50 minutes? What were your assumptions?

RESPONSE

The stated assumption is conservative; i.e., continued HPCI operation without containment heat removal results in containment failure prior to core melt (Class IV). In the Limerick PRA, another conservative assumption was made: that in all cases, containment failure would lead to loss of coolant injection (See p. 3-27 and Footnote #11, p-3-172) and eventual core melt. For completeness, the PRA modelled the possibility that HPCI would continue running until containment failure.

QUESTION C.03

What is the basis for assuming that the diaphragm floor fails at 2/3 of the floor penetration (70 cm) (p. C-19)? What happens to the core after floor failure?

RESPONSE

(a) The diaphragm floor was assumed to fail when the rebar, the major load bearing structural member, was contacted by the melt. There are two groupings of the rebar in the floor, at about 1/3 and 2/3 of the floor thickness. It was felt that contact of the second grouping would result in complete loss of load capability. This could result from rapid heating of the steel and loss of function. The steel would previously have been insulated by the concrete which has low thermal conductivity. INTER calculations, (core concrete attack model used in the Limerick PRA), indicated that a very short thermal boundary layer existed (a few centimeters) such that heating of the rebar was unlikely to occur before this time. Had the floor been assumed to fail earlier, the non-condenssible gas generation loading on the containment would have ceased. Additionally the vaporization release mechanism would have terminated, no longer providing the only mechanism for the release of the long lived Lanthanum group (includes Pu). If the floor had been assumed to fail at 1/3 penetration, or the time of first rebar contact, the timing of the accident would have been accelerated only 1/2 hour. Therefore, warning times for emergency response would have changed only 1/2 hour out of 6 hours. This can be demonstrated to be insignificant. Therefore, it is felt by the analysts that this 2/3 floor penetration failure assumption is conservative with respect to non-condensible gas loading on containment, and the release of some long lived radionuclides. Further it is insignificant with respect to overall warning times and accident response times.

(b) After floor failure, the combined mass of core debris and concrete slag is assumed to fall into the wetwell pool. This results in rapid steam generation. No credit, however, was taken for the impact of the concrete slag on fragmentation. Increasing the viscosity, which occurs from intermixing of the concrete, has been shown to suppress rapid fragmentation. Therefore, the resulting containment response is likely to be conservative. In the CORRAL analysis, a puff release of the cotnainment inventory of radionuclide occurs at this point in the accident sequence. Since the total core debris is assumed to fall into the wetwell pool, the vaborization release

from core-concrete interaction would be terminated, and radionuclide release to the environment would correspondingly cease.

QUESTION C.04

Please provide the modification that was done to INCOR which tracks the water level in the vessel and assigns 30% power to covered nodes and decay heat power to uncovered nodes (p. C-15).

RESPONSE

The water level in the vessel as calcualted in BOIL was brought into the main routine as a common block. During core uncovery, the water level, Y, relative to the BAF, was used as the measure of the fraction of core which would be covered; thereby producing ~ 30% power. The core power level above the decay heat level is input as a table of energy source vs. time in CONTEMPT. This energy source corrected by the fraction of the core which is covered, is then added directly to the primary coolant.

QUESTION C.05

Provide all data for input decks of RACAP including documentation of calculations performed in order to obtain the required inputs.

RESPONSE

These proprietary data supplied under separate cover on June 4 , 1982.

QUESTION D.01

Provide the basis for the assumption that 98% of the secondary containment building air flow is filtered and 2% is not (p. D-13)?

RESPONSE

The values of 98% and 2% were obtained from modeling using the INCOR code package. The SGTS was modeled as a normal leakage by the CONTEMPT containment analysis subroutine and the design basis leak rate was modeled as a flow through an orifice. In this way a large break in the primary containment lead to a calculated pressure driven flow through the secondary containment which is partitioned as flow through the SGTS and through the secondary enclosure leaks. This was found to result in 98% and 2% partitioning, for flows which do not lead to secondary enclosure pressures beyond the pressure capacity of the blowout panels.

QUESTION D.02

a) Justify the use of the CORRAL Value instead of the REACT value for the Tellurium release fraction.

b) How sensitive are the consequences to this assumption (p. D-28)?

RESPONSE

a) Radionuclide release fractions from the containment to the atmosphere for all classes of accident sequences and failure modes were obtained from CORRAL as stated on page 3-129 of the LGS PRA. The REACT calculations were used for only two purposes: (1) to compare REACT with the CORRAL calculations, and (2) to estimate the slow leakage cases and the effect of the secondary enclosure.

In general, as stated on page D-28 of the LGS PRA, the iodine and tellurium release magnitudes are comparable. For one accident class and containment failure mode, TQUV γ ($C_1\gamma$) the tellurium release for REACT was a factor of 30 higher. The CORRAL value was used in the CRAC calculations since the Te release fraction as calculated by REACT was considered to be over-estimated.

The REACT calculations assumed average containment conditions and intercompartment flow rates, as well as leakage rates to the atmosphere. On the other hand, CORRAL interpolates between data points to get instantaneous reactor conditions and flow rates as a function of time. Therefore, for cases where the flow rates are initially high, and rapidly decreasing, the REACT calculations would tend to overestimate the predicted radionuclide release.

b) Since the Class I accident (dominated by TQUV) is probabilistically significant, there could be an effect on the consequences by using the CORRAL values instead of the REACT value for Tellurium. However, as stated above the REACT calculations for TQUVγ overestimate the Te release.

The WASH-1400 consequence analysis indicates that Te can be a significant contributor to short term dose, (i.e., early and continuing effects). However, a factor of 30 difference on the release magnitudes between REACT and CORRAL calculations for Te would not result in the same order of magnitude difference in the results. It is estimated that a factor of 3 increase in early and continuing dose effects may result by using the REACT value instead of CORRAL value.

QUESTION D.03

(a) In the tabulation of nuclide species, iodine is listed as elemental and/or organic. In the fission product transport calculations, however, iodine is assumed to be CsI (Appendix D). In the estimation of SGTS effectiveness, different DF values for elemental and organic forms are quoted (Appendix D). Please identify what forms of iodine were assumed in what proportions, and why; then determine decontamination factors consistently for this form(s).

(b) Indicate the applicability of the decontamination factors of Table 3.6.4 with respect to fission product element and physical/chemical form.

(c) Section D.2.3.1 states that the SGTS was assumed to achieve certain decontamination factors independent of the accident sequence. The evaluation of filtration systems as ESFs in NUREG-0772, in contrast, indicates susceptability of these systems to plugging as a result of high aerosol loading for some sequences. Discuss the particulate loading capability of the SGTS, and compare with the expected aerosol loading (including non-radioactive materials) for the various accident sequences.

(d) What is the basis of the statement that the three conditions listed on page D-8 "dictate" the degree of suppression pool decontamination? Indicate the relative importance of such variables as degree of subcooling, gas composition (non-condensible gas fraction), gas flow rate, and iodine concentration.

(e) On page D-9 it is stated that the reason for increasing the saturated pool DF for CsI is the greater solubility of CsI. Since saturated pool DFs are limited by reduced surface interaction, as stated on the previous page, explain how a difference in solubility of highly soluble compounds can produce an order of magnitude change in DF.

(f) Since any cesium iodide reaching the suppression pool is in particulate form, explain why CsI is treated differently than other particulates.

(g) Quantify the "additional credit" in decontamination factors discussed on p. D-9 and explain how this additional credit is achieved by pH, particularly in view of the discussion of CsI in the previous paragraph.

RESPONSE

(a) Since the question of the form of iodine was raised during the study an examination of both forms was made. The CORRAL calculations assumed elemental iodine. A study of the

decontamination factors expected for subcooled and saturated suppression pools and for the SGTS indicated that elemental iodine and CsI, or all particulates, should receive the same DF's. While CsI and I would involve different natural removal mechanisms, natural removal was secondary to the above mentioned considerations. Therefore, the study results on the conservative calculational basis employed should be assumed to be independent of iodine form. If a best estimate analysis were performed one would expect the iodine form to be more important and CsI would be likely to further reduce the release fraction, particularly for an intact containment during release.

(b) As was indicated by the above response, the decontamination factors were applicable to all physical/chemical forms of iodine. Data was used for elemental iodine and particulates to determine those DF's. The DF's apply for all iodine and particulates, but not to the inert gases.

