Cost Benefit Analysis

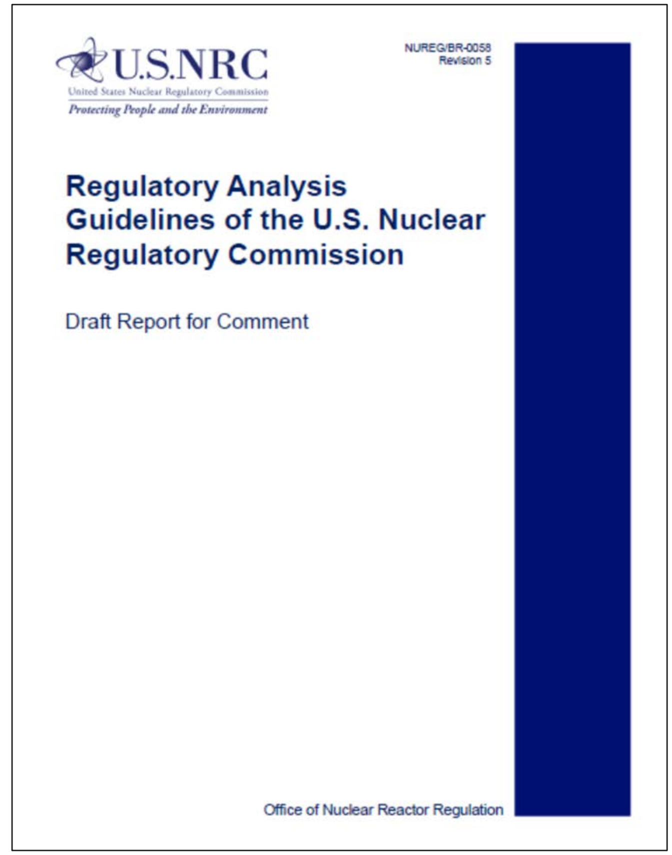
Tina Ghosh, PhD
Senior Reactor Systems Engineer
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Outline

- Background on cost-benefit analyses at the NRC and the role of the MACCS code
- Example from a detailed regulatory analysis for potential rule-making
- Example from a severe accident mitigation alternative analysis, for environmental impact statement supporting initial license renewal of a nuclear power plant (NPP)

Background

- NRC began performing cost-benefit analyses in 1976 to evaluate costs and benefits of proposed regulatory actions.
- These evaluations help the staff provide an adequate basis for the proposed action and document a clear explanation of why the staff recommended the proposed action.
- Cost-benefit analyses are used in regulatory decision-making documents such as regulatory analyses and environmental impact statements.
- Regulatory analyses may include safety goal evaluations too.



Background (continued)

- The MACCS code has been used to quantify some of the attributes for cost-benefit analysis, for example, public doses and offsite economic costs that would be averted by a particular severe accident prevention or mitigation measure.
- The net value of a potential nuclear power plant enhancement to mitigate severe accidents, for example, can be calculated with the following basic equation:

Net Value = (APE + AOC + AOE + AOSC) – COE

where

APE = present value of averted public exposure ($)

AOC = present value of averted offsite economic costs ($)

AOE = present value of averted occupational exposure costs ($)

AOSC = present value of averted onsite costs ($)

