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Subject: **Changes to NEDO-33201 in Response to the NRC Audit of the Probabilistic Risk Assessment Supporting Chapter 19 of the ESBWR Design Certification Application**

From May 6-8, 2009, the NRC staff conducted an audit of the Probabilistic Risk Assessment (PRA), which supports Chapter 19 of the ESBWR Design Certification Application (Ref. 1). The purpose of this letter is to submit markups to NEDO-33201 Draft Rev 5, *ESBWR Probabilistic Risk Assessment* (Enclosure 1).

If you have any questions or require additional information, please contact me.

Sincerely,

Richard E. Kingston

Richard E. Kingston
Vice President, ESBWR Licensing

D068
NRO

Reference:

1. Summary of the Audit of the Probabilistic Risk Assessment Supporting Chapter 19 of the ESBWR Design Certification Application, Dennis Galvin (NRC) to GEH, dated July 27, 2009.

Enclosure:

1. Markups to NEDO-33201 Draft Rev 5, ESBWR Probabilistic Risk Assessment

cc:

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Enclosure 1

Markups to NEDO-33201 Draft Rev 5,

ESBWR Probabilistic Risk Assessment

1 INTRODUCTION

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1 INTRODUCTION

1.1 PURPOSE

The purpose of this analysis is to describe the methodology and the results of the ESBWR probabilistic risk assessment (PRA) and severe accidents.

The PRA has been performed in an iterative manner with the ESBWR design development to evaluate and improve the risk aspects of the ESBWR design.

The overall objectives of the ESBWR PRA are:

- Provide an integrated and systematic assessment of the ESBWR design in response to transient and accident events (including severe accidents),
- Assess the capability of the ESBWR design to meet, with sufficient margin, the NRC safety goals for new plant designs,
- Identify design and analysis areas where further investigation and/or improvement is needed to meet the safety goals,
- Assess the sensitivity of the ESBWR risk profile to human interactions,
- Identify the importance of individual systems and components to the ESBWR risk profile, and
- Develop an analytic tool for use in investigating alternatives in design and operational strategies to optimize ESBWR plant safety.

The specific objectives of the plant-specific PRA and severe accident evaluations are to demonstrate that the ESBWR has been designed with state-of-the-art safety features, incorporating highly reliable and available passive safety functions with significant redundancy and diversity.

The design-specific PRA results and insights are compared against the following goals (note: these are goals and not regulatory requirements) and address how the plant features properly balance severe accident prevention and mitigation:

- Determine how the risk associated with the design compares against the Commission's goals of less than 1E-4/yr for core damage frequency (CDF),
- Determine how the risk associated with the design compares against the Commission's goals of less than 1E-6/yr for large release frequency (LRF),
- A deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges, and
- A probabilistic goal that the conditional containment failure probability (CCFP) is ~~less than~~ approximately 0.1 for the composite of ~~all~~at-power core damage sequences assessed in the PRA.

The ESBWR design PRA uses the current information available from the ESBWR plant design, Technical Specifications, and procedures. Component failure data and initiating event frequencies are based on generic industry data with consideration of the ESBWR design. Given the state of the ESBWR plant design, the PRA analyses contain conservative elements

(e.g., pre-initiator and post-initiator human error probabilities; maintenance unavailabilities; component failure rates; flood and fire initiation, propagation and effects; ground level release with no evacuation warning assumed in consequence analysis). As such, the actual ESBWR risk profile is judged to be lower than the current quantitative results described in this report.

1.2 SCOPE

The ESBWR PRA is a full scope (Level 1, Level 2, and Level 3) PRA, that covers both internal and external events, full power and shutdown. Where applicable, ASME-RA-Sb-2005 capability category 2 attributes are included in the analysis. Some of these attributes are not achievable at the certification stage of a nuclear power plant. For example, many aspects of assessing human actions cannot be analyzed in absence of a physical, operating plant and operation staff. In these cases, a bounding approach is taken to encompass potential sites, configurations, and operating organizations. In addition, any analyses requiring site-specific characteristics are treated in a bounding manner.

1.3 PRA OVERVIEW

1.3.1 Internal Events

The PRA quantification modeling methodology used in the ESBWR Level 1 PRA is a linked fault tree approach.

Fault trees have been developed and evaluated for the major ESBWR front line and support systems to determine the unavailability on demand of emergency core cooling and decay heat removal systems. Transient and loss-of-coolant accident events have been consolidated into major accident event sequences that are described by the accident event trees. These event trees are used to calculate the frequency of core damage sequences by directly linking the fault trees and solving for the minimal cutsets.

Outcomes of the event trees are transferred to containment event trees for further treatment to determine frequencies of radioactive releases to the environment.

Results of the containment event tree analyses provide the necessary input to model and assess fission product transport through the drywell and containment; calculate fission product release fractions associated with containment release paths; and determine potential consequences associated with each fission product release category.

The characteristics of the internal events PRA are as follows:

- Initiating Events

  Transients, Loss of Preferred Power, Loss of Coolant Accidents, and special initiator categories are identified based on review of industry PRAs and guidance documents. These are modified based on specifics of the ESBWR design and expected operation.

  Initiating event frequencies are estimated based on generic industry data for operating BWRs.

- Accident Sequence Analysis

  Accident sequence event tree structures and end states are defined for each initiator category based on review of industry PRAs and guidance documents. These are modified based on specifics of the ESBWR design and expected operation.

  Event tree nodal inputs are system fault tree logic or nodal point estimates, as appropriate.

  Functional success criteria are based on analysis of ESBWR design and expected operation.

- Systems Analysis

  System fault trees are developed based on standard industry techniques and reflect the ESBWR system design. Systemic success criteria are based on analysis of the ESBWR design and expected operation.