(c) It was expected that aerosol loadings to the SGTS would be substantially less than the estimates in NUREG-0772. These values were derived for an in containment aerosol loading. The aerosol loading for these accident sequences, is where the flow path from the primary system to the containment is through the suppression pool, expected to be lower due to suppression pool scrubbing. Additionally leakage out of the containment could be limited due to plugging. Using a crack width and length used in the study for containment failure and equation ? in reference 1, a few tens of kilograms of aerosol will leak prior to plugging of the hole. Also, aerosol diameters would be increased. The additional path to the SGTS filters would likely result in small amounts of aerosols reaching the filters. Therefore, if they did plug, it is likely that much aerosol would subsequently escape since plugging the filters could account for much of the aerosol mass. However, a specific analysis of this was not performed. Qualitatively, it is felt that the filters would be likely to be effective and actual release fractions reduced using an aerosol model.

1. Leakage of Aerosols from Containment Buildings
H. A. Morewitz, Health Physics, May 29, 1981.

(d) The three conditions listed on page D-8 were derived from references 1, 2, and 3.

The relative importance of gas composition (non-condensible gas fraction), gas flow rate, and iodine concentration were considered in the above analysis. However, the specific accident sequence analyses did not identify parameter ranges for the above values that would result in a reduction of DF's, i.e., the non-condensible gas fraction gas flow rates and iodine concentrations were not high enough to impact the DF's. It should be noted that these are identical to WASH-1400 assumptions. The most important parameter assumed in the Limerick PRA for differentiating among decontamination

factors was the degree of pool subcooling. Each of the accident classes, I through IV, were defined such that the suppression pool was either subcooled or saturated as follows:

Class I	- Subcooled
Class II	- Saturated
Class III	- Subcooled
Class IV	- Saturated

Based upon these conditions, the DF's were assigned. The degree of subcooling was treated as subcooled or not with a saturated pool receiving a smaller DF. Since pool saturation resulted from containment failure (TW) or occurred rapidly (TC) the degree of subcooling was not considered important.

(e) The greater solubility of CsI was only one of many factors in the analysis. In actuality the particulate nature was the most important and lead to the DF chosen (3).

(f) CsI particulates were treated in the same manner as other particulate forms.

(g) The additional credit lead to the choice of increasing the DF from 2-5 to 10 for a saturated pool and elemental iodine. Devel indicated that for a pH increase associated with about 50% of the Cs in the pool, the DF's for elemental iodine were measured at 45 and 120. It is felt that 10 is conservative.

REFERENCES

(1) Diffey, H. R. et. al., "Iodine Cleanup in a Steam Suppression System," CONF-650407 2 (1965) 776.

(2) Hillary, J. J. et. al., "Iodine Removal by a Scale Model of the S.G.H.W. Reactor Vented Steam Suppression System," TRG Report 1256 (1966).

(3) Devell, L. et. al., "Trapping of Iodine in Water Pools at 100 C," Sweden 1967.

QUESTION D.04

The natural deposition analysis of WASH-1400 assumed iodine predominantly in the elemental form. In view of the assumption of CsI discussed in the previous section, explain how the WASH-1400 model is applicable.

RESPONSE

The CORRAL analysis assumes that iodine is airborne in containment as elemental iodine. The removal mechanisms for natural deposition of WASH-1400 are therefore applicable. The particular form of iodine, elemental or inorganic aerosol, was considered in relation to the more significant removal mechanisms associated with the suppression pool scrubbing. (See response to D.03).

Provide the "data from the TMI accident" which indicate that CsI is a "much larger constituent than previously believed," and discuss this data with respect to the expected partitioning of elemental iodine.

RESPONSE

The reference material is Miller's paper entitled, "Radiation Source Terms and Shielding at TMI-2" presented at the ANS meeting in Las Vegas in June, 1980. It was concluded that the radioiodine release from the TMI-2 accident was very much lower than predicted by safety codes and Cs is a much larger constituent than previously believed. (Experiments indicate that Cs and iodine release should be similar).

The expected partitioning of elemental iodine between the gas and aqueous phases should not be affected by these data. Accidents in which water was present tend to have the iodine absorbed. NUREG-0772 concludes that "The assumed chemical form of iodine can influence the attenuation within the reactor coolant system, where, in general the attenuation factor may be greater for CsI than for elemental iodine." If the predominant form is CsI, then by implication less iodine will escape the containment.

QUESTION D.06

The first paragraph of p. D-12 states that RB overpressurization results in ground level releases, while the last paragraph states that pressurization of the RB would result in release via the SGTS exhaust stack. Please clarify.

RESPONSE

RB overpressurization above the differential pressure capacity of the blowout panels results in a direct leakage path to the outside, elevated or a ground level release. On the other hand, pressurization of the RB (without reaching the blowout panels pressure capacity) will result in an elevated release via the SGTS exhaust vent. All releases, except λ", are assumed to be elevated releases. That is, the blowout panels on the refueling floor (i.e., at the top of the Reactor Building) and the vent are treated as approximately equivalent in height.

QUESTION D.07

The discussion of radioactive material inventory and risk associated with the spent fuel pool is inaccurate in several respects:

a) The spent fuel pool inventories quoted from WASH-1400 are not necessarily applicable to LGS. Past experience indicates that inventories of many discharged cores must be expected to be stored in the pool As a result, the inventory of several radiologically significant long-lived isotopes (e.g. Sr-90) may be substantially larger than the core inventory. The text should be revised accordingly.

b) NUREG-CR/0603 discusses risks from Classes 3 - 8 only, and therefore, provides no basis for the claim that risks (including Class 9 events) from spent fuel pool events are negligible. This section should be revised to provide a basis for the claims made concerning accidents involving the spent fuel pool or, if no such basis is provided, the conclusions should be revised accordingly.

RESPONSE

The LGS PRA in Section 3 cites the WASH-1400 estimate of the spent fuel storage pool radionuclide inventory as a relative comparison for an order of magnitude approximation. As noted in part (a) of the question this comparison may not be precisely correct. However, the conclusion should remain the same; that is, the quantity of radionuclides present in spent fuel storage is not negligible. The reason it was not explicitly analyzed in WASH-1400 is because of the low probability of fuel melt release from the spent fuel pool. Since the LGS PRA was prepared for comparison with WASH-1400, the accidents involving the core were the only ones explicitly assessed as was done in WASH-1400.

It can be further noted that the release of radionuclides to the environment from the spent fuel pool would tend to be contributors to latent effects and not early fatalities due to the relatively slow release.

Appendix E

QUESTION E.01

(a) What effect does the formation of CsI have on the postulated accident sequences? How much Tellurium is oxidized during the various sequences?

(b) Why is there no Co-58 or Co-60 at the Limerick plant? Why are Cs-134, Cs-136 and Cs-137 inventories so much smaller than WASH-1400 (p. E-30)?

RESPONSE

(a) Since the more significant attenuation factors used for CsI and I are equal, the formation of CsI would not have a mojor impact on the consequences of the postulated accident sequences. It should be noted, however, that for an accident where the containment is failed prior to meltdown release from the fuel, the removal rate calculated for I is slightly higher than those for particulates. For this case, the results may be conservative.

The Tellurium released upon fuel oxidation for various sequences follows the WASH-1400 formulation for oxidation release following fuel fragmentation due to steam explosion. WASH-1400 considered that 60% of the Te radionuclide species remaining in that portion of the fuel which participates in steam explosion is released due to oxidation.

(b) The fission product inventory was calculated using the RADCICINDER code system.

Cesium variations are due to the manner in which the core was modeled and the BWR fuel loading. The core was modeled into separate bundle groups* and each bundle group followed through its respective power history and fuel down time.

The activation products were not calculated for this system of codes, therefore, CO-58 and CO-60 were not included. However, activation products are not significant in comparison to the fission products and transurancies.

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* The WASH-1400 core inventory was calculated by means of the ORIGEN program for a 2300 MWT PWR core, where the core was represented by three regions only. The BWR model considered more fuel bundle groups than this.

QUESTION E.02

In accident sequences in which the containment building fails due to steam overpressurization, there is the chance of a large amount of "fog" or vapor formation around the building due to the expanding steam. What effect would this have on the release fractions as calculated by CORRAL?

RESPONSE

A large amount of "fog" around the building would allow the released radioactive aerosols to form nucleation sites for condensation. Therefore, aerosol mean diameters would increase. This would result in a much more rapid fallout rate. Reference E.02-1 indicates that fogs are characterized by aerosol diameters in the range of 5 to 40 with mists resulting in diameters greater than 40 . The CORRAL calculations performed assumed aerosol diameters in the range of 5 at this time in the accident. If 40 was used the natural removal rates would increase to a value similar to the largest leakage rate and four times the smallest leakage rate assumed for a major reactor building failure. Therefore, the existence of a fog could decrease environmental release fractions as much as a factor of 2 to 5. (Mist would be <u>much</u> more.) Furthermore, the fog would reduce the mean transport distance away from the reactor. Assuming a larger aerosol mean diameter could easily limit transport such that the consequences would be significantly reduced.