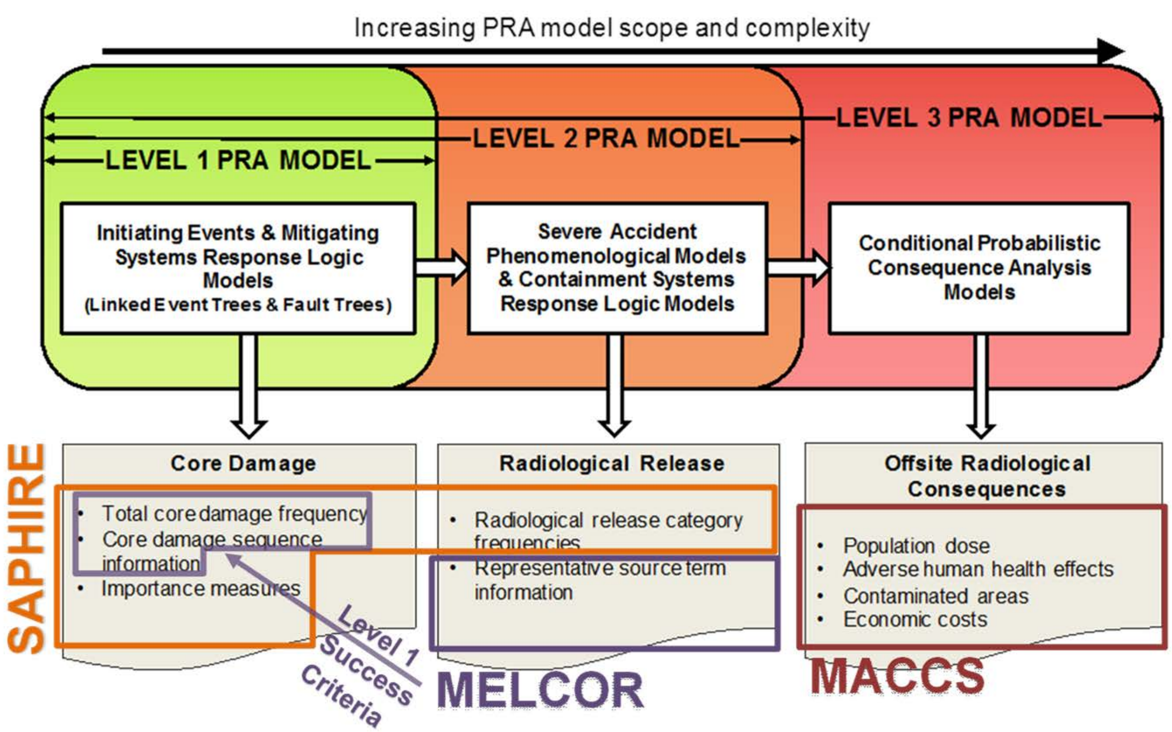
COE = cost of enhancement ($)

Source: NUREG-1437, Supplement 45, F.6.1

Example from a Detailed Regulatory Analysis

- As a result of the events at the Fukushima Dai-ichi Nuclear Power Plant in 2011, the NRC Near Term Task Force created a series of recommendations for enhancing the ability of nuclear power plants to respond to beyond-design-basis events.
- In response to Commission direction, NRC staff began developing the regulatory basis for a containment protection and release reduction (CPRR) rulemaking for boiling water reactors (BWRs) with Mark I and Mark II containments.
- The objective of the CPRR regulatory basis was to determine what, if any, additional requirements were warranted on filtering strategies and severe accident management for BWRs with Mark I and Mark II containments, assuming the installation of severe-accident-capable hardened vents.