- Human Reliability Analysis

  Pre-initiator and post-initiator human error probabilities are defined based on the ESBWR design and expected operation. The human error probabilities used in the model are screening values based on the time available to perform the various actions.

- Data Analysis

  Component failure probabilities are estimated based on generic industry data.

- Containment Performance Analysis

  Severe accident phenomena are explicitly addressed and are quantitatively treated. The Risk Oriented Accident Analysis Methodology is used to assess the containment response to severe accident phenomena. A linked fault tree approach is used to address the containment systems and the ability to prevent overpressurization from loss of decay heat removal.

  In order to support the consequence analysis, multiple radionuclide release categories are modeled.

- Consequence Analysis

  Source terms are defined based on ESBWR thermal hydraulic analysis. Offsite consequence analyses are performed, showing that the ESBWR design meets NRC safety goals with sufficient margin.

1.3.2 External Events

The external events portion of the PRA explicitly analyzes core damage accidents initiated during power and shutdown operation for the following hazards:

- Internal floods,
- Internal fires,
- High winds, and
- Seismic events.

The external events analyses are bounding assessments that are meant to show significant design margin for these hazards. The frequencies of initiating events are based on generic industry data, and are applied in a bounding manner. The fault trees and event trees developed for the internal events evaluations are used in the external events analyses to the maximum extent possible, using logic flags that account for the common failures induced by the external hazard events.

The ESBWR seismic assessment is a seismic margins analysis. The analysis demonstrates the ESBWR plant and equipment can withstand an earthquake with a magnitude at least 1.67 times the safe shutdown earthquake.

1.3.3 Shutdown Risk

The shutdown and transition risk analysis includes an assessment of the internal event initiated core damage accidents occurring during shutdown operations. Initiator categories and

frequencies are based on a review of industry studies, generic industry data, and consideration of ESBWR design and operation. In addition, a typical refueling outage time line is assumed.

1.4 PRA DOCUMENTATION ORGANIZATION

The ESBWR PRA is documented as follows:

Introduction

- Introduction (Section 1)

Level 1 Analysis

- Initiating Events (Section 2)
- Accident Sequence Analysis (Section 3)
- Systems Analysis (Section 4)
- Data Analysis (Section 5)
- Human Reliability Analysis (Section 6)
- Core Damage Frequency Quantification (Section 7)

Level 2 Analysis

- Containment Performance (Section 8)
- Source Terms (Section 9)

Level 3 Analysis

- Consequence Analysis (Section 10)

Uncertainty and Sensitivity

- Uncertainty and Sensitivity Analysis (Section 11)

External Event Analysis

- Probabilistic Fire Analysis (Section 12)
- Probabilistic Flood Analysis (Section 13)
- High Wind Risk (Section 14)
- Seismic Margins Analysis (Section 15)

Low Power/Shutdown Analysis

- Shutdown Risk (Section 16)

Results and Insights

- Results Summary (Section 17)
- PRA Insights Affecting ESBWR Design (Section 18)
- Reliability and Maintainability (Section 19)
- Regulatory Treatment of Non-Safety Systems (Section 20)

Severe Accident Analysis

- Severe Accident Management (Section 21)

Changes to the PRA Model

- ESBWR PRA Changes (Section 22)

Section 22 provides a process to document and evaluate changes to PRA parameters that occur after the PRA modeling details are frozen and incorporated into NEDO-33201. New information from technical reviews, operating experience, or other feedback mechanisms is evaluated for its effect on PRA parameters. Subsections correspond to the main report, for example, changes to the flooding PRA would be found in subsection 22.13.

The site-specific PRA model is completed during the construction phase prior to fuel load, and it incorporates the Section 22 changes.

11.3.6 Transportation and Nearby Facilities Sensitivity

~~A series of s~~Sensitivit~~ies~~y studies were conducted to evaluate other external events (Rev. 2) on the L1 PRA model. These types of external events include in this evaluation are as follows:

- ~~Aircraft impact~~Airports and Airways hazards,
- Industrial accidents,
- Pipeline accidents,
- Hydrogen storage failures, and
- Transportation accidents.

Each of the external events was evaluated using the L1 PRA model (Rev. 2). To facilitate the quantification of the risk impact associated with transportation and nearby facilities, a number of assumptions and simplifications were made in support of the sensitivities and are shown in Table 11.3-40.

11.3.6.1 ~~Aircraft Impact~~Airports and Airways Hazards

~~An aircraft impact~~ A sensitivity study on unintentional aircraft hazards (Rev. 2) was performed to evaluate the significance of this event on CDF and the L1 PRA model. The evaluation of the aircraft accidents including commercial, military and small private aircraft has been previously conducted within the industry. For the purpose of the ~~aircraft impact~~ sensitivity study, a screening probability for unintentional aircraft accidents was calculated to be 1.52E-07/year and is shown in Table 11.3-41 (ref. 11-1, 11-4 and 11-5). With the assumption that an aircraft ~~impact event~~accident impacting the plant facility would result in a loss of preferred power (LOPP), the Level 1 PRA CDF for the aircraft ~~impact~~ accident would be 1.31E-14/year. In the event that the aircraft ~~impact~~accident results in more extensive damage of the plant site impacting all nonsafety-related components (equivalent to a focused PRA), a more conservative CDF value of 1.94E-11/year is obtained.

The robust ESBWR design with its high redundant passive systems greatly help to mitigate the effect~~impact~~ of these events. ~~In addition, the NRC has an agreement with NORAD that enables reactor operators to quickly learn of imminent aviation threats and to swiftly place the reactor into a safe state.~~ An assessment of intentional aircraft impacts to the site is discussed in DCD Appendix 19D.

