Session I-B

PSA Methodology

Session I-B Summary

Chair: Katsunori OGURA (JNES), Hak-Kyu LIM (KOPEC)

Six papers were presented and discussed about PSA Methodology in this session.

I-B-1. Jong-Soo CHOI (KINS-Korea): Truncation Error Evaluation for a PSA Model,
The truncation error in quantification was discussed. A simple and practical method to measure the truncation error was suggested. It was discussed that the truncation value had to be decided in context of the essential figure or the number of significant figures of the core damage frequency (CDF).

I-B-2. Akira YAMAGUCHI (Osaka Univ.-Japan): Usage of Information Criterion for Reducing Modeling Uncertainty in Reactor Safety
This paper from Osaka University of Japan, suggested usage of information criterion for reducing modeling uncertainty. The entropy was introduced to judge the model and information as the effective parameter.

I-B-3. Pi-Lin HSU (INER-Taiwan): The Development of a 3-D Risk Matrix for Qualitative Maintenance Risk Management
This paper from INER of Taiwan, showed the 3-D Risk Matrix for qualitative maintenance risk management, which was pre-solved for many configurations. The developed tool is planned to be applied by plant operators to nuclear power plants in Taiwan from next year.

I-B-4. Man Cheol KIM (KAERI-Korea): Some approaches for quantification of important factors in PSA for digital I&C systems
I-B-5. Seung Jun LEE (KAERI-Korea): Effect Estimation of an Automatic Periodic Tests in NPP Digital I&C Systems by Fault Injections
These two papers were presented about digital I&C system by KAERI of Korea, which had lots of hot issues.

I-B-6. Yu YU (INET-China): An Approach for Accident Event Sequence Analysis by Different Phases in Nuclear Power Plant
This paper showed an approach for accident event sequence analysis by different phases in a nuclear power plant (NPP) by Tsinghua University of China.

In this session, the participants, including audiences, talked about current issues of PSA, such as uncertainty, risk management, and digital I&C system analysis. While they discussed in the session, they might understand what others have done and how they cooperate each other.

I -B-1

The 10th KJPSA (제주 서귀포 해비치 호텔&리조트, 5.18-20)

Truncation Error Evaluation for a PSA Model

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KOREA INSTITUTE OF NUCLEAR SAFETY

Jong-Soo Choi

Introduction

- PSAs are based on Minimal Cut Set (MCS) quantification.
- It is impossible to get all MCSs of PSA problems due to memory limitation and computing time.
  (To determine MCSs with manageable size, truncation neglecting low-frequency cutsets is applied.)
- Truncation error (TE) is uncertainty in all PSA results (Risk measures and Importance measures).
  - Incompleteness of PSA quantification method
  - No tools available to evaluate TE for PSA problems
- This paper proposes an approach to TE evaluation of real CDF problems and presents application results .

Introduction

- **For fault tree problems, the TE evaluation method using Monte Carlo techniques and characteristics of fault trees and MCSs was developed.**
  1) J. S. Choi and N. Z. Cho, Truncation Error Evaluation Method for Minimal Cut-Set-Based Fault Tree Analysis, Journal of Nuclear Science and Technology, Vol.42, p.854, 2005.
  2) J. S. Choi and N. Z. Cho, A Practical Method for Accurate Quantification of Large Fault Trees, Reliability Engineering and System Safety, Vol.92, p.971, 2007.
- **The proposed TE evaluation method for real PSA problems is developed on the basis of the previous study regarding FTs.**

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Introduction

- **Proposed TE Evaluation Method for truncated CDFs**
  1) **Develop CCDP model for each initiating event**
     - **Convert Frequency-type problem into Probability-type problem**
  2) **Develop a large # of MCSs for each CCDP under low cut-off value**
     - **Get many MCSs of each CCDP which is convinced that its TE is very small enough to ignore**
     - **Overcome the computing limit, especially of memory size**
  3) **Evaluate TEs by the proposed Monte Carlo method**
     - **Quantify the TEs of CCDP models by comparing each CCDP model and the corresponding MCS set**
     - **TE measure: "MCS Coverage" (defined as the ratio of MCS-covered probability to the actual top-event probability)**
  4) **Evaluate "uncut CDF" and "TE of CDF"**

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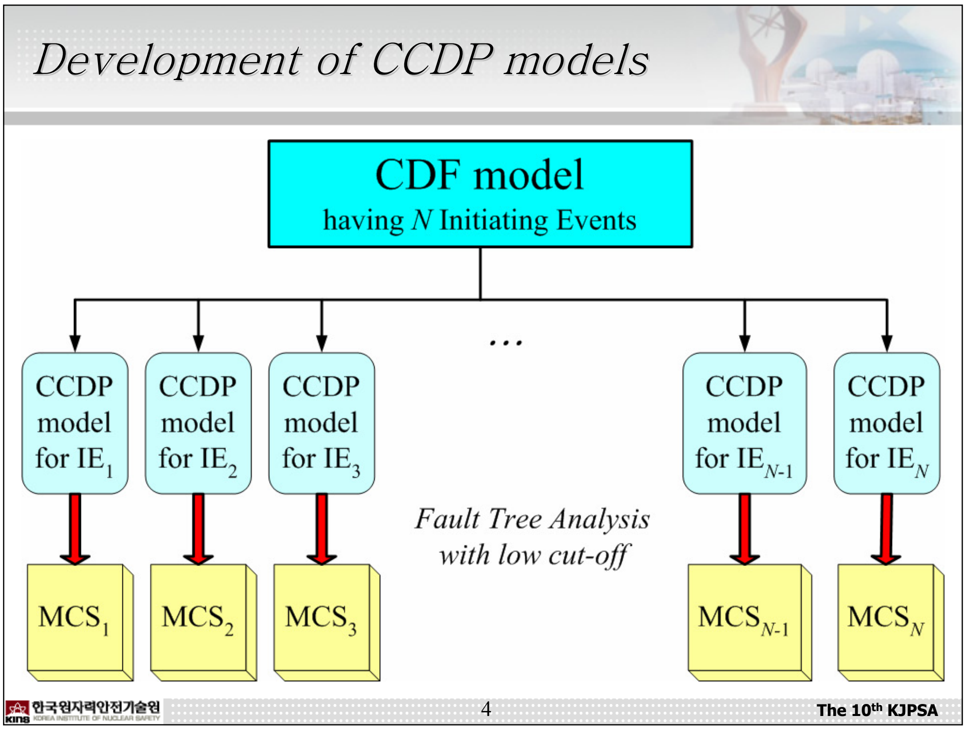
Development of CCDP models
CDF model
having N Initiating Events
CCDP model for IE_1
CCDP model for IE_2
CCDP model for IE_3
…
CCDP model for IE_{N-1}
CCDP model for IE_N
Fault Tree Analysis
with low cut-off
MCS_1
MCS_2
MCS_3
MCS_{N-1}
MCS_N
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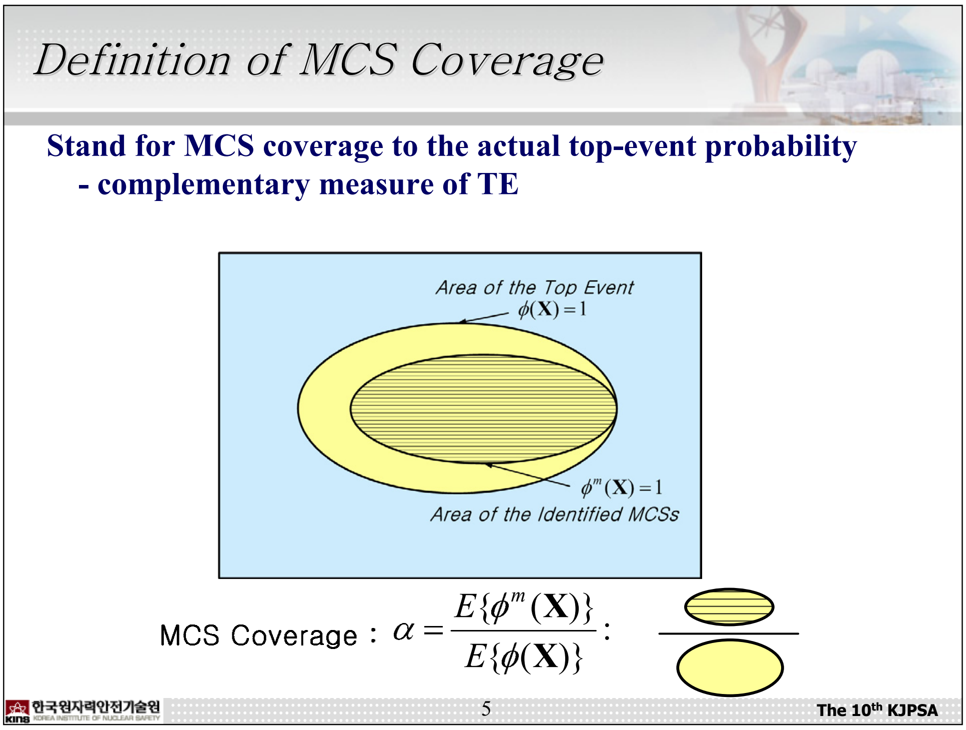
Definition of MCS Coverage
Stand for MCS coverage to the actual top-event probability
- complementary measure of TE
Area of the Top Event
$\phi(\mathbf{X})=1$
$\phi^m(\mathbf{X})=1$
Area of the Identified MCSs
MCS Coverage : $\alpha = \dfrac{E\{\phi^m(\mathbf{X})\}}{E\{\phi(\mathbf{X})\}}$:
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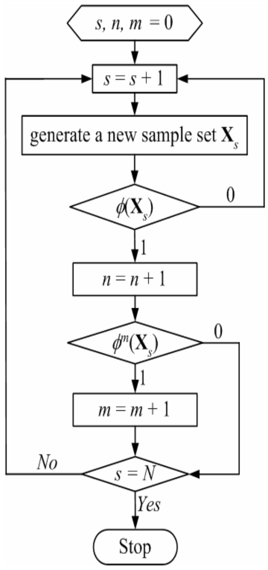
Evaluation of MCS Coverage



$$n = \sum_{s=1}^{N} \phi(\mathbf{X}_s)$$

$$m = \sum_{s=1}^{N} \phi^m(\mathbf{X}_s)$$

$$\longrightarrow \quad \boxed{\hat{\alpha} = \frac{m}{n}}$$

Using random sampling

$$Var(\hat{\alpha}) = \frac{\alpha(1-\alpha)}{N \cdot E\{\phi(\mathbf{X})\}}$$

As α is close up to 1, its variance is close to 0.

⇩

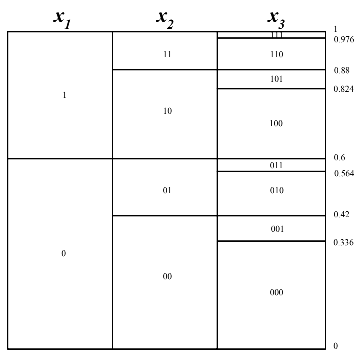
As # of MCSs increases, the efficiency of MC simulation increases.

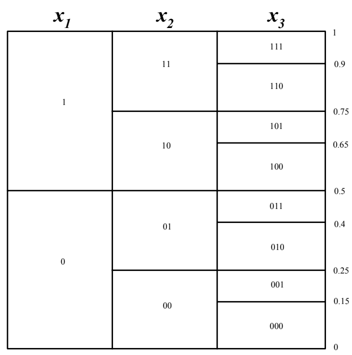
Evaluation of MCS Coverage

- **Importance Sampling Technique (IS)**
  - One of variance reduction techniques
  - Very useful in rare event problems

i	p_i	w_ip_i	w_i
1	0.4	0.5	1.25
2	0.3	0.5	1.667
3	0.2	0.4	2.0

$f(\mathbf{X})$ $\longrightarrow$ $f^*(\mathbf{X})$





Evaluation of MCS Coverage

Applying Importance Sampling Technique

- **Joint probability distribution for X**

$$f(\mathbf{X}) = \prod_{i \in \mathbf{B}_1(\mathbf{X})} p_i \times \prod_{i \in \mathbf{B}_0(\mathbf{X})} (1 - p_i).$$

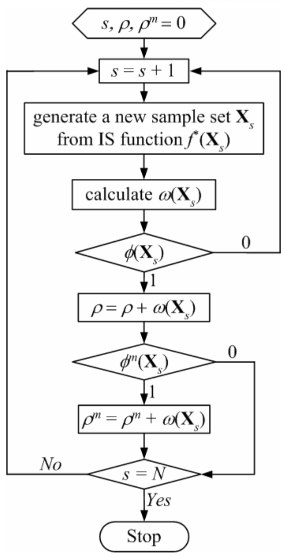
- **If the probability p_i is altered into $w_i p_i$ to apply IS**

$$f^*(\mathbf{X}) = \prod_{i \in \mathbf{B}_1(\mathbf{X})} w_i p_i \times \prod_{i \in \mathbf{B}_0(\mathbf{X})} (1 - w_i p_i).$$

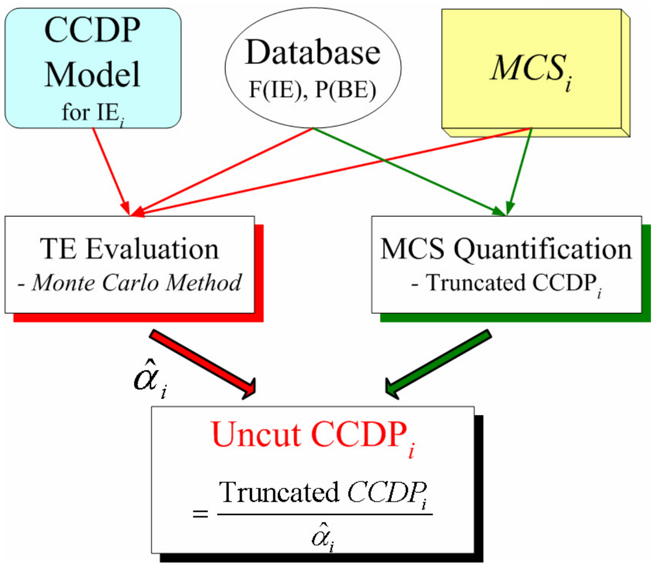
$$\omega(\mathbf{X}_s) \equiv \frac{f(\mathbf{X}_s)}{f^*(\mathbf{X}_s)} = \prod_{i \in \mathbf{B}_1(\mathbf{X}_s)} \frac{1}{w_i} \times \prod_{i \in \mathbf{B}_0(\mathbf{X}_s)} \frac{(1 - p_i)}{(1 - w_i\ p_i)}$$

- **The MCS coverage can be estimated by**

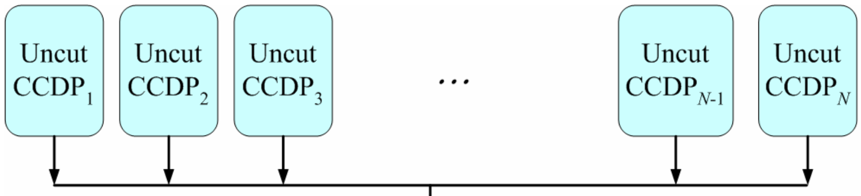
$$\hat{\alpha} = \frac{\rho}{\rho^m}$$



Evaluation of Uncut CCDP for each IE



Truncation Error of PSA results



$$\mathit{Uncut\ CDF} = \sum_{i=1}^{N} F(IE_i) \times \mathit{Uncut\ CCDP}_i$$

$$\text{Truncation Error of PSA rusult} = \mathit{Uncut\ CDF} - \mathit{Truncated\ CDF}$$

Example PSA model

PSA results for a PSA model (YGN-3/4 Risk Monitor Model)

# of IEs: 19 **# of BEs: 1944** **# of logic equations: 6456**

IE	*F(IE)*	CCDP	# MCSs
1: A: LLOCA	5.0E-6	2.82E-03	85
2: S1: MBLOCA	4.0E-5	4.57E-03	378
3: S2: SBLOCA	5.0E-4	2.74E-03	885
4: TR: SGTR	7.0E-3	2.81E-05	2,697
5: IISL: IS-LOCA	3.03E-7	1.0	1
6: IRVR: RV rupture	2.70E-7	1.0	1
7: T1A: transient	3.60E-1	5.25E-08	1,605
8: T1B: transient	7.02E-1	3.30E-07	4,441
9: T2 : transient	9.56E-2	2.24E-06	10,546
10: TP: LOOP	3.78E-2	8.17E-05	16,117

IE	*F(IE)*	CCDP	# MCSs
11: TS1: MSLB	6.50E-3	2.57E-06	387
12: TS2: MSLB	6.50E-3	3.48E-08	29
13: TC1: CCW	8.92E-4	8.50E-07	147
14: TF1: G2&4 power	9.29E-6	0.	0
15: TF2: G3 power	5.87E-6	0.	0
16: LODCA: DC bus A	3.34E-4	6.65E-04	1,572
17: LODCB: DC bus B	3.34E-4	9.03E-05	2,383
18: TA-1: ATWS	1.39	1.82E-08	222
19: TA-2: ATWS	0.1286	3.44E-09	14

Total CDF = 6.1841E-6 **# of MCSs : 41,510**

Cut-off value used in the PSA report : 10^{-12}

Sensitivity of cut-off value

PSA results for different cut-off values

V_c 1)	10^{-12}	10^{-14}	10^{-16}	$\leq 10^{-17}$
CDF	6.184E-6	6.239E-6	6.245E-6	? 3)
# MCSs	~ 42K	~ 638K	~ 6.26M	?
# SEs 2)	307	327	335	?
CPU time	69.5	691	3370	

1) Truncation limit or Cut-off value
2) Number of significant events (RRW>1.005 or RAW>2.0)
3) Not calculated due to memory limitation

From the sensitivity calculation,
we don't know the CDF without truncation (uncut CDF) any more.

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Development of CCDP models

CCDP Quantification with low V_c

IE	CCDP	V_c*	# MCSs**
A	2.85E-3	2.0E-17	~ 6.76M
S1	4.58E-3	2.5E-17	~ 8.54M
S2	2.74E-3	2.0E-16	~ 4.76M
TR	2.87E-5	1.4E-16	~ 4.37M
ISL	1.0		1
IRVR	1.0		1
T1A	6.87E-8	2.8E-18	~ 7.60M
T1B	3.46E-7	1.4E-17	~ 4.10M
T2	2.36E-6	1.0E-17	~ 7.08M
TP	8.23E-5	2.6E-17	~ 8.62M

IE	CCDP	V_c*	# MCSs **
TS1	2.82E-6	1.5E-16	~ 4.16M
TS2	7.39E-8	1.5E-17	~ 4.07M
TC1	9.48E-7	1.1E-18	~ 1.29M
TF1	6.87E-8	1.1E-17	~ 4.17M
TF2	6.87E-8	1.7E-17	~ 3.39M
LODCA	6.69E-4	3.0E-20	~ 3.54M
LODCB	9.67E-5	3.0E-18	~ 1.94M
TA-1	1.84E-8	7.2E-20	~ 1.98M
TA-2	3.57E-9	7.8E-22	~ 3.52M
Total CDF	6.2452E-6		~ 79.9M

* Cut-off values used in quantification of the CCDP models

** Sets of MCSs enough to cover actual CCDPs

❖ **in order to get more accurate estimates of MCS coverage**

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MCS Coverages of CCDPs

Evaluated by Monte Carlo Simulations

IE	MCS Coverage (α)		
	Mean	Variance	95% CL
A	1.000000	0.0	1.000000
S1	1.000000	1.54E-13	1.000000
S2	1.000000	0.0	1.000000
TR	0.999988	1.79E-10	0.999984
IISL	1.0		
IRVR	1.0		
T1A	0.999813	2.58E-07	0.999660
T1B	0.999959	1.28E-08	0.999925
T2	0.999993	1.53E-10	0.999990
TP	0.999666	1.79E-08	0.999626
TS1	0.999903	7.10E-09	0.999878

IE	MCS Coverage (α)		
	Mean	Variance	95% CL
TS2	0.999470	1.26E-06	0.999134
TC1	0.999993	5.39E-10	0.999986
TF1	0.999618	1.12E-06	0.999301
TF2	0.999666	1.30E-07	0.999557
LODCA	1.000000	0.0	1.000000
LODCB	1.000000	0.0	1.000000
TA-1	0.999932	6.49E-08	0.999856
TA-2	0.999999	3.29E-11	0.999997

Uncut CDF = SUM (uncut CCDPs)	
Mean	6.2463E-06
Upper Bound	6.2464E-06

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Truncation Error of Example PSA Model

For PSA results calculated with 10^{-12} cut-off

	Uncut CDF	Truncated CDF
CDF	6.2463E-6 (mean) 6.2464E-6 (UB)	6.1841E-6
# MCSs	~ 79.90M (CCDP models)	~ 42K
MCS coverage of truncated CDF		99.0043% (mean) 99.0021% (UB)
Truncation error of truncated CDF		**< 1% of Real CDF**

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Conclusions

- Until now, there is no practical tools which quantify PSA models without truncation error. (So, we need to evaluate truncation errors.)
- This paper shows that the proposed method can successfully quantify the truncation error of an example CDF model.
- The proposed TE evaluation method
  - Can be easily implemented in PSA Problems
  - Will be useful to confirm that the TEs of PSA results are small enough to ignore.
  - Estimates TEs of PSA results with accuracy.