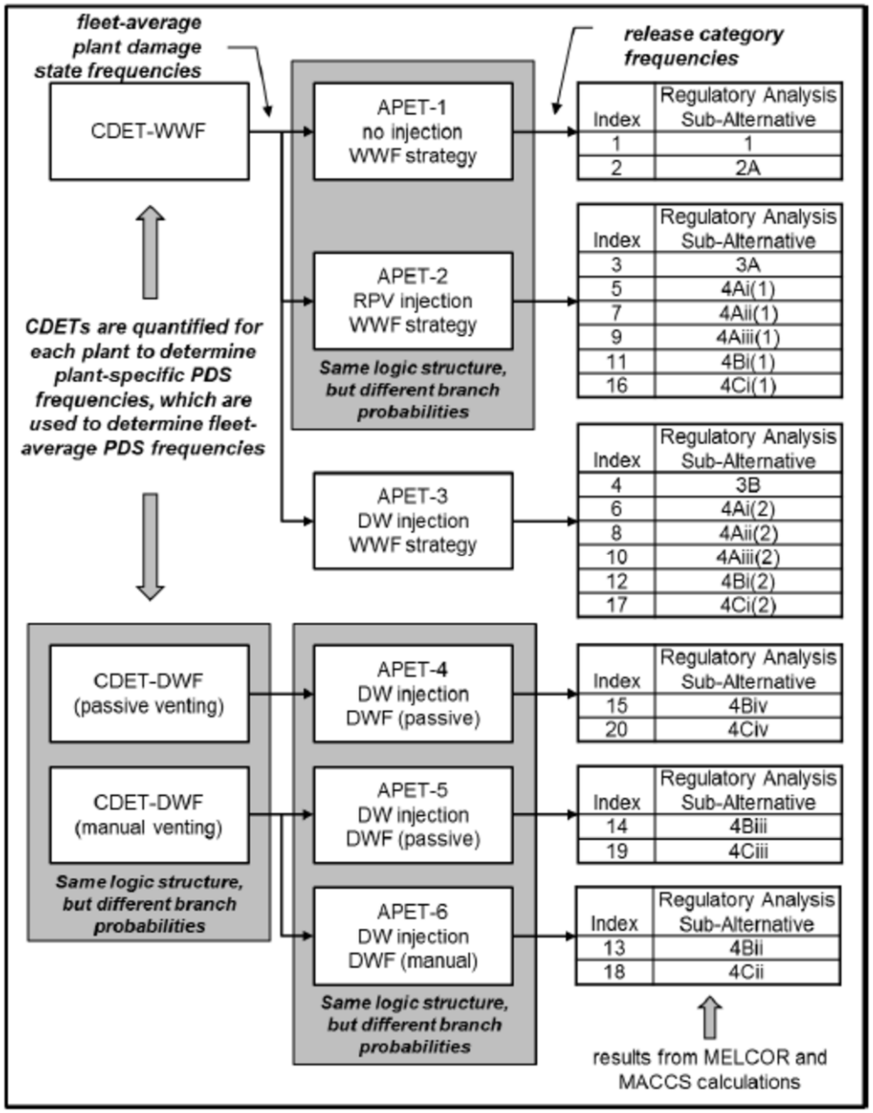
Overall Structure of Traditional NPP PRA Models and Role of SAPHIRE, MELCOR, and MACCS Code Suites



Source: NUREG/BR-0058, Rev. 5, Appendix H

Modular Approach to Event Tree Development

(NUREG-2206, “Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments,” Figure 2-1)

