APPENDIX H

INITIAL PRA RESULTS FOR THE BWR 3/4 PLANT GROUP

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H.1 INTRODUCTION

For initial evaluations of the potential for generic changes to emergency core cooling system (ECCS)-related regulatory requirements for the boiling water reactor (BWR) 3/4 plant group, Peach Bottom 2&3 units were selected. The Peach Bottom units have been examined extensively using probabilistic risk assessment (PRA) -- the Reactor Safety Study, NUREG-1150, the Peach Bottom Individual Plant Examination (IPE), and the Peach Bottom SPAR model. The Peach Bottom units are representative of plants with 4 core spray pumps, 4 AC divisions per unit and above average emergency power reliability. Pending the outcome of this initial evaluation, units such as Brunswick 1&2 may need to be evaluated as representative of plants with 2 core spray pumps and 2 AC power divisions per unit.

The analyses performed for Peach Bottom 2&3 are discussed in detail in this appendix. The Level 1, Revision 3i ("i" stands for "interim") SPAR model for the Peach Bottom units was first changed to incorporate the best available pipe break frequencies, to specifically address the conditional probability of loss of offsite power (LOOP) given a loss of coolant accident (LOCA), and to eliminate nonrepresentative plant-specific features. The modified base case model obtained in this manner was examined to test the quantitative significance of alternative injection via non-safety systems. Studies were then performed to examine the feasibility of changing ECCS technical specifications, reducing ECCS redundancy, eliminating core spray pumps, or eliminating LOOP as a design basis event for large-break LOCAs and medium-break LOCAs.

H.2 PEACH BOTTOM 2&3 SPAR 3i MODEL

The SPAR 3i model for Peach Bottom 2&3 includes 22 event trees. Seven event trees delineate interfacing system LOCA sequences that lead to a core-damage-vulnerable (CD-V) endstate. The remaining 15 event trees delineate sequences leading to the core damage (CD) endstate. These core-damage event trees are listed in Table H.1. The event trees named with numerical suffixes are referred to as continuation trees, transfer trees or subtrees. For example, LOOP-1, LOOP-2, and LOOP-3 are subtrees of the LOOP tree. The LOOP tree transfers to the LOOP-1, the LOOP-2, or the LOOP-3 subtree given one, two, or more stuck-open safety-relief valves (SRVs), respectively. Similarly, the LODCB, LOSW, and TRAN event trees transfer to the TRAN-1, the TRAN-2, or the TRAN-3 subtree given one, two, or more stuck-open SRVs, respectively.

Table H.1 Peach Bottom 2&3 SPAR 3i Event Trees Leading to Core Damage Endstate

Event Tree	Description
ATWS	Anticipated transients without scram (transfer from TRANS for reactor protection system [RPS] failure)
ATWS-1	ATWS continuation tree
LODCB	Loss of DC bus
LLOCA	Large break-induced LOCA
LOOP	Loss of offsite power
LOOP-1	Transfer tree for LOOP sequences with one stuck-open SRV
LOOP-2	Transfer tree for LOOP sequences with two stuck-open SRVs
LOOP-3	Transfer tree for LOOP sequences with three or more stuck-open SRVs
LOSWS	Loss of service water
MLOCA	Medium break-induced LOCA
SBO	Station blackout (transfer from LOOP given emergency AC power failure)
SLOCA	Small break-induced LOCA
TRAN	Transients
TRAN-1	Transfer tree for transients with one stuck-open SRV
TRAN-2	Transfer tree for transients with two stuck-open SRVs
TRAN-3	Transfer tree for transients with three or more stuck-open SRVs

Excluding the ATWS tree and the ATWS-1 subtree, the Peach Bottom 2&3 event trees listed in Table H.1, utilize 22 fault trees. These fault trees and their base case quantifications are summarized in Table H.2. The fault trees are listed in the order they occur in the event trees. This order is depicted in Table H.3. The effective failure probability for any of the listed fault trees for a given accident sequence path on an event tree may be greater than that listed in Table H.2 due to the occurrence of failure events earlier in the sequence. For example, the failure probability of AC powered systems is greater in sequences involving LOOP. Note that the Peach Bottom 2&3 SPAR 3i model does not credit alternate injection via the firewater system.

Table H.2 Peach Bottom 2&3 SPAR 3i Fault Trees

Fault Tree	Description	Minimum Pt.Est. Base Case Top Event Probability
L	Small LOCA recovery fails	5.0E-1
RPS	Reactor protection system fails	2.4E-6
EPS	Emergency AC power system fails	4.1E-4
PC1	Power conversion system fails in LODCB, LOSWS, or SLOCA	7.0E-2
PCS	Power conversion system fails in transient (TRAN) event tree	3.7E-1
SRV	One or more SRVs fail to close	1.8E-1
EARLY COOLANT INJECTION		
RCI	Reactor core isolation cooling (RCIC) fails	6.6E-2
HCI	High pressure coolant injection (HPCI) fails	2.5E-1
HC1	HPCI fails (1 SORV)	6.7E-2
DEP	Depressurization	5.0E-4
DE1	Depressurization (early, 1 SORV)	5.0E-4
DE2	Depressurization (early, 2 SORVs)	5.0E-4
CR1	Control rod drive (CRD) hydraulic system fails (single-pump mode)	2.4E-3
CRD	CRD hydraulic system fails (enhanced flow mode)	1.1E-2
CDS	Condensate system fails (transients)	3.8E-2
CD1	Condensate system fails (LOSW & SLOCA)	4.6E-3
LCS	Core spray system fails	6.2E-4
LCI	Low pressure coolant injection (LPCI) fails	3.9E-4
VA	Alternate injection via high-pressure service water (HSW) fails early	1.1E-2
LONG-TERM DECAY HEAT REMOVAL & ALTERNATE INJECTION		
DEP	Depressurization	5.0E-4
SPC	Suppression pool cooling mode of residual heat removal (RHR) fails	1.6E-3
CSS	Containment spray mode of RHR fails	1.6E-3
SDC	Shutdown cooling mode of RHR fails	7.7E-3
CV1	Containment vent system fails (non-ATWS)	1.2E-3
CVS	Containment vent system fails (ATWS)	9.1E-3
CR1	Control rod drive hydraulic system fails (delayed, 1 pump)	2.4E-4
VA1	Alternate injection via high-pressure service water (HS1) fails late	1.1E-2
VA2	Alternate injection via high-pressure service water (HS1), LPCI, or core spray fails late	1.0E-3