I -B-2

Usage of Information Criterion for Reducing Modeling Uncertainty in Reactor Safety

Akira Yamaguchi and Takashi Takata

Osaka University
Department of Energy and Environment

Introduction

- In the reactor safety analysis, we predict the future event
  - We try to find data generation mechanism
- The best model simulates the data generation mechanism the best
  - Past data are used for model development
  - We do not know the true model
- What is the best model when several models are available?
- How can we reduce the model uncertainty?
  - Future data are used for model update and uncertainty reduction

What is model?

- Model is a mathematical expression of phenomenon

$$y_i = M_i(\mathbf{x}|\boldsymbol{\xi}) \qquad \mathbf{x} : input,\ \ \boldsymbol{\xi} : parameter\,(bias)$$

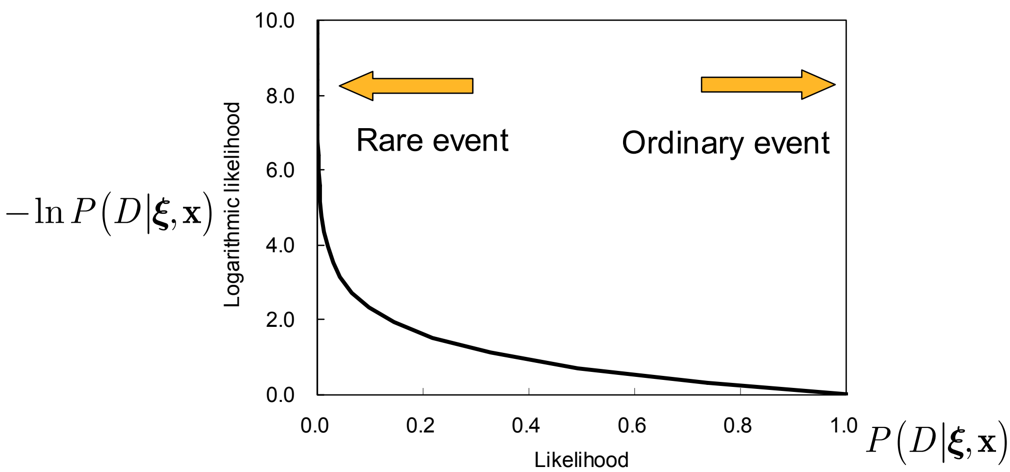
  - Input data : stochastic vector with PDF $h(\mathbf{x})$
  - Bias : stochastic vector with PDF, reflect our knowledge $g(\boldsymbol{\xi})$
  - Model output: stochastic value with PDF $f_i(y_i|\mathbf{x},\boldsymbol{\xi})$
- If m models are available, the mixture?

$$\bar{y} = \sum_{i=1}^{m} y_i\beta_i \Big/ \sum_{i=1}^{m} \beta_i \qquad f(\bar{y}|\mathbf{x},\boldsymbol{\xi})$$

Logarithmic Likelihood

$$-\ell n P(D|\boldsymbol{\xi},\mathbf{x}); \qquad D : data$$

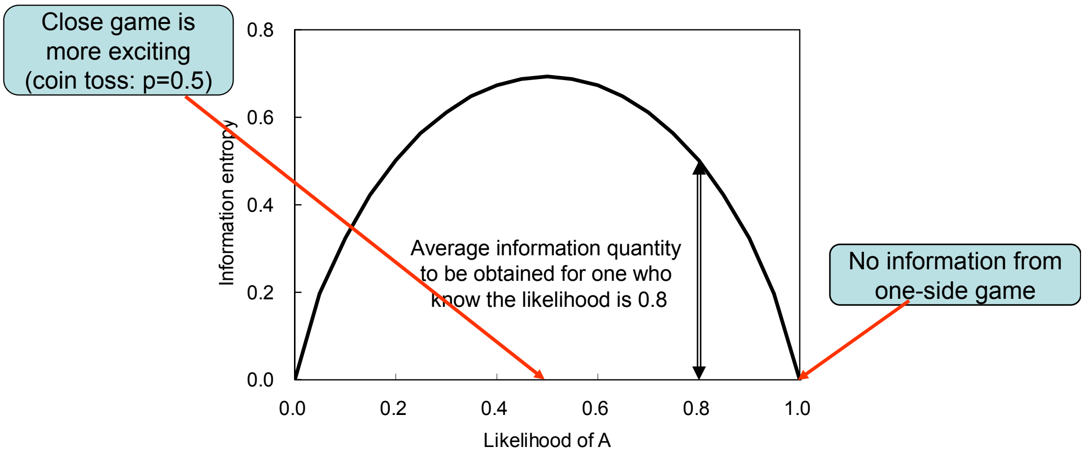
- Information quantity of an observation
  - Rare event has more information quantity



Information Entropy

$$l(\boldsymbol{\xi},\mathbf{x}) = -\int P\left(D\middle|\boldsymbol{\xi},\mathbf{x}\right)\ell n P\left(D\middle|\boldsymbol{\xi},\mathbf{x}\right)dD = -E\left[\ell n P\left(D\middle|\boldsymbol{\xi},\mathbf{x}\right)\right]$$

- Average of information quantity
  - Exclusive events *A* and *B*
  - Likelihoods of *A* and *B* are *p* and *1-p*, respectively

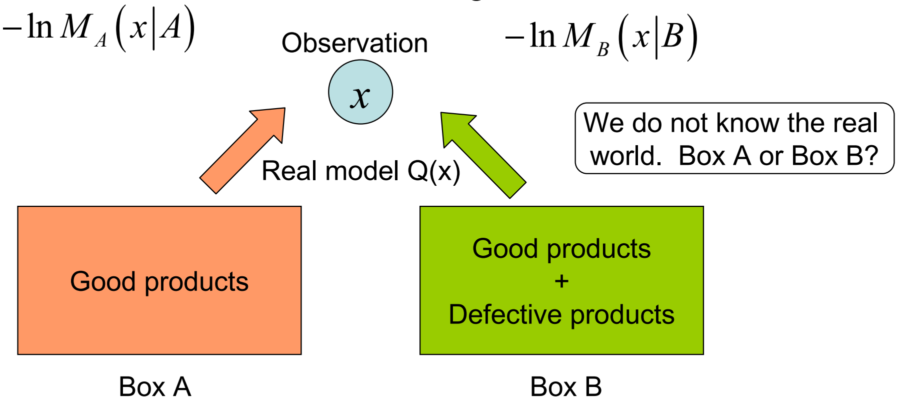


Which model is more likely?

- Likelihood of *A* $M_A\left(x\middle|A\right)$
- Likelihood of *B* $M_B\left(x\middle|B\right)$

Model M⇔Real world Q	Distance of M and Q $I\left(Q:M\right)$

- Logarithmic likelihood $-\ln M_A\left(x\middle|A\right)$
- Logarithmic likelihood $-\ln M_B\left(x\middle|B\right)$



Information criterion

- Kullback–Leibler divergence (KL)
  - Q: true distribution of x
  - P: predicted distribution of x
  - KL=0 if P=Q (prediction is true), otherwise always positive (Gibbs' inequality)

$$KL\left[Q(x):P(x)\right]=\int Q(x)\ln\left\{\frac{Q(x)}{P(x)}\right\}dx=\underbrace{\boxed{E\left[\ln\left\{Q(x)\right\}\right]}}_{\text{Constant}}-\boxed{E\left[\ln\left\{P(x)\right\}\right]}\geq 0$$

  - "Best prediction of P" minimizes

$$L=-\int Q(x)\ln\left\{P(x)\right\}dx=-E\left[\ln\left\{P(x)\right\}\right]$$

Entropy

Q is unknown. Model is necessary to predict P(x).

Information criterion (cont.)

- If the model has parameters $\hat{\xi}_k$

$$KL\left(Q(x):P_k\left(x,\hat{\xi}_k\right)\right)=\int Q(x)\ln Q(x)\,dx-\int Q(x)\ln P_k\left(x,\hat{\xi}_k\right)dx$$

$$=E\left[\ln\left\{Q(x)\right\}\right]-E\left[\ln\left\{P_k\left(x,\hat{\xi}_k\right)\right\}\right]$$

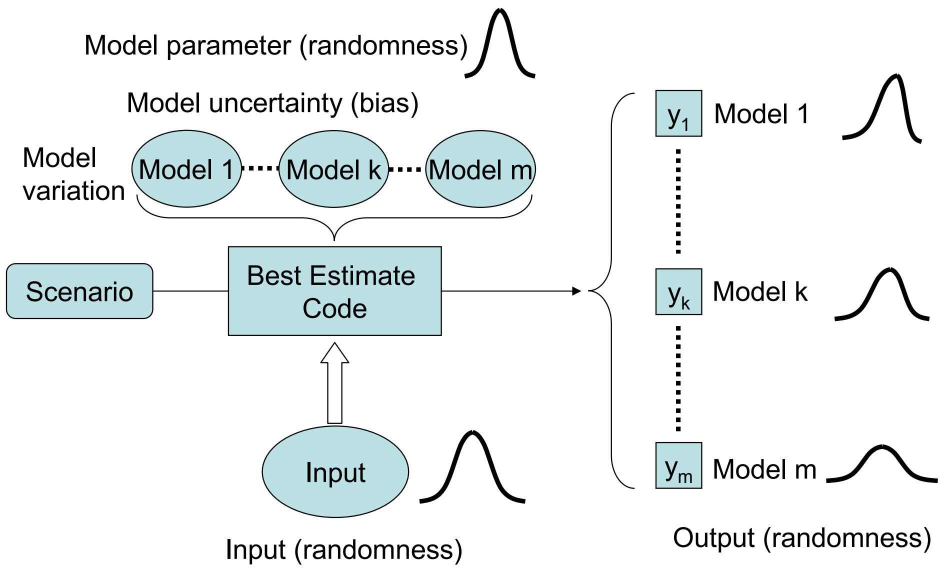
  - Problem is to find a model M_k that minimizes

$$l_k=-\int Q(x)\ln P_k\left(x,\hat{\xi}_k\right)dx$$

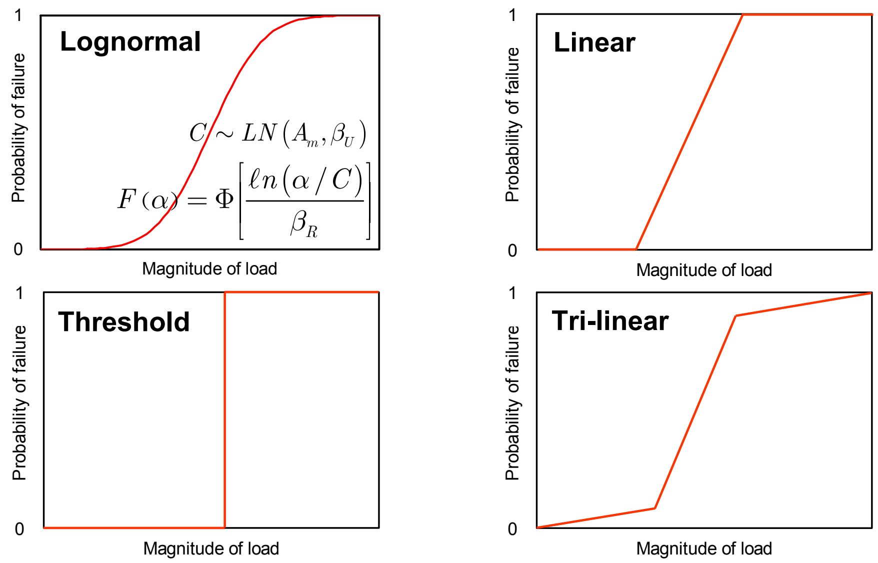
  - Parameter $\hat{\theta}_k$ is subjective and stochastic.
  - Obtain average with regard to $\hat{\xi}_k$

$$L_k=E\left[l_k\left(\hat{\xi}_k\right)\right]$$

Uncertainty of prediction
BEPU approach



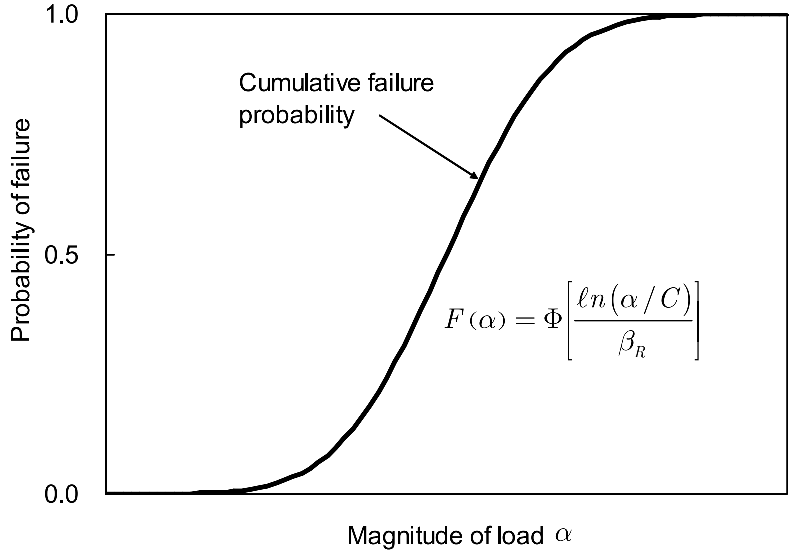
Fragility model variation



Which model is the best?

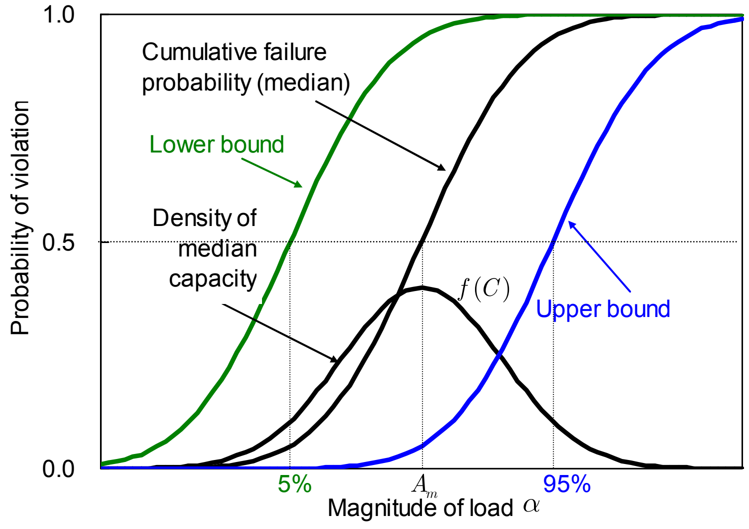
What the parameter values should be?

Failure probability model (model parameter is definite)



Failure probability model (model uncertainty)

$$f(C) = \frac{1}{\sqrt{2\pi}\beta_U C}\exp\left[-\frac{1}{2}\left\{\frac{\ell n(C/A_m)}{\beta_U}\right\}^2\right] \qquad F(\alpha) = \Phi\left[\frac{\ell n(\alpha / C)}{\beta_R}\right]$$



Reduce model uncertainty Bayesian update

- Model gives the likelihood $l(y|\boldsymbol{\xi},\mathbf{x}) = f(y|\mathbf{x},\boldsymbol{\xi},M_i)$
  - Model M has input data $\mathbf{x}$ and parameter $\boldsymbol{\xi} \sim g(\boldsymbol{\xi})$
- We select the model with minimum information criterion
- How can we reduce the model uncertainty?
  - Obtain new evidence (data)!

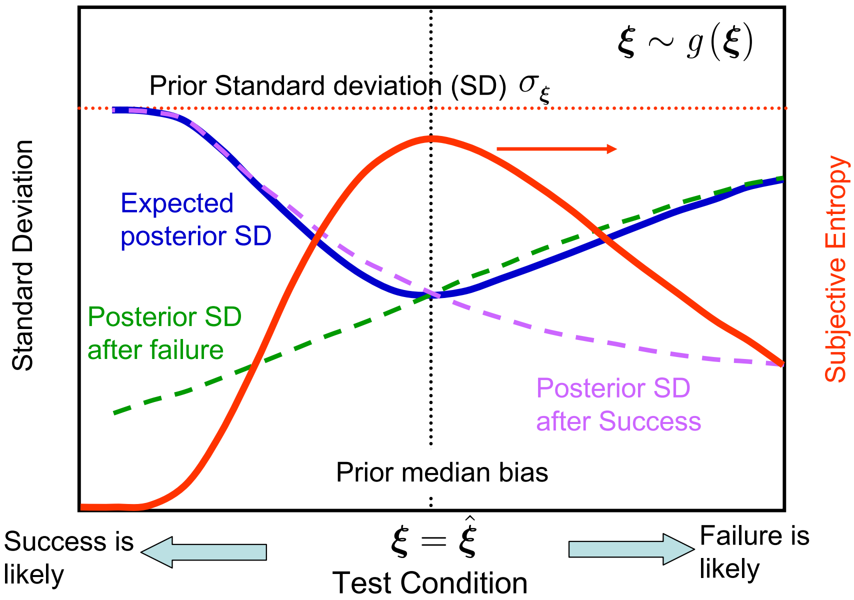
We update the model uncertainty with Bayes theorem

$$g(\boldsymbol{\xi}|Evidence) \propto f(Evidence|\boldsymbol{\xi},\mathbf{x})\, g(\boldsymbol{\xi})$$

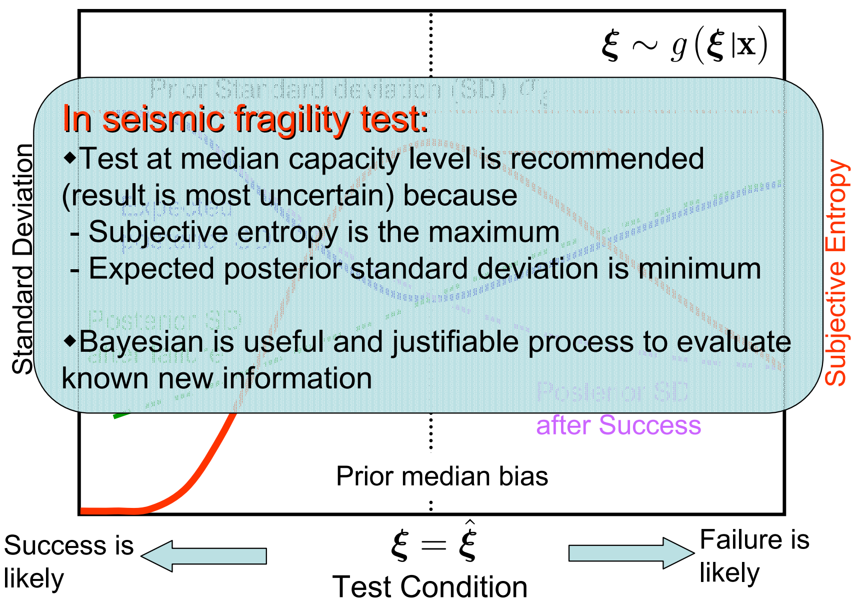
Information criterion and fragility

Model	M	*Double Lognormal*
Parameter	$\hat{\xi}$	C
pdf of parameter	$f(\hat{\xi})$	$f(C)$
mean	$\hat{\xi}_m$	A_m
SD	σ	β_U
Likelihood	$P(x_i,\hat{\xi})$	$F(\alpha)$
Log − likelihood	$-\ln\{P(x_i,\hat{\xi})\}$	$-\ln\{F(\alpha)\}$
Entropy	$l = -\int Q(x)\ln P(x,\hat{\xi})\,dx$	$l = -F(\alpha)\ln F(\alpha) - \{1-F(\alpha)\}\ln\{1-F(\alpha)\}$
Subjective entropy	$L = E\left[l(\hat{\xi})\right]$	$L = \int l\, f(C)\,dC$
Data	$D = \{x_1,\cdots,x_n\}$	*Success / failure*

Entropy and Bayesian Uncertainty Reduction



Entropy and Bayesian Uncertainty Reduction



Conclusions

- Model with minimum entropy is the “good” model
- Information with maximum entropy is the “good” information
- Entropy and uncertainty are equivalent
- For uncertainty reduction in PSA
  - Select the model that minimize entropy
  - Collect evidence that maximize entropy
- Coupled use of Bayesian and information criterion is recommended for uncertainty reduction.
  - Bayesian evaluate “known” information
  - Information criteria evaluate model goodness
  - Entropy evaluate “future” information

Introduction

- We want to predict the future event (data)
  - We need to know the data generation mechanism
  - But we do not know the true model
- The best model simulates the data generation mechanism the best
- We observed past events (data)
  - Data is used for model development
- We have several models
  - Data is used for model selection
  - Data is used for model update

Model selection and uncertainty reduction

- Which model is to be selected?
  - Smallest information criterion is the best

    $l_k = -\int Q(x) \ln P_k\left(x, \hat{\theta}_k\right) dx \qquad \hat{l}_k = -\frac{1}{n}\sum_{i=1}^{n} \ln\left\{P_k\left(x_i, \hat{\theta}_k\right)\right\} \Leftarrow$ model M_k
  - Average with regard to model parameters $\hat{\theta}_k$

    $L_k = E\left[l_k\left(\hat{\theta}_k\right)\right]$
- How to reduce uncertainty?
  - Uncertainty of model parameter $\hat{\theta}_k$
  - Use data $D = \{x_1, \cdots, x_n\}$

I -B-3

Session I-B : PSA Methodology

The Development of a 3-D Risk Matrix for Qualitative Maintenance Risk Management

Pi-Lin Hsu, Chung-Kung Lo

PRA Group

Institute of Nuclear Energy Research

Presented by: Pi-Lin Hsu

ISO 9001
BSMI

The 10th KJPSA

18-20 May 2009

PRA/INER

Outline

- Introduction
- System Design
- Conclusion

PRA/INER

Introduction

- The 3-D Risk Matrix provides plant personnel for qualitative assessment of operational risk. It is a sub-function of the MR tool.
- **M**aintenance **R**ule (MR) is an American regulation, 10 CFR 50.65, it declares the "Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants ".
- The purpose to implement MR in Taiwan
  - American nuclear power plants begin to implement MR since the middle age of 1990. Several Operating indicators and the performance of power generation has made great progress.
  - In view of the fruitful outcomes, the Taiwan Power Company (TPC) determines to implement MR in August of 2004, and plans to put it into practice for the basis of executing rolling on-line maintenance in the end of 2009.