Table 11.3-40

Transportation Sensitivity - Assumptions

External Event	Assumption
~~Aircraft Impact~~Airports and Airways Hazards	Accident rate for aircraft is 4.0E-10 per mile.
	A total of approximately 980,000 flights per year (Atlanta Hartsfield Jackson International Airport, 2006).
Transportation	Accident conditional release probability of 0.09 for trucks, 0.2 for rail and 0.023 for barges.
	A total of four major highways are within an approximate width of 9 miles.
All Industrial Accidents	A 10 mile diameter area of interest for chemical storage. Materials stored or situated at a distance of greater than 5 miles from the plant site need not be considered. (RG 1.78)
General Siting	ESBWR facilities occupy approximately 10% of total site or 0.014 square miles.

Table 11.3-41

Transportation Sensitivity – ~~Aircraft Impacts~~Airports and Airways Hazards

~~Probability~~Frequency	(Accident Rate * # Flights * Area)/ Airway Width		
Accident Rate for Aircraft	=	4.00E-10	per mile
Number of flights	=	980,000	
Area of ESBWR facility	=	0.014	sq. mi.
Airway width	=	9	mi
	Frequency	1.52E-07	per year
Scenario 1	Aircraft ~~impact~~ accident results in station blackout		
	Frequency	1.52E-07	per year
Level 1 PRA CCDP	=	8.61E-08	per year
CDF	=	1.31E-14	per year
Scenario 2	Aircraft ~~impact~~ accident results in station blackout and loss of non-safety systems		
	Frequency	1.52E-07	per year
Level 1 PRA CCDP	=	1.27E-04	per year
CDF	=	1.94E-11	per year

22 ESBWR PRA Changes

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22 ESBWR PRA CHANGES

Section 22 provides a process to document and evaluate changes to PRA parameters that occur after the PRA modeling details are frozen and incorporated into NEDO-33201. New information from technical reviews, operating experience, or other feedback mechanisms is evaluated for its effect on PRA parameters. Subsections correspond to the main report, for example, changes to the flooding PRA would be found in subsection 22.13.

22.13 CHANGES TO FLOODING PRA MODEL

A review of the dominant flooding risk contributors shows that conservatisms in the current model skew the results significantly. The contribution from bypass and filtered release sequences are conservatively high, and this misrepresents the conditional containment failure probability from flooding events. The bypass (BYP) release category contributes about 42% of total CDF and about 70% of LRF. The filtered release (FR) release category contributes about 16% of total CDF and 26% of LRF.

Table 22.13-1
At-Power Flooding Results

Release Category	**Frequency**	**% Total**	**% nTSL**
TSL	2.799E-09	40.507%	-
FR	1.082E-09	15.654%	26.312%
OPW2	1.297E-10	1.876%	3.154%
OPW1	6.708E-13	0.010%	0.016%
OPVB	2.008E-12	0.029%	0.049%
BYP	2.894E-09	41.884%	70.403%
CCIW	1.436E-12	0.021%	0.035%
CCID	7.567E-13	0.011%	0.018%
EVE	-	-	-
BOC	5.276E-13	0.008%	0.013%
CDF	6.910E-09	100.000%	100.000%
nTSL	4.111E-09	CCFP	0.595

"TSL" = Allowable containment leakage
"nTSL" = Large release frequency

The F-V importance of flooding sequences contributing to LRF also clearly demonstrates the skewed results as shown below: T-IORV065 (~65%) and T-GEN004A (~26%).

Table 22.13-2
Fussell-Vesely Importance of At-Power Flooding Sequences

L1AP Sequence Flag	F-V
FL_T-GEN004A	2.62E-01
FL_T-GEN015	2.52E-02
FL_T-GEN017	1.14E-04
FL_T-GEN019	4.87E-04
FL_T-GEN021	1.19E-06
FL_T-GEN022	1.49E-05
FL_T-GEN026	4.78E-03
FL_T-GEN027	6.37E-05
FL_T-GEN030	1.05E-04
FL_T-GEN034	4.24E-04
FL_T-GEN035	3.91E-06
FL_T-GEN067	4.35E-06
FL_T-GEN068	1.02E-04
FL_T-GEN069	1.11E-02
FL_T-IORV011	1.12E-03
FL_T-IORV022	1.65E-04
FL_T-IORV063	2.82E-03
FL_T-IORV064	2.61E-05
FL_T-IORV065	6.47E-01

Two dominant conservatisms embedded in the flooding model have been identified:

1. Cutsets associated with flooding initiators %FL_TBC-B21A/B-S & %FL_TBC-B21A/B-L contribute to release category BYP and sequence T-IORV065. These cutsets assume that a break in the B21 Main Steam lines results in all function failures of the entire B21 system, including either failure or inadvertent actuations of the SRVs and DPVs. However, the DPVs and SRVs are inside the containment and their control signals do not pass through the flooding area of concern. Therefore, the depressurization function should not be failed by these flooding scenarios.
2. Cutsets associated with flooding initiators with fire protection system U43 pipe breaks (e.g., %FL_TB1-U43-L) contribute to release category FR and sequence T-GEN004A. There are a number of sequences modeled with U43 pipe breaks. The flooding model adopted a bounding assumption that the a break in the U43 piping would not be isolated and then, over a long period of time, would result in the total failure of the system. It is reasonable to account for the ability of plant personnel to diagnose a piping or tank rupture and then isolate the break so the entire fire protection system is not failed.

The removal of the conservatism associated with the B21 system failure is straightforward. The postulated flooding in the flooding area TBC (Turbine Building General Area) does not result in the failure of any DPVs or SRVs. The postulated flooding scenarios do not impact the depressurization functions.

The removal of the conservatism associated with the U43 system failure requires additional considerations. Since the fire protection function itself of U43 is not credited in the internal event models, the impact to the internal event models is due to the modeled IC/PCCS pool makeup function from U43.