QUESTION E.03

It is not made clear in Section 3.7 or in the Appendix E as to what type of meteorological sampling scheme was actually used in the Limerick site-specific consequences analysis. Clarify whether it was the stratified meteorological sampling of WASH-1400 using 91 start times, which is the normal sampling method for the CRAC analysis, for which 8760 consecutive hourly met-data must be input; or was it a non-standard sampling scheme of invariant meteorology (for the Start Code 9 of the CRAC Manual) generated by joint frequency distribution of meteorological data over any non-specific period (continuous, or with gaps) of time? Use of the latter sampling scheme of CRAC is known to result in acute fatality CCDF about an order of magnitude lower compared to the acute fatality CCDF generated by the former (the standard) sampling. Mention of features of both sampling schemes in Item II in Table E.1 has led to this lack of clarity.

RESPONSE

The WASH-1400 stratified meteorological sampling method was used for the CRAC analysis. This used 91 start times chosen every 4 days and 13 hours of the 8760 consecutive hourly met-data input to CRAC. Five years worth of data was "sampled" in this scheme, with the postulated accident assumed equally likely to occur in any one year.

QUESTION E.04

Provide justification for use of the numerical values of 25 miles, 1.2 mph, and 0.0 days respectively (See Table E.6) for the evacuation distance, effective speed of evacuation and time lag before evacuation in Limerick site-specific consequence analysis. Values of these parameters should have been obtained from evaluation of Limerick site's plans for emergency response within the plume exposure pathway Emergency Planning Zone which is approximately a circular region, centered at the reactor, of about 10-mi radius.

What is the basis for assuming that 95% of the people participate in the emergency evacuation? How sensitive are the consequences to this assumption (p. E-17, 18, 19)?

RESPONSE

The WASH-1400 evacuation model was used to provide a direct comparison of results. Figure 4.13 of Revision 3 of the Limerick Generating Station PRA shows the results of a ten mile vs twenty-five mile radius evacuation zone.

QUESTION E.05

Provide bases for (a) the value of 0.29 used for the ground shielding factor wihtout sheltering in Table E.6 in contrast to 0.33 assumed in WASH-1400 for situations of normal activities of people, and (b) the breathing rate of 1.1 x 10^{-4} m^3/sec rather than the standard value of 2.3 x 10^{-4} m^3/sec.*

RESPONSE

(a) The value of 0.29 for ground shielding was chosen from WASH-1400, Table VI 11-12 (WASH-1400) page 11-28. It applied to region IV in which Figure VI 11-9 classifies Pennsylvania. It should be noted that actual site specific data indicated that 90 percent of the homes were brick. Therefore, the shielding factor would actually be less than 0.26, the value used in Region V for 80 to 90 percent bricks. It is felt that 0.29 adequately represents the population within 200 miles (the range of dominant contributions from latent effects).

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* Revision 3 of the LGS PRA used the WASH-1400 standard breathing rate.

QUESTION E.06

Clarify as to which is the apatial interval over which the shielding factors for sheltering in Tables E.2a, E.2b and E.3 were used in CRAC analyses and as to the impact on the Limerick site-specific consequences.

RESPONSE

All spatial intervals for non evacuating region received the shielding factors for sheltering. This might impact the amount of latent cancer fatalities beyond the region of expected sheltering. However, it should be noted that latent cancer fatalities are dominated by chronic health effects in the CRAC model. Acute fatalities would not extend beyond the region of expected sheltering. Therefore, there should be only a minor impact in the Limerick site-specific consequences.

QUESTION E.07

In situations where the emergency response would be the sheltering mode rather than the evacuation or the no-response modes, there would still be a time lag before people would actually be in the sheltering mode (due to delay in notification advising people to shelter). During this time-lag the shielding factors only for the situation of normal activities of people as assumed in WASH-1400 would be appropriate. Further, for deriving any benefit from the improved shielding factors for inhalation (given the sheltering mode) it is also necessary to advise the people to open the windows and enhance ventilation to expel the contaminated air trapped inside the buildings for exchange with the outside fresh air after the radioactive plume has left the area. Unless this latter action were taken, the dose from prolonged inhalation of the contaminated air trapped in the buildings would result in higher doses from plume inhalation exposure pathway (see WASH-1400, Appendix VI page 11-8 and Figure VI II-5). Therefore, provide a discussion of the emergency response scenario used and matching analysis of how the shielding protection factors in Tables E.?a, E.2b and E.3 have been factored-in in the Limerick site-specific consequence analysis.

RESPONSE

The emergency response scenario used considers sheltering of people beyond the evacuation zone. This involves no special communications other than the normal communication media. It includes advising the population to remain indoors with the doors and windows closed and ventilation systems turned off. Further they are advised to go to their basements if available. Little physical activity is also recommended.

The shielding factors contained in Tables E.2a, E.2b and E.3 are applied at all times because it was felt that adequate warning time would be available prior to plume passage. The accidents of interest allowed for approximately six hours warning time based on initiation of the emergency response plan when high radiation existed in containment and safety systems were not functional. Significant additional time would be available prior to plume front arrival. Given these long warning times it was felt that the normal vigilence of the communications media would be sufficient. It should further be noted that the tables include the relative percentages for normal activity. People outdoors or commuting were not assumed to seek shelter. (11% of the population) Only people at home or at work were assumed to know to seek shelter. The inclusion of normal activities into the sheltering model effectively decreased dose reduction from a factor of 2.31 to 1.85. Finally it was assumed that 5 percent of the population ignored the

sheltering order. (Reference 1) It should be noted that the values quoted for Limerick in Tables E.2a and E.2b are higher than values quoted in Reference 2.

The values for inhalation dose sheltering were derived from Reference 3. It assumes no "air-out" strategy as is noted in WASH-1400 and question E.07. The sheltering values used in the LGS PRA were derived from Figures 3 and 5 on pages 32 and 34, using a value of $\mu = 0.5\ hr^{-1}$. This is more conservative than the recommended value of $\mu = 10\ hr^{-1}$ (Reference 3). The value chosen was felt to be more representative of closed doors and windows as recommended in the emergency plan scenario already discussed. Furthermore, these values reflect weather stripped windows or other energy conservation measures implemented since the energy crisis. These values are still felt to be conservative based on the value of 0.25 for the ingress fraction. Finally, while some houses may exhibit higher ingress values, some will exhibit lower values and the shielding factor will represent that range. The values for basements were conservatively estimated as a ten percent reduction based on Figure 8, page 41.

In summary, the values chosen for shielding factors resulting from sheltering were felt to be applicable to a normal communications network warning system. Normal activity was factored in and warning times beyond the evacuation zone are long. Inhalation dose sheltering assumes no "air-out" procedures. Finally, the factors represent a range of factors and it has been shown previously that a range of factors are adequately represented by an average value. (Reference 4.)

REFERENCES

[1] Evacuation Risks - An Evaluation, Joseph M. Hans Jr. and Thomas C. Sell. June, 1974. USEPA.

[2] Examination of Offsite Radiological Emergency Protective Measures for Nuclear Reactor Accidents Involving Core Melt, D. C. Aldrich, P. McGrath, N. C. Rasmussen, Sandia Laboratory NUREG/CR-1131, SAND78-9454. June, 1978.

[3] Public Protection Strategies in the Event of a Nuclear Reactor Accident: Multicompartment Ventilation Model for Shelters, D. C. Aldrich, David M. Ericson, Jr. January, 1978, SAND7701555, Sandia Laboratories.

[4] Public Protection Strategies for Potential Nuclear Reactor Accidents: Sheltering Concepts With Existing Public and Private Structures, D. C. Aldrich, D. M. Ericson, Jr., Jay D. Johnson, February, 1978, SAND77-1725, Sandia Laboratories.

QUESTION E.08

Provide the following additional information for use in staff's confirmatory Limerick site-specific consequence calculations:

a) Population input for CRAC standard spatial grid, i.e. for each area element generated by the 16 direction sectors of the compass and the 34 rings of outer radii as specified in the description of the Subgroup SPATIAL in page 49 of the CRAC Manual, for the year 1970 and the year 2000.

b) State code and habitable land fraction for each area element of the CRAC spatial grid, and

c) Estimates of evacuation times including the notification times, and travel (response times for clearing a 10-mile plume exposure pathway Emergency Planning Zone under normal and adverse conditions, consistent with the expected traffic loading on the existing road net-works and for various segments of the population (in schools, factories, hospitals, etc.).