Table H.3 Peach Bottom 2&3 SPAR 3i Order of Fault Trees in Non-ATWS Event Trees

Event Tree	Fault Trees																			
LODCB		RPS		PC1	SRV	RCI	HCI	DEP	CRD	CD1	LCS	LCI	VA	SPC	DEP	SDC	CSS	CV1	CR1	VA1 or VA2
LLOCA		RPS									LCS	LCI		SPC			CSS	CV1		VA1
LOOP		RPS	EPS		SRV	RCI	HCI	DEP	CRD		LCS	LCI	VA	SPC	DEP	SDC	CSS	CV1	CR1	VA1 or VA2
LOOP-1						RCI	HC1	DE1			LCS	LCI	VA	SPC			CSS	CV1	CR1	VA1 or VA2
LOOP-2							HC1	DE2			LCS	LCI	VA	SPC			CSS	CV1		VA1
LOOP-3											LCS	LCI		SPC			CSS	CV1		VA1
LOSWS		RPS		PC1	SRV	RCI	HCI	DEP	CRD	CD1	LCS	LCI	VA	SPC	DEP	SDC	CSS	CV1	CR1	VA1 or VA2
MLOCA		RPS					HC1	DEP			LCS	LCI	VA	SPC			CSS	CV1		VA1
SLOCA	L	RPS		PC1		RCI	HC1	DEP		CD1	LCS	LCI	VA	SPC	DEP		CSS	CV1		VA1 or VA2
TRANS		RPS		PCS	SRV	RCI	HPI	DEP	CRD	CDS	LCS	LCI	VA	SPC	DEP	SDC	CSS	CV1	CR1	VA1 or VA2
TRAN-1						RCI	HC1	DE1		CDS	LCS	LCI	VA	SPC			CSS	CV1	CR1	VA1 or VA2
TRAN-2							HC1	DE2		CDS	LCS	LCI	VA	SPC			CSS	CV1		VA1
TRAN-3											LCS	LCI		SPC			CSS	CV1		VA1

Two fault trees, HCI and HC1, are used for the HPCI system. The HCI fault tree model assumes that HPCI is initially aligned to take suction from the condensate storage tank and that realignment of suction to the suppression pool will not be required, but is available as an option. The HCI fault tree, which gives a point-estimate failure probability of 2.4×10^{-1}, is used in the LOOP, LOSWS, and TRANS event trees for sequences that do not involve stuck-open relief valves. In such sequences, on-off cycling of HPCI can occur due to changes in water level. The HC1 model, which gives a point-estimate failure probability of 6.74×10^{-2}, is used in LOCA or SORV sequences which may require transfer of the suction from the CST to the suppression pool before low steam pressure fails the HPCI turbine, and therefore requires both suction paths to be available. The difference between the HCI and HC1 failure probabilities is mainly due to a single event cutset HCI-MOV-CC-IVFRO (HPCI injection valve fails to reopen), with a mean failure probability of 2.0×10^{-1}. This event is associated with on-off cycling of HPCI in transient sequences that do not involve stuck-open relief valves.

Two fault trees, CDS and CD1, are used for the condensate system. The CDS fault tree, which gives a point estimate failure probability of 3.8×10^{-2}, is used in the TRANS, TRANS-1, & TRANS-2 event trees. The CD1 fault tree, which gives a point estimate failure probability of 4.6×10^{-3}, is used in the LODCB, LOSWS, & SLOCA event trees. The difference between the two failure probability estimates is attributable to a single-event cutset, CDS-SCRAM-NOCDS (Reactor trip results in a loss of condensate in general transients), with a mean failure probability of 3.0×10^{-2}. This event accounts for the fact that based on operational experience, approximately 3% of general transients directly result in loss of the condensate system. The probability of loss of condensate coincident with reactor trip is assumed to be negligible for the LODCB, LOSWS, and SLOCA sequences (CD1 fault tree).

Two fault trees, CRD and CR1, are used for the control rod drive hydraulic system. The CRD fault tree is used to model the failure probability of the system in the enhanced flow mode. Success is defined as 2-of-2 pumps running, and an injection path available. The injection path is from the condensate storage tank (CST) and through either the cooling water header or charging header. To achieve full enhanced flow the operator must place the standby pump in service and make sure one flow control valve is fully open. The use of CRD in this model follows the NUREG/CR-4550 model for Grand Gulf. CRD is asked only if depressurization fails, based on the assumption that enhanced CRD flow would not be sufficient to prevent reactor water level from reaching TAF. The current model credits enhanced injection though the orificed charging water header which limits flow to about 180 gpm. Should new calculations show this to be overly optimistic, setting event CRD-PSF-HW-CWHDR to TRUE would force the model to credit only the cooling water header. The CRD fault tree is used in the LOOP, LOSWS, and TRANS event trees for high-pressure sequences in which RCIC, HPCI, and depressurization fail.

The CR1 fault tree is used to model the failure probability of CRD injection using one CRD pump. Success is defined as 1-of-2 pumps running, and 1 injection path available. The injection path is from the CST to either the cooling water header or charging header. The CR1 model is used for long-term injection in LOOP, LOSW, and TRANS sequences in which depressurization fails. No other high-pressure injection system is credited for such sequences.

Four fault trees, VA, VA1, VA2, and VA3, are used to model failure of the alternate low-pressure injection systems to provide a low-pressure injection source. VA is used in the early timeframe.

VA1, VA2, or VA3 is used in the late timeframe. VA and VA1 only credit high pressure service water (HPSW) as an injection source. VA2 credits core spray (CS), LPCI, and HPSW injection. In the most limiting case, long-term alternate low pressure injection is modeled by fault tree VA1, which only credits HPSW injection. In cases where ECCS suction on the suppression pool remains possible, fault tree VA2 is used to credit alternate injection via core spray, LPCI, or HPSW. Fault tree VA3 credits firewater or any other AC-independent injection sources, but VA3 only appears in the SBO event tree and its failure probability is set to 1.0 in the SPAR 3i model for Peach Bottom.

H.3 PEACH BOTTOM 2&3 BASE CASE

The base case for Peach Bottom 2&3 analyses was the original SPAR 3i model with five changes. The changes and their individual and collective impacts on predicted core damage frequency (CDF) are summarized in Table H.4. In this and future tables, CDFs are stated per reactor critical year rather than per reactor calendar year. To convert CDFs stated per reactor critical year to CDFs per reactor calendar year, multiply by the industry average criticality factor of 75%. The changes to the original SPAR 3i model for Peach Bottom are discussed in the following paragraphs.