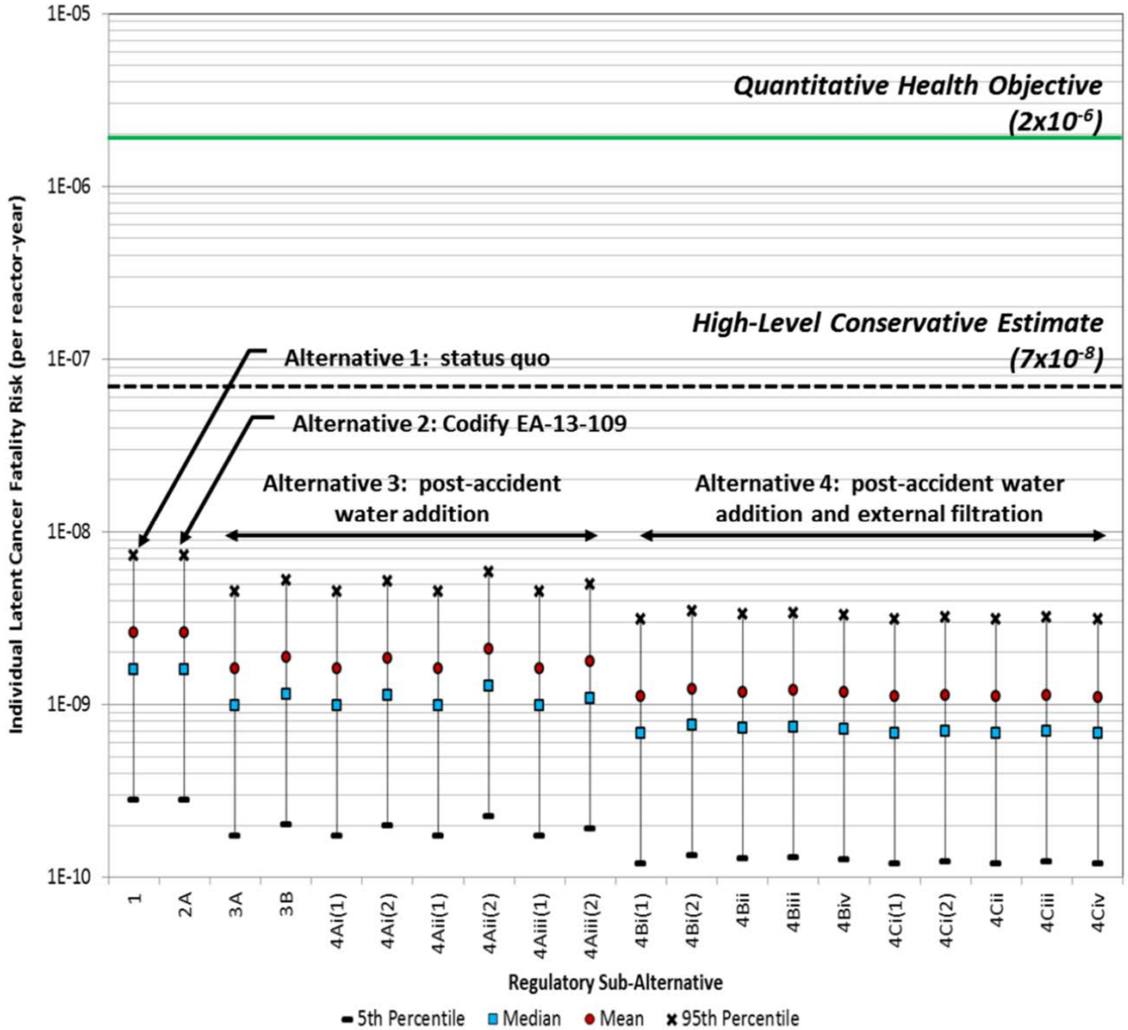


Example - MACCS Results for 9 Mark II Source Term Bins
(SECY-15-0085, Enclosure, Table 4-22)

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
						0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	11DF1000	0.00004%	0.0005%	20.3	20	0	9.72E-08	1.03E-08	3.45E-09	282	345
2	5DF1000	0.0006%	0.005%	32.2	20	0	1.15E-06	1.81E-07	6.35E-08	4,340	5,440
3	42DF100	0.0043%	0.037%	14.3	13	0	6.58E-06	8.67E-07	3.02E-07	20,700	26,700
4	11	0.042%	0.45%	20.3	20	0	7.90E-05	9.68E-06	3.27E-06	202,000	261,000
5	51DF10	0.23%	2.01%	16.6	9	0	1.35E-04	3.39E-05	1.21E-05	689,000	888,000
6	5	0.55%	4.94%	32.2	20	0	2.29E-04	1.05E-04	4.01E-05	2,160,000	2,900,000
7	3	1.09%	10.26%	14.3	20	0	3.08E-04	1.88E-04	7.43E-05	4,140,000	5,580,000
8	1	2.46%	19.81%	22.8	25	0	4.70E-04	3.17E-04	1.25E-04	6,110,000	8,260,000
9	52	3.57%	28.67%	16.6	10	0	4.03E-04	2.46E-04	1.01E-04	5,430,000	7,440,000

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
						0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	11DF1000	0.00004%	0.0005%	20.3	20	381,000,000	381,000,000	-	-	-	-
2	5DF1000	0.0006%	0.005%	32.2	20	381,000,000	381,000,000	0	0	-	-
3	42DF100	0.0043%	0.037%	14.3	13	393,000,000	393,000,000	2	2	0	0
4	11	0.042%	0.45%	20.3	20	844,000,000	846,000,000	44	47	1,030	1,030
5	51DF10	0.23%	2.01%	16.6	9	4,250,000,000	4,380,000,000	130	221	15,400	15,400
6	5	0.55%	4.94%	32.2	20	24,000,000,000	28,000,000,000	303	551	62,400	62,400
7	3	1.09%	10.26%	14.3	20	80,800,000,000	105,400,000,000	698	1,200	619,000	649,000
8	1	2.46%	19.81%	22.8	25	85,500,000,000	109,300,000,000	854	1,680	721,000	741,000
9	52	3.57%	28.67%	16.6	10	53,600,000,000	63,800,000,000	618	1,400	414,000	449,000

Figure 3-3: Uncertainty Bounds for Individual Latent Cancer Fatality Risk



Example from SECY-15-0085, "Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities" Enclosed Draft Regulatory Basis

Severe Accident Mitigation Alternative (SAMA) analysis

- As part of the environmental review performed for initial license renewal for U.S. nuclear power plants, licensees performed a severe accident mitigation alternative (SAMA) analysis.
- A SAMA analysis is a systematic search for potentially cost beneficial enhancements to further reduce nuclear power plant risk.
- The first step of a SAMA evaluation is to identify and characterize the leading contributors to core damage frequency (CDF) and offsite risk based on a plant-specific risk study.
- The MACCS code was used to estimate site-specific offsite consequences.
  - Plant-specific input to the code includes the source terms for each source term category, reactor core radionuclide inventory, site-specific meteorological data, projected population distribution within a 50-mi (80-km) radius for future year, emergency evacuation modeling, and economic data.