RESPONSE

a) Attachment 1 lists the Limerick population used as CRAC input in the Limerick PRA. This is 1970 population data. The data is presented with radial distance vertically down the page and with columns for each 22 1/2° sector oriented on the Limerick Grid Structure. The data is given for intervals and cumulatively.

b) Attachment 2 lists the state code used for each area element in the Limerick PRA CRAC runs. THe states corresponding to the codes are as follows:

1.	Marine	13.	Michigan
2.	New Hamphsire	22.	Delaware
4.	Massachusetts	23.	Maryland
6.	Connecticut	24.	Virginia
7.	New York	25.	West Virginia
8.	New Jersey	26.	North Carolina
9.	Pennsylvania	30.	Kentucky
10.	Ohio	51.	Ontario

Attachment 3 lists the habitable land fraction for each grid area.

c) Site specific maximum exacution time estimates were made for a 10 mile plume exposure pathway EPZ. These range from 9 hours in normal weather to 11.25 hours in

adverse weather. The notification time portion of these estimates is 5 hours.

Attachment 1
Page 1

LIMERICK BASE SITE POPULATION

DISTANCE (MI)	N	NNE	NE	ENE	E	ESE	SE	SSE
.5	12.	28.	5.	6.	5.	14.	0.	3.
1.0	36.	83.	16.	19.	14.	43.	0.	8.
1.5	241.	86.	31.	30.	47.	55.	174.	128.
2.0	338.	120.	43.	41.	66.	76.	243.	180.
2.5	190.	118.	100.	124.	143.	150.	529.	597.
3.0	233.	144.	123.	152.	175.	133.	646.	729.
3.5	248.	167.	150.	171.	220.	135.	2052.	1677.
4.0	286.	193.	173.	197.	254.	155.	2367.	1935.
4.5	369.	72.	147.	85.	178.	155.	583.	707.
5.0	413.	80.	164.	95.	198.	173.	652.	791.
6.0	1022.	458.	437.	270.	1759.	2107.	468.	3716.
7.0	1208.	541.	517.	319.	2079.	2490.	553.	4392.
8.5	2160.	968.	924.	570.	3718.	4453.	989.	7854.
10.0	2579.	1155.	1103.	680.	4438.	5315.	1181.	9375.
12.5	1562.	3674.	2056.	3012.	5643.	25610.	14780.	3788.
15.0	1909.	4491.	2512.	3681.	6898.	31300.	18060.	4630.
17.5	1292.	5104.	7813.	14590.	18350.	34820.	26800.	7491.
20.0	1491.	5889.	9015.	16840.	21170.	40180.	30920.	8643.
25.0	14620.	21790.	9071.	17270.	21440.	172000.	392000.	43570.
30.0	25630.	166800.	5797.	21350.	73250.	552200.	864000.	206800.
35.0	25180.	81190.	5595.	5144.	39870.	444400.	316600.	14100.
40.0	17770.	89240.	16640.	12040.	124900.	165900.	250000.	11460.
45.0	21170.	12180.	14080.	13420.	206500.	31810.	88880.	11690.
50.0	6740.	23010.	24470.	15290.	132100.	41850.	14260.	9593.
55.0	2171.	16940.	19190.	71090.	26290.	31280.	13310.	29360.
60.0	2378.	18550.	21020.	77860.	28790.	34260.	14580.	32160.
65.0	3500.	5701.	72540.	257500.	55890.	5875.	7955.	30480.
70.0	3779.	6158.	78340.	278100.	60360.	6346.	8592.	32920.
85.0	129600.	22110.	287600.	5892000.	156700.	12580.	86130.	27030.
100.0	154700.	26390.	343300.	7033000.	187000.	15010.	102800.	32260.
150.0	158900.	143700.	682300.	1132000.	0.	0.	0.	0.
200.0	937400.	996900.	1382000.	502700.	0.	0.	0.	0.
350.0	721400.	863200.	902800.	89520.	0.	0.	0.	0.
500.0	228200.	714000.	426900.	0.	0.	0.	0.	0.
	2468726.	3231228.	4316972.	15459166.	1178444.	1660925.	2260104.	538067.

Attachment 1
Page 2

LIMERICK BASE SITE POPULATION

Distance (MI.)	S	SSW	SW	WSW	W	WNW	NW	NNW
.5	1.	0.	14.	11.	12.	2.	6.	3.
1.0	2.	0.	41.	32.	37.	5.	17.	8.
1.5	163.	192.	78.	85.	25.	32.	194.	281.
2.0	228.	268.	109.	120.	34.	44.	272.	394.
2.5	110.	131.	73.	213.	536.	1465.	1502.	500.
3.0	134.	160.	90.	260.	655.	1791.	1836.	612.
3.5	31.	146.	131.	184.	553.	5141.	3938.	508.
4.0	36.	168.	151.	213.	639.	5931.	4543.	586.
4.5	159.	105.	126.	448.	895.	1569.	938.	385.
5.0	178.	118.	140.	500.	1001.	1754.	1049.	430.
6.0	593.	297.	468.	164.	45.	1212.	160.	914.
7.0	701.	351.	553.	194.	53.	1433.	189.	1081.
8.5	1253.	627.	989.	347.	94.	2563.	338.	1933.
10.0	1496.	749.	1181.	414.	113.	3059.	404.	2307.
12.5	6476.	2911.	606.	807.	456.	6211.	1250.	1748.
15.0	7915.	3558.	741.	987.	557.	7592.	1528.	2137.
17.5	21180.	10030.	1260.	2636.	118.	50750.	2330.	2754.
20.0	24440.	11570.	1453.	3042.	137.	58560.	2689.	3177.
25.0	11460.	8849.	22960.	7044.	7765.	58790.	10160.	11140.
30.0	17560.	15000.	11220.	12670.	7241.	12520.	6754.	3361.
35.0	243400.	5803.	3799.	18260.	23420.	8434.	9091.	3156.
40.0	88790.	30870.	6177.	44040.	18300.	10330.	5462.	2636.
45.0	7996.	35400.	8791.	92560.	18020.	8756.	23640.	27100.
50.0	10250.	9959.	5528.	33870.	52640.	17260.	38330.	7784.
55.0	6325.	21280.	13540.	71090.	31350.	9278.	23870.	33120.
60.0	6928.	23310.	14830.	77860.	34330.	10160.	26140.	36280.
65.0	5935.	10260.	25340.	257500.	100500.	7439.	24860.	36210.
70.0	6409.	11080.	27370.	278100.	108600.	8034.	26850.	39110.
85.0	28410.	54160.	910200.	5892000.	44050.	29900.	50270.	14850.
100.0	33910.	64640.	1086000.	7033000.	52570.	35690.	59990.	17730.
150.0	49640.	97800.	268000.	165700.	374500.	129100.	78670.	284400.
200.0	23080.	650900.	191600.	133500.	603300.	176400.	212700.	448800.
350.0	64760.	1530000.	1509000.	833800.	3221000.	5512000.	9274000.	852900.
500.0	0.	961400.	3481000.	1394000.	7662000.	4345000.	113600.	322100.
	669948.	3562090.	7593557.	16355651.	12365544.	10528205.	10007570.	2160433.

THE TOTAL POPULATION FOR 16 SECTORS IS 94356632.