PRA/INER 2

Introduction (Cont.)

- How to implement?
  - The TPC initiated a corporative project with Institute of Nuclear Energy Research to setup MR for three NPPs in Taiwan in November 2005. In this project, INER developed two MR tools,
  - MEMOS (the Maintenance Effectiveness Monitor System)
    - It serves as a platform for the plants to scope the monitoring program, and to monitor plant daily maintenance activities. **(a1, a2, a3, b1, b2)**
  - The MIRU (the Maintenance Integrated Risk Utilities)
    - It serves as a platform for the plants to plan their daily maintenance activities and perform the risk assessment in the future rolling on-line maintenance. **(a4)**
    - The 3-D Risk Matrix is a sub-function of the MIRU for qualitative assessment of operational risk.

a(1): SSC not meeting goals…… corrective actions needed

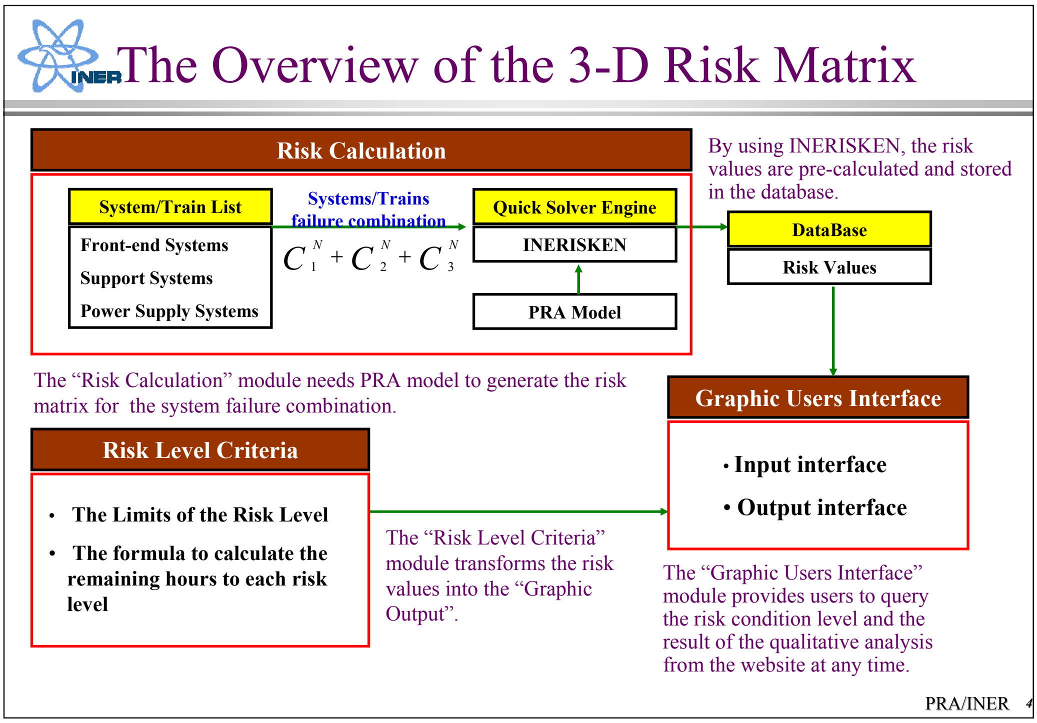
a(2): SSC remains capable of performing intended function

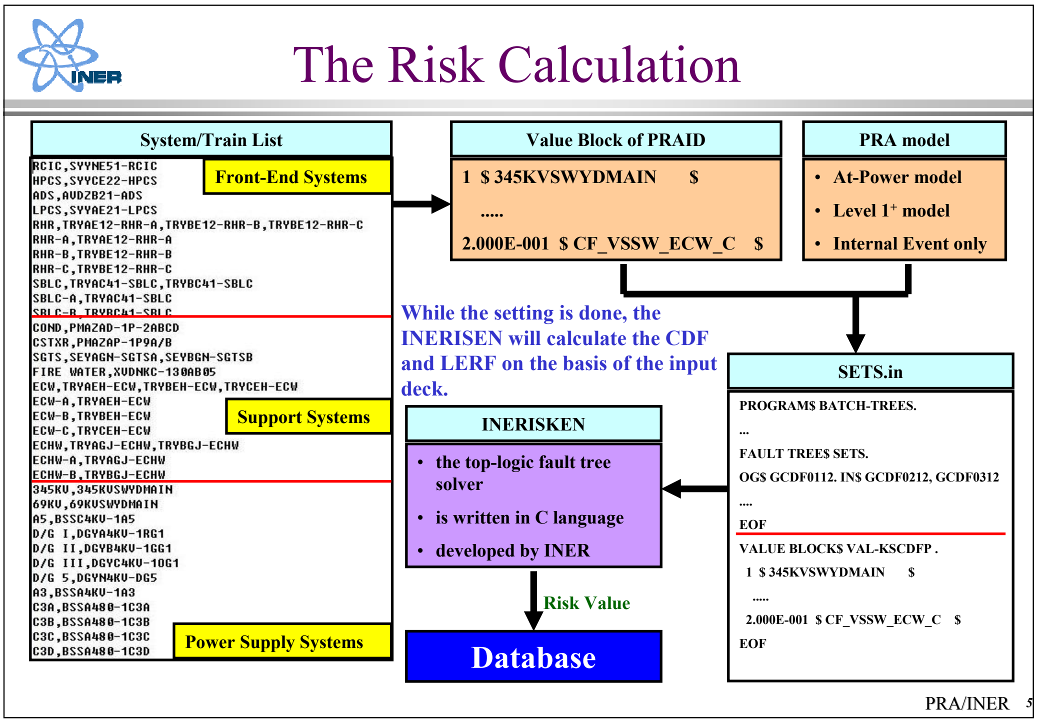
a(3): SSC goals and PM activities are periodically evaluated

a(4): Before maintenance, the licensee shall assess and manage the increased risk

b(1): Safety-related SSCs

b(2): Nonsafety related SSCs


The Overview of the 3-D Risk Matrix
Risk Calculation
System/Train List
Front-end Systems
Support Systems
Power Supply Systems
Systems/Trains failure combination
$C_1^N + C_2^N + C_3^N$
Quick Solver Engine
INERISKEN
PRA Model
By using INERISKEN, the risk values are pre-calculated and stored in the database.
DataBase
Risk Values
The "Risk Calculation" module needs PRA model to generate the risk matrix for the system failure combination.
Graphic Users Interface
• Input interface
• Output interface
Risk Level Criteria
• The Limits of the Risk Level
• The formula to calculate the remaining hours to each risk level
The "Risk Level Criteria" module transforms the risk values into the "Graphic Output".
The "Graphic Users Interface" module provides users to query the risk condition level and the result of the qualitative analysis from the website at any time.
PRA/INER 4


The Risk Calculation
System/Train List
Front-End Systems
RCIC,SYYNE51-RCIC
HPCS,SYYCE22-HPCS
ADS,AVDZB21-ADS
LPCS,SYYAE21-LPCS
RHR,TRYAE12-RHR-A,TRYBE12-RHR-B,TRYBE12-RHR-C
RHR-A,TRYAE12-RHR-A
RHR-B,TRYBE12-RHR-B
RHR-C,TRYBE12-RHR-C
SBLC,TRYAC41-SBLC,TRYBC41-SBLC
SBLC-A,TRYAC41-SBLC
SBLC-B,TRYBC41-SBLC
Support Systems
COND,PMAZAD-1P-2ABCD
CSTXR,PMAZAP-1P9A/B
SGTS,SEYAGN-SGTSA,SEYBGN-SGTSB
FIRE WATER,XVDNKC-130AB05
ECW,TRYAEH-ECW,TRYBEH-ECW,TRYCEH-ECW
ECW-A,TRYAEH-ECW
ECW-B,TRYBEH-ECW
ECW-C,TRYCEH-ECW
ECHW,TRYAGJ-ECHW,TRYBGJ-ECHW
ECHW-A,TRYAGJ-ECHW
ECHW-B,TRYBGJ-ECHW
Power Supply Systems
345KV,345KVSWYDMAIN
69KV,69KVSWYDMAIN
A5,BSSC4KV-1A5
D/G I,DGYA4KV-1RG1
D/G II,DGYB4KV-1GG1
D/G III,DGYC4KV-1OG1
D/G 5,DGYN4KV-DG5
A3,BSSA4KV-1A3
C3A,BSSA480-1C3A
C3B,BSSA480-1C3B
C3C,BSSA480-1C3C
C3D,BSSA480-1C3D
Value Block of PRAID
1 $ 345KVSWYDMAIN $
.....
2.000E-001 $ CF_VSSW_ECW_C $
PRA model
• At-Power model
• Level 1+ model
• Internal Event only
While the setting is done, the INERISEN will calculate the CDF and LERF on the basis of the input deck.
SETS.in
PROGRAM$ BATCH-TREES.
...
FAULT TREE$ SETS.
OG$ GCDF0112. IN$ GCDF0212, GCDF0312
....
EOF
VALUE BLOCK$ VAL-KSCDFP .
1 $ 345KVSWYDMAIN $
.....
2.000E-001 $ CF_VSSW_ECW_C $
EOF
INERISKEN
• the top-logic fault tree solver
• is written in C language
• developed by INER
Risk Value
Database
PRA/INER 5

The Risk Level Criteria

Risk matrix is a table of numerical values which represent the risk levels of systems/trains failure combinations. It is usually expressed as the multiplier of the baseline CDF. By checking the matrix, an operator can quickly determine the risk condition level of the current plant configuration and then take actions accordingly as required by the plant risk management process.

- Typically, risk matrix is used for two-system/train failure combination. All the system/trains are listed in the top row and the left column.
- The diagonal cells are represented the single failure of the system. The upper and lower triangle matrix, represents two-system failure, is symmetrical to the diagonal line, and the risk value is the same both sides.
- Usually the risk matrix displays the lower triangle matrix, and the color of the risk matrix square directly correlates to the current risk condition level.

	SYS-1	SYS-2	SYS-3	SYS-4	....	SYS-N
SYS-1	xxxxx					
SYS-2	xxxxx	xxxxx				
SYS-3	xxxxx	xxxxx	xxxxx			
SYS-4	xxxxx	xxxxx	xxxxx	Xxxxx		
....	xxxxx	xxxxx	xxxxx	Xxxxx	xxxxx	
SYS-N	xxxxx	xxxxx	xxxxx	Xxxxx	xxxxx	xxxxx

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The Risk Level Criteria (Cont.)

The color definition referred the administrative regulation of American nuclear power plants is divided four risk condition.

- low risk awareness level : CDF <= 3 times of the baseline
- medium risk awareness level : 3 times <= CDF <= 10 times
- high risk awareness level : CDF > 10 times
- not allowed by technical specifications and are treated as risk high level

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The Risk Level Criteria (Cont.)

The qualitative analysis, according to the specific failure combination, estimates the remaining hours before reaching to the potentially more significant risk level. It reminds users to make proper compensatory measures in time.

- The ΔCDP upper limit of the green is 1.0E-6, 1.0E-5 for white, 1.0E-4 for yellow.
- The ΔLERP limit is one order lower than the corresponding ΔCDP.

ΔCDP	1.0E-6	1.0E-5	1.0E-4	
	G	W	Y	R
ΔLERP	1.0E-7	1.0E-6	1.0E-5	

The Risk Level Criteria (Cont.)

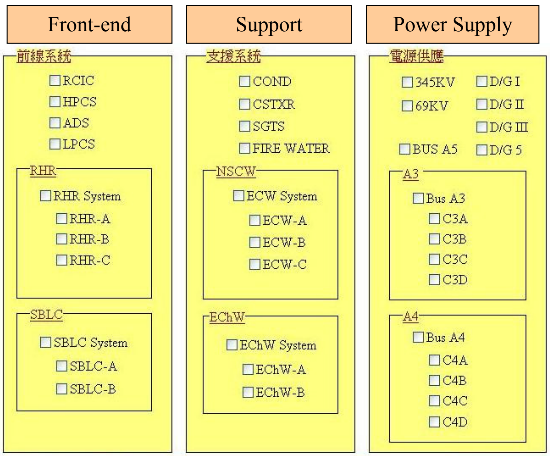
The formula to calculate the remaining hour
before reaching to the potentially more significant risk level

	Reaching to White	Reaching to Yellow	Reaching to Red
ΔCDP	HRwcdp= 1.0E-6 / $DiffCDF_i$ * 8760	HRycdp= 1.0E-5 / $DiffCDF_i$ * 8760	HRrcdp= 1.0E-4 / $DiffCDF_i$ * 8760
ΔLERP	HRwlerp= 1.0E-7 / $DiffCDF_i$ * 8760	HRylerp= 1.0E-6 / $DiffCDF_i$ * 8760	HRrlerp= 1.0E-5 / $DiffCDF_i$ * 8760
Result	Min(HRwcdp , HRwlerp)	Min(HRycdp , HRylerp)	Min(HRrcdp , HRrlerp)

- At first, the risk value for the specific system failure (CDFi) is query from the database and then minus the baseline risk value (CDFbase) to get the DiffCDFi. (DiffCDFi = CDFi - CDFbase)
- Since the total hours for one year is 8760, the ΔCDP and ΔLERP remaining hours reaching to each color is the upper limit divide DiffCDFi times 8760. (remaining hour = upper limit / DiffCDFi * 8760)
- Finally, the minimum remaining hours between ΔCDP and ΔLERP for each risk level will be shown in the output display.

The Input Interface

- The Input Interface provides users to specify at most three systems/trains failure combination.
- If one system/train is failed, the operator clicks the checkbox in front of the system/train to set them as failed.
- In this example, the input interface is a BWR plant, the systems/trains in PRA model are 39. Therefore the risk combination is equal to 9919.



The Input Interface of a BWR Plant "A"

$$C_1^{39} + C_2^{39} + C_3^{39} \quad = \mathbf{9919}$$

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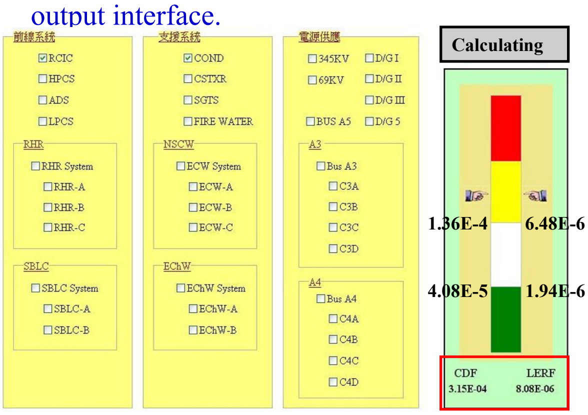
The Output Interface – Risk Level

While pressing the calculating button, the risk value is retrieved from the DB. And then the risk level, risk matrix and qualitative analysis will be displayed in the output interface.



- For example, the front-end system RCIC and support system COND are failed in the same time.
- As we know,
  - the Baseline for Plant "A"
    - CDF is 1.36E-5
    - LERF is 6.48E-7
  - the ten-time-baseline
    - CDF is 1.36E-4
    - LERF is 6.48E-6
  - the three-time-baseline
    - CDF is 4.08E-5,
    - LERF is 1.94E-6.
- Since the risk value of CDF and LERF are greater than the ten times of baseline, and will be all in the yellow region.

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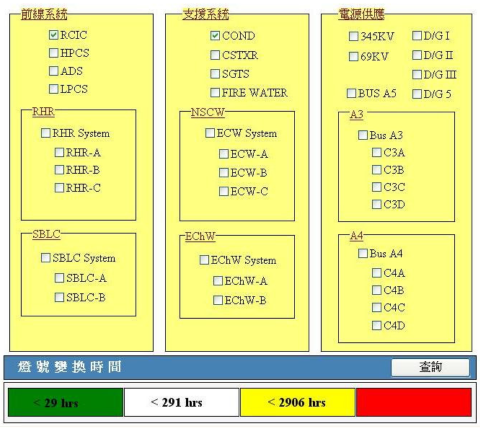
The Output Interface – Risk Matrix

失效系統1	失效系統2	失效系統3	CDF	LERF
RCIC	COND	C4D	4.47E-03	9.61E-06
RCIC	COND	C4C	3.39E-03	1.16E-05
RCIC	COND	C3C	2.27E-03	1.12E-05
RCIC	COND	RHR-A	2.16E-03	1.08E-05
RCIC	COND	C3D	2.13E-03	8.64E-06
RCIC	COND	CSTXR	1.87E-03	8.24E-06
RCIC	COND	ECW-B	1.44E-03	1.12E-05
RCIC	COND	C4B	1.42E-03	1.12E-05
RCIC	COND	ECW-A	1.32E-03	1.12E-05
RCIC	COND	C3B	1.31E-03	1.13E-05
RCIC	COND	ECHW-B	1.14E-03	8.88E-06

The 3-D Risk Matrix lists the risk value of the third system failure given RCIC and COND has already failed. It helps users to handle the risk condition and establish the compensatory measures or emergency plans as needed.

The Output Interface - The Qualitative Analysis

The qualitative analysis informs operator the remaining hours before reaching to the potentially more significant risk level in accordance with the specific failure combination.



- In this case, the RCIC and COND are failed, showing that the operators have
  - 29 hours to do the proper action before entering the white light,
  - 291 hours reaching to yellow,
  - 2906 hours to red.
- It provides operators to oversee maintenance risk impact and evaluation of risk due to emergent issues so that the risk levels of scheduling remain acceptable.

Conclusion

- Generally speaking, the implementation of the two-dimensional risk matrix is a time-consuming work. It needs a large amount of PRA engineers and man-hours to involve the task. INER, using the quick solver engine - INERISKEN, makes it possible to generate three-dimensional risk matrix. It really shortens the execution time and reduces the developing cost.

- The 3-D Risk Matrix is a simple and practical approach which is embedded in the MIRU for the qualitative analysis. It is now used in the plants and will play an important role to enhance the MR risk management during the on-line maintenance in Taiwan.

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Thanks for your attention

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ISO 9001 BSMI

PRA/INER

I -B-4

국가 미래 에너지를 책임지는 연구원

Some Approaches for Quantification of Important Factors in PSA for Digital I&C Systems

Presented by:

Man Cheol KIM (金滿哲) Ph.D.

Integrated Safety Assessment Division

Korea Atomic Energy Research Institute

KAERI Korea Atomic Energy Research Institute

Outline

1. Introduction
2. Quantification of Software Reliability
3. Quantification of Fault Coverage
4. Quantification of Human Reliability
5. Summary

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Introduction

- Important parameters in digital I&C PSA
  - Software reliability
    - Software failure = common cause failure
  - Fault coverage
    - $(U=10^{-3}) + (C=0.99) = (U=10^{-5})$
  - Common cause failures
  - Human reliability

Reference
H. G. Kang and T. Sung, "An Analysis of Safety-Critical Digital Systems for Risk-Informed Design," Reliability Engineering and System Safety, vol.78, pp.307-314 (2002)

How can we quantify the parameters ?