Table H.4 Changes to SPAR 3i for Peach Bottom 2&3 Base Case

Case	Description	Pt.Est. CDF (crit yr)$^{-1}$	Pt.Est. ΔCDF (crit yr)$^{-1}$	Ratio Case / Case 1
0	Peach Bottom 2&3 SPAR 3i model	5.1E-6	9.8E-8	1.018
1	Modify model to use VA2 fault tree to credit core spray and LPCI for alternate injection where appropriate in transients with stuck-open relief valves (SORVs)	5.0E-6	--	1
2	Require only one of four 18-inch vent lines to open for successful venting in non-ATWS accidents	3.1E-6	-2.0E-6	0.606
3	Modify AC power fault trees to model the probability that a break-initiated LOCA induces LOOP[a]	5.0E-6	1.3E-9	1.000
4	Modify AC power fault trees to model the probability that a transient induces LOOP.[a]	5.1E-6	7.6E-8	1.015
5	Change initiating break frequencies from NUREG/CR-5750 to NRC-updated values[a]	5.1E-6	9.3E-8	1.018
6	Fail backup offsite power from Conowingo dam[a]	6.2E-6	1.2E-6	1.229
Base Case	Apply all of the preceding changes to the original Peach Bottom SPAR 3i model.[a]	5.2E-6	1.7E-8	1.021

[a] The VA2 correction made in Case 1 was applied to all subsequent cases.

First the event tree logic for alternative injection was modified to credit core spray and LPCI consistent with the plant's design features. For small LOCAs and transients with one stuck-open relief valve, as indicated in the SPAR 3i model documentation, the VA2 tree should be instead of the VA1 tree when ECCS suction on the suppression pool remains possible. Case 1 correctly credits core spray and LPCI for long-term alternative injection. As indicated in Table H.4, the CDF predicted by the original Peach Bottom SPAR 3i model is slightly greater than that predicted in Case 1. The Case 1 correction was applied in all other calculations, and the point-estimate ΔCDFs in Table H.4 are with respect to Case 1.

At Peach Bottom, four 18-inch lines would be required to successfully vent containment in an ATWS, but only a single 6-inch line is required for other accidents. The original SPAR 3i model applies the ATWS success criteria to all sequences. To avoid the conservatism inherent in this approximation, an alternative containment venting fault tree (CV1) was developed. The CV1 fault tree requires only one of the four 18-inch lines for successful venting in non-ATWS sequences. As indicated in Case 2 of Table H.4, using the CV1 fault tree to replace CVS in non-ATWS event trees reduces the point-estimate CDF by nearly 40% relative to Case 1.

In Case 3, the electric power fault trees were modified to permit the occurrence of LOOP given a break-initiated LOCA to be modeled. The mean probability is 2.3×10^{-2} based on the analysis presented in Appendix G for units that supply AC power to plant components during normal operation from the main generator via an auxiliary transformer. A slightly lower mean of 2.0×10^{-2} applies when AC power is supplied to plant components from offsite power through a startup transformer during normal operation. Because the BWR 3/4 group includes both configurations, the larger of the two values was selected for the analyses presented here.[1] The point-estimate increase in CDF associated with adding the LOCA-LOOP event to Case 1 is 1.3×10^{-9} per critical year. A lognormal distribution with mean of 2.3×10^{-2} an error factor of 3.7 was used to model the uncertainty in the LOCA-LOOP probability. The 95-th percentile of this distribution is a LOCA-LOOP probability of 6.2×10^{-2}.

In Case 4, the electric power fault trees were modified to model the occurrence of LOOP given a transient initiator. The mean probability was based on 8 occurrences of LOOP in 3415 reactor trips. A beta distribution with a mean of 2.342×10^{-3} and parameter b=3407 was used to model the uncertainty. When multiplied by the frequency (1.5 per critical year) of transient initiators, this gives 3.5×10^{-3} transient-initiated LOOP events per critical year. In comparison, the frequency of LOOP as an initiator is 4.6×10^{-2} per critical year. The point-estimate increase in CDF associated with adding the TRAN-LOOP event to Case 1 is 7.6×10^{-8} per critical year. As currently implemented, power recovery is not modeled for transient-induced LOOP.

In Case 5, the pipe break frequencies were changed from the BWR values provided in NUREG/CR-5750 to more recent U.S. Nuclear Regulatory Commission (NRC) values, which are developed in Appendix F and summarized in Table H.5. This change increased point-estimate CDF by 9.3×10^{-8} per critical year relative to Case 1. Virtually all of this increase is attributable to the increase in the small-break initiator frequency by a factor of ~3.7 relative to NUREG/CR-5750.

Backup AC power from the Conowingo dam is a Peach Bottom feature that is not representative of other plants in the BWR 3/4 group. Accordingly, in Case 6, failure of the backup power source was set to TRUE. As indicated in Table H.4, this change by itself increased point-estimate CDF by 1.2×10^{-6} per critical year relative to Case 1.

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[1]Subsequent to the calculations described here, the mean conditional probability of LOOP given a LOCA was revised upward in Appendix G from 2.3×10^{-2} to 2.4×10^{-2}. This change has no significant impact on the results provided in this appendix.

Table H.5 BWR Break Frequencies

		Large	Medium	Small	Units
Break Area		>0.1	0.004-0.1	<0.004	ft^2
Equivalent Inside Diameter		>5	liquid 1-5 steam 4-5	liquid <1 steam <4	inches
NRC Interim Update to BWR Break Frequencies[a]	5% **Mean** 95% **Error Factor**	2.0E-6 **5.2E-5** 2.0E-4 **10**	3.4E-6 **9.1E-5** 3.4E-4 **10**	3.7E-4 **1.5E-3** 3.7E-3 **3.2**	per calendar year **per calendar year** per calendar year
Criticality Factor[b]		0.75			
NRC Interim Update to BWR Break Frequencies[c]	5% **Mean** 95% **Error Factor**	2.6E-6 **6.9E-5** 2.6E-4 **10**	4.6E-6 **1.2E-4** 4.6E-4 **10**	5.0E-4 **2.0E-3** 5.0E-3 **3.2**	per critical year **per critical year** per critical year
hours/year		8760			
NRC Interim Update to BWR Break Frequencies[d]	5% **Mean** 95% **Error Factor**	3.0E-10 **7.915E-09** 3.0E-08 **10**	5.2E-10 **1.385E-8** 5.2E-8 **10**	5.7E-8 **2.283E-7** 5.7E-7 **3.2**	per critical hour **per critical hour** per critical hour

[a] NRC interim update to NUREG/CR-5750 values documented in Attachment F.
[b] Industry average criticality factor used in NUREG/CR-5750.
[c] Break frequency per calendar year from NRC interim update divided by industry average annual criticality factor (0.75 critical years per calendar year).
[d] Break frequency per calendar year from NRC interim update divided by average annual criticality factor and 8760 hours per year. Saphire7 requires frequencies be stated per critical hour. Four significant digits are carried in the mean frequencies to avoid compounding roundoff errors when converting frequencies for and from Saphire7.