Attachment 1
Page 3

LIMERICK BASE SITE POPULATION (SUMMED)

Distance (Mi.)	N	NNE	NE	ENE	E	ESE	SE	SSE
.5	12.	28.	5.	6.	5.	14.	0.	3.
1.0	48.	110.	21.	25.	18.	57.	0	11.
1.5	289.	196.	52.	55.	65.	112.	174.	139.
2.0	627.	316.	95.	96.	131.	188.	417.	319.
2.5	618.	434.	195.	220.	274.	338.	946.	916.
3.0	1050.	578.	318.	372.	449.	521.	1592.	1645.
3.5	1298.	745.	468.	543.	669.	656.	3644.	3322.
4.0	1583.	938.	640.	740.	923.	811.	6011.	5257.
4.5	1953.	1010.	787.	825.	1101.	966.	6594.	5964.
5.0	2365.	1090.	951.	920.	1299.	1139.	7246.	6755.
6.0	3387.	1548.	1389.	1190.	3058.	3246.	7714.	10471.
7.0	4595.	2089.	1905.	1509.	5137.	5736.	8267.	14863.
8.5	6755.	3056.	2830.	2079.	8855.	10189.	9256.	22717.
10.0	9334.	4211.	3933.	2759.	13293.	15504.	10437.	32092.
12.5	10896.	7885.	5989.	5771.	18936.	41114.	25217.	35880.
15.0	12805.	12376.	8501.	9452.	25834.	72414.	43277.	40510.
17.5	14097.	17480.	16314.	24042.	44184.	107234.	70077.	48001.
20.0	15588.	23369.	25329.	40882.	65354.	147414.	100997.	56644.
25.0	30208.	45159.	34400.	58152.	86794.	319414.	492997.	100214.
30.0	55838.	211959.	40197.	79502.	160044.	871614.	1356997.	307014.
35.0	81018.	293149.	45792.	84646.	199914.	1316014.	1673597.	321114.
40.0	98788.	382389.	62432.	96686.	324814.	1481914.	1923597.	332574.
45.0	119958.	394569.	76512.	110106.	531314.	1513724.	2012477.	344264.
50.0	126698.	417579.	100982.	125396.	663414.	1555574.	2026737.	353857.
55.0	128869.	434519.	120172.	196486.	689704.	1586854.	2040047.	383217.
60.0	131247.	453069.	141192.	274346.	718494.	1621114.	2054627.	415377.
65.0	134747.	458770.	213732.	531846.	774384.	1626989.	2062582.	445857.
70.0	138526.	464928.	292072.	809946.	834744.	1633335.	2071174.	478777.
85.0	268126.	487038.	579672.	6701946.	991444.	1645915.	2157304.	505807.
100.0	422826.	513428.	922972.	13734946.	1178444.	1660925.	2260104.	538067.
150.0	581726.	657128.	1605272.	14866946.	1178444.	1660925.	2260104.	538067.
200.0	1519126.	1654028.	2987272.	15369646.	1178444.	1660925.	2260104.	538067.
350.0	2240526.	2517228.	3890072.	15459166.	1178444.	1660925.	2260104.	538067.
500.0	2468726.	3231228.	4316972.	15459166.	1178444.	1660925.	2260104.	538067.
	2468726.	3231228.	4316972.	15459166.	1178444.	1660925.	2260104.	538067.

Attachment I
Page 4

LIMERICK BASE SITE POPULATION (SUMMED)

Distance (MI.)	S	SSW	SW	WSW	W	WNW	NW	NNW
.5	1.	0.	14.	11.	12.	2.	6.	3.
1.0	3.	0.	55.	42.	49.	7.	23.	10.
1.5	166.	192.	133.	127.	74.	39.	217.	291.
2.0	393.	460.	241.	247.	108.	83.	489.	685.
2.5	503.	591.	314.	460.	644.	1548.	1991.	1186.
3.0	637.	750.	404.	720.	1298.	3339.	3827.	1797.
3.5	668.	896.	535.	904.	1851.	8480.	7765.	2305.
4.0	704.	1064.	685.	1117.	2490.	14411.	12308.	2890.
4.5	863.	1169.	811.	1565.	3385.	15980.	13246.	3275.
5.0	1041.	1287.	951.	2065.	4386.	17734.	14295.	3705.
6.0	1634.	1584.	1419.	2229.	4431.	18946.	14455.	4619.
7.0	2335.	1934.	1972.	2424.	4484.	20379.	14644.	5700.
8.5	3588.	2562.	2961.	2771.	4578.	22942.	14983.	7633.
10.0	5084.	3310.	4142.	3185.	4690.	26001.	15386.	9940.
12.5	11560.	6221.	4749.	3992.	5146.	32212.	16636.	11688.
15.0	19475.	9779.	5489.	4979.	5703.	39804.	18164.	13825.
17.5	40655.	19909.	6749.	7615.	5822.	90554.	20494.	16579.
20.0	65095.	31379.	8202.	10657.	5958.	149114.	23183.	19756.
25.0	76555.	40228.	31162.	17701.	13723.	207904.	33343.	30896.
30.0	94115.	55228.	42382.	30371.	20964.	220424.	40097.	34257.
35.0	337515.	61031.	46181.	48631.	44384.	228858.	49188.	37413.
40.0	426305.	91901.	52358.	92671.	62684.	239188.	54650.	40049.
45.0	434301.	127301.	61149.	185231.	80704.	247944.	78290.	67149.
50.0	444551.	137260.	66677.	219101.	133344.	265204.	116620.	74933.
55.0	450876.	158540.	80217.	290191.	164694.	274482.	140490.	108053.
60.0	457804.	181850.	95047.	368051.	199024.	284642.	166630.	144333.
65.0	463739.	192110.	120387.	625551.	299524.	292081.	191490.	180543.
70.0	470148.	203190.	147757.	903651.	408124.	300115.	218340.	219653.
85.0	498558.	257350.	1057957.	6795651.	452174.	330015.	268610.	234503.
100.0	532468.	321990.	2143957.	13828651.	504744.	365705.	328600.	252233.
150.0	582108.	419790.	2411957.	13994351.	879244.	494805.	407270.	536633.
200.0	605188.	1070690.	2603557.	14127851.	1482544.	671205.	619970.	935433.
350.0	669948.	2600690.	4112557.	14961651.	4703544.	6183205.	9893970.	1838333.
500.0	669948.	3562090.	7593557.	16355651.	12365544.	10528205.	10007570.	2160433.
	669948.	3562090.	7593557.	16355651.	12365544.	10528205.	10007570.	2160433.

THE TOTAL POPULATION FOR 16 SECTORS IS 94356632.

ATTACHMENT 2
PAGE 1

LIMERICK STATE CODE FOR SECTOR

DISTANCE (MI.)	N	NNE	NE	ENE	E	ESE	SE	SSE
.5	9	9	9	9	9	9	9	9
1.0	9	9	9	9	9	9	9	9
1.5	9	9	9	9	9	9	9	9
2.0	9	9	9	9	9	9	9	9
2.5	9	9	9	9	9	9	9	9
3.0	9	9	9	9	9	9	9	9
3.5	9	9	9	9	9	9	9	9
4.0	9	9	9	9	9	9	9	9
4.5	9	9	9	9	9	9	9	9
5.0	9	9	9	9	9	9	9	9
6.0	9	9	9	9	9	9	9	9
7.0	9	9	9	9	9	9	9	9
8.5	9	9	9	9	9	9	9	9
10.0	9	9	9	9	9	9	9	9
12.5	9	9	9	9	9	9	9	9
15.0	9	9	9	9	9	9	9	9
17.5	9	9	9	9	9	9	9	9
20.0	9	9	9	9	9	9	9	9
25.0	9	9	9	9	9	9	9	9
30.0	9	9	9	9	9	9	8	8
35.0	9	9	9	9	9	9	8	8
40.0	9	9	8	8	9	8	8	8
45.0	9	9	8	8	9	8	8	8
50.0	9	9	8	8	8	8	8	8
55.0	9	9	8	8	8	8	8	8
60.0	9	9	8	8	8	8	8	8
65.0	9	9	8	8	8	8	8	8
70.0	9	9	8	8	8	8	8	8
85.0	9	9	8	8	8	8	8	8
100.0	9	9	7	8	8	8	8	8
150.0	9	9	7	7	7	7	22	22
200.0	7	7	6	6	6	6	6	6
350.0	51	7	2	4	4	4	4	4
500.0	50	50	1	1	1	1	1	1