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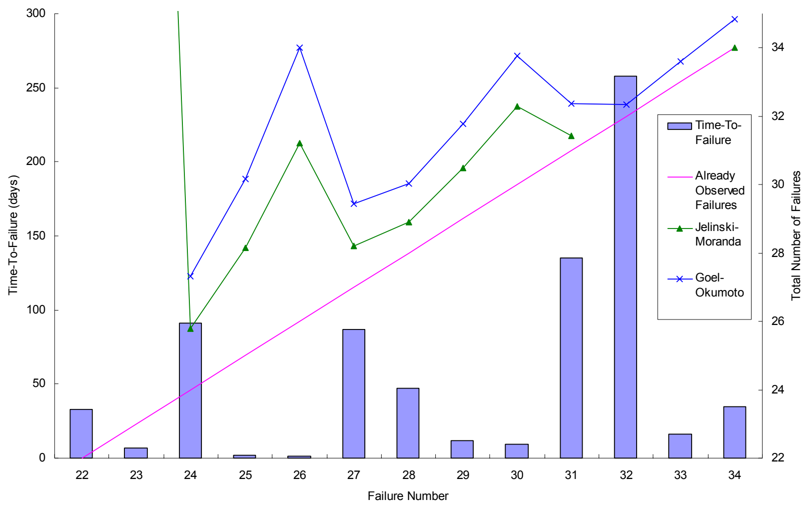
Quantification of Software Reliability

- Software reliability growth model



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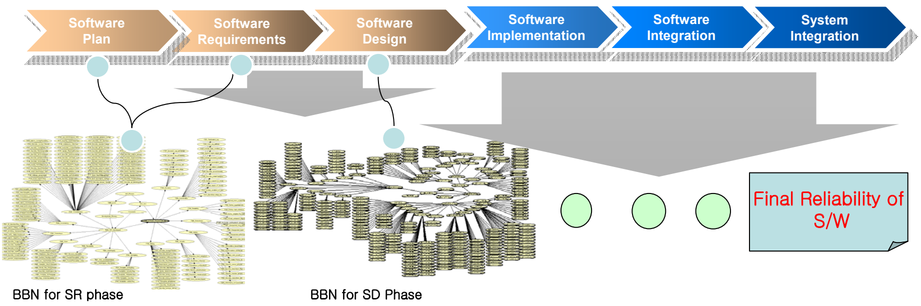
Quantification of Software Reliability

- Bayesian network-based approach
  - Development of a Bayesian network (BN) model
    - Based on verification and validation (V&V) checklists
    - Expert knowledge elicitation
    - Quantification of qualitative information



Scheme of software reliability estimation considering the software development process

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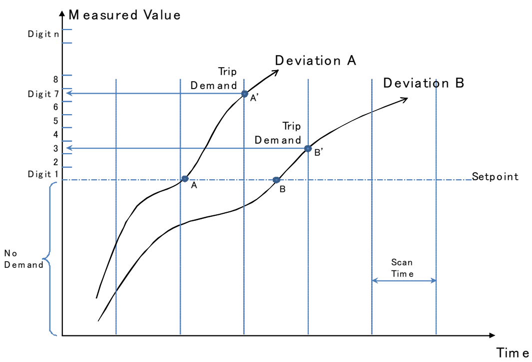
Quantification of Software Reliability

- Software test-based approach
  - Physical parameters vary continuously
  - Analog-to-Digital converter has limited resolution and scan time
  - Demand arrives when parameter goes beyond setpoint



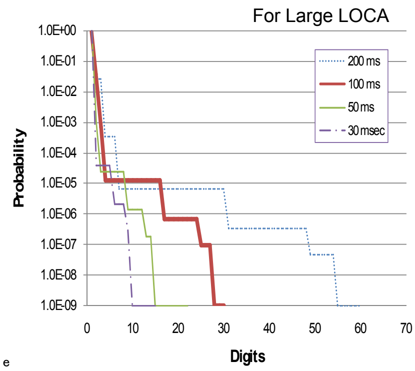


Illustration for Input-profile-based software failure probability quantification

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Quantification of Fault Coverage

- What is fault coverage ?
  - Probability that a system properly processes an occurring fault in the system
- How to estimate fault coverage ?
  - Fault injection experiments
- Various methods for fault injection experiments
  - Heavy-ion irradiation
  - Power supply disturbance
  - Pin-level fault injection
  - **Software-implemented fault injection (SWIFI)**

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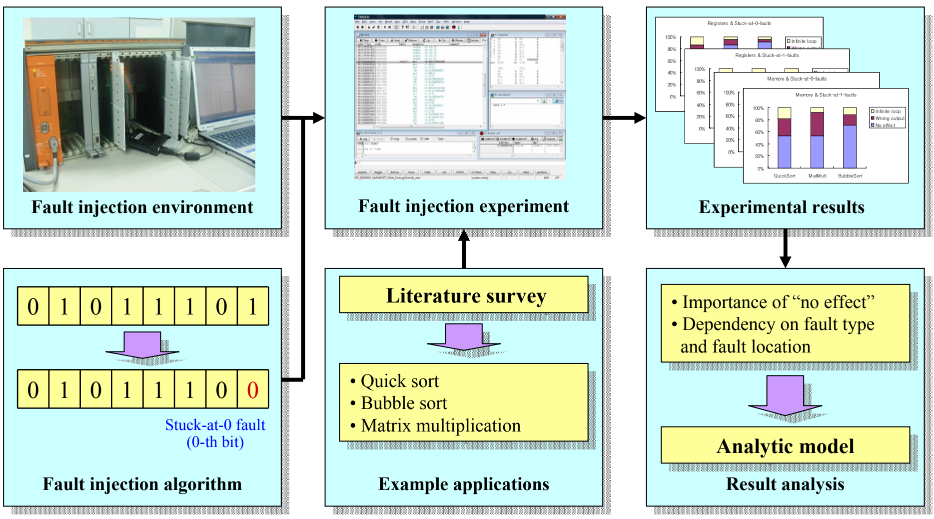
Quantification of Fault Coverage



Process of fault injection experiments for fault injection estimation of digital I&C systems

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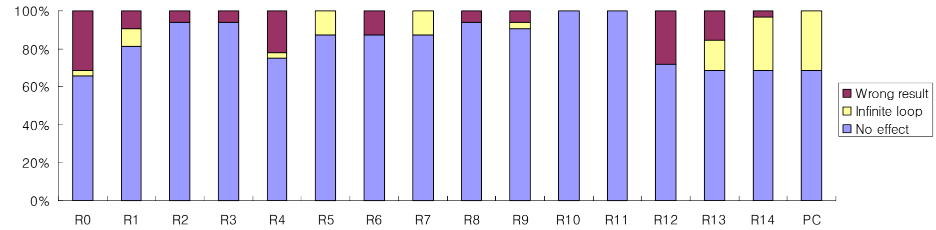
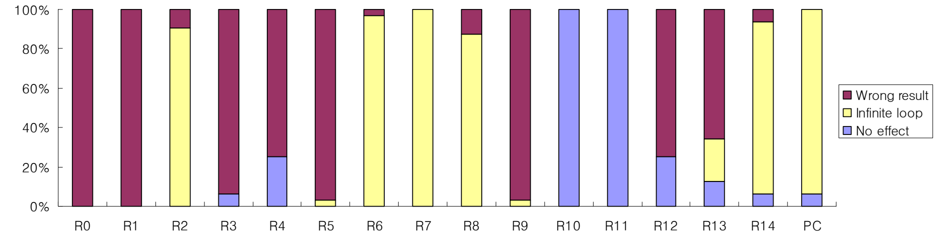
Fault Injection Experiment



Effects of the injection of stuck-at-0 faults

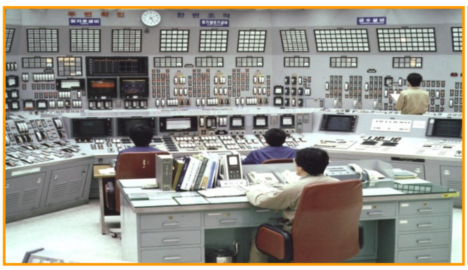


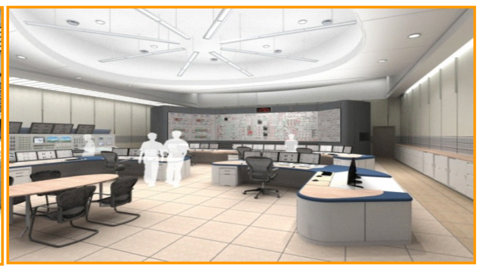
Effects of the injection of stuck-at-1 faults

Quantification of Human Reliability

- New HRA method for digital environment
  - Different operational environment
    - Analog-based conventional MCRs
    - Digital-based advanced MCRs

MCR: main control room





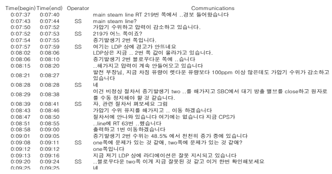
Analog-based conventional MCR Digital-based advanced MCR

Quantification of Human Reliability

– Data collection
  - Simulated accident situations
    - LOCA
    - SGTR
    - ESDE
    - LOOP
  - Audio-video recording
  - Post experiment debriefing

An operator team during an experiment

– Analysis
  - Identification of human error events
  - Behavior pattern analysis
  - Protocol and timeline analysis



Communication log

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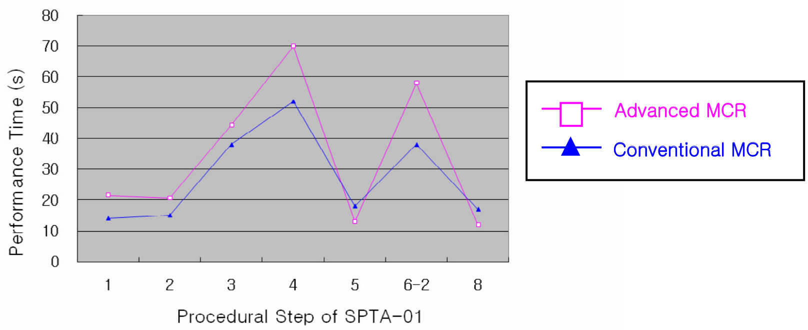
Quantification of Human Reliability

– Results
  - Faster performance in the digitalized MCR for simple tasks
  - Comparable performance in the digitalized MCR for complex tasks composed of several task steps

⇨ Possible effect of insufficient training or unfamiliarity in digitalized MCR ??



Comparison of step performance time for procedural steps

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Summary

- Quantification of important factors in digital I&C PSA
  - Software reliability
  - Fault coverage
  - Human reliability
- Future works
  - Application to real safety-critical digital systems
    - Safety-critical digital I&C system developed by KNICS
    - Digitalized MCR of APR-1400

KNICS : Korea Nuclear Instrumentation and Control Systems
APR : Advanced Power Reactor

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I -B-5

Integrated Safety Assessment Division

Research on Risk/Performance Assessment & Management

Effect Estimation of an Automatic Periodic Tests in NPP digital I&C Systems by Fault Injections

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Integrated Safety Assessment Division
Korea Atomic Energy Research Institute

KAERI KOREA ATOMIC ENERGY RESEARCH INSTITUTE

Table of Contents

KAERI KOREA ATOMIC ENERGY RESEARCH INSTITUTE

Introduction

- As digital technologies have been improved, new NPPs have adapted various kinds of digital systems including digital I&C systems for safer and more efficient operations.
- The development of a methodology for the probabilistic safety assessment (PSA) of digital I&C systems is a critical issue because conventional PSA techniques cannot adequately evaluate all features of digital systems.
- In fact, digital I&C systems have more various fault-tolerant techniques including automatic inspection functions than conventional analog I&C systems.
- Even though these fault-tolerant techniques in digital I&C systems are designed to ensure and improve the safety of systems, the effects of them have not been properly considered yet in most system PSA models.
- Therefore, it is necessary to develop an evaluation method which can describe the features of digital I&C systems.

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Introduction

- There are several issues to be solved in order to obtain accurate system safety.
  - It is important to quantify the error coverage of the specific fault-tolerant technique, because the specific fault-tolerant technique cannot detect and recover all possible faults to occur in the system.
  - It is important to exclude duplicated effect of fault-tolerant techniques since various fault-tolerant techniques such as component-level fault detection algorithm, board-level self-diagnostics, and system-level error detection mechanisms are implemented simultaneously at each level of system's hierarchy.
  - Each fault-tolerant technique has a different detection period.
  - Some fault-tolerant techniques do not make the system automatically generate fail-safe signals but just warn the abnormal situation to system's human operators, In this case, the probability for human operators to fail to detect and recover the warning should be considered.
- In this work, a method to quantify the error coverage with consideration of duplicated effects of fault-tolerant techniques in digital I&C systems is suggested using fault injection experiments.

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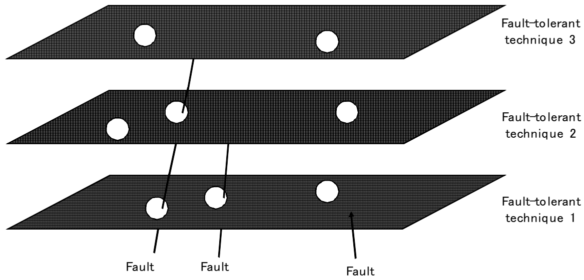
Fault Coverage Quantification of Fault-Tolerant Techniques

- The digital I&C systems adopt multiple barriers consisting of various fault-tolerant techniques to increase fault detection rate.

- Even though the fault is not detected by the fault-tolerant technique implemented in lower level of system, it could be detected by higher level fault-tolerant technique in the system.



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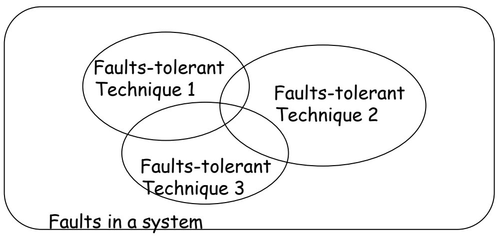
Fault Coverage Quantification of Fault-Tolerant Techniques

- Overall fault coverage of fault-tolerant techniques implemented in system is not the simple summation of fault coverage of each fault-tolerant technique, but union set of faults which can be detected by each fault-tolerant technique.



- In this work, the overall fault coverage was identified based on the following three steps.
  - All the possible fault are defined using FMEA (Failure Mode and Effects Analysis).
  - An input table for fault injection experiments is constructed.
  - Fault injection experiments are performed using the input table and the overall fault coverage is obtained based on the results.

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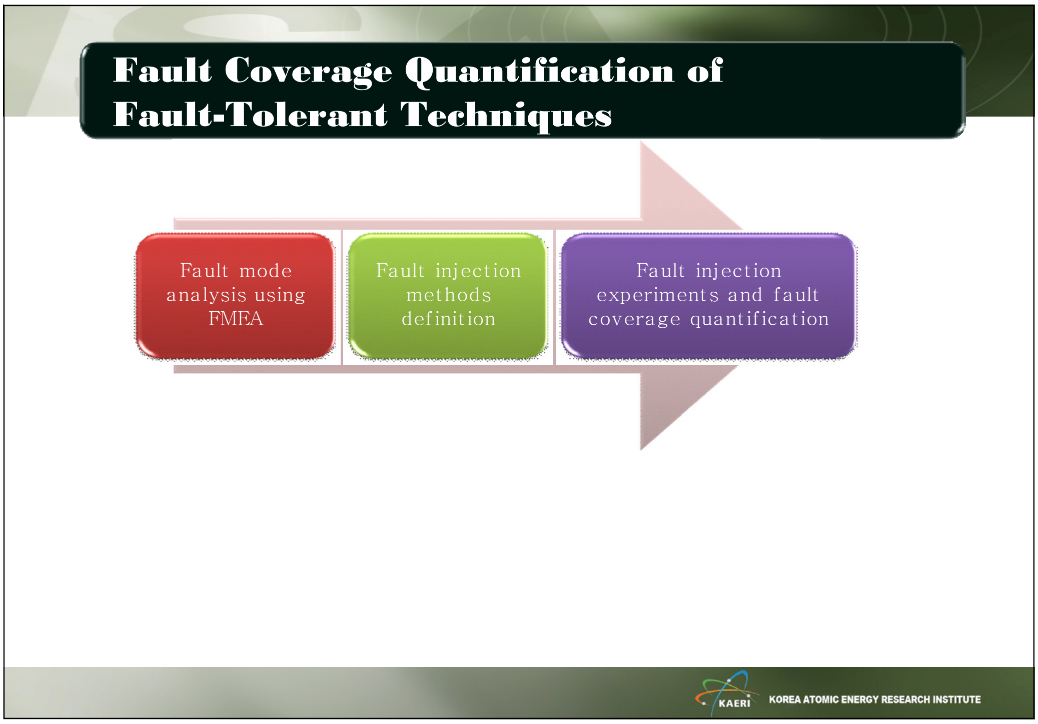
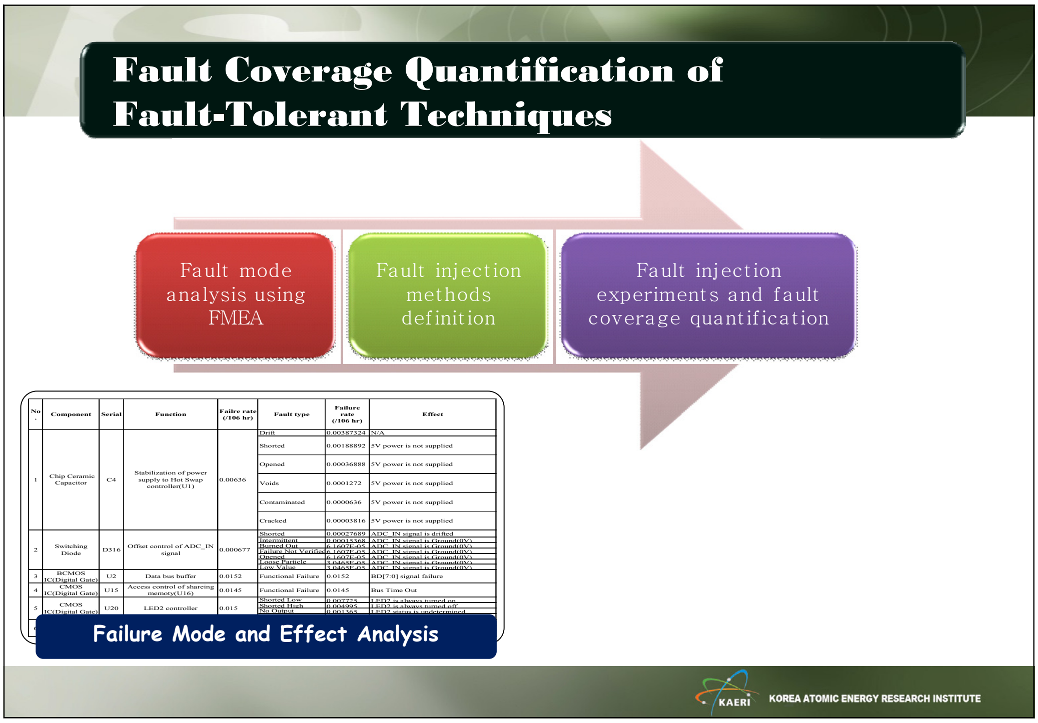
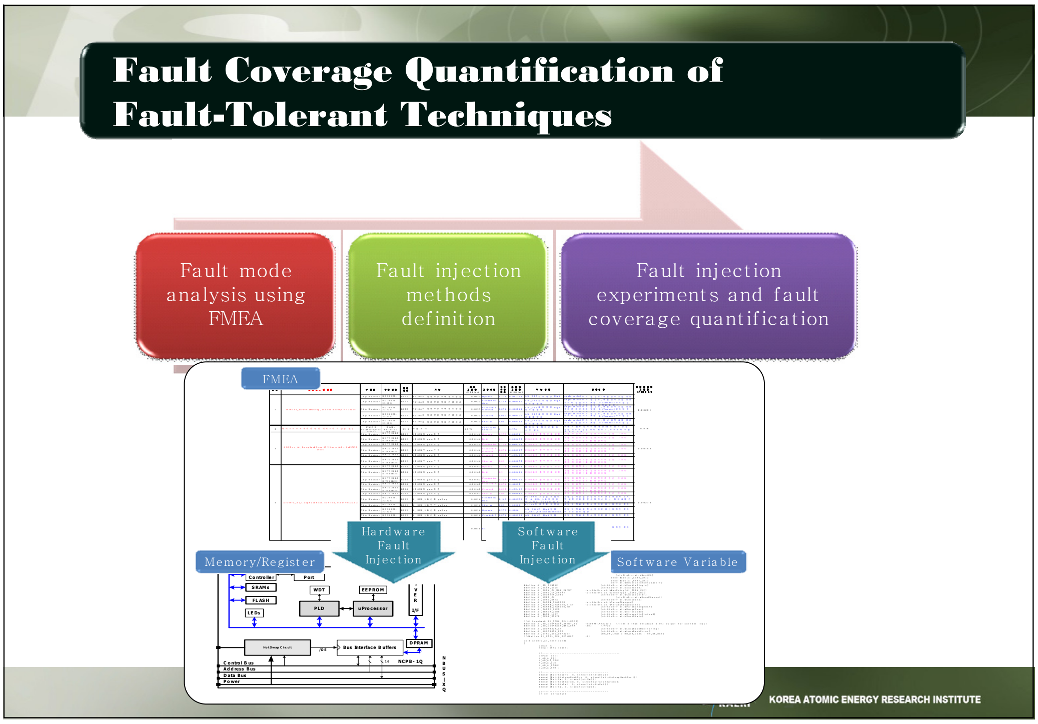