As indicated in the bottom row of in Table H.4, the point estimate core damage frequency increased from 5.0×10^{-6} per critical year for Case 1 to 5.2×10^{-6} per critical year with all five of the preceding changes implemented together. A Monte Carlo sample of size 99,999 was used to estimate the following parameters of the base case CDF distribution:

5-th percentile:	8.7×10^{-7} per critical year
median:	3.0×10^{-6} per critical year
point estimate:	5.2×10^{-6} per critical year
mean:	6.0×10^{-6} per critical year
95-th percentile:	2.0×10^{-5} per critical year

This case is adopted as the Peach Bottom base case in the discussions that follow.

Table H.6 summarizes the point estimate initiator frequencies, core damage frequencies, and conditional probabilities of core damage for the base case by event tree. The interfacing system LOCA results are conservative in that the event trees are only taken to a core vulnerable state. LOOP and general transients (TRAN) are the dominant contributors to the base-case CDF. The conditional probabilities of core damage are less than 0.01 for all accident classes except SBO.

Table H.6 Initiator Frequency, CDF, and CPCD by Main Event Tree

Event Tree	Pt.Est. Init.Freq. (yr^{-1})	Pt.Est. CDF (yr^{-1})	Pt.Est. CPCD
ATWS	3.6E-6[a]	2.2E-8	6.1E-3
ISLOCA	2.7E-2	1.2E-9	4.6E-8
LLOCA	6.9E-5	2.2E-8	3.2E-4
LODCB	2.1E-3	5.3E-7	2.5E-4
LOOP	4.6E-2	2.2E-6	4.9E-5
LOSWS	9.6E-4	7.8E-9	8.1E-5
MLOCA	1.2E-4	3.1E-8	2.6E-4
SBO	1.9E-5[b]	5.7E-7	3.1E-2
SLOCA	2.0E-3	1.2E-7	6.2E-5
TRAN	1.5E+0	1.6E-6	1.1E-6
TOTAL	1.6E+0	5.2E-6	3.3E-06

[a] Estimated from TRAN tree branch fractions
[b] Estimated from LOOP tree branch fractions

Table H.7 summarizes base-case initiating frequencies, core damage frequencies, and conditional probabilities of core damage (CPCD) by LOCA type. The LOCA contribution to CDF is dominated by transients with one stuck-open SRV. Even with very conservative initiating frequencies, the interfacing system LOCAs are not significant contributors. Collectively, the LOCA sequences contribute 4.8×10^{-6} per critical year to the point-estimate CDF. This is two thirds of the total point-estimate CDF for the base case.

The Peach Bottom SPAR model delineates 327 core damage sequences. The base case core-damage expression contains 33,270 cutsets; however, five cutsets account for 40% of the point-estimate CDF. Table H.8 summarizes these five cutsets. The events listed in the dominant cutsets in Table H.8 are described in Table H.9.

The first cutset, which accounts for 22.3% of the base-case point-estimate CDF, is for the LOOP-54-3 sequence. Following a LOOP initiator, the reactor scrams, the power conversion system (PCS) fails due to LOOP, one SRV sticks open, and RCIC succeeds but trips on low steam pressure caused by the stuck-open SRV. The suppression pool cooling (SPC) mode of RHR succeeds, but alternate injection fails. In the dominant cutset operator errors fail both single-pump CRD and alternate injection systems.

Table H.7 Initiating/Branch Frequencies, CDFs and CPCD Values by LOCA Type

Core Damage Sequences	Pt.Est. Initiator Freq.(yr^{-1})	Pt.Est. CDF (yr^{-1})	Pt.Est. CPCD
Break-Initiated LOCAs			
LLOCA	6.9E-5	2.2E-8	3.2E-4
MLOCA	1.2E-4	3.1E-8	2.6E-4
SLOCA	2.0E-3	1.2E-7	6.2E-5
Subtotal	2.2E-3	1.8E-7	8.1E-5
Transients with Stuck-Open SRVs			
One Stuck-Open SRV	1.1E-1	2.9E-6	2.7E-5
Two Stuck-Open SRVs	7.8E-4	1.8E-7	2.3E-4
Three or More Stuck-Open SRVs	1.3E-4	4.2E-8	3.1E-4
Subtotal	1.1E-1	3.1E-6	2.8E-4
Interfacing System LOCAs	2.7E-2	1.2E-9	4.6E-8
LOCA SUBTOTAL	1.4E-1	3.3E-6	2.4E-5
Non-LOCA Sequences	1.4E-0	1.9E-6	1.3E-6
TOTAL	1.6E-0	5.2E-6	3.3E-6

Table H.8 Dominant Sequences and Example Cutsets for Peach Bottom Base Case

ACCIDENT SEQUENCE		DOMINANT CUTSET(S)	Freq/Prob[a]
LOOP 54-03, CDF=1.2E-6, 40 CUTSETS			
IE-LOOP	Initiating event is LOOP	IE-LOOP	4.6E-2
/RPS	Reactor Protection System succeeds		
/EPS	Emergency AC power succeeds		
SRV(P1)	One SRV sticks open	*PPR-SRV-00-1VLV	0.18
/RCI	RCIC injection succeeds		
/SPC	RHR succeeds in SPC mode		
CR1	Single-pump CRD injection fails	*CRD-XHE-XM-BRKRS	1E-3
VA2	Alternate injection fails	*OPR-XHE-XM-ALPI5	0.14
		Product	1.2E-6