Attachment Z
Page 2

LIMERICK   STATE CODE FOR SECTOR

DISTANCE (MI.)	S	SSW	SW	WSW	W	WNW	NW	NNW
.5	9	9	9	9	9	9	9	9
1.0	9	9	9	9	9	9	9	9
1.5	9	9	9	9	9	9	9	9
2.0	9	9	9	9	9	9	9	9
2.5	9	9	9	9	9	9	9	9
3.0	9	9	9	9	9	9	9	9
3.5	9	9	9	9	9	9	9	9
4.0	9	9	9	9	9	9	9	9
4.5	9	9	9	9	9	9	9	9
5.0	9	9	9	9	9	9	9	9
6.0	9	9	9	9	9	9	9	9
7.0	9	9	9	9	9	9	9	9
8.5	9	9	9	9	9	9	9	9
10.0	9	9	9	9	9	9	9	9
12.5	9	9	9	9	9	9	9	9
15.0	9	9	9	9	9	9	9	9
17.5	9	9	9	9	9	9	9	9
20.0	9	9	9	9	9	9	9	9
25.0	9	9	9	9	9	9	9	9
30.0	22	9	9	9	9	9	9	9
35.0	22	9	9	9	9	9	9	9
40.0	22	9	9	9	9	9	9	9
45.0	22	23	9	9	9	9	9	9
50.0	22	23	9	9	9	9	9	9
55.0	22	23	23	9	9	9	9	9
60.0	22	23	23	9	9	9	9	9
65.0	22	23	23	9	9	9	9	9
70.0	22	23	23	9	9	9	9	9
85.0	22	23	23	9	9	9	9	9
100.0	22	23	23	23	9	9	9	9
150.0	23	23	23	23	9	9	9	9
200.0	24	24	24	25	9	9	9	7
350.0	26	26	24	25	10	9	7	51
500.0	26	26	26	30	10	13	51	51

ATTACHMENT 3
PAGE 1

LIMERICK FRACTION OF HABITABLE LAND

DISTANCE (MI.)	N	NNE	NE	ENE	E	ESE	SE	SSE
.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
1.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
1.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
2.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
2.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
3.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
3.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
4.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
4.5	1.00000	1.00000	1.00000	00000	1.00000	1.00000	1.00000	1.00000
5.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
6.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
7.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
8.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
10.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
12.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
15.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
17.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
20.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
25.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
30.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	.97000	.97000
35.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
40.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
45.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
50.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
55.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
60.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
65.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
70.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	.93000
85.0	1.00000	1.00000	1.00000	.68000	.70000	.60000	.65000	.53000
100.0	1.00000	1.00000	1.00000	.72000	0.00000	0.00000	0.00000	.16000
150.0	1.00000	1.00000	1.00000	.65000	0.00000	0.00000	0.00000	.05000
200.0	1.00000	1.00000	1.00000	.55000	0.00000	0.00000	0.00000	0.00000
350.0	.92000	.93000	.85000	.25000	0.00000	0.00000	0.00000	0.00000
500.0	1.00000	1.00000	.70000	0.00000	0.00000	0.00000	0.00000	0.00000

LIMERICK FRACTION OF HABITABLE LAND

DISTANCE (MI.)	S	SSW	SW	WSW	W	WNW	NW	NNW
.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
1.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
1.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
2.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
2.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
3.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
3.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
4.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
4.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
5.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
6.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
7.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
8.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
10.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
12.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
15.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
17.5	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
20.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
25.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
30.0	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
35.0	.97000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
40.0	.97000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
45.0	.93000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
50.0	.89000	.75000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
55.0	.82000	.60000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
60.0	.75000	.60000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
65.0	.72000	.60000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
70.0	.80000	.63000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
85.0	.80000	.63000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
100.0	.91000	.63000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
150.0	.90000	.63000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
200.0	.20000	.75000	1.00000	1.00000	1.00000	1.00000	1.00000	1.00000
350.0	.20000	.95000	1.00000	1.00000	1.00000	.80000	1.00000	.70000
500.0	.04000	.65000	1.00000	1.00000	1.00000	.86000	.42000	.93000

ATTACHMENT 3
PAGE 2

QUESTION E.09

Provide a basis for the "conservative estimates of saturated pool DF", as well as a reference for the "other data evaluations." Provide the data for these evaluations and discuse their applicability to the accident conditions at LGS.

RESPONSE

The referenced quotations are on p. D-8 of the PRA.

During the course of performing calculations for the Limerick PRA, an extensive literature search to define suppression pool decontamination factors appropraite for use in BWR risk assessments was being conducted. Based on results of that review, (1) the following conclusions were made:

1. Suppression pool decontamination factors of at least 100 for elemental iodine and particulates and 1000 for cesium iodide are justifiable for subcooled pools.

2. For saturated pools, decontamination factors of at least 30 for elemental iodine and 100 for particulates and cesium iodide are currently justified.

3. The above values were recognized as minimum values which could justafiably be increased several orders of magnitude by testing at conditions more representative of degraded core conditions.

Subsequent to the PRA and the review discussed above, a predictive first-principle model has been formulated for the calculation of accident sequence-dependent decontamination factors for use in PRA's. Suppression pool scrubbing tests have been performed by General Electric. These tests have provided data which directly verifies the model. The model and the tests confirm the use of suppression pool decontamination factors of 10^2 - 10^4 for saturated pools. The value used in the Limerick PRA for saturated pools (10) is conservative, based on the above results.

REFERENCE

(1) Rastler, D. M., "Suppression Pool Scrubbing Factors for Postulated Boiling Water Reactor Accident Conditions," NEDO 25420, June, 1981.

QUESTION F.01

Most of Appendix F is spent in justifying why Gamma distributions were used as priors instead of lognormals. On the other hand, it is stated that the prior distributions were discretized. Given that, it should not have made any difference (from a mathematical convenience point of view) whether Gamma or lognormal distributions are used. A list of all the prior distributions (for each input parameter) along with the important characteristics (like mean, median, or other parameters) should be provided.

RESPONSE

Appendix F discusses the choice of probability distributions which may be used as priors if a Bayesian analysis is to be used to construct a probability distribution. One conclusion from the discussion is that: "Two obvious choices for a distribution are gamma and lognormal; both have advantages and disadvantages, but there is no inherent reason to choose one over the other. Both are equally 'correct'." It is further stated that: "there is no theoretical foundation for or against either the gamma or lognormal distributions, and there is little data for either choice, ..."WASH-1400 did employ lognormal distributions to represent the distributions of some parameters, however, both gamma and lognormal distributions appear as possible alternatives of <u>continuous</u> prior distributions.

In the practical application of Bayesian analysis to update the priors, discrete formulation would probably be used to construct the priors. The main advantages of discrete formulation are discussed in Section F.4. It is noted that the discrete priors are <u>not</u> established through discretizing some other prior distribution (e.g., gamma or lognormal); instead, they have in the past been defined by the opinions of a panel of "experts"; this is primarily due to the lack of data responses on which to base a distribution. It is recognized that "the method of defining a discrete prior is subject to much debate."

For the LGS PRA the Bayesian analysis technique was applied in selective cases. The decision not to perform extensive data manipulation using Bayesian Techniques was based upon three principal factors:

1. There is a limited amount of plant specific data applicable to Limerick.

2. Bayesian manipulation of generic data sources (see Appendix A Section A.2.3) does not have large

effect on the calculated mean values of a failure rate.

3. The time and money to perform such data manipulations was judged not to produce a commensurate benefit.

Lognormal distributions are accepted to represent the uncertainty in the fault tree input for the component failure rates in the Limerick PRA similar to the assumption made in WASH-1400.

QUESTION G.01

It is stated that the use of mean values in point estimates of a fault tree will result in the mean value of the top event, provided that the basic events are independent. This is not the case, however, if the basic events of the fault tree (or any other technique) include identical components that fail dependently, but are characterized by the same failure rates. Was this eff[illegible]ct taken into consideration in estimating the mean values of the accident sequences?

RESPONSE

The LGS PRA, as stated in Appendix G. assumes that all failure probabilities for components are characterized by random independent variables. This approach is similar to that taken in the following PRA studies:

- WASH-1400 (2)
- RSSMAP (3, 4, 5)
- IREP (to be published)
- Oconee (EPRI) (to be published)

While the assumption used in these studies of independent failure probabilities may be an over-simplificaiton, the assumption cited by Apostolakis (1) is an equal simplification of the real world. The truth undoubtedly lies somewhere between these two assumptions.

Thus far in the evolution of probabilistic risk assessment techniques as applied to nuclear power plants, no completely adequate treatment of the propagation of uncertainties and the incorporation of statistical coupling among components has been established.

The Limerick PRA has used the formalisms established by WASH-1400 to characterize the conditional probabilities of component failures and which is the current state of the technology. To include the statistical correlation, even if it were known, would be a monumental job which has not heretofore been accomplished in any PRA.

To be less precise about the point value that was calculated in the LGS PRA, one might refer to it as a point estimate (i.e., not the mean). This is a similar argument to that made in the IREP studies. It is judged that the use of the term mean best approximates the value, however, in the strict statistical sense this may not always be valid.

This question is of theoretical interest to establish a consistent formalism for quantitative evaluation to be used in PRA; however, because of the lack of data to support either extreme, or a position in between, the decision on an established format may need to be arbitrary. It should be noted that the draft PRA Guide does not resolve this issue and does not make recommendations on how to treat it in the future. While the formalism is important, it is not of central importance to PRA. In other words, this is the wrong question to focus on.