With the conservative approach used in the flooding analysis, the system / train with the postulated break is assumed to be failed. Therefore, the postulated U43 pipe break (or spurious actuation of the FPS) was assumed to result in the failure of all U43 functions. However, this approach does not consider the following factors, which reduce the flooding risks and remove the conservatisms that skew the CCFP value:

(1) There are multiple firewater storage tanks. DCD Subsection 9.5.1.4 states that there are a minimum of two sources: (i) at least one "primary" source to the suctions of primary fire pumps and corresponding jockey fire pump and (ii) at least one "secondary" source to suctions of secondary fire pumps and corresponding jockey fire pump. The postulated U43 break may result in a total depletion of one tank, but may not deplete all tanks since the tank water level or switching to other sources prompts the operator to investigate whether there is an actual demand (accompanied by a highly noticeable fire), a leak, or a spurious actuation. It is reasonable to assume that the operator would isolate the break instead of switching to other firewater storage tanks then depleting all the firewater.

(2) DCD Table 9.5-2 lists the FPS Component Design Characteristics. The combined total primary firewater is 3900 m^3 (1,030,000 gallons). The minimum secondary storage firewater is 2082 m^3 (550,000 gallons). The largest firewater demand is 967 m^3/hr (4256 gpm) for Turbine Building (TB), including hose stream. Assuming the most limiting U43 break in the TB, the time to deplete all firewater storage would be about 6.2 hours. This makes the failure to isolate the U43 break an unlikely event.

(3) As shown in DCD Figure 9.5-1, the firewater to the TB, Electrical Building, and Service Building is connected from the yard piping while the TB connections to the Nuclear island (NI) are normally isolated. Per DCD Subsection 9.5.1.4, the primary, Seismic Category I, firewater storage tanks and Seismic Category I diesel-driven pump and fire protection piping provide post-accident makeup water to the IC/PCCS pools and Spent Fuel Pool using FAPCS piping. FPS components located outside the Reactor Building supporting FAPCS makeup do not fulfill a fire protection function. Fire hydrants, stand pipes, or other large lines will not be attached to the dedicated portion of the FPS designed to provide long term makeup to pools in the Reactor Building. Also as shown, the dedicated connections to IC/PCCS pools branch off the main headers to the NI before the isolation valves. Therefore, the isolation of the postulated U43 breaks should have no impact on the dedicated connections to IC/PCCS pools.

With the above considerations, it is reasonable to to assume that, for a postulated flooding scenario caused by an FPS (U43) pipe break or spurious actuation, operators should be able to isolate the FPS break or spurious actuation and the failure to isolate is unlikely with sufficient response time before the firewater storage tanks are depleted even with the most limiting firewater demands. To validate this assumption, the following steps are taken:

1. Assume a new operator action, unique to the internal events at-power flooding model, to isolate U43 pipe breaks or spurious actuations. This event is named U43-XHE-FO-ISOLATE, OPERATOR FAILS TO ISOLATE FPS BREAKS. The probability of similar operator actions, such as U43-XHE-FO-2ND, OPERATOR FAILS TO ALIGN FPS CROSSTIE, is 1.61 E-2. However, for simplicity, and to account for uncertainties, a probability of 1.61E-1 is assumed in this analysis.
2. Since the loss of firewater storage tank inventory does not impact the service water system functions (i.e., component cooling), the event tree that is used to show flood damage states can be simplified as the following figure. Note the spurious actuation is treated similarly as a pipe break.

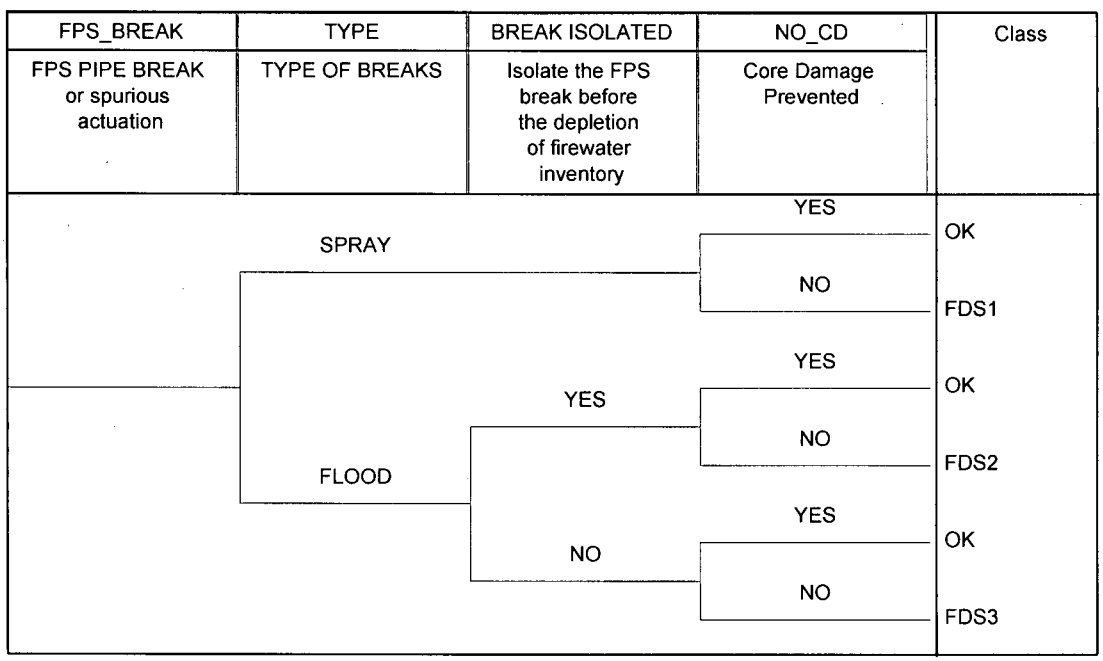

Figure 22.13-1 Flooding Damage States

3. The flooding model includes spraying effects and major flooding initiators separately, and the flooding sequences are plugged into the Level 2 internal events PRA model file, (a single top fault tree) for quantifications. The above

Step 2 flood damage states result in the following changes to the flooding model files:

(a) For FPS spray flooding scenarios (Class FDS1), U43 system function for IC/PCCS pool makeup is not affected. Therefore, the modified flag files without U43 IC/PCCS pool makeup failures are used for the subject sequences in the updated quantification file.