Table H.8 Dominant Sequences and Example Cutsets for Peach Bottom Base Case

ACCIDENT SEQUENCE		DOMINANT CUTSET(S)	Freq/Prob[a]
SBO-04, CDF=3.4E-7, 109 CUTSETS			
IE-LOOP	Initiating event is LOOP	IE-LOOP	4.6E-2
/RPS	Reactor Protection System succeeds		
EPS	Emergency AC power fails	*EPS-DGN-CF-DGNS	3.9E-4
/SRV	No SRVs stick open		
/RCI	RCIC succeeds		
/DEP	Manual depressurization succeeds		
VA3	Alternate injection via firewater fails	*TRUE	1.0
AC	AC power is not recovered in 4 hours	*ACP-XHE-NOREC-4H	2.3E-2
		Product	3.4E-7
TRAN 60-10, CDF=8.4E-7,[a] 1841 CUTSETS			
IE-TRAN	Initiating event is general transient	IE-TRAN	1.5
/RPS	Reactor Protection System succeeds		
PCS	Power Conversion System fails	*TRAN-LOOP	2.3E-3
SRV(P1)	One SRV sticks open	*PPR-SRV-00-1VLV	0.18
SPC	RHR fails in SPC mode		
CSS	RHR fails in containment spray mode	*EPS-DGN-CF-DGNSR	3.9E-4
CVS	Containment vent system fails		
		Product	2.5E-7
LODCB-59, CDF=3.8E-7, 23 CUTSETS			
IE-TDCB	Initiator is loss of DC bus	IE-TDCB	2.1E-3
/RPS	Reactor Protection System succeeds		
/SRV	No SRVs stick open	*/SRV	0.8188
RCI	RCIC fails		
HCI	HPCI fails	*HCI-MOV-CC-IVFOR	0.2
DEP	Depressurization fails	*ADS-XHE-XE-MDEPR	5.E-4
CRD	Enhanced-mode CRD injection fails		
		Product	1.7E-7
IE-TDCB	Initiator is loss of DC bus	IE-TDCB	2.1E-3
/RPS	Reactor Protection System succeeds		
/SRV	No SRVs stick open	*/SRV	0.8188
RCI	RCIC fails		
HCI	HPCI fails	*DCP-BDC-LP-DIV	9.0E-5
DEP	Depressurization fails		
CRD	Enhanced-mode CRD injection fails		
		Product	1.5E-7

[a] CDF, initiator frequencies, and cutset frequencies are per reactor critical year.

Table H.9 Top-Events by Base-Case Fussell-Vesely Importance

Event Name	Event Description	Fussell-Vesely
PPR-SRV-OO-1VLV	One SRV fails to close	55.8%
IE-LOOP	LOOP initiator	54.5%
IE-TRANS	General transient initiator	31.5%
OPR-XHE-XM-ALPI5	Operator fails to start/control alternate injection	25.5%
CRD-XHE-XM-BRKRS	Operator fails to close CRD-reactor building cooling water (RBCW) breakers	24.1%
EPS-DGN-CF-DGNS	Diesel generators fail from common cause	17.0%
TRANS-LOOP	General transient induces LOOP	14.9%
OEP-XHE-NOREC-10H	Operator fails to recover offsite power in 10 hours	12.3%
PCS-SCRAM-NOPCS	Reactor trip occurs with a loss of power conversion	11.5%
IE-TDCB	Loss of vital DC bus initiator	10.4%
RHR-XHE-XM-ERROR	Operator fails to start/control RHR	9.8%
EPS-DGN-FC-DGD	Diesel generators D fails to start and run	9.8%
EPS-DGN-FC-DGC	Diesel generators C fails to start and run	8.7%
ACP-XHE-NOREC-4H	Operator fails to recover offsite power in 4 hours	8.6%
ADS-XHE-XE-MDEP1	Operator fails to depressurize the reactor	7.5%
EPS-DGN-FC-DGB	Diesel generators B fails to start and run	6.8%
EPS-DGN-FC-DGA	Diesel generators A fails to start and run	6.6%
RCI-RESTART	Restart of RCIC is required	6.3%
RCI-TDP-FS-RSTRT	RCIC fails to restart given start and short-term run	6.3%
RCI-XHE-XL-RSTRT	Operator fails to recover RCIC failure to restart	6.3%
HCI-MOV-CC-IVFRO	HPCI injection valve fails to reopen	6.2%
ADS-XHE-XE-MDEPR	Operator fails to depressurize the reactor	6.1%
CVS-XHE-XM-VENT	Operator fails to vent containment	4.8%
HCI-TDP-FR-TRAIN	HPCI pump fails to run given it started	4.4%
EPS-DGN-TM-DGD	Diesel generator D is unavailable because of maintenance	4.1%
DCP-BDC-LP-DIV	Division IV 125V DC bus fails	4.0%

The second cutset, which accounts for 6.5% of the base-case point-estimate CDF, is for the SBO-4 sequence. Emergency AC power systems fail following a LOOP initiator. No SRVs stick open. RCIC succeeds. The reactor is manually depressurized, but AC power is not recovered within 4 hours. It should be noted that recovery events were not been modeled for cutsets involving transient-induced LOOP.

The third cutset, which accounts for 4.8% of the base-case point-estimate CDF, is for the TRAN 60-10 sequence. This sequence is initiated by a general transient that causes reactor trip. The power conversion system fails, one SRV sticks open, the suppression pool cooling and containment spray modes of RHR fail, and containment venting fails. In the cutset shown the system failures result from a transient-induced LOOP and common cause failures of the diesel generators. As mentioned previously, power recovery is not currently modeled for transient-induced LOOP.

The fourth and fifth cutsets, which accounts for 6.3% of the base-case point-estimate CDF, is for the LODCB-59 sequence. In this sequence, PCS fails as a result of an initiating loss of a vital DC bus. No relief valves stick open, but RCIC, HPCI, depressurization and CRD injection all fail leading to core damage. In the fourth cutset, RCIC and CRD fail due to the loss of DC bus initiator. In the fifth cutset, the system failures are caused by loss of a second DC bus.

Table H.9 gives the top 20 Fussell-Vesely importance measures for the Peach Bottom 2&3 base case. The event with the greatest Fussell-Vesely value is the sticking open of a single relief valve, which appears in the LODCB, LOSWS, and TRAN event trees. As previously indicated in Table H.7, this event appears in cutsets that contribute about 56% of the base case, point-estimate CDF. Several operator errors appear in Table H.9. These include operator failure to start/control RHR, operator failure to start/control alternate injection, and operator failure to close the CRD-RBCW breakers. The latter precludes CRD injection in both the enhanced-flow (early) and single-pump-flow (late) modes. The appearance of the TRANS-LOOP event and various diesel generator failure events indicates the importance of emergency electrical power systems (though these importances would be reduced if credit were given for electric power recovery in transient-induced LOOP scenarios). Operator failure to open the containment vent path also appears in Table H.9, indicating the importance of preventing containment failure in order to maintain RHR functions. Finally, the appearance of the initiating events for loss of offsite power (IE-LOOP), general transient (IE-TRAN), and loss of a vital DC bus (IE-TDBC) is consistent with their CDF contributions listed in Table H.6.