It is desirable to have a unified and agreed upon formalism which will make the treatment of data and its application readily apparent to the reader, but the main purpose of PRA is to establish a systematic methodology for the assessment of the ultimate capability of the plant to ensure the public's safety. This is an engineering task.

REFERENCES

[1] Apostolakis, G. and Kaplan, S., Pitfalls in Risk Calculations, Reliability Engineering Vol 2, p. 135-145, 1981.

[2] WASH-1400 "Reactor Safety Study" U.S. NRC, October 1975.

[3] G. J. Kolb, S. W. Haten, P. Cybulskis, and R. O. Wooten, Reactor Safety Study Methodology Applications Program, Oconee #3 PWR Power Plant, NUREG/CR-1659/2 of 4, dated January 1981.

[4] D. D. Carlson, W. R. Cramond, J. W. Hickman, S. V. Asselin, and P. Cybulskis, Reactor Safety Study Methodology Application Program: Sequoyah #1 PWR Power Plant, NUREG/CR-1659/1 of 4, dated February 1981.

[5] S. W. Hatch, P. Cybulskis, and R. O. Wooten, Reactor Safety Study Methodology Applications Program: Grand Gulf #1 BWR Power Plant, NUREG/CR-1659/4 of 4, dated October 1981.

QUESTION G.02

In order to compare results of the Limerick PRA to those of WASH-1400, median values were estimated for the Limerick results based on mean values. What distributions were assumed in this process? Are the Limerick median results shown in the report?

RESPONSE

Median valves were not used for comparisons, and, infact, were not calculated in the LGS PRA. In order to compare the calculated frequency of core melt between the Limerick PRA and WASH-1400, mean values were estimated for WASH-1400. Figure 3.5.4 shows the resulting comparison.

The mean values for WASH-1400 were estimated using the following procedures:

1. The median core melt frequency was taken as that presented in the Appendix V summary Table V.

2. Release Category 3 has a frequency of core melt * containment failure of $2x10^{-5}$/Reactor Year where the containment failure prob. = 1.

   Therefore, core melt median = $2x10^{-5}$/Reactor Year

3. The probability distribution of the core melt frequency is taken to be lognormal similar to that for the impact quantities.

4. The median and mean of a lognormal distribution can be related by the following:

   $$\text{mean} = \text{median} \ *\text{EXP}\ (\sigma^2/2)$$

   where

   $$\sigma = \frac{\ln\ (\text{Error Factor})}{1.64}$$

   From Table V in WASH-1400 the error factor on the core melt frequency can be conservatively estimated to be determined by the ratio

   $$\frac{\text{upper bound (95\% value)}}{\text{median (50\% value)}} = \frac{8E-5}{2E-5} = 4$$

QUESTION H.01

The Core Dispersal Model described in Appendix H involves a molten jet exiting the vessel and attacking the concrete. How does the erosion of concrete influence the strength of the diaphragm floor and potentially the pedestal wall?

RESPONSE

The erosion of the concrete floor as calculated in Appendix H would not substantially change the strength of the diaphragm floor. In general, the structural importance of the diaphragm floor is in the annular region between the pedestal and the wall of the primary containment which would not be eroded by the jet of molten core debris. The attack as calculated overestimates the penetration since the accumulation of material is assumed to have no influence on the jet.

QUESTION H.02

The Core Dispersal Model, as described in Appendix H, involves the rapid cooling of 50% of the core materials. This appears to be inconsistent with Appendix C, which considers the steam spike associated with vessel failure to be uncertain. Hence, containment failure is based on a gradual pressure rise and is predicted to occur several hours after vessel failure. Explain this apparent inconsistency.

RESPONSE

The steam spike calculated in Appendix C is governed by the vessel failure model in the CONTEMPT code. This model assumes the entire vessel head fails at the equator and the primary system exhausts into the drywell faster than the downcomers and suppression pool can respond. Given the extensive number of penetrations in the lower head of a BWR, the vessel failure would be via one or more of the limited depth welds joining the penetrations to the vessel wall. Since these welds are on the inside of the vessel, they would experience essentially the same thermal transient as the wall and with their smaller thickness (~1/5 of the vessel wall), they would fail before the vessel. As a result of this failure and the ablation experienced as the molten material is discharged, the hole size governing the depressurization of the primary system would be about 40 cm in diameter as discussed in Appendix H. The resulting pressure history in the drywell is a slow rise and coupled with response of the suppression pool, the maximum pressure is far below the containment ultimate. If the decay heat cannot be removed from the primary containment, the pressure will increase slowly over several hours until containment failure occurs.

QUESTION H.03

There appears to be drains directly below the vessel through the diaphragm floor covered only by thin steel plates. These steel plates would offer little resistance to the attack of a molten jet of core materials at vessel failure. Failure of these plates would open up a direct path between the wetwell and the drywell. The core materials could then mix with water increasing the potential for steam explosions and/or rapid steam generation. Would this increase the potential for containment failure at vessel failure?

RESPONSE

Equipment and floor drains are located in the CRD room and are connected to piping leading to collecting tanks mounted below the diaphragm floor. Failure of the drain plates would only allow material to flow into these pipes. However, assuming failure of the drain plates and the associated piping would allow molten debris to enter the suppression pool and be quenched. Potential steam explosions resulting from debris falling through about 10 m and into a suppression pool would not threaten the containment integrity for two major reasons: (1) Given the 10 cm dia. initial failure size for the drain plates with the driving force being the static head of the accumulated debris, the discharge of significant quantities of material would require tens of seconds as compared to an interval of a few seconds required to quench debris coarsely fragmented to sizes of concern in steam explosion analyses. (2) Large scale experiments have shown explosions to be initiated when the molten material contacts the water container boundary. In addition, these same tests have shown that explosions do not occur for deep water pools. For the 7 m depth of the suppression pool, debris would be solidified by the time it reached the bottom of the pool, i.e., and explosion would not be initiated.

QUESTION H.04

The Core Dispersal Model involves large dispersal forces. What is the effect on the integrity of the containment of such large dispersal forces?

RESPONSE

As evaluated in the Limerick Study, the specific geometry of the pedestal, specifically the windows for the CRD hydraulic lines, provides sufficient bypass for primary system blowdown in the high pressure sequences that substantial dispersal of material would be unlikely. If such dispersion were to occur, the principal result would be the transmission of debris into the suppression pool where it would be quenched. Dispersal forces resulting from the blowdown would be small compared to the structural capabilities of the primary containment boundary.

QUESTION H.05

The above questions imply (for those accident sequences with the containment intact at vessel failure) that there would be the potential for containment failure at vessel failure rather than due to gradual overpressurization failure. What is the impact of this on the appropriateness of the release categories and its influence on risk?

RESPONSE

As discussed in the response to H.02, the calculated containment pressurization at vessel failure results from the assumed vessel failure mechanism in the code. Such global failure is not a credible mechanism for the BWR vessel and its associated penetrations for control rod drives and in-core instrument tubes. Consequently, it is not a mechanism to be included in the realistic assessments for evaluating the risk due to these low probability events.

QUESTION I.01

Provide a description of the process used to discover Limerick plant-specific intersystems dependencies and common cause failures. For example, those compromises in redundancy due to maintenance and testing procedures, HVAC dependencies, AC power dependence upon DC control (DC control of EDG start), EDG support systems dependencies, the assumption that equipment can perform in the hostile environment resulting from the accident initiator, and location dependent failures. Include a summary of the contrasts between the process used to discover intersystems dependencies at Limerick versus the process used in WASH-1400. Provide a description of the mathematical and graphical (ET/FT) methods used to accommodate the increased probabilities of failure due to the discovered dependencies.

RESPONSE

(a) WASH-1400 identified common cause failures of redundant systems as a possible contributor to core melt frequency. Subsequently the incident at TMI-2 demonstrated that adverse effects on safety systems could be cause[illegible]forseen system interaction. In recognition of [illegible] potential for important accident sequences to be dominated by common cause or systems interaction problems, the LGS PRA affords PECo the following:

1. A quantitative assessment of the risk associated with the plant.

2. A logic model of plant systems which can be evaluated both qualitatively and quantitatively.

3. A framework within which systems interactions can be identified and assessed.

The PRA LGS methodology affords a systematic approach to the identification of postulated accident sequences and the failures which can cause accident sequences including those attributed to systems interaction or common cause.