No.	Component	Serial	Function	Failre rate (/106 hr)	Fault type	Failure rate (/106 hr)	Effect
1	Chip Ceramic Capacitor	C4	Stabilization of power supply to Hot Swap controller(U1)	0.00636	Drift	0.00387324	N/A
					Shorted	0.00188892	5V power is not supplied
					Opened	0.00036888	5V power is not supplied
					Voids	0.0001272	5V power is not supplied
					Contaminated	0.0000636	5V power is not supplied
					Cracked	0.00003816	5V power is not supplied
2	Switching Diode	D316	Offset control of ADC_IN signal	0.000677	Shorted	0.00027689	ADC_IN signal is drifted
					Intermittent	0.00015368	ADC_IN signal is Ground(0V)
					Burned Out	6.1607E-05	ADC_IN signal is Ground(0V)
					Failure Not Verified	6.1607E-05	ADC_IN signal is Ground(0V)
					Opened	6.1607E-05	ADC_IN signal is Ground(0V)
					Loose Particle	3.0465E-05	ADC_IN signal is Ground(0V)
					Low Value	3.0465E-05	ADC_IN signal is Ground(0V)
3	BCMOS IC(Digital Gate)	U2	Data bus buffer	0.0152	Functional Failure	0.0152	BD[7:0] signal failure
4	CMOS IC(Digital Gate)	U15	Access control of shareing memoty(U16)	0.0145	Functional Failure	0.0145	Bus Time Out
5	CMOS IC(Digital Gate)	U20	LED2 controller	0.015	Shorted Low	0.007725	LED2 is always turned on
					Shorted High	0.004995	LED2 is always turned off
					No Output	0.001365	LED2 status is undetermined


Fault Coverage Quantification of Fault-Tolerant Techniques
Fault mode analysis using FMEA
Fault injection methods definition
Fault injection experiments and fault coverage quantification
FMEA
Hardware Fault Injection
Software Fault Injection
Memory/Register
Software Variable
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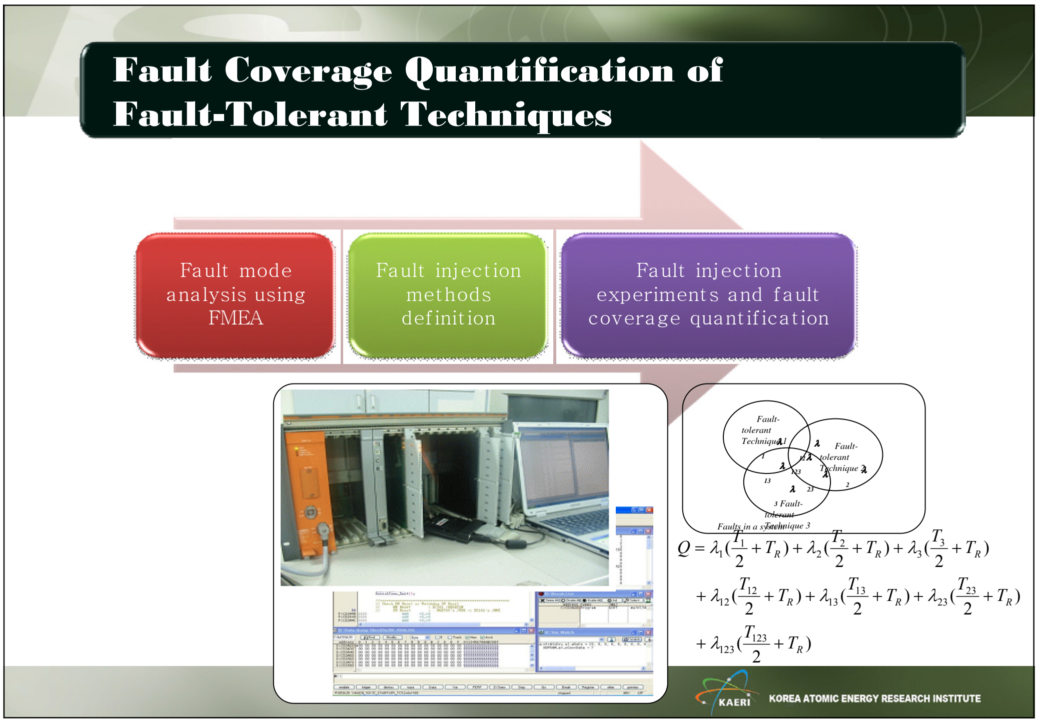
Fault Coverage Quantification of Fault-Tolerant Techniques
Fault mode analysis using FMEA
Fault injection methods definition
Fault injection experiments and fault coverage quantification
Fault-tolerant Technique 1
Fault-tolerant Technique 2
Fault-tolerant Technique 3
Faults in a system
$Q = \lambda_1(\frac{T_1}{2}+T_R)+\lambda_2(\frac{T_2}{2}+T_R)+\lambda_3(\frac{T_3}{2}+T_R)$
$+\lambda_{12}(\frac{T_{12}}{2}+T_R)+\lambda_{13}(\frac{T_{13}}{2}+T_R)+\lambda_{23}(\frac{T_{23}}{2}+T_R)$
$+\lambda_{123}(\frac{T_{123}}{2}+T_R)$
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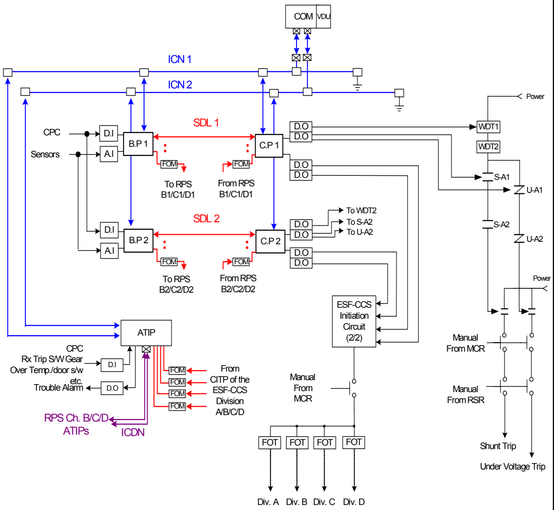
Application

- IDiPS (Integrated Digital Protection System) RPS
  - Four independent channels consist of bi-stable processors (BPs), coincidence processors (CPs), automatic test and interface processors (ATIPs), cabinet operator modules (COMs), and other analog hardware components.
  - The IDiPS RPS has been developed with the 2-out-of-4 redundant architecture, and every channel is implemented with the same architecture.
  - The ATIP module monitors the operation status of the RPS, and conducts the automatic periodic test to ensure a reliable operation of the BP and the CP module in the same channel.



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Application

Test Type	Test Kind	Function
Passive Testing	Board-level self-diagnostics	- HW self-diagnostics - OS self-diagnostics - Support mean of surveillance Test
	System-level on-line status diagnostics	- Status Comparison - Processor Integrity Monitoring - Support mean of surveillance Test
Active Testing	Automatic periodic test	- Protection logic test - I/O HW test - A mean of surveillance test
	Manual test	- I/O test - Protection path test - Protection logic test - Initiated circuits test - A mean of surveillance test

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Application

- For the first step, the target module was analyzed using the FMEA.
  - From the FMEA, totally 1788 possible failures were found for 344 components in the module.
  - The failures of the FMEA are categorized into safe failures and dangerous failures. 1099 safe failures and 689 dangerous failures were found from the FMEA.

No.	Component	Serial	Function	Failre rate (/106 hr)	Fault type	Failure rate (/106 hr)	Effect
1	Chip Ceramic Capacitor	C4	Stabilization of power supply to Hot Swap controller(U1)	0.00636	Drift	0.00387324	N/A
					Shorted	0.00188892	5V power is not supplied
					Opened	0.00036888	5V power is not supplied
					Voids	0.0001272	5V power is not supplied
					Contaminated	0.0000636	5V power is not supplied
					Cracked	0.00003816	5V power is not supplied
2	Switching Diode	D316	Offset control of ADC_IN signal	0.000677	Shorted	0.00027689	ADC_IN signal is drifted
					Intermittent	0.00015368	ADC_IN signal is Ground(0V)
					Burned Out	6.1607E-05	ADC_IN signal is Ground(0V)
					Failure Not Verified	6.1607E-05	ADC_IN signal is Ground(0V)
					Opened	6.1607E-05	ADC_IN signal is Ground(0V)
					Loose Particle	3.0465E-05	ADC_IN signal is Ground(0V)
					Low Value	3.0465E-05	ADC_IN signal is Ground(0V)
3	BCMOS IC(Digital Gate)	U2	Data bus buffer	0.0152	Functional Failure	0.0152	BD[7:0] signal failure
4	CMOS IC(Digital Gate)	U15	Access control of shareing memoty(U16)	0.0145	Functional Failure	0.0145	Bus Time Out
5	CMOS IC(Digital Gate)	U20	LED2 controller	0.015	Shorted Low	0.007725	LED2 is always turned on
					Shorted High	0.004995	LED2 is always turned off
					No Output	0.001365	LED2 status is undetermined
					Opened	0.000915	LED2 is always turned off
6	DIP Switch	SW4	Selection of operation mode	0.000308	Damaged	0.00012844	VI_CH8~VI_CH1 are stuck at High
					Opened	0.00010256	VI_CH8~VI_CH1 are stuck at High
					Mechanical Failure	0.000077	VI_CH8~VI_CH1 are stuck at High

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Application

- The dangerous failures could be categorized into six types.
  - The target module works abnormally or generates incorrect signals by these dangerous failures.
  - Some failures such as LED failures, however, can be ignorable because they do not have any effects on system performance. There are 52 ignorable failures in the FMEA.
  - In the experiment, we considered only dangerous failures except the ignorable failures.

Failure types	Examples
Spurious error signal	Generation of a loopback error signal Generation of a set-point error signal Etc.
Function halt	Halt of a function Etc.
Transmission error	Transmission error of an input signal to a processor Etc.
Stuck value	Stuck at high of an input signal Stuck at low of an input signal Stuck at current value of an input signal Etc.
Wrong value	Drift of an channel input signal Undetermined value of an channel input signal Etc.
Ignorable error	Wrong LED signal Etc.

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Application

- In this work, fault injection experiments were performed through simulated fault injection using TRACE32.
  - Faults could be injected only to memory and register in the experiment environment.
  - Two kinds of fault injections were used; fault injections using hardware faults (memory and register faults) and software variables.
    - A fault which sticks a variable assigned at fixed address could be correspond to a memory or register fault representing its address.
    - The other faults which are difficult to match to a fault on memory or register of specified address are mapped on corresponding software variables and the values of the variables are modified when specific conditions are satisfied.

Part	Function	Fault Mode	Effects	Fault Mapping
Chip Ceramic Capacitor	DC 5V over-voltage attenuation	Shorted	It cannot supply power for AD converter and photo coupler because of over-current protection circuit	The value of variable, wData0, in AIODrv_AiShiftIn is stuck to 0xFFFF
Switching Diode	CO+CH0 over-voltage protection	Shorted	The voltage of CO+CH0 is stuck to +15V or -15V	The value of variable, wAD, in AIODrv_Ai_NormalScan(709 line) is stuck to 0xFFFF when the value of variable, AI_CONV_CH, is 0x0000
Quartz Crystal/Crystal	Provide microcontroller clock	No Output	Microcontroller halts	The value of global variable, XDPRAM.wMem, is stuck to the latest updated value
Chip Resistor	Supply power for DCDC1	Opened	It cannot supply power for analog circuits(D5V, A5V, A-5V, +15V, -15V, +15V2) in module	The value of variable, wData0, in AIODrv_AiShiftIn is stuck to 0xFFFF

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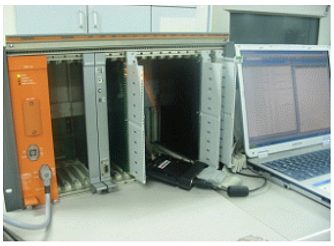
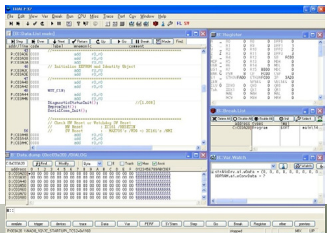
Application

- Based on the input table, the fault injection experiments were performed using the constructed experiment environment .





Component	Serial	Fault type	Failure rate (/106 hr)	Self-diagnostics	Automatic periodic test
BCMOS IC(Digital Gate)	U2	Functional Failure	0.0152	O	O
CMOS IC(Linear Gate)	U3	Functional Failure	0.0739	O	O
CMOS IC(Linear Gate)	U5	Functional Failure	0.0739	O	O
Varistor Diode	ZD1	Shorted	0.0021594	O	O
Zener Diode	ZD2	Shorted	0.0021594	O	O
CMOS IC(Digital Gate)	U15	Functional Failure	0.0145	O	O
FET/N-Channel	Q1	Electrical Overstress	0.0021879	O	O
FET/N-Channel	Q1	Opened	0.0019305	O	O
FET/N-Channel	Q1	Voids	0.0011583	O	O
Chip Ceramic Capacitor	C4	Shorted	0.00188892	O	O
Chip Ceramic Capacitor	C4	Opened	0.00036888	O	O
Chip Ceramic Capacitor	C4	Voids	0.0001272	O	O
Chip Ceramic Capacitor	C4	Contaminated	0.0000636	O	O
Chip Ceramic Capacitor	C4	Cracked	0.00003816	O	O
Tantalum Capacitor	C108	Shorted	0.0187902	O	O
Network Resistor	RA1	Contamination	0.00004182	O	O
Network Resistor	RA1	Shorted	0.000037995	O	O
Quartz Crystal/Crystal	X1	No Output	0.0264	O	O
Quartz Crystal/Crystal	X1	Degraded	0.0088	O	O
Quartz Crystal/Crystal	X1	Fractured/Cracked	0.0088	O	O
Network Resistor	RA12	Opened	0.000168394	O	X
Chip Resistor	R115	Contamination	0.0002384	O	X
Chip Resistor	R115	Shorted	0.00008	O	X
Chip Resistor	R115	Opened	0.0006	O	X
Chip Resistor	R115	Cracked/Fractured	0.0001184	O	X
Chip Ceramic Capacitor	C300	Shorted	0.00165726	X	X
Chip Resistor	R1	Opened	0.00042375	X	X
Chip Resistor	R1	Cracked/Fractured	0.00008362	X	X
Chip Resistor	R300	Contamination	0.00021456	X	X
Chip Resistor	R300	Cracked/Fractured	0.00010656	X	X
Chip Resistor	R301	Shorted	0.000072	X	X
Chip Resistor	R301	Opened	0.00054	X	X

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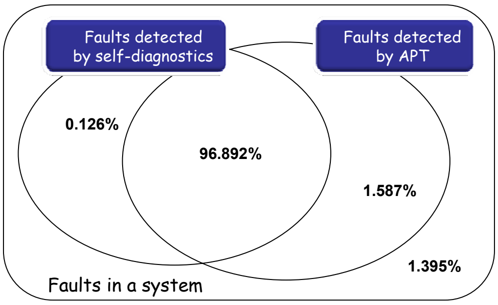
Application

- Based on the result we can identify independent fault coverage for each fault-tolerant technique and overall fault coverage.

	Fault detection rate (%)
Detected by self-diagnostics	97.018
Detected by automatic periodic test	98.479
Detected by both functions	96.892
Undetected by both functions	1.395
Fault detection rate using both functions	98.605



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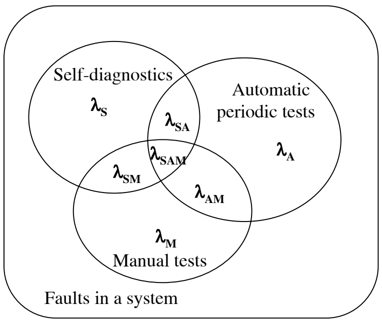
Application

- If we assume that the system checks its availability through a self-diagnostics, automatic periodic tests and manual tests and that the manual test detects all the faults which are not detected by other fault-tolerant techniques, then the unavailability of the system could be calculated using the following equation.

$$Q = \lambda_S(\frac{T_S}{2}+T_R)+\lambda_A(\frac{T_A}{2}+T_R)+\lambda_M(\frac{T_M}{2}+T_R)$$
$$+\lambda_{SA}(\frac{T_{SA}}{2}+T_R)+\lambda_{SM}(\frac{T_{SM}}{2}+T_R)+\lambda_{AM}(\frac{T_{AM}}{2}+T_R)+\lambda_{SAM}(\frac{T_{SAM}}{2}+T_R)$$



T_S = Time interval of a self-diagnostics (50 msec)
T_A = Time interval of an automatic periodic test (8 hours)
T_M = Time interval of manual periodic tests (720 hours)
T_R = Time required for maintenance (24 hours)

$$Q = \lambda_{S'}T_R+\lambda_{A'}(\frac{T_A}{2}+T_R)+\lambda_M(\frac{T_M}{2}+T_R)$$

$\lambda = 4.559\times10^{-6}\,/\,hour$
$\lambda_S = 4.422\times10^{-6}\,/\,hour$
$\lambda_A = 0.072\times10^{-6}\,/\,hour$

The unavailability of the system: 1.859×10^{-3}

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Discussions

- The evaluation method in this work is performed based the FMEA and only the faults in the FMEA are simulated. Therefore, accurate and detailed FMEA is necessary. Inaccurate or wrong information of the FMEA may cause inadequate or incorrect results.
- For constructing an input table for the experiment, all the faults are matched with hardware or software fault injection. In the application of this work, the relations were defined by engineering judgments or experts. If a simulation tool which can simulate more various types of hardware faults is used, we can obtain more reliable results by minimizing the mapped hardware or software faults. Moreover, other methods for how to match faults to fault injections need to be considered.
- Since this work is focused on the issue for duplicated effect reflections, the method needs to be extended considering other issues. For example, even though a fault-tolerant technique detects a fault successfully and provide the fault information to operators, the system could stay in abnormal status because of human error. Therefore, extension of the proposed method, including human reliability analysis (HRA) and so on, should be performed for further works.

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Conclusions

- Even though new NPPs have adapted digital I&C systems including various fault-tolerant techniques, the effects of them have not been properly considered yet in most PSA models.
- Among the issues to be solved in order to obtain accurate reliability of digital I&C systems, this work focused on the issue to exclude duplicated effect reflection when various fault-tolerant techniques are implemented simultaneously.
- In order to exclude a duplicated effect consideration, exact definitions of relations between faults and fault-tolerant techniques is required.
- In this work, the relation between faults and fault-tolerant techniques are defined using fault injection experiments.
- As an application, independent fault coverage of each fault-tolerant technique in a module and overall fault coverage were identified using the proposed methods and the experiment showed reasonable results.