The SPAR 3i models for the BWR 3/4 group assign very low probabilities to operator errors to manually depressurize. Specifically, this probability is set to 10^{-3} everywhere it occurs in an event tree. This deemphasizes high-pressure meltdown scenarios that were found to be important in some IPEs. It is, on the other hand, conservative from the point of view of the analyses presented later because these analyses examine the possibility of relaxing existing regulatory requirements related to highly-redundant low-pressure ECC systems.

H.4 SENSITIVITY CASES

To determine the potential for changes to the ECCS-related regulatory requirements for the BWR 3/4 plant group, various sensitivity studies were performed using the Peach Bottom SPAR 3i base case model. The set of sensitivity studies completed to date include:

- Sensitivity of base case to injection via non-ECC systems
- Analysis of low-pressure ECCS technical specification requirements
- Sensitivity of single-outage risks to LPCI success criterion
- Analysis of low-pressure ECCS redundancy
- Analysis of design basis LOOP event for large- and medium-break LOCAs

A discussion of each of these sensitivity studies is provided in the following sections.

H.4.1 Sensitivity of Base Case to Injection via Non-ECC Systems

When they are available, credit is given for injection by several non-ECCS systems in the base case. High-pressure injection via the RCIC system and the PCS is credited early in transients and small-break LOCAs. Enhanced flow via both CRD trains is credited early, and single-pump CRD injection is credited late. Alternate injection via the high-pressure service water system is credited both early (fault tree VA) and late (fault trees VA1 and VA2). Low-pressure injection via the condensate system is credited early in transients and small LOCAs in which depressurization is successful. Low-pressure injection using the firewater system is not credited in the Peach Bottom 2&3 SPAR 3i model. Of the non-ECC systems that are credited, only RCIC is independent of AC power. The rest have motor-driven pumps.

The relative importance of non-ECCS systems was examined by sensitivity analyses. The non-ECCS injection systems were removed from the base case model individually by setting top events in the corresponding fault trees to TRUE. The results are depicted in increasing order of impact on CDF in Table H.10. These results indicate that the most important non-ECC injection mode is late, single-pump CRD injection. The predicted CDF increases by a factor of ~29 when credit for this injection mode is removed from the model.

When credit for all non-ECC injection systems is removed, the point-estimate, base case CDF increases to 4.7×10^{-4} per reactor critical year. A baseline frequency of this magnitude would preclude changes involving risk increases for a plant group according to the current framework for risk-informing 10 CFR Part 50 [Ref. H.1]. That is, credit for at least some non-ECC systems is needed to justify risk-informed changes to current regulatory requirements for ECC systems. Some increased level of regulatory treatment for non-ECC systems may, therefore, have to be part of any proposed change. In this appendix, credit is given for all of the non-ECC systems listed in Table H.10. Credit is not given for injection via the fire water system. Available PRA models for BWR 3/4 plants vary in the extent to which injection via non-ECC is credited. The extent of credit appropriate for other plants in the group requires further analysis.

Table H.10 Non-ECC Injection System Sensitivities

Case Description	Individually		Cumulative	
	Pt.Est. CDF (crit yr)$^{-1}$	Pt.Est. Ratio case/base	Pt.Est CDF (crit yr)$^{-1}$	Pt.Est. Ratio case/base
Base Case	5.2E-6	1	5.2E-6	1
No credit for early low-P condensate injection	5.2E-6	1.005	5.2E-6	1.005
No credit for early alternate injection via HPSW	5.5E-6	1.07	6.5E-6	1.27
No credit for late alternate injection via HPSW	7.0E-6	1.37	8.5E-6	1.64
No credit for early high-P PCS injection	7.5E-6	1.45	1.6E-5	3.05
No credit for early high-P enhanced CRD injection	1.0E-5	1.98	2.8E-5	5.37
No credit for early high-P RCIC injection	1.2E-5	2.30	2.2E-4	43.6
No credit for late high-P single pump CRD injection	1.5E-4	29.0	4.7E-4	91.3

H.4.2 Analyses of Low-Pressure ECCS Technical Specification Requirements

For 10 of the 22 BWR 3/4 units, including Peach Bottom 2&3, 4 core spray (CS) pumps provide flow to the 2 CS injection lines; that is, there are 2 CS subsystems each having two CS pumps. Subsystem A contains CS pumps A and C. Subsystem B contains pumps B and D. Core spray success requires 2 of the 4 CS pumps to function; that is, full flow from one subsystem or half flow from each subsystem.

For 19 of the 22 units in the BWR 3/4 group, including Peach Bottom 2&3, 4 RHR pumps can provide flow to 2 LPCI injection lines; that is, there are two LPCI subsystems each having two pumps. Subsystem A contains RHR pumps A and C. Subsystem B contains RHR pumps B and D. In the other units (Hope Creek and Limerick 1&2), each of the 4 RHR pumps has its own injection line. For Peach Bottom 2&3 and the other BWR 3/4 units, LPCI and other modes of RHR system operation require 1-of-4 RHR pumps to operate.

The period of time during which LPCI and CS subsystems or pumps can be out of service is limited by the technical specification requirement listed below.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable. OR One low pressure coolant injection (LPCI) pump in each subsystem is inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	7 days
B. Required Action and associated Completion Time if Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

For continued power operation (Condition A), no two subsystems of low pressure injection/spray can be out of service at the same time. Because the CS success criterion is 2 out of 4 pumps, Condition A occurs if CS subsystem A is inoperable, that is, if CS pump A is inoperable, CS pump C is inoperable, or CS pumps A and C are both inoperable. Similarly, Condition A occurs if CS subsystem B is inoperable, that is, if CS pump B is inoperable, CS pump D is inoperable, or CS pumps B and D are both inoperable. If two CS pumps in different subsystems are inoperable, Condition B occurs. With respect to LPCI, Condition A exists if any two RHR pumps are inoperable at the same time. Condition B exists if any three RHR pumps are inoperable at the same time. This is consistent with the 1-out-of-4 success criterion for LPCI. With respect to mixed CS and RHR pump outages, Condition B exists if either CS subsystem is inoperable at the same time as

any other low pressure injection/spray subsystem. The quantitative analyses below deal exclusively with condition A, not with Condition B.