The event tree/fault tree methodology augments the other more deterministic approaches used by PECo to ensure plant safety. A major advantage of the use of the event tree methodology is that it provides an overview of the postulated accident sequences and the plant functions required to maintain public safety. This overview is useful in assessing the impact of sequence dependent functional failures of systems. In addition to the functionally induced failures there are also

system interfaces, support systems, human interactions, and other system interactions which are accounted for in accident sequences via the system level fault trees. Their importance is provided quantitatively through their contribution to risk.

The types of systems interactions which have been classified thus far include the following (2):

1. <u>Functionally coupled</u>

2. <u>Spatially coupled</u>

3. <u>Humanly coupled</u>

These effects have been investigated in the LGS PRA using the fault tree event tree approach as limited by the groundrules and are discussed below and in the LGS PRA.

Event tree/fault tree techniques are one of the mature methodologies available today to gain additional insight and perspective regarding the potential for as yet unidentified systems interactions affecting the public safety. The use of fault tree/event tree logic models are the best available method for identification and evaluation of systems interaction (5). This methodology represents a highly structured framework that has the following desirable attributes; it is

- systematic
- flexible
- reproducible
- simple
- understandable

Within the context of the PRA the following techniques were used to isolate system dependencies and interactions:

1. System walkdown and plant familarization.

2. Previous PRA/Systems Interactions Study reviews.

3. Implementation of containment interactions as they affect plant systems.

4. Review of LWR operating experience, e.g., LERS.

5. Incorporation of the above in a fault tree event tree framework for consistent evaluation.

(b) Specific Items Requested to be Discussed:

(c) Comparison of the Limerick PRA Approach to Systems Interactions Versus Other Studies

In this part of the response a brief summary comparison is made among the analysis options which are available. Principally these methods include:

1. Event tree/fault tree methodology as described in the PRA Guide (1), or IREP,

2. Dependency diagrams

3. WASH-1400

Table 2 summarizes some of the techniques utilized in each analysis method. The purpose of Systems Interaction analysis (3) is the identification of potential functional and human coupling or interfaces which could defeat multiple systems. The PRA on the other hand has a much broader charter since it must identify failure modes of systems and multiple system, evaluate these (qualitatively and quantitatively), and assess public risk associated with such failures. However, in order for the PRA to effectively accomplish its purposes it must treat the systems interactions issue in a thorough manner. (See d below)

(d) The description of methods used to account for intersystem dependencies is presented in Appendix B of the LGS PRA.

The principal methods used are the following:

- dependency matrices for identification of system dependencies

- system fault trees for incorporation of the intrasystem auxiliary system, and support system dependencies

- functional fault trees and event trees for the incorporation of the inter-system dependencies, interfaces, and sequence dependent effects

Boolean Algebra is used to confirm that the systems within an accident sequence do correctly account for the systems interaction effects. In many instances the effects inter-system dependencies are minimal because of the degree of separation designed into the plant.

REFERENCES

1. PRA Procedures Guides, A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants, NUREG/CR-2300, dated September, 1981.

2. Coffman, F., et. al., The Development of Interim Guidance on Systems Interactions, Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, dated September, 1981.

3. Allesso, H. P., et. al., On Issues Important to the Development of a Systems Interaction Evaluation Procedure, Proceedings of the International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, dated September, 1981.

4. Cybulskis, P., et. al., Review of Systems Interaction Methodologies, Battelle Columbus Laboratories, NUREG/CR-1896, dated January, 1981.

(b) Specific Items Requested to be Discussed:

ITEM	METHOD USED IN PRA
Maintenance Procedures and Test	A two fold approach was taken to on-line maintenance: (a) Those cares in which maintenance occurs on-line are modeled explicitly along with the dependencies of other systems based upon LCOs. (b) The possibility that LCOs would be violated and multiple systems would be unavailable as also incorporated. In addition the possibility that maintenance actions would lead to disabling of equipment required for safe shutdown and also included. Further, test and maintenance errors which could disable multiple channels of instrumentation were accounted for in the construction and quantification of the fault trees. However other components do not have intercomponent dependencies due to common mode maintenance errors were modeled as stated in the Groundrules.
HVAC Dependencies	HVAC is not required for the operation of Limerick equipment, in fact HVAC is isolated in accident signals so that it will not be in general available. Component cooling is provided through the emergency service water system which is explicitly modeled in the fault trees.
AC Power Dependence Upon DC Power DC Control of Diesel Generator Start Emergency Diesel Support Systems	Each system is directly dependent upon both AC and DC power. This direct dependency is included within each system fault tree. DC power depends upon the power and therefore the diesel generators through battery charging and inverter circuits. AC power depends upon DC power. These dependencies could result in fault tree circular logic unless care is taken to implement the dependencies within the system level fault trees with some care. Therefore the DC power dependency of each system is explicitly modeled at a higher level within each system fault tree. It should be noted that for

	the diesels a detailed fault tree was not performed but rather operating experience data was used to model these systems. The resulting probability of single and multiple diesel failure which was used in the event tree quantification is far higher than the calculated contribution from loss of a single DC bus or multiple AC buses.
Equipment Peformance in Hostile Environment	Several cases of equipment required to operate in potentially hostile environment are included in the LGS PRA. These cases include: (1) If containment is breached due to overpressure because of the inability to remove heat from containment, (this could be either a Class II* or IV** accident) then the environment present in the reactor building following this has been assumed to lead to an unacceptable environment in the reactor building. Therefore continued coolant makeup to the reactor is conservatively assumed as unavailable. (the result is the same as WASH-1400 assumptions) (2) The environment inside containment may exceed the qualification envelope for APS values and electrical support systems. This has been accounted for. (3) No recovery from degraded core conditions has been treated since the environmental conditions existing following severely degraded cores are not easily identifiable. (same as WASH-1400)

*TW

**ATWS

Table 2

COMPARISON OF SYSTEM INTERACTION METHODOLOGIES

DEFINITION OF SYSTEM INTERACTIONS OR DEPENDENT - FAILURES		METHODS OF TREATING SYSTEM INTERACTIONS**			
Systems Interactions (2)	PRA Guide (1)	IREP (Not Published)	System Interaction (2)	Limerick PRA	WASH-1400
Functionally Coupled	Intersystem Dependencies - Functional - Shared	- Event Trees - Fault Trees - Dependency Matrix	Dependency Matrices Diagraphs	- Event Trees - Fault Trees - Dependency Matrix	- Event Trees - Fault Trees
Spatially Coupled	- Physical - External Events	NA	Methodology not Developed to point where this has been demonstrated	- Room cooling evaluation - Walkdowns - Containment Interaction	- Walkdown - Containment Interaction
Humanly	Human-Interaction	Limited Development	Dependency Matrices Diagraphs	- Fault Tree Input - Swain Guttman	- Fault Tree Input
	Intercomponent dependencies	Not performed	Not addressed	Not addressed	- Not addressed

* Excludes human error and sabotage (2).

** A walkdown was used to augment these methods.

Appendix J

QUESTION J.01

Provide documentation showing that the suppression pool water static load does not affect the containment failure pressure.

Provide documentation showing how the dynamic loading, due to blowdown of steam into the suppression pool (either through the downcomers or from SRV's), has been factored into the failure pressure. Additionally, how do the dynamic forces on the diaphragm floor, during blowdown of the core through the vessel, effect the containment failure pressure?

RESPONSE

The containment design includes provision for the static load of the water in the suppression pool. This loading is included in all the structural analysis associated with the containment design. Since the containment was designed to meet the requirements of appropriate codes and specifications, the PRA did not verify the containment design analysis.

The structural analysis performed on the Limerick PRA was a static analysis to determine the ultimate pressure capability of the containment. Pressure loading input to the structural analysis was provided by the INCOR analysis and is discussed in Appendix C, with bounding pressure-time curves shown on Figures C.3, C.4, C.5, C.6, C.8, C.10, and C.12. Areas of uncertainty in the analysis (associated with RPV failure) are shown on the curves and discussed in the text. For Class II and Class IV accidents, the containment is assumed to have failed prior to RPV failure. For Class I and Class III accidents, the containment is intact at the time of RPV failure. Containment failure was assumed to occur at the time of failure of the diaphragm floor due to dynamic forces and weakening of the containment structure.

Dynamic loading due to blowdown of steam into the suppression pool was not evaluated as a contributor to containment failure. Blowdown loads would be localized in the suppression pool and would be attenuated so as to be insignificant in their effect on containment structure to the static pressure load and capability.