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I -B-6

AN APPROACH FOR ACCIDENT EVENT SEQUENCE ANALYSIS BY DIFFERENT PHASES IN NUCLEAR POWER PLANT

Yu Yu, Tong Jiejuan, Liu Tao, Zhao Jun and Zhang Aling
Institute of Nuclear and New Energy Technology
Tsinghua University
Beijing 100084, China

CONTENT

- Introduction
- Accident sequence
- Model of the sequence
- Result
- Conclusion

INTRODUCTION

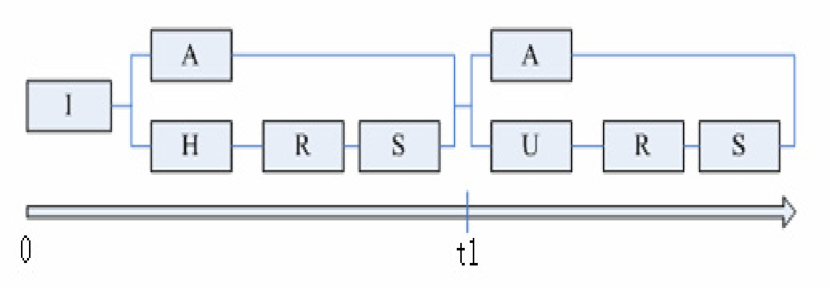
- PSA is a systematic engineering evaluating technology
- Mission in different phases being executed by different systems or function modules is a common phenomena in engineering project
- ET/FT are static analysis methods
- Idea of the method

ACCIDENT SEQUENCE

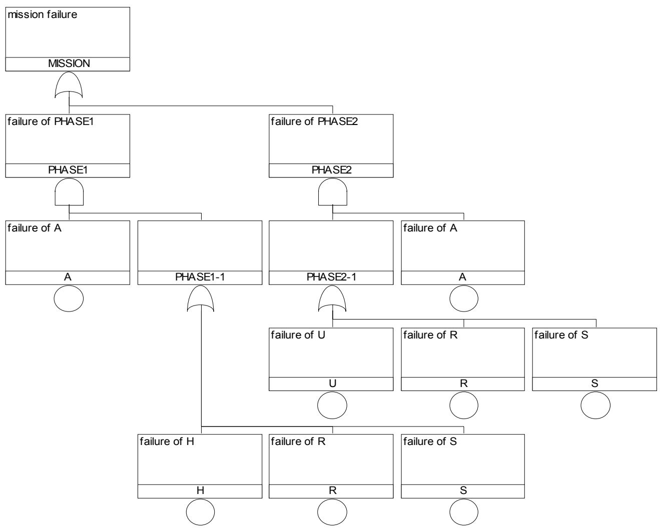
- Initiating event : loss of main feed water in PWR
- Three mitigating ways:
- ✓ Auxiliary feed water system operated
- ✓ feed-bleed is executed according to procedure H2 successfully
- ✓ feed-bleed is executed according to procedure U1 successfully

MODEL OF THE SEQUENCE

- reliability diagram for accident sequence of loss of main feed water



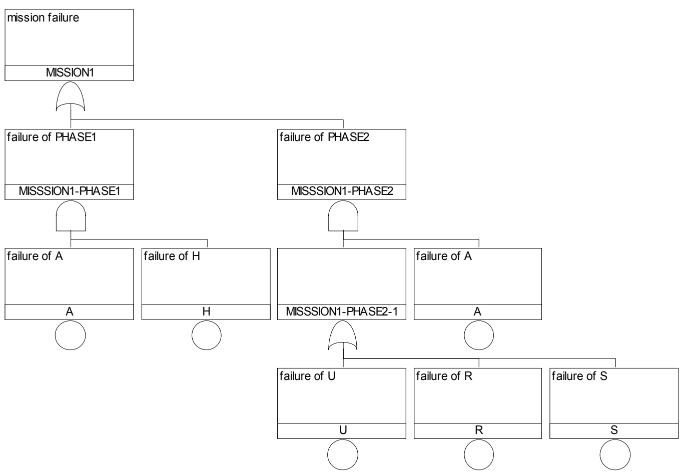
MODEL OF THE SEQUENCE(CONT.1)



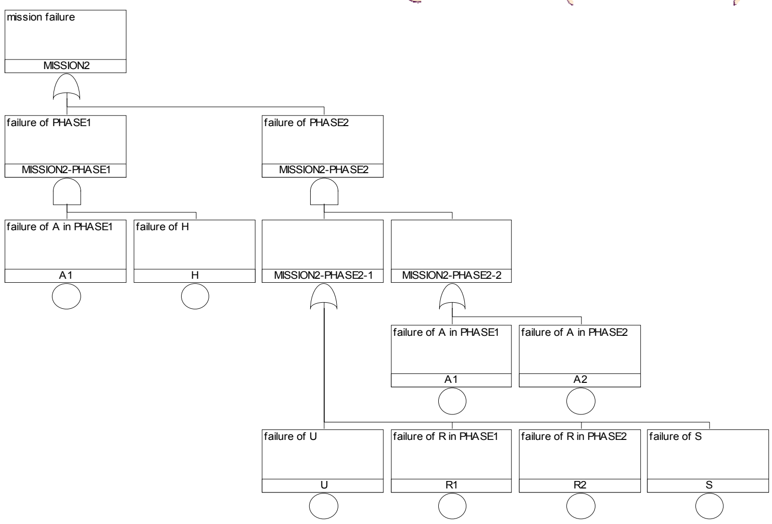
MODEL OF THE SEQUENCE(CONT.2)

- Cut-set reduce
- Basic event transform
- equivalent single-phase system
- Calculate the cut-set of the new fault tree

MODEL OF THE SEQUENCE(CONT.3)



MODEL OF THE SEQUENCE(CONT.4)



RESULT

- CDF= 8.58*10^{-6} /(reactor*year),
  the cut-sets are: {A2,U}、{A1,U}、{A1,H}、{A2,R2}、{A1,R2}、{A2,R1}、{A1,R1}、{A2,S}and{A1,S}
- Delete the illogical cut-set {A1,U} ,
  CDF= 2.57*10^{-6}/(reactor*year)
- {A2,R2}+{A2,R1}+{A1,R2}+{A1,R1}={A,R}
  CDF = 5.8*10^{-6}/(reactor*year)

CONCLUSION

- The method of evaluating the accident sequence by different phases can be used to handle the dynamic problems.
- In following condition, correction should be made according to the engineering system:
  - ✓ The logical model is changed according to the condition, e.g. example in this paper.
  - ✓ The front systems share the common support systems.

THANKS

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Session I-C

Severe Accident Management & Level 3 PSA (I)

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Session I-C Summary

Chair: Toshimitsu HOMMA (JAEA), Kwang-Il AHN (KAERI)

I-C-1. Dong-ha KIM (KINS-Korea): Development of Simplified Risk Measure Based on Dose (SiRD)
The paper proposed a simplified risk measure based on a combination of frequency and dose as an alternative to the existing probabilistic performance goals such as CDF and LERF. This measure can be obtained mainly from Level 2 PSA without performing the complicated Level 3 PSA. It considers the whole-body and thyroid doses as consequences. It was suggested from the participant that other exposure pathways might be important from the release of radionuclides for the core damage accidents.

I-C-2. Kyungmin KANG (KINS-Korea): The Development of a Relationship Framework between LERF and Level-3 PSA
This study also developed a method for identifying correlations between LERF and individual early fatality without a detailed Level 3 PSA. The results of this study may contribute to defining LERF and finding a measure for risk-informed regulations and risk-informed decision-making.

I-C-3. Toshimitsu HOMMA (JAEA-Japan): Risk-informed Evaluation of Off-site Response Planning for Nuclear Emergencies
The paper presented the methodology and the results of the risk-informed evaluation of off-site emergency planning using the Level 3 PSA code, OSCAAR developed at JAEA. The analysis has been made to evaluate the effectiveness of protective action strategy involving a combination of evacuation, sheltering and administration of stable iodine. The results of this study will be expected to form a basis for the future technical guidance for protective action strategy.

I-C-4. Shogo TAKAHARA (JAEA-Japan): Optimization of Relocation Decisions using the Method of Probabilistic Accident Consequence Assessment
The paper described the application of a probabilistic accident consequence assessment model to the planning of relocation. Calculations of the consequence have been made of a postulated accident with source terms derived from a generic level 2 PSA. The results provided the insights for the development optimum dose criteria for introducing and terminating relocation.

I-C-5. Jongtae JEONG (KAERI-Korea): Development of an Off-site Risk Assessment Tool for the Risk-Informed Application
This paper presented a development plan for the offsite consequence assessment for use in the risk-informed application. In order to reduce the uncertainties in the offsite consequence assessment, the puff trajectory model for atmospheric dispersion modeling will be used to take account of the terrain effect on the dispersion of the release of radionuclides and the appropriate meteorological sampling method will be developed.

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I -C-1

10th KJPSA Workshop – May 18-20, 2009, Jeju

Development of Simplified Risk Measure based on Dose (*SiRD*)

Dong-Ha Kim*, Kwang-Il Ahn

10th Korea-Japan Joint Workshop on PSA

2009. 5. 18

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Purpose

- To propose a simple risk measure (*SiRD*) based on a combination of frequency & dose as an alternative to the existing probabilistic performance goals (CDF and LERF)

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Contents

- Current Practices
- Overview of *SiRD*
  - Whole-body/Thyroid dose calculation
  - Formulation of *SiRD*
- Application of *SiRD*
- Conclusions

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Nuclear Safety Goals

- Safety Goals
  - NRC issued a Policy Statement regarding the establishment of Safety Goals for the operation of Nuclear Power Plants which defined acceptable levels of radiological risk to the public.
  - NRC enunciated 2 qualitative safety goals, one for the individual risk and the other for societal risk.
- Quantitative Health Objectives (QHOs)
  - These qualitative safety goals were implemented by 2 Quantitative Health Objectives (QHOs) which set numerical goals for the individual risk of early fatalities and the risk of latent cancers.
  - "The overall mean frequency of a large release of radioactive material to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation"

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Current Practices in Implementing Safety Goals

- Practical Surrogates*
  - Early Fatality
  - Late Fatality
  - Frequency of a release large enough to require prolonged relocation
  - Frequency of Large Early Release (LERF)
  - Frequency of Core Melt (CDF)
  - Frequency of exceeding a given public dose
  - Frequency of release of a given quantity of radio-nuclides
  - Cumulative frequency for the release of Cs-137

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* P. Hessel, CNSC, IAEA-AECL-CNSC TM on SAA, AM and PSA, Mississuga, 2008

Overview of *SiRD* (1)

- *SiRD* stands for *Si*mplified *R*isk measure based on *D*ose [rem/yr].
- Characteristics:
  - As risk is defined as frequency * consequence, SiRD adopts dose for consequence.

$$SiRD = \sum_{i}^{sequences} \sum_{j}^{FPs} \sum_{k}^{TH/WB} dose(i,j,k) \bullet Freq(i)$$

  - Whole-body and Thyroid doses are considered.
  - Hence *SiRD* considers frequency as well as dose.
  - Can be calculated mainly from Level 2 PSA

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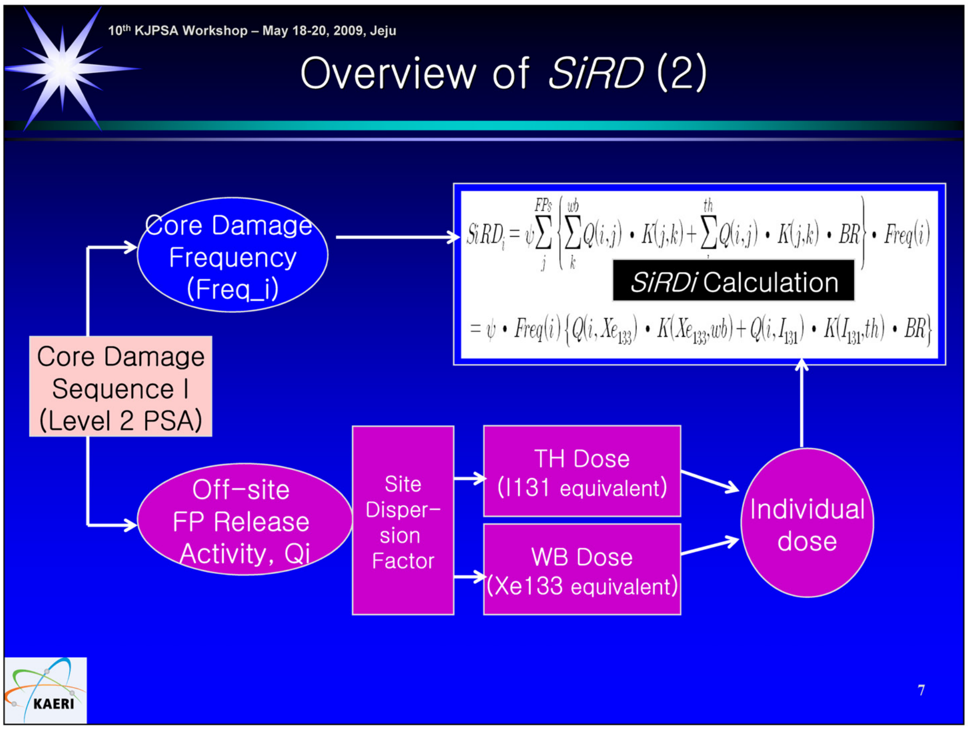

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Dose calculation (1)

- Assumptions
  - All radioactivity releases are treated as ground-level releases
  - The dose receptor is a standard man
  - No credit is taken for cloud depletion by ground deposition and radioactive decay during transportation to the exclusion area boundary (EAB)

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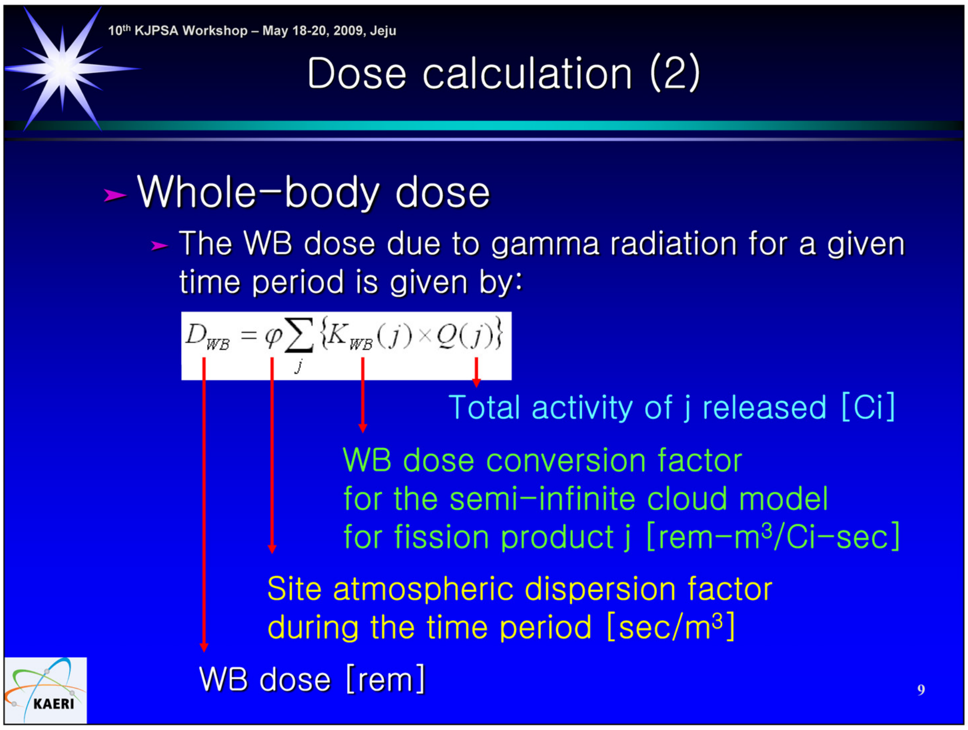
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Dose calculation (2)
Whole-body dose
The WB dose due to gamma radiation for a given time period is given by:
$D_{WB} = \varphi \sum_j \{K_{WB}(j) \times Q(j)\}$
Total activity of j released [Ci]
WB dose conversion factor for the semi-infinite cloud model for fission product j [rem-m^3/Ci-sec]
Site atmospheric dispersion factor during the time period [sec/m^3]
WB dose [rem]
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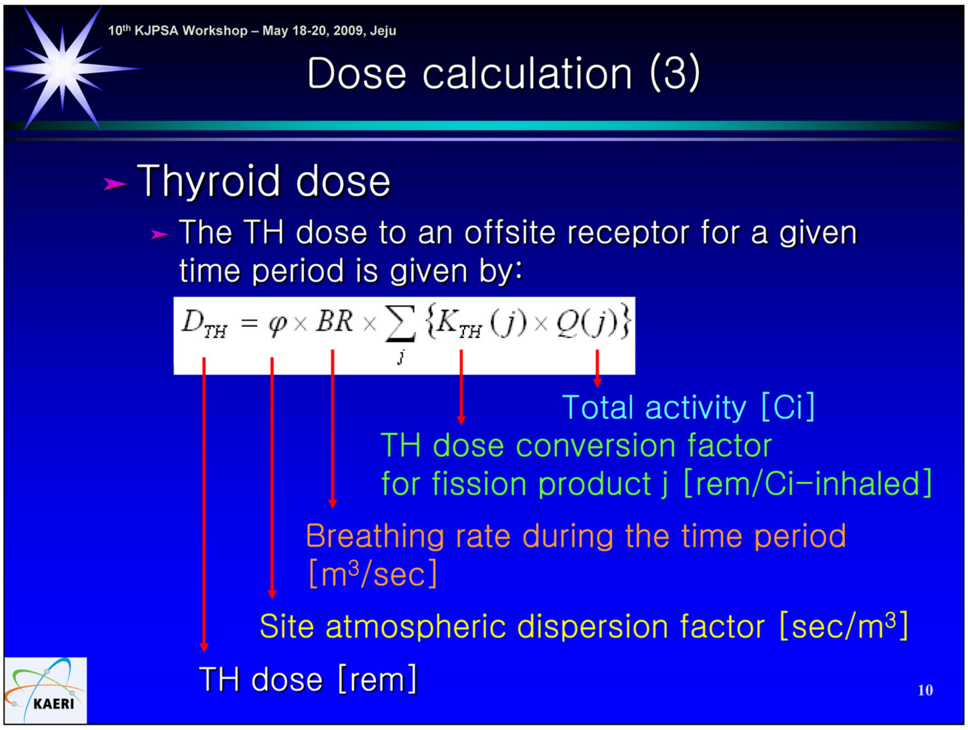
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Dose calculation (3)
Thyroid dose
The TH dose to an offsite receptor for a given time period is given by:
$D_{TH} = \varphi \times BR \times \sum_j \{K_{TH}(j) \times Q(j)\}$
Total activity [Ci]
TH dose conversion factor for fission product j [rem/Ci-inhaled]
Breathing rate during the time period [m^3/sec]
Site atmospheric dispersion factor [sec/m^3]
TH dose [rem]
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Formulation of *SiRD* (1)

- *SiRDi*, a risk index for the *i*th core damage accident, is defined as the frequency of an initiating event *i* multiplied by the dose from the released radioactive isotope *j* :

$$SiRD_i = F(i)\cdot\sum_{j}^{FPs}\sum_{k}^{WB/TH} dose(i,j,k)$$
$$= F(i)\cdot\varphi\sum_{j}^{FPs}\{Q(i,j)\cdot K_{WB}(j)+Q(i,j)\cdot K_{TH}(j)\cdot BR\}$$

Frequency of the core damage sequence I [/sec]

- *SiRD* can be expressed by a sum of *SiRDi* for the core damage sequence i

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Formulation of *SiRD* (2)

- When the early and late containment failure modes (ECF/LCF) are concerned, *SiRD* can be expressed by:

$$SiRD = SiRD_{ECF} + SiRD_{LCF}$$
$$= \sum_{i_1}^{ECF\ sequences} SiRD_{i_1} + \sum_{i_2}^{LCF\ sequences} SiRD_{i_2}$$

- At present, the WB dose from Xe_{133} and the TH dose from I_{131} are taken into account:

$$SiRD_i = F(i)\cdot\varphi\cdot\{Q(i,Xe_{133})\cdot K_{WB}(Xe_{133})$$
$$+Q(i,I_{131})\cdot K_{TH}(I_{131})\cdot BR\}$$

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Application of *SiRD* (1)

- Based on the following formulation, *SiRD* was calculated for each source term release category (STC) group at typical Korean PWRs (UCN 3&4)

$$SiRD_i = \psi \sum_{j}^{FPs} \left\{ \sum_{k}^{wb} Q(i,j) \cdot K(j,k) + \sum_{k}^{th} Q(i,j) \cdot K(j,k) \cdot BR \right\} \cdot Freq(i)$$

$$= \psi \cdot Freq(i) \{ Q(i, Xe_{133}) \cdot K(Xe_{133}, wb) + Q(i, I_{131}) \cdot K(I_{131}, th) \cdot BR \}$$

  - ψ: site dispersion factor
    (=1.963x10^{-4} [sec/m^3] at EAB (UCN 3&4))
  - K(j,k): conversion factors for fission product j
    (K(Xe_{133}, whole body)=9.96x10^{-3} [rem-m^3/Ci-sec],
    K(I_{131}, thyroid)=1.48 x10^6 [rem/Ci-inhaled])
- BR: Breathing rate (=3.47x10^{-4} [m^3/sec])

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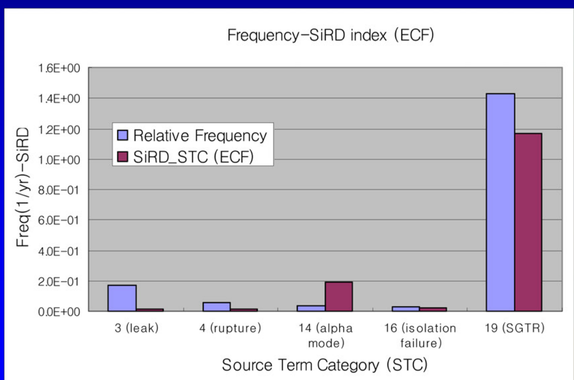
Application of *SiRD* (2)–ECF

- The SGTR sequence (STC19) has a larger frequency as well as higher risk index (1.17)
- The alpha mode of containment failure (STC14) shows the 2nd largest index in spite of its relative low frequency.



ECF STC definition

STC3: leak
STC4: rupture
STC14: alpha mode
STC16: isolation
STC19: SGTR

Fig. 1 ECF STC relative frequencies & *SiRD* indices

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Application of *SiRD* (3)–ECF

- The reason for the high risk index in STC14 comes from the larger release of I-131than in other STCs.

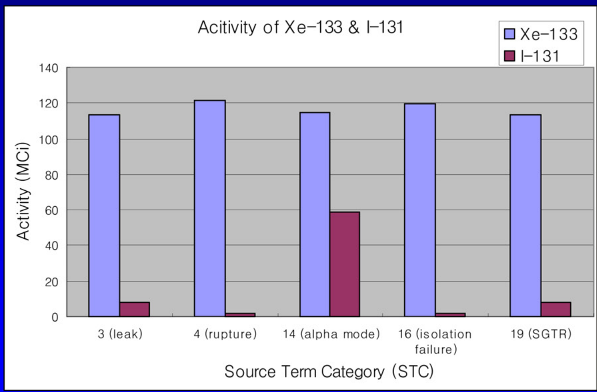

Fig. 2 ECF STC activities of Xe-133 & I-131

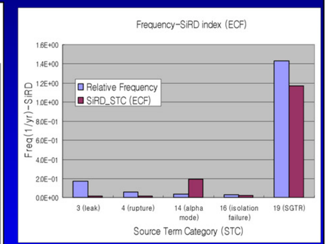

Fig. 1 ECF STC relative frequencies & *SiRD* indices

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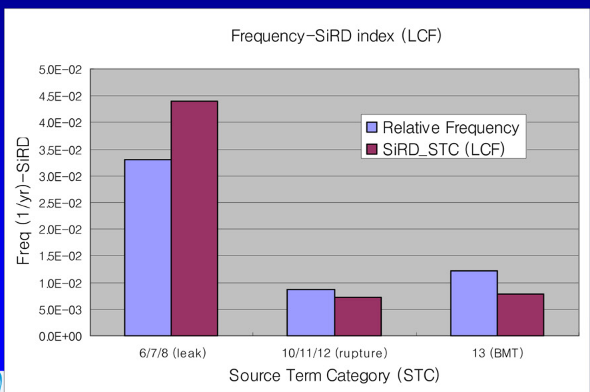
Application of *SiRD* (4)–LCF

- For the late containment failure STCs, the leak failure mode (STC6/7/8) has a larger frequency and higher SiRD.
- Comparing $SiRD_{ECF}$ (~1.37) and $SiRD_{LCF}$ (~0.06),ECF has a larger risk impact than LCF has.