Table H.11 summarizes point-estimate single outage risks for outages of all possible pump pairs within the CS and LPCI systems. The method employed follows that described in NUREG/CR-6141 [Ref. H.2] and Regulatory Guide 1.177 [Ref. H.3]. To illustrate the computation of single outage risk, consider the case in which CS pump A and CS pump B are both inoperable. The point-estimate CDF is first computed with all test-maintenance events for the low pressure injection/spray systems set to FALSE. Then the point-estimate CDF is computed with events LCS-MDP-TM-TRNA and LCS-MDP-TM-TRNB set to TRUE and with appropriate increases in common cause failure probabilities. The single outage risk for an outage that is 1 hour in duration is simply the difference between the two computations, which is 3.1×10^{-12}. The single outage risk for an outage that is 7 days in duration is simply $(7)(24)(3.1\times10^{-12})$ or 5.2×10^{-10}.

Table H.11 Single Outage Risks for Low Pressure Injection/Spray Pump Pairs

	Outage Time				
Inoperable Pump Pair	1 hour	12 hours	36 hours	7 days	31 days
CS Pumps A&B	3.1E-12	3.7E-11	1.1E-10	5.2E-10	2.3E-09
CS Pumps A&C[a]	2.2E-12	2.6E-11	7.7E-11	3.6E-10	1.6E-09
CS Pumps A&D	3.2E-12	3.8E-11	1.1E-10	5.3E-10	2.4E-09
CS Pumps B&C	1.1E-11	1.3E-10	4.0E-10	1.9E-9	8.3E-09
CS Pumps B&D[b]	1.1E-11	1.3E-10	4.0E-10	1.9E-9	8.3E-09
CS Pumps C&D	1.1E-11	1.3E-10	4.0E-10	1.9E-9	8.4E-09
RHR Pumps A&B	8.1E-11	9.7E-10	2.9E-9	1.4E-8	6.0E-08
RHR Pumps A&C[a]	1.6E-9	1.9E-8	5.8E-8	2.7E-7	1.2E-06
RHR Pumps A&D	8.8E-10	1.1E-8	3.2E-8	1.5E-7	6.6E-07
RHR Pumps B&C	9.3E-10	1.1E-8	3.4E-8	1.6E-7	7.0E-07
RHR Pumps B&D[b]	1.8E-9	2.2E-8	6.6E-8	3.1E-7	1.4E-06
RHR Pumps C&D	2.1E-9	2.5E-8	7.5E-8	3.5E-7	1.5E-06

[a] Subsystem A
[b] Subsystem B

The results in Table H.11 show that, for the same outage duration, the maximum single outage risk associated with having any pair of RHR pumps inoperable exceeds the maximum single outage risk associated with having any pair of CS pumps inoperable by a factor of >190. Current technical specifications permit any two RHR pumps to be inoperable for 7 days. Table H.11 shows that 3.5×10^{-7} per critical year bounds the CDF contribution associated with this requirement (postulating the limiting RHR outage occurs once per year). This estimate is bounding because the average outage would be expected to be much shorter than 7 days. In fact, the SPAR 3i models postulate a single pump test-maintenance outage probability of 0.002, which corresponds to an average single-pump outage duration of only (365)(0.002) = 0.73 days.

Current technical specifications only permit two core spray pumps within the same subsystem to be inoperable for up to 7 days. A possible change that is supported by the results in Table H.11 would be to extend this technical specification to pairs of CS pumps (A&B, A&D, B&C, C&D) in different CS subsystems. Table H.11 shows that 1.9×10^{-9} per critical year bounds the CDF contribution associated with either the current CS requirement or the proposed change (again postulating the limiting CS outage occurs once per year). Alternatively, the technical specifications could be changed to permit any pair of low pressure injection/spray pumps to be inoperable for up to 7 days; however, the single outage risks associated with having various pairs of CS and RHR pumps inoperable would have to be computed to support this alternative.

Another change that could be contemplated would be to lengthen the maximum duration for having low pressure injection/spray pumps inoperable. The single outage risks for a 31 day limit are included in Table H.11. The bounding CDF values cited above simply increase by a factor of 4.43 (31/7). Additional variations in technical specifications for low pressure injection/spray systems could also be considered.

H.4.3 Sensitivity of Single-Outage Risks to LPCI Success Criterion

An issue exists regarding the LPCI success criteria for a large break in a recirculation line. The core may only be reflooded to two-thirds core height for large LOCAs with break flows equivalent to the capacity of one LPCI (RHR) pump. If only one or two RHR pumps operate and if flow is not controlled, the water in the vessel may be subcooled and there will not be steam cooling of the top third of the core. This may be less of a problem for LPCI systems that inject into the shroud area. Current ECCS evaluation models do not proceed this far in time.

To address this issue, a variation of the LPCI fault tree was developed to apply a 3-out-of-4 pump success criterion for large LOCAs. When this alternative LPCI fault tree was applied to large-break LOCAs, the point estimate total plant CDF increased by less than 0.2%. Table H.12 presents the single outage risks for low pressure injection/spray pump pairs with the 3-out-of-4 pump LPCI success criteria applied to large-break LOCAs. Although the large-break LOCA CDF increases by 40% with the alternative success criteria, the overall single outage risks for pairs of low pressure injection/spray pumps do not increase dramatically from those presented in Table H.11. The maximum outage risk associated with having a pair of RHR pumps inoperable now exceeds the maximum outage risk associated with having a pair of CS pumps inoperable by a factor of >123.

Table H.12 Single Outage Risks for Low Pressure Injection/Spray Pump Pairs with 3/4 LPCI Success Criterion for LBLOCA

Inoperable Pump Pair	Outage Duration				
	1 hour	12 hours	36 hours	7 days	31 days
CS Pumps A & B	7.7E-12	9.3E-11	2.8E-10	1.3E-09	5.7E-09
CS Pumps A & C[a]	6.6E-12	8.0E-11	2.4E-10	1.1E-09	4.9E-09
CS Pumps A & D	8.8E-12	1.1E-10	3.2E-10	1.5E-09	6.5E-09
CS Pumps B & C	1.7E-11	2.0E-10	6.0E-10	2.8E-09	1.2E-08
CS Pumps B & D[b]	1.6E-11	1.9E-10	5.6E-10	2.6E-09	1.2E-08
CS Pumps C & D	1.7E-11	2.1E-10	6.2E-10	2.9E-09	1.3E-08
RHR Pumps A & B	8.6E-11	1.0E-09	3.1E-09	1.4E-08	6.4E-08
RHR Pumps A & C[a]	1.6E-09	2.0E-08	5.9E-08	2.7E-07	1.2E-06
RHR Pumps A & D	8.9E-10	1.1E-08	3.2E-08	1.5E-07	6.6E-07
RHR Pumps B & C	9.4E-10	1.1E-08	3.4E-08	1.6E-07	7.0E-07
RHR Pumps B & D[b]	1.8E-09	2.2E-08	6.6E-08	3.1E-07	1.4E-06
RHR Pumps C & D	2.1E-09	2.5E-08	7.5E-08	3.5E-07	1.5E-06

H.4.4 Analyses of Low-Pressure ECCS Redundancy

Three change cases were developed to investigate the feasibility of decreasing the redundancy of low-pressure ECCS (LPCI/RHR and core spray). Each change was evaluated using the base case (1-out-of-4 pumps) and alternative (3-out-of-4 pumps) success criteria for LPCI in large-break LOCAs. These change cases and their quantitative impact on predicted CDF are summarized in Table H.13 and discussed below.