(b) For FPS major flood scenarios with successful isolation of the postulated break (Class FDS2), U43 system function for IC/PCCS pool makeup is not affected. Therefore, the modified flag files without U43 IC/PCCS pool makeup failures are used for the subject sequences in the updated quantification file.

(c) For FPS major flood scenarios with the failure to isolate the postulated break (Class FDS3), U43 system function for IC/PCCS pool makeup is affected. New sequences are added for this flooding model sensitivity study. The original flag files with U43 IC/PCC pool makeup failures are used for the new sequences in the updated quantification file. The new sequences are named as "FL_*-U43-L-FI" (note "*" in the sequence name represents the flood zone). Twenty new sequences are created as follows in the modified fault tree "FL-AP-R4_1-Modified.caf":

Table 22.13-3
Additional Sequences for Failure to Isolate

FL_CBSW-U43-L-FI	AND	%FL_CBSW-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_CTA-U43-L-FI	AND	%FL_CTA-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_CTB-U43-L-FI	AND	%FL_CTB-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_EB-U43-L-FI	AND	%FL_EB-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_EB1-U43-L-FI	AND	%FL_EB1-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_FB-3-U43-L-FI	AND	%FL_FB-3-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_FB-P-U43-L-FI	AND	%FL_FB-P-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_FB-SW-U43-L-FI	AND	%FL_FB-SW-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_FW1-U43-L-FI	AND	%FL_FW1-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_FW2-U43-L-FI	AND	%FL_FW2-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_PH-U43-L-FI	AND	%FL_PH-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_RB-3A-U43-L-FI	AND	%FL_RB-3A-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_RB-3B-U43-L-FI	AND	%FL_RB-3B-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_RB2-U43-L-FI	AND	%FL_RB2-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_RW-U43-L-FI	AND	%FL_RW-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_TB-1-U43-L-FI	AND	%FL_TB-1-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_TB-U43-L-FI	AND	%FL_TB-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_TB1-U43-L-FI	AND	%FL_TB1-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_TB2-U43-L-FI	AND	%FL_TB2-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP
FL_TBSW-U43-L-FI	AND	%FL_TBSW-U43-L	U43-XHE-FO-ISOLATE	L2-ONETOP

The following summary is obtained by updating the CDF value and BE importance report for the flooding model results.

Table 22.13-4
Revised At-Power Flooding Results

	Frequency	% total	% nTSL
TSL	2.824E-09	85.478%	-
FR	2.098E-10	6.350%	43.729%
OPW2	1.983E-11	0.600%	4.134%
OPW1	6.345E-13	0.019%	0.132%
OPVB	1.810E-12	0.055%	0.377%
BYP	2.464E-10	7.458%	51.356%
CCIW	5.922E-13	0.018%	0.123%
CCID	2.754E-13	0.008%	0.057%
EVE			
BOC	4.357E-13	0.013%	0.091%
CDF	3.304E-09	100.000%	100.000%
nTSL	4.798E-10	CCFP	0.145

These results show a CDF of 3.30E-9/yr, a LRF of 4.80E-10/yr and a CCFP of 0.145. These updated results demonstrate that no new vulnerabilities are identified from the updated flooding model. In addition, the operator action that is included in the flooding model sensitivity study does not have significant impact on the CDF or LRF values, but has some impact on the CCFP, which is discussed as follows.

Using the same merged cutset files, the probability of the operator action U43-XHE-FO-ISOLATE is changed to demonstrate its significance in the following two cases:

- Case A: Set to 0 (i.e., operator always isolate the break; Or, the non-isolated break would not result in loss of all firewater inventory).
- Case B: Set to 1.0 (i.e., operator always fail to isolate the break).

Case A results are listed as follows:

Table 22.13-5
Sensitivity Case A

	Frequency	% total	% nTSL
TSL	2.822E-09	88.920%	-
FR	8.892E-11	2.802%	25.285%
OPW2	1.508E-11	0.475%	4.289%
OPW1	6.327E-13	0.020%	0.180%
OPVB	1.778E-12	0.056%	0.506%
BYP	2.440E-10	7.687%	69.382%
CCIW	5.643E-13	0.018%	0.160%
CCID	2.613E-13	0.008%	0.074%
EVE			
BOC	4.343E-13	0.014%	0.124%
CDF	3.174E-09	100.000%	100.000%
nTSL	3.517E-10	CCFP	0.111

Case B results are listed as follows:

Table 22.13-6
Sensitivity Case B

	Frequency	% total	% nTSL
TSL	2.833E-09	71.105%	-
FR	8.430E-10	21.160%	73.230%
OPW2	4.457E-11	1.119%	3.872%
OPW1	6.566E-13	0.016%	0.057%
OPVB	1.984E-12	0.050%	0.172%
BYP	2.594E-10	6.512%	22.537%
CCIW	7.342E-13	0.018%	0.064%
CCID	3.488E-13	0.009%	0.030%
EVE			
BOC	4.366E-13	0.011%	0.038%
CDF	3.984E-09	100.000%	100.000%
nTSL	1.151E-09	CCFP	0.289

A comparison of the above results are listed in the following table:

Table 22.13-7
Sensitivity Results Comparisons

	CDF (/yr)	% Change from Baseline	LRF (/yr)	% Change from Baseline	CCFP	% Change from Baseline
Baseline	3.30E-09	N/A	4.80E-10	N/A	0.145	N/A
U43-XHE-FO-ISOLATE set to 0	3.17E-09	-3.9%	3.52E-10	-26.7%	0.111	-23.4%
U43-XHE-FO-ISOLATE set to 1	3.98E-09	20.6%	1.15E-09	139.6%	0.289	99.3%

It should be noted that even the Case B CCFP value of 0.289 is lower than the original result since the added operator action only affects the new sequences associated with U43 breaks. The removal of the conservatism associated with the B21 breaks is straightforward. It has reduced the CCFP value from 0.595 to about 0.3. The changes to the U43 sequences reduce the CCFP to 0.145 even when new scenarios are included.