LCF STC definition

STC 6/7/8 : leak
STC10/11/2: rupture
STC 18 : BMT

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Fig. 3 LCF STC relative frequencies & *SiRD* indices

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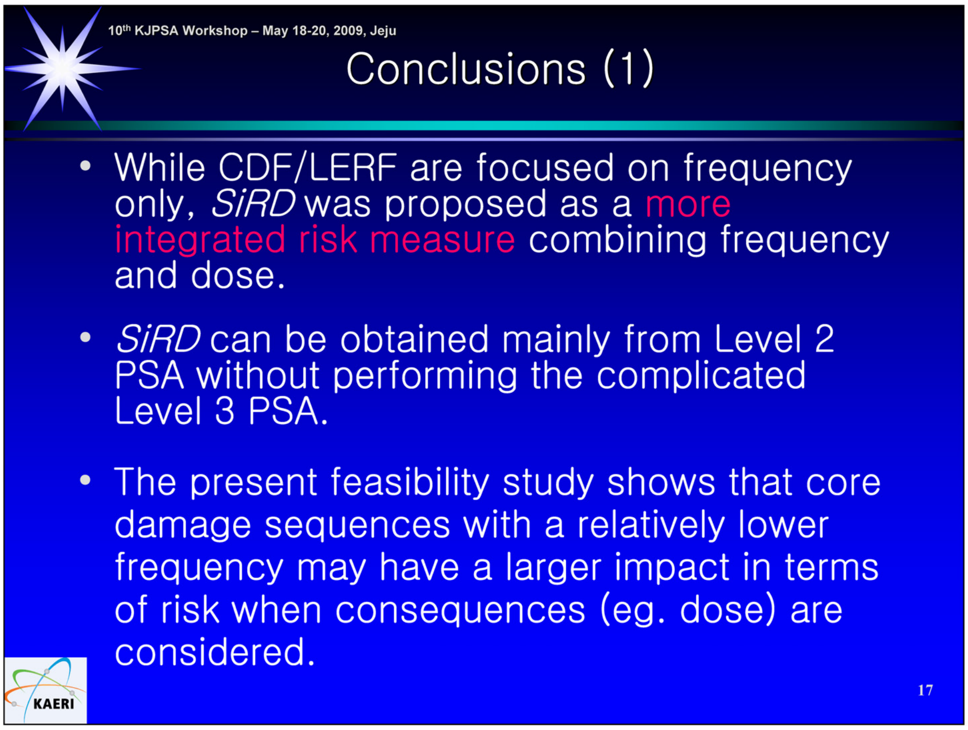
10th KJPSA Workshop – May 18-20, 2009, Jeju
Conclusions (1)
• While CDF/LERF are focused on frequency only, SiRD was proposed as a more integrated risk measure combining frequency and dose.
• SiRD can be obtained mainly from Level 2 PSA without performing the complicated Level 3 PSA.
• The present feasibility study shows that core damage sequences with a relatively lower frequency may have a larger impact in terms of risk when consequences (eg. dose) are considered.
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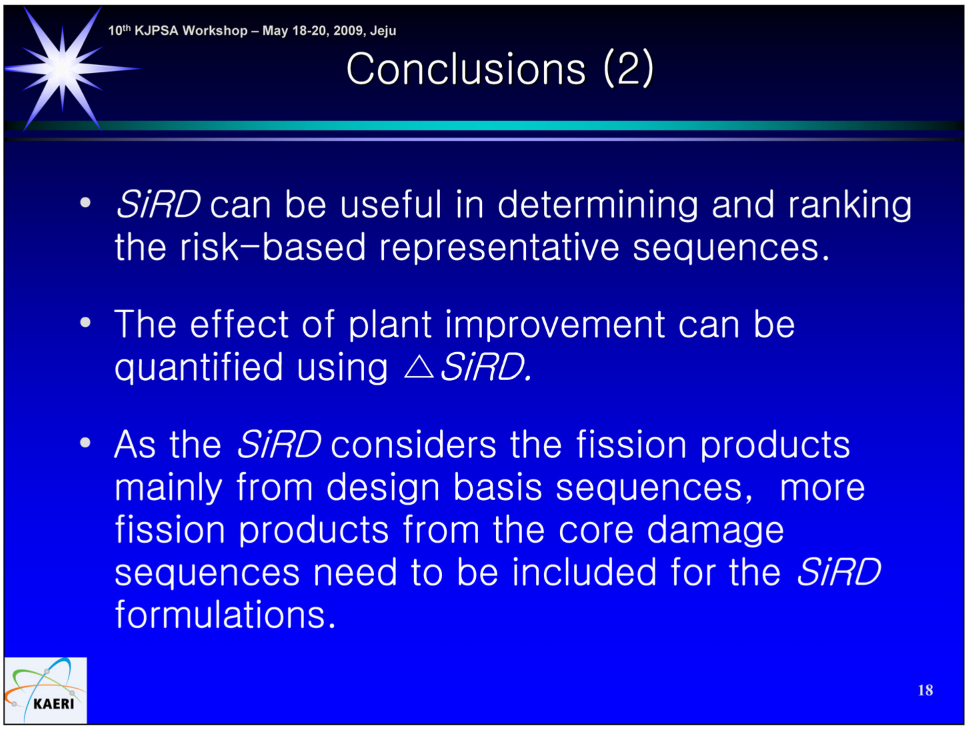
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Conclusions (2)
• SiRD can be useful in determining and ranking the risk-based representative sequences.
• The effect of plant improvement can be quantified using △SiRD.
• As the SiRD considers the fission products mainly from design basis sequences, more fission products from the core damage sequences need to be included for the SiRD formulations.
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I -C-2

The Development of a Relationship Framework between LERF and L3 PSA

2009. 5. 18
KJPSA, Jeju

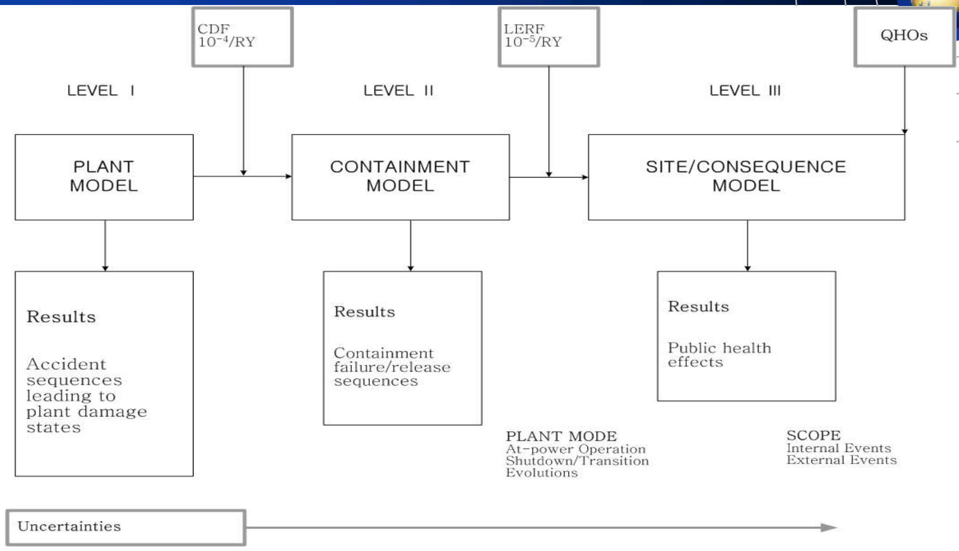
Kyungmin Kang (KINS)
Kwang-Il Ahn (KAERI)
Moosung Jae (Hanyang University)

Contents

- Introduction
- background
- Alternative/option
- Results
- Conclusions

2

PSA Model Overview and Subsidiary Objectives



- The subsidiary objectives have been formulated in terms of the frequency of core damage accidents and the frequency of large early release.
- The QHO's and subsidiary objectives have been used by the NRC staff only in the context of generic regulatory decisions.

3

Early Fatalities QHO

The risk to an average individual in the vici nity of a nuclear power plant of prompt fata lities that might result from reactor accident s should not exceed one-tenth of one perce nt (0.1 percent) of the sum of prompt fatali ty risks resulting from other accidents to which members of the U.S. population are generally exposed.

-Advantages: It provides for a more efficient, objective regulatory process. And the defining a specific minimum level of safety will promote stability and predictability in the regulatory structure.

-Disadvantages: This option increase the needed resources.(Level 3 PSA: a probabilistic consequence assessment in terms of health effects).

4

LERF(Large Early Release Frequency*)

Defined as the frequency of those accident s leading to significant, unmitigated releases from containment in a time frame prior to ef fective evacuation of the close-in population such that there is a potential for early health effects

- Advantages: This option would eliminate the need for carrying out probabilistic assessments.

- Disadvantages: There are the significant uncertainties regarding the feasibility of this approach.

* W. T. Pratt, V. Mubayi, T.L. Chu, "An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," U.S. Nuclear Regulatory Commission, NUREG/CR-6595, 2004.

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Objectives

Difficulties exist in defining "large early release frequency"

The release definition constitute s the link between the level - 2 PSA results and an indirect atte mpt to assess health effects fro m the release.

A simple and easy to use approach for providing rea sonably robust estimates f or the large early release f requency for PSA analyses lacking a detailed Level 3 offsite consequence.

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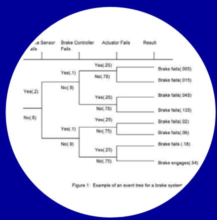
ALTERNATIVES/OPTIONS



A-1: Any release that occur because of severe accidents that would entail early containment failure (including containment isolation failure) and containment bypass conditions

A-2: Any release that would exceed specific thresholds in terms of fractional releases and timing of release

A-3: A collection of all releases that would result in one or more early fatalities offsite

- Mohsen Khatib-Rahbar., "An Approach to Definition of Large Release", PSAM7/ESREL'04 Conference, 2004.

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ALTERNATIVES 1

Any release that occur because of severe accidents that would entail early containment failure (including containment isolation failure) and containment bypass conditions

Advantages:

- Only requires Level-2 PSAs
- Not subject to interpretation

Disadvantages:

- Does not includes site impact (weather & population)
- Can not include impact of EP
- Driven largely by core damage profile (i.e., little impact from consideration of containment performance)

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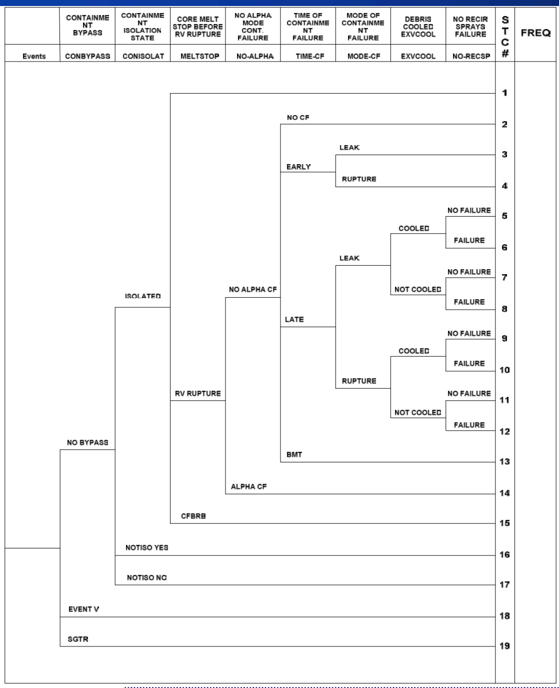
UCN3&4 source term logic diagram*



STC NO	격납건물 파손 유형	Frequency(/RY)
1	원자로용기파손전 노심용융물 냉각	1.12E-06
2	원자로용기파손 격납건물 비파손	3.111E-06
3	조기 격납건물 파손 (LEAK)	9.212E-11
4	조기 격납건물 파손 (RUPTURE)	7.382E-09
6	후기 격납건물 파손 (LEAK)	3.383E-07
7	후기 격납건물 파손 (LEAK)	1.945E-09
8	후기 격납건물 파손 (LEAK)	3.011E-07
10	후기 격납건물 파손 (LEAK)	1.70E-07
11	후기 격납건물 파손 (LEAK)	6.485E-10
12	후기 격납건물 파손 (LEAK)	1.477E-07
13	원자로공동기초판용융관통(BMT)	2.306E-07
14	알파모드 파손	8.225E-09
15	원자로용기 파손전 격납건물 파손	3.787E-07
16	격리 실패	6.194E-10
17	격리 실패	3.915E-09
18	저압경계부 냉각재상실사고 (우회사고)	5.35E-08
19	증기발생기 세관 파단 사고 (우회사고)	9.538E-07

* KOPEC, "Probabilistic Safety Assessment for Ulchin Units 3&4," Korea Electric Power Cooperation, 2004.

ALTERNATIVES 2

Any release that would exceed specific thresholds in terms of fractional releases and timing of release(>2.5% I,Te)

Advantages:

- Requires Level-2 PSAs

Disadvantages:

- Defining threshold level
- Implications associated with the definition of time of release

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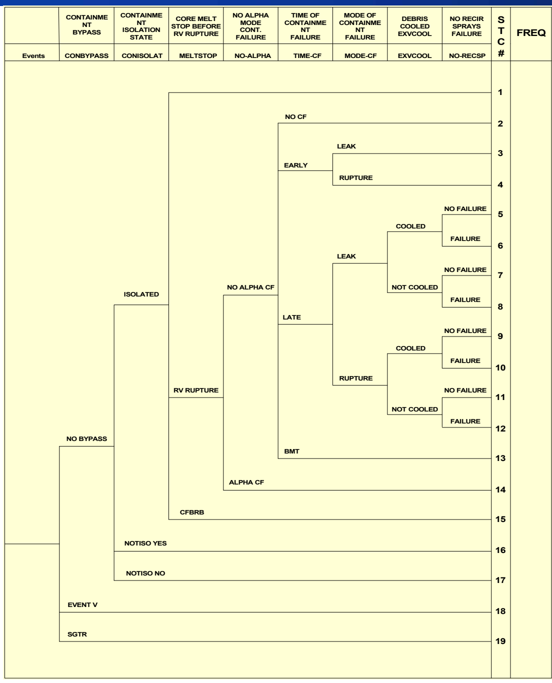
UCN3&4 source term logic diagram



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ALTERNATIVES 3

A collection of all releases that would result in one or more early fatalities offsite

Advantages:

- Includes power-dependence
- Includes site impact (weather & population)
- Can include impact of EP

Disadvantages:

- Focused on early fatalities only
- Mitigates impacts of large late releases

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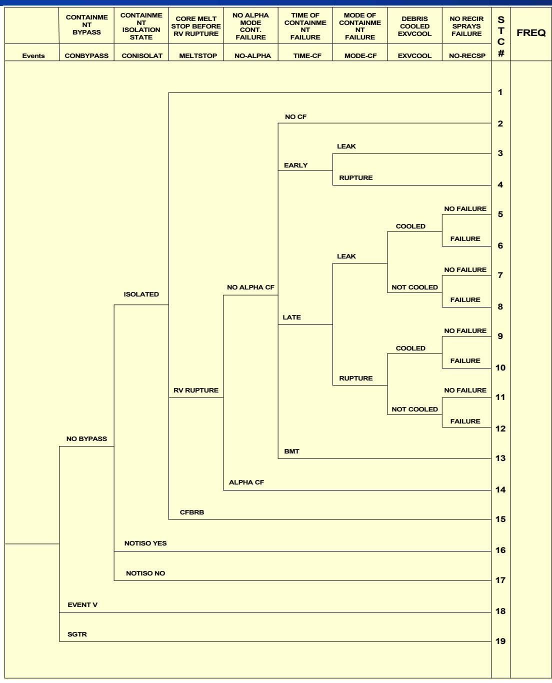
UCN3&4 source term logic diagram



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Risk

Level 1&2 PSA　　Level 3 PSA

$$R = \sum_{i=1}^{n} P_i \cdot C_i$$

P_i : Estimation of the likelihood or frequencies of events
C_i : Early fatality consequences given the sequence i which has associated with it a source term, that may be defined in terms of the equivalent release of iodine to the outside environment.
i : 1, 2, ●●●, n (Accident Scenario)

$$Mean\ of\ Early\ Fatality = \sum STCF_i \cdot C_i$$

$STCF_i$: source term category frequency i

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Individual Early Fatality Risk

$$IEFR=\sum STCF_i \cdot (C_i/P)$$

$$IEF=(C_i/P)=1-exp(-0.693[H_i/D_{50}]^{\beta})$$

H : effective acute dose to the target organ
P: population within that same one mile region
D_{50}: dose required for producing an effect in 50 percent of the exposed individuals

simple estimates for the effective acute dose

$$IEFR=\sum STCF_i \cdot \{1-exp(-0.693[H_i/D_{50}]^{\beta})\}$$

$STCF_i$: source term category frequency i

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The method for estimating the radiation doses

$$\frac{dC}{dt}=-\alpha pC$$

α: 0.1% per day
p: the leakage rate in percent
C: the amount of the nuclide remaining in the building at time t

$$\bar{Q}=\lambda_l C=\lambda_l C_0 e^{-\lambda_c t}$$

$$\lambda_c=\lambda_l+\lambda \quad (\lambda_l=0.01p)$$

λ_l: the rate at which the fission product is released from the building
λ: the radioactivity decay constant

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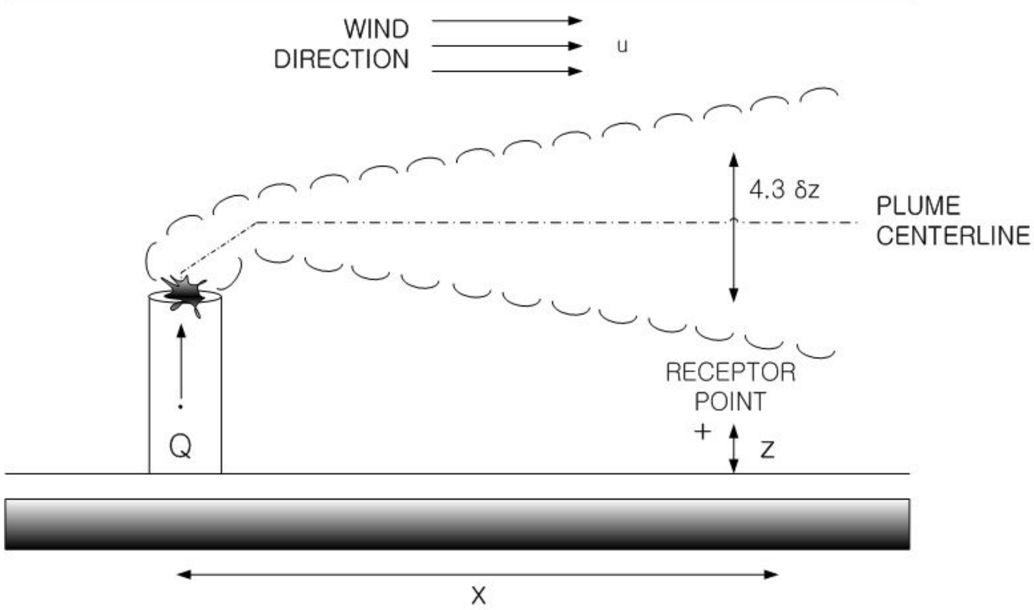
Gaussian Plume Model



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The method for estimating the radiation doses

χ : The concentration of some effluent as function of space and time

$$K\ \nabla^2 \chi = \frac{\delta \chi}{\delta t}$$

$$K_x \frac{d^2 \chi}{dx^2} + K_y \frac{d^2 \chi}{dy^2} + K_z \frac{d^2 \chi}{dz^2} = \frac{d\chi}{dt}$$

The effluent moving along the x-direction spreads out in Gaussian distributions in the y-and z-directions.
σ_y, σ_z**: The horizontal and vertical dispersion coefficients**

- **With ground level release, the value is largest along the centerline of the plume (i.e., where y=0).**

$$\chi = \frac{\lambda_l C_0 e^{-\lambda_c t}}{\pi \bar{v} \sigma_y \sigma_z}$$

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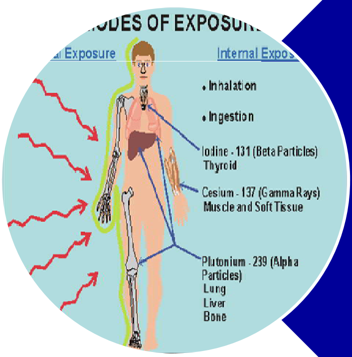
Modes of exposure