Example SAMA Results - Salem Nuclear Generating Station Breakdown of Population Dose by Containment Release Mode

Containment Release Mode	Population Dose (Person-Rem[1] Per Year)	Percent Contribution[2]
Containment over-pressure (Late)	42.9	55
Steam Generator Rupture	31.9	41
Containment isolation failure	2.3	3
Containment intact	0.2	<1
Interfacing system Loss-of-Coolant Accident (LOCA)	0.6	<1
Catastrophic isolation failure	0.4	<1
Basemat Melt-Through (Late)	negligible	negligible
Total[3]	**78.2**	**100**

[1]One person-rem = 0.01 person-Sv

[2]Derived from Table E.3-7 of the ER (PSEG 2009)

[3]Column totals may be different due to round off.

Source: NUREG-1437, Supplement 45, Table F-2

Example SAMA Results - SAMA Cost/Benefit Screening Analysis for Salem Nuclear Generating Station

SAMA	Assumptions	% Risk Reduction		Total Benefit ($)		Cost ($)
		CDF	Population Dose	Baseline (Internal + External)	Baseline With Uncertainty[e]	
9 – Connect Hope Creek Cooling Tower Basin to Salem Service Water System as Alternate Service Water Supply	**Reduce failure probabilities for all service water fouling events by a factor of 10.**	**13**	**11**	**1.7M**	**4.3M**	**1.2M**
10 – Provide Procedural Guidance for Faster Cooldown Loss of RCP Seal Cooling	**The probability that operators would fail to reduce reactor coolant system (RCS) pressure was reduced to 0.1 from 1.0.**	**1**	**<1**	**110K**	**280K**	**100K**
11 – Modify Plant Procedures to Make use of Other Unit's PDP for RCP Seal Cooling	**The probability that operators would fail to respond short/long-term seal injection demand was reduced to 0.1 from 1.0.**	**13**	**12**	**2.0M**	**5.0M**	**100K**
12 – Improve Flood Barriers Outside of 220/440VAC Switchgear Rooms	**Reduce likelihood that the drains would fail to remove the volume of water assumed in the flooding analysis from 1.0E-01 to 1.0E-03.**	**3**	**3**	**550K**	**1.4M**	**475K**
13 – Install Primary Side Isolation Valves on the Steam Generators	Reduce likelihood of a SGTR in each steam generator from 1.75E-03 to 1.75E-05.	6	30	5.2M	13M	18M

Source: NUREG-1437, Supplement 45, Table F-6

References

- NUREG/BR-0058, Rev. 5, Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission, Draft Report for Comment; https://www.nrc.gov/reading-rm/doc-collections/nuregs/brochures/br0058/index.html
- SECY-15-0085, Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling Water Reactors Rulemaking Activities; https://www.nrc.gov/docs/ML1502/ML15022A218.html
- NUREG-2206, Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling Water Reactors with Mark I and Mark II Containments; https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr2206/index.html
- Generic Environmental Impact Statement for License Renewal of Nuclear Plants (NUREG-1437) and supplements; https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1437/index.html

ABBREVIATIONS AND ACRONYMS

APE = averted public exposure
AOC = averted offsite economic costs
AOE = averted occupational exposure
AOSC = averted onsite costs
APET accident progression event tree
BWR boiling water reactor
CDET core-damage event tree
CDF core damage frequency
COE = cost of enhancement
CPRR containment protection and release reduction
DF decontamination factor
DW drywell
DWF drywell first strategy
LOCA loss of coolant accident
NPP nuclear power plant
PDS plant damage state
RCP reactor coolant pump
RPV reactor pressure vessel
SAMA severe accident mitigation alternative
SGTR steam generator tube rupture
WWF wetwell first strategy

Thank you