The shutdown flooding CDF / LRF value is low (5.21 E-9/year) and the dominant risk contributors are different than those identified in the at-power flooding model. Therefore, the model changes described in this task are not applied to the shutdown model.

The following summary is obtained by updating the CDF value and Basic Event importance report for the updated focus / RTNSS flooding model results.

Table 22.13-8
Revised Results for Focus PRA

	Frequency	% total	% nTSL
TSL	1.624E-06	1.729%	-
FR	-	-	-
OPW2	4.020E-05	42.811%	43.565%
OPW1	2.539E-10	0.000%	0.000%
OPVB	3.809E-11	0.000%	0.000%
BYP	5.205E-05	55.429%	56.404%
CCIW	1.665E-08	0.018%	0.018%
CCID	1.168E-08	0.012%	0.013%
EVE	-	-	-
BOC	2.696E-12	0.000%	0.000%
CDF	9.391E-05	100.000%	100.000%
nTSL	9.229E-05	CCFP	0.983

Summary of the Focus PRA with RTNSS sensitivity study results:

Table 22.13-9
Revised Results for Focus PRA with RTNSS Components

	Frequency	% total	% nTSL
TSL	7.317E-08	23.002%	-
FR	-	-	-
OPW2	2.345E-07	73.714%	95.735%
OPW1	1.091E-11	0.003%	0.004%
OPVB	2.635E-11	0.008%	0.011%
BYP	1.036E-08	3.258%	4.231%
CCIW	2.708E-11	0.009%	0.011%
CCID	1.620E-11	0.005%	0.007%
EVE	-	-	-
BOC	3.709E-12	0.001%	0.002%
CDF	3.181E-07	100.000%	100.000%
nTSL	2.449E-07	CCFP	0.770

The Revision 4 of NEDO-33201 Section 10 calculated offsite radiological consequences with the contribution from the external events and shutdown models. The results meet NRC goals and the contribution from the at-power flooding model would be reduced. Therefore, the current results are acceptable.

Specifically, Subsection 13.6.1, At-Power Flooding Scenarios, need to be updated to include the discussion on the newly added operator action to isolate the FPS piping breaks. Currently it states, "No operator actions have been included in the internal events PRA for the purpose of isolating or mitigating the consequences of at-power flooding scenarios." This statement needs to be modified to address the above changes; or Section 22.13 should capture the changes if determined to be more appropriate.

NEDO-33201 Subsection 17.3.2, External Events – Flood, is evaluated to consider the results and insights from this sensitivity study. The following tables apply:

- Table 17.1-1 Results
- Table 17.1-3 ESBWR Risk-Significant Operator Actions
- Table 17.7-11 Flood CDF At-Power Operator Actions – RAW
- Table 17.7-12 Flood CDF At-Power Operator Actions – FV
- Table 17.7-13 Flood LRF At-Power Operator Actions - RAW
- Table 17.7-14 Flood LRF At-Power Operator Actions – FV

The risk importance of the new operator action used in the sensitivity study for the at-power flooding model is negligible. Using the threshold of 1E-7 /yr for change in LRF with respect to its RAW value, this event does not qualify as a key operator action. The delta LRF for operator action U43-XHE-FO-ISOLATE is calculated as:

$$\text{Delta LRF} = (\text{RAW} - 1) * \text{Updated Baseline LRF}$$
$$\text{Delta LRF} = (2.42 - 1) * 4.80\text{E-}10 = 6.82\text{E-}10\ /\text{yr} << 1\text{E-}7\ /\text{yr}$$

Therefore, the revised flooding results from this sensitivity study have a negligible effect on the Section 17 tables listed above and provide no additional insights on risk significance.

NEDO-33411 Tables 1 and 4 contain at-power flooding model results. However, the results from this sensitivity study add no new risk significant basic events.

22.16 CHANGES TO SHUTDOWN PRA MODEL

The RWCU/SDC Break Outside of Containment (BOC) event trees have been modified to include a top event for four DPVs actuating prior to GDCS actutation. This modification makes the RWCU/SDC BOC trees consistent with other shutdown (SD) PRA event trees. This change slightly increases the SD risk values since new sequences are added. However, the contribution from these new sequences is negligible because they involve multiple failures of the injections systems. The Mode 6 event trees do not require depressurization, so this change has no impact on event trees "M6F-RWCU-BOC.eta" and "M6U-RWCU-BOC.eta".