People are exposed to radiation in mainly two modes



- From radiation sources outside the body (extern al exposure)
- From radioactive subst ances that are inhaled or ingested into the body (internal exposure)

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The method for estimating the radiation doses

The total external dose to a person who stands in the plume given by Equation.

$$H = 0.262\bar{E}_\gamma \int_0^{t_0} \chi dt \quad \Rightarrow \quad H = \frac{0.262\bar{E}_\gamma \lambda_l C_0 t_0}{\pi \bar{v} \sigma_y \sigma_z}$$

parameters	values	distribution
σ_y (horizontal dispersion coefficient)	70	uniform [min:55 ~ max: 85]
σ_z (vertical dispersion coefficient)	21	uniform [min:15 ~ max: 30]
C_0 (the amount of the nuclide [C_i])	7.68E7	normal [mean:7.6E7 ~ s.d:7.6E6]
λ_l (release fraction [sec^{-1}])	5.8E-6	normal [mean:5.8E-6 ~ s.d:5.8E-7]
t_0 (period of time [sec])	14400	uniform [min:12960 ~ max: 15840]
$\bar{U}$ (wind speed [m/sec])	1	uniform [min:0.5 ~ max: 1.5]
E_r (average energies per disintegration)	0.371	constant
D_{50}(lethal dose [rem])*	400	uniform [min:250 ~ max: 550]
β (shape factor)	6	uniform [min:4 ~ max: 8]

***J.S. Evans, "Health Effects Models for Nuclear Power Plant Accident Consequence Analysis, NUREG/CR- 4214," SAND85-7185, Rev. 1, Sandia National Laboratories, 1990.**

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The method for estimating the radiation doses

The calculation of the dose received by an organ over a given period of inhalation

$$\dot{H} = \frac{592B\xi}{M}\int_0^{t_0}\chi(\tau)R(t_0-\tau)e^{-\lambda(t_0-\tau)}d\tau \quad \Rightarrow \quad H = \frac{592B\xi q\lambda_l C_0}{\pi\bar{v}\sigma_y\sigma_z M\lambda_e\lambda_c}(1-e^{-\lambda_c t_0})$$

parameters	values	distribution
σ_z, σ_y (horizontal, vertical dispersion coefficients)	70, 21	Uniform
B (average breathing rate [m^3/sec])	2.32E-4	normal [mean:2.3E-4 ~ s.d:2.3E-5]
M (mass of the organ [grams])	20	normal [mean: 20 ~ s.d: 2]
ξ (effective energy equivalent [Mev])	0.23	constant
q (fraction of the radionuclide)	0.23	constant
λ_e (effective decay constant[sec^{-1}])	1.06E-6	constant
λ_c (totoal decay constant[sec^{-1}])	1.00E-6	constant
D_{50}(lethal dose [Gy])*	200	uniform [min:150 ~ max: 250]

*J.S. Evans, "Health Effects Models for Nuclear Power Plant Accident Consequence Analysis, NUREG/CR- 4214," SAND85-7185, Rev. 1, Sandia National Laboratories, 1990.

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The method for estimating the radiation doses

The individual early fatality risk(IEFR) provides a measure of the average probability that a specific individual within one mile of the plant would be exposed to a lethal radiation dose.

$$\sum_i (STCF)_i\left\{1-e^{-(I+E)}\right\}_i = IEFR$$

$$I = \ln 2\left(\frac{0.262\bar{E}_\gamma\lambda_l C_0 t_0}{\pi\bar{v}\sigma_y\sigma_z D_{50}}\right)^\beta \qquad E = \ln 2\left(\frac{592B\xi q\lambda_l C_0 t_0}{\pi\bar{v}\sigma_y\sigma_z M\lambda_e D_{50}}\right)^\beta$$

The relationship between the STCF and the LERF is defined.

$$LERF = \sum_{k=1}^{n}(STCF)_k \quad (\text{expected value of early fatality} \geq 1)$$

n : number of STC

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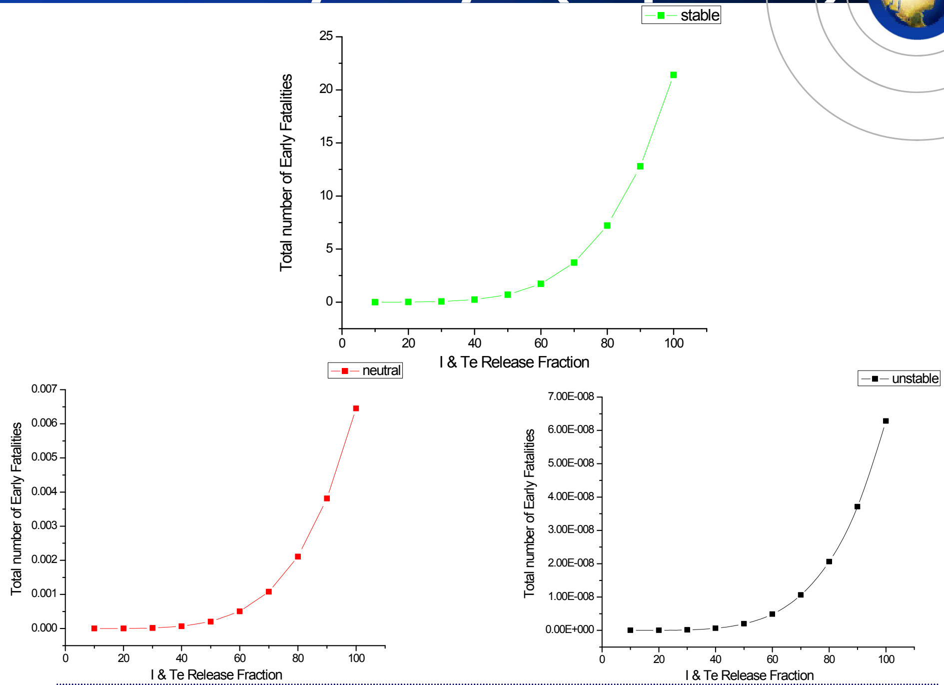
Sensitivity Analysis(dispersion)



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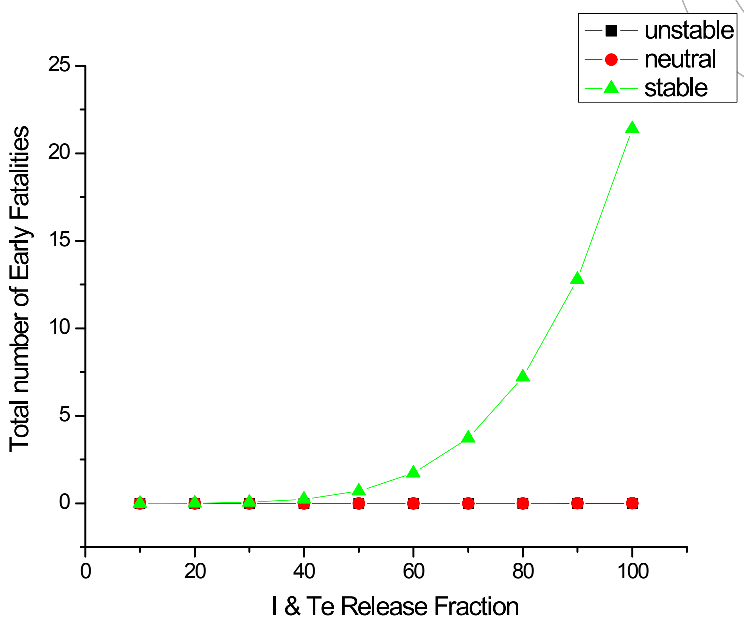
Sensitivity Analysis(dispersion)



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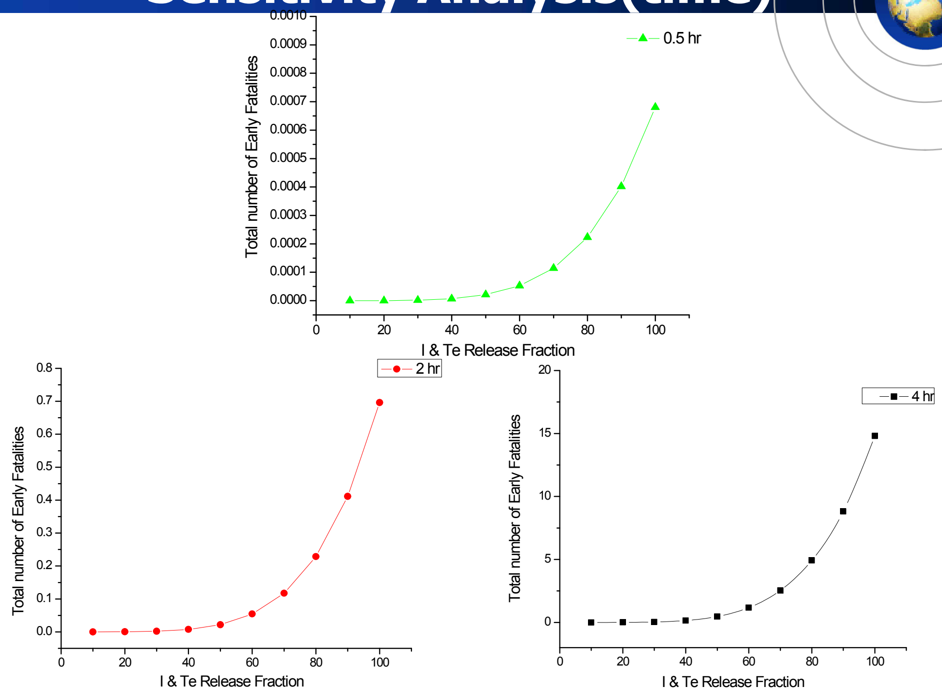
Sensitivity Analysis(time)



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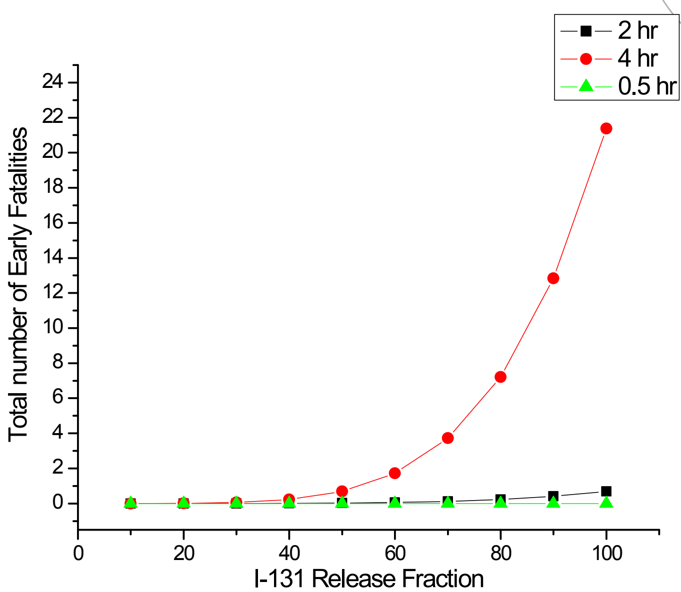
Sensitivity Analysis(time)



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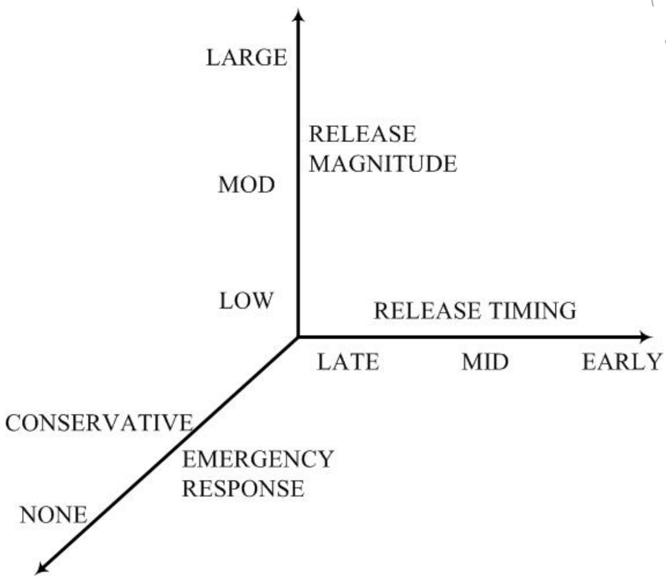
FACTORS AFFECTING EARLY FATALITY CALCULATION



*A. L. Hanson, R. E. Davis, V.Mubayi, "Calculations in Support of a Potential Definition of Large Release," U.S. Nuclear Regulatory Commission, NUREG/CR-6094, 1994.

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Conclusions

To assess the implication of using various definiti ons for "Large Early Release Frequency (LERF)"

To define the link between the level - 2 PSA re sults and an indirect attempt to assess health effects from the release.

LERF is strongly affected by site population, EP, release characteristics.

This study may contribute to defining LERF and finding a measure for Risk-informed Regulations and Risk-informed application (RIR&RIA).

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Thank you

for your listening

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I -C-3

Risk-Informed Evaluation of Off-site Response Planning for Nuclear Emergencies

T. Homma, M. Kimura, S. Takahara, and J. Ishikawa
Japan Atomic Energy Agency

KJPSA10 Jeju, May 18 - 20, 2009

Introduction

- ICRP Publ.60 (1990) + ICRP Publ.63 (1992)
  - Recommends values for the AVERTED dose for SINGLE protective actions where intervention is almost always justified
  - Optimization of a protective action in terms of averted dose using a cost-benefit consideration
- ICRP Publ. 103 (2007) + ICRP TG report (2009)
  - Recommends an upper value of the PROJECTED dose (called REFERENCE LEVEL) received via ALL pathways below which optimisation is applied.
  - Optimization of an overall strategy in terms of RESIDUAL dose using a reference level
- Objective
  - ➢ To perform a risk-informed evaluation of off-site protective actions such as sheltering, evacuation, and administration of stable iodine, to develop practical guidance for protective action strategies.

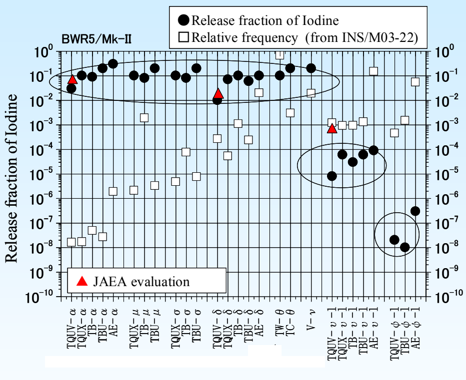
2

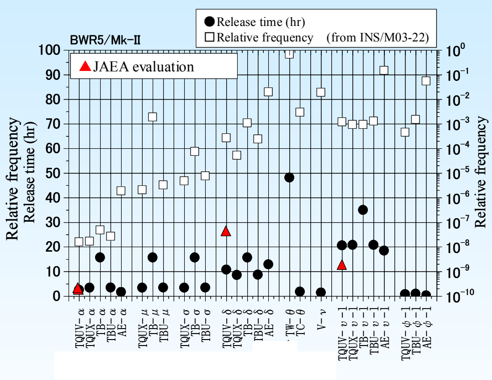
Risk-Informed Methods

1. Define source terms to be used as the bases for EP
2. Develop a R-I framework for evaluation of EP effectiveness
   - Application of Level 3 approach
   - All weather conditions, all exposure pathways
3. Investigate site information and define parameters
4. Develop countermeasure models
   - Dose reduction from protective actions (Shielding and filtering factors for sheltering)
   - Estimation of evacuation time
   - Administration of stable iodine (Introduction of iodine metabolic model)
5. Assess protective action strategies

3

Source Terms Development



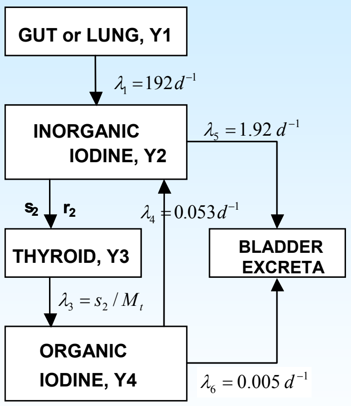


- 10^{-1} Iodine release fraction, early release : Energetic events
- 10^{-1} late release : Overpressure
- 10^{-4}～10^{-5} : Containment vent
- 10^{-7}～10^{-8} : Accident termination

4

Metabolic Model of Iodine in Man

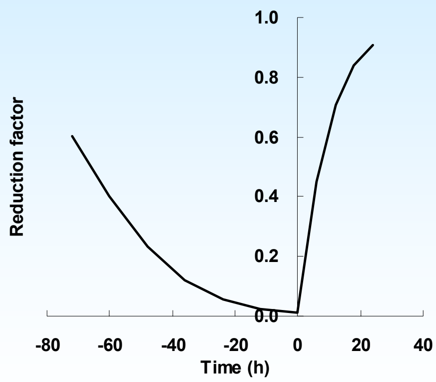
Johnson model(1981)



$r_2 = s_2 (Y_2^r / Y_2^s)$

r_2, s_2 : the rate of uptake of radioactive and stable iodine by the thyroid

Reduction in the committed thyroid dose to man (for adult, 100mg)



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Source Terms and PM strategy

Three source term scenarios

Release scenario	*Release time (hr after scram)*	*Duration of release (h)*	*Release fraction of iodine (%)*
Large early release	***3***	***1***	***7.9***
Large late release	***27***	***7***	***3.3***
Control release	***12***	***22***	***0.09***

Strategies of protective measures

- Large early release: precautionary evacuation with stable iodine intake
- Large late releases: evacuation and sheltering with stable iodine intake
- Control releases: sheltering with stable iodine intake

Site data

- A reference site is assumed to be located at a coastal area facing the Pacific Ocean (JAEA site at Tokai)

6

Steps for Consequence Evaluation

Calculation of dose from each pathway and time-dependent iodine concentrations in air at receptor points using OSCAAR

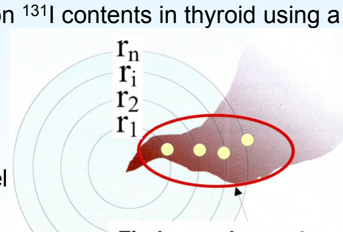
- 248 weather sequences selected by a stratified sampling method

Calculation of dose reduction effects by various combinations of protective measures

- Intervention levels for implementing each protective measure
  Sheltering : 10 mSv, Evacuation : 50 mSv (effective dose)
  Administration of stable iodine : 100 mSv (thyroid equivalent dose)
- Inhalation dose due to iodine intake based on ^{131}I contents in thyroid using a metabolic model by Johnson

Calculation of maximum dose at each distance from the site and its probability of weather sequence

- Probability of exceeding a specific dose level
- Dose at each distance from the site at a specific cumulative probability of weather sequences



Find a maximum dose at each distance and each weather sequence

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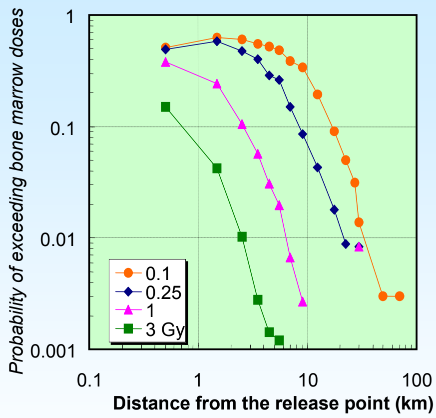
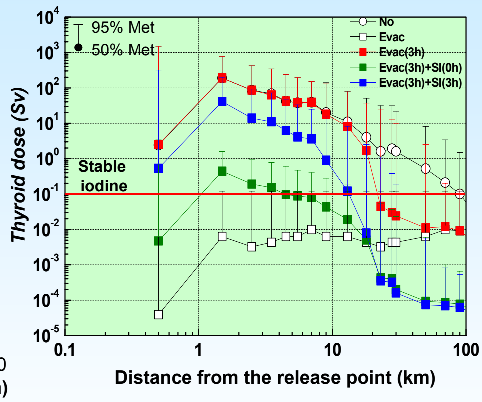
Large Early Release





- Even for large early release without protective measures, mortality would be very unlikely to occur beyond about 5 km.
- Early stable iodine intake can be very effective to reduce the thyroid dose for the people close to the site even the delay of evacuation.

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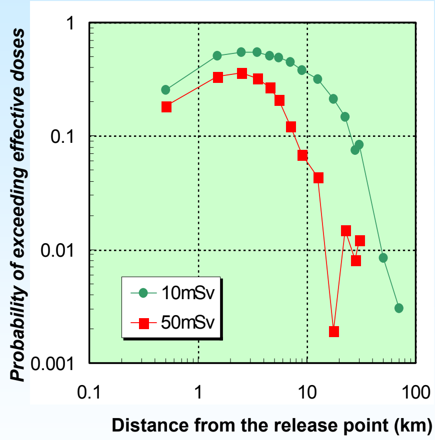