In Case 1, core spray pumps C and D were removed from operation. Pumps C and D were chosen because they have the highest single outage probability for CS pump pairs in Tables 11 and 12. Removal of CS pumps C and D was modeled by setting the events corresponding to core spray pumps C and D being out for test or maintenance (LCS-MDP-TM-TRNC and LCS-MDP-TM-TRND) to TRUE, and increasing the common cause failure probabilities to reflect a 2-pump instead of a 4-pump core spray system. This case increased point estimate CDF by approximately 3% relative to the base case. The increase in point-estimate CDF relative to the base case is 1.5×10^{-7} per critical year with either large-break LPCI success criterion.

Table H.13 Impact of ECCS Changes on Point-Estimate CDF

Cases with 1-out-of-4 pump LPCI success criteria for LBLOCAs		Pt.Est. CDF per critical year	Pt.Est. ΔCDF per critical year	Ratio case/base
	Base Case	5.2E-6		
1	Remove CS pumps C&D	5.3E-6	1.5E-7	1.03
2	Remove CS system	5.6E-6	4.7E-7	1.1
3	Remove LPCI (RHR) pumps C&D	2.4E-5	1.9E-5	4.7
Cases with 3-out-of-4 pump LPCI success criteria for LBLOCAs		**Pt.Est. CDF per critical year**	**Pt.Est. ΔCDF per critical year**	**Ratio case/base**
	Base Case with 3-out-of-4 pump LPCI success criteria for LBLOCAs	5.2E-6		
1	Remove CS pumps C&D	5.3E-6	1.5E-7	1.03
2	Remove CS system	6.2E-6	1.0E-6	1.2
3	Remove LPCI (RHR) pumps C&D	2.4E-5	1.9E-5	4.7

In Case 2, all four core spray pumps were removed from service. This was modeled by setting all of the events corresponding to core spray pumps being out for test or maintenance to TRUE. For this case, point estimate CDF increased by approximately 10% relative to the original base case. The increase in point estimate CDF relative to the base case was 4.7×10^{-7} per critical year. With the alternative (3-out-of-4 pump) LPCI success criterion for large-break LOCAs, the increase in point estimate CDF is 10^{-6} per critical year.

In Case 3, LPCI/RHR pumps C and D were removed from service. Pumps C and D were chosen because they have the highest single outage probability for RHR pump pairs as indicated in Tables 11 and 12. Removal of LPCI/RHR pumps C and D was modeled by setting the events corresponding to core spray pumps C and D being out for test or maintenance (RHR-MDP-TM-TRNC and RHR-MDP-TM-TRND) to TRUE, and increasing the common cause failure probabilities to reflect a 2-pump instead of a 4-pump RHR system. For this case, point estimate CDF increased by a factor of approximately 4.7 relative to the base case. The corresponding increase in point estimate CDF is 1.9×10^{-5} per critical year. This increase is considerably greater than that associated with completely removing the core spray system. This is because the other modes of RHR provide long-term decay heat removal, which is important for a broad spectrum of accidents.

Removal of all RHR pumps was not investigated because of the diverse modes of operation required of the RHR system.

Based on the above results, with credit for injection by non-ECC systems, eliminating or reducing the redundancy of the core spray system appears consistent with the CDF acceptance guidelines specified in Section 4.1 of the main report. However, this conclusion does not reflect any potential impacts from current work on the pressurized thermal shock (PTS) issue, nor is it known whether there are any unanalyzed scenarios for which a vessel spray system may be of some unique benefit (e.g., a massive failure of the bottom of the vessel).

H.4.5 Analyses of Design-Basis LOOP Event for Large- and Medium-Break LOCAs

General Design Criterion 35 of 10 CFR Part 50, Appendix A, requires that the ECCS provide sufficient core cooling during a LOCA when it is powered by either onsite or offsite power. To comply with this requirement, ECCS evaluations generally assume a pipe break with a coincident LOOP. This assumption generally results in requirements for fast diesel generator starting and loading times that provide significant stresses on the diesels and may reduce their reliability.

To investigate the feasibility of eliminating this design requirement for large-break LOCAs, the base case model was changed to force all large-break LOCAs in which LOOP occurs to core damage, consistent with the guidance in Section 3.1.2 of the main report. This case increases point estimate CDF by a factor of 1.8 relative to the base case. The increase in mean and point estimate CDF relative to the base case is 1.6×10^{-6} per critical year. This is the mean frequency of large breaks (6.9×10^{-5} per critical year) times the mean probability of LOOP given a break initiator (2.3×10^{-2}).

The feasibility of eliminating the LOCA-LOOP design requirement for large and medium LOCAs was investigated by forcing all large and medium break LOCAs that induce LOOP to core damage. This case increases point estimate CDF by a factor of 7.5 relative to the base case. The increase in mean and point estimate CDF relative to the base case is 4.8×10^{-6} per critical year. This is the mean frequency of large and medium breaks ($6.9\times10^{-5} + 1.2\times10^{-4}$ per critical year) times the mean conditional probability of LOOP given a break initiator (2.3×10^{-2}).

REFERENCES

H.1 U.S. Nuclear Regulatory Commission (USNRC), SECY-00-0198, “Status Report on Study of Risk-informed Changes to The Technical Requirements of 10 CFR Part 50 (Option 3) and Recommendations on Risk-informed Changes to 10 CFR 50.44 (Combustible Gas Control),” September 14, 2000.

H.2 Samanta, P.K., et al., “Handbook of Methods for Risk-Based Analyses of Technical Specifications,” NUREG/CR-6141, BNL-NUREG-52398, Brookhaven National Laboratory, December 1994.

H.3 USNRC, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Regulatory Guide 1.177, August 1998.