The updated Figures 22.16.4-31 and 22.16.4-32 for the RWCU BOC event trees for Mode 5 and 5 Open are shown as below. In the Mode 5 and Mode 5 Open event trees, top event XD-TOPDPV has been added before top SD-GDCS. Four new core damage sequences are generated, which are also classified as CD-V (i.e., bypassing the containment):

- M5-RWCU-BOC006A
- M5-RWCU-BOC-018A
- M5O-RWCU-BOC005A
- M5O-RWCU-BOC016A

Based on the updated event trees, the new top logic associated with these four sequences is:

```
M5-RWCU-BOC006A    AANB M5-RWCU-BOC006A_F  M5-RWCU-BOC006A_S

M5-RWCU-BOC006A_F AND  %M5_RWCU_BOC  B32-3LOOPSFAIL    UD-TOPINJ2    VL-
TOPINJ    VM-TOPINJXD-TOPDPVFL_M5-RWCU-BOC006

M5-RWCU-BOC006A_S OR   BC-TOPRWCU    MS-TOP18 MS-TOP2

M5-RWCU-BOC018A    AANB M5-RWCU-BOC018A_F  M5-RWCU-BOC018A_S

M5-RWCU-BOC018A_F AND  %M5_RWCU_BOC  BC-TOPRWCU    B32-3LOOPSFAIL    UD-
TOPINJ2  VL-TOPINJVM-TOPINJXD-TOPDPVFL_M5-RWCU-BOC018

M5-RWCU-BOC018A_S OR   IM-TOPSDCMS-TOP18 MS-TOP2

M5O-RWCU-BOC005A   AANB M5O-RWCU-BOC005A_FM5O-RWCU-BOC005A_S

M5O-RWCU-BOC005A_FAND  %M5O_RWCU_BOC UD-TOPINJ2    VL-TOPINJVM-TOPINJXD-
TOPDPV    FL_M5O-RWCU-BOC005

M5O-RWCU-BOC005A_SOR   BC-TOPRWCU    MS-TOP18 MS-TOP2

M5O-RWCU-BOC016A   AANB M5O-RWCU-BOC016A_FM5O-RWCU-BOC016A_S

M5O-RWCU-BOC016A_FAND  %M5O_RWCU_BOC BC-TOPRWCU    UD-TOPINJ2    VL-
TOPINJ    VM-TOPINJXD-TOPDPVFL_M5O-RWCU-BOC016

M5O-RWCU-BOC016A_SOR   IM-TOPSDCMS-TOP18 MS-TOP2
```

Since the new top logic does not bring in any new system model changes, the updated SD baseline internal events model fault tree simply incorporated the above new sequences M5-RWCU-BOC006A, M5-RWCU-BOC-018A, M5O-RWCU-BOC005A, and M5O-RWCU-BOC016A under gates "~M5_-RWCU-BOC" and "~M5O-RWCU-BOC", respectively.

For this sensitivity study, only the new sequences are quantified since the total impact to the SD baseline result is negligible. For convenience, the total contribution from the new sequences are quantified with a new added top gate "SD_CDF_NEW", which is defined as follows:

```
SD_CDF_NEW    OR   M5-RWCU-BOC006A    M5-RWCU-BOC018A    M5O-RWCU-BOC005A
     M5O-RWCU-BOC016A
```

No evaluation to the SD focus/RTNSS studies is performed with the updated model files described in above subsections because the impact to the baseline model is negligible.

The above model changes do not impact the SD external events models (fire, flood and high wind analyses) and their associated focus/RTNSS studies. Since the SD external events initiators do not follow the RWCU BOC sequences, the added new sequences have no impact on the SD external events models.

The updated SD internal events model generates a total additional CDF / LRF contribution of 2.03E-12 /yr for the new SD RWCU BOC sequences, which is 0.012% of the baseline SD CDF / LRF of 1.70E-8 /yr in Section 16. As discussed above, the CDF / LRF increase associated with the addition of new sequences is negligible to the SD baseline CDF/LRF value.

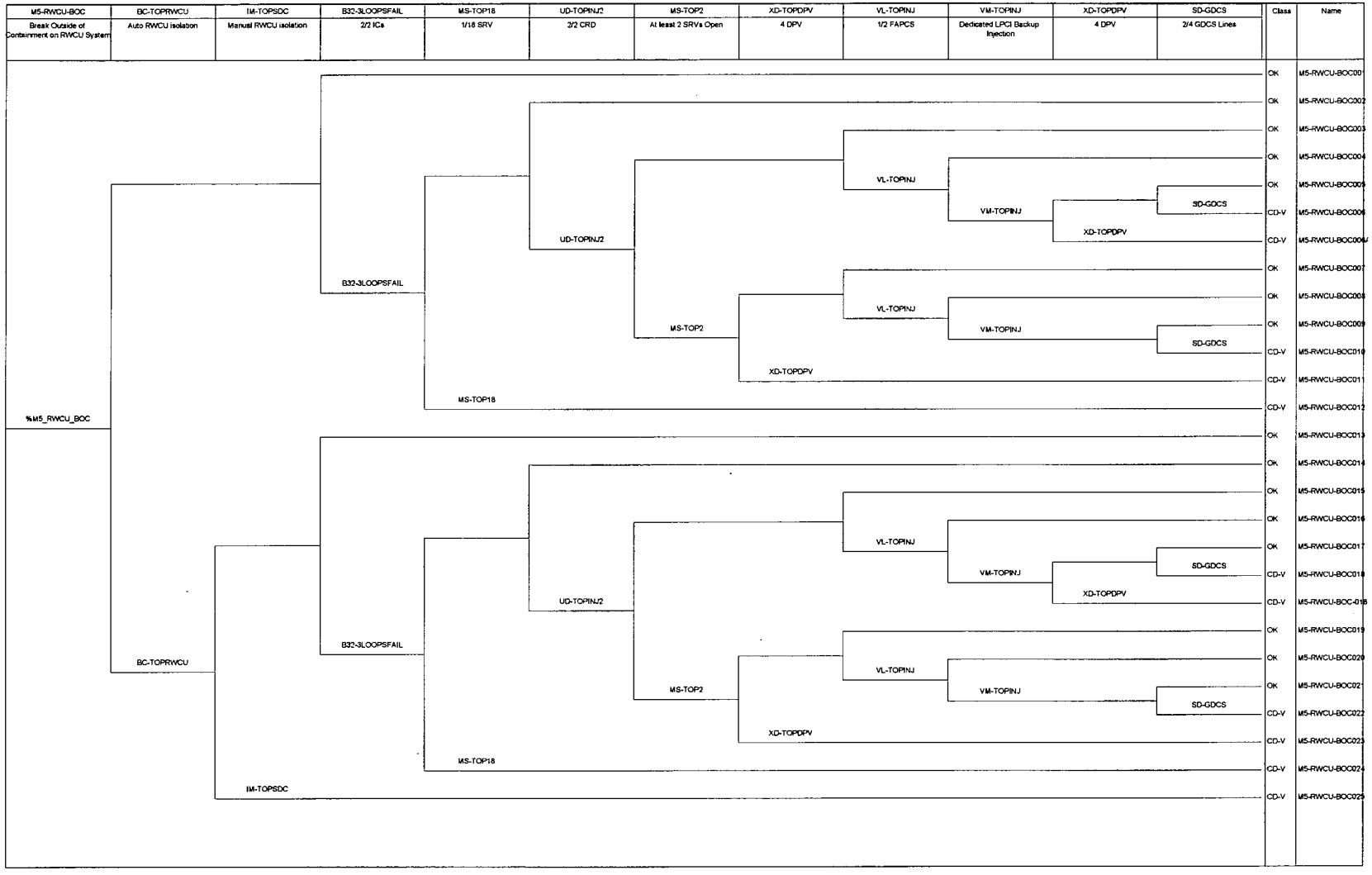

Figure 22.16.4-31. (Updated) LOCA – RWCU Break Outside Containment (Mode 5)

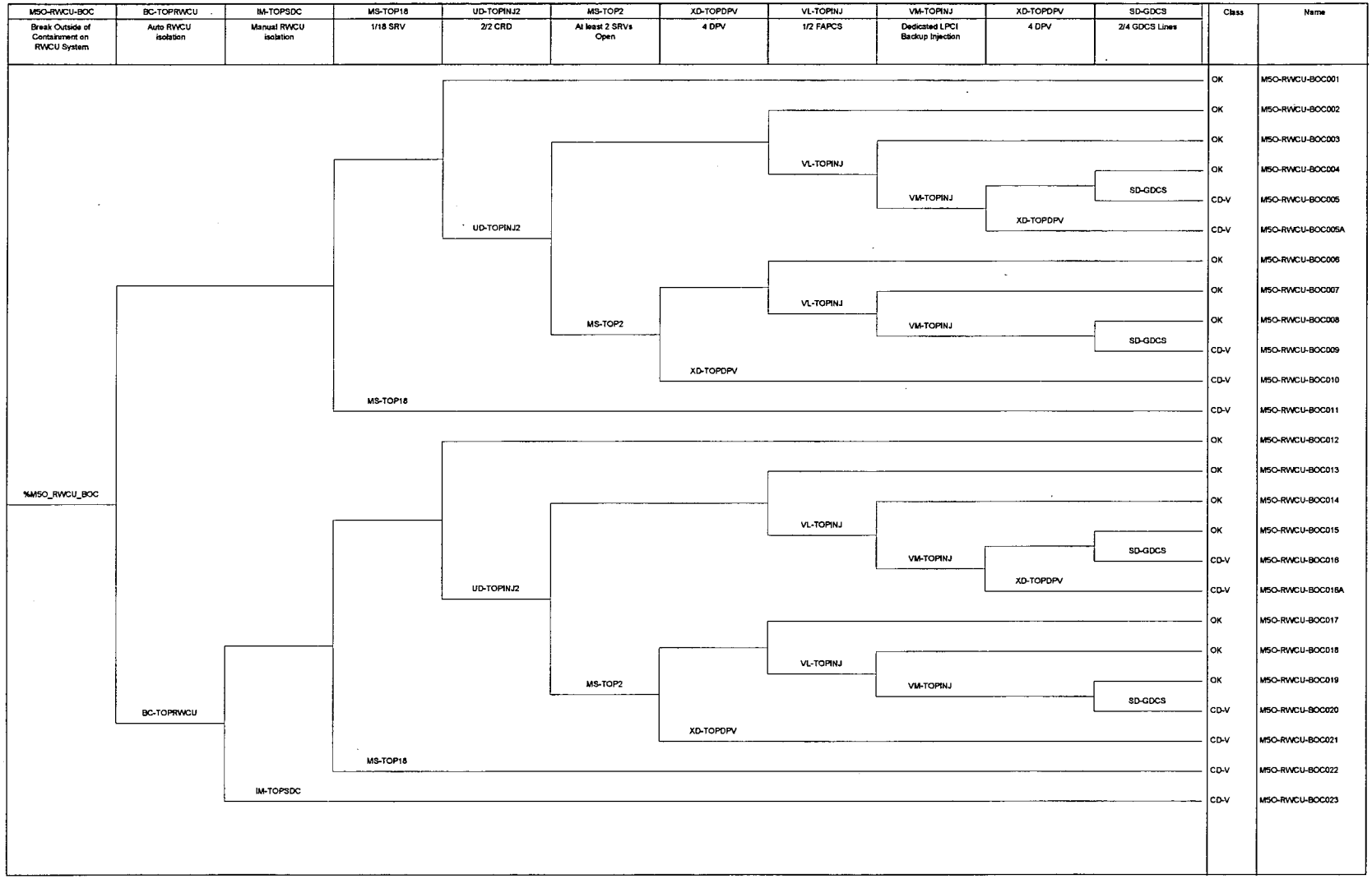

Figure 22.16.4-32. (Updated) LOCA – RWCU Break Outside Containment (Mode 5 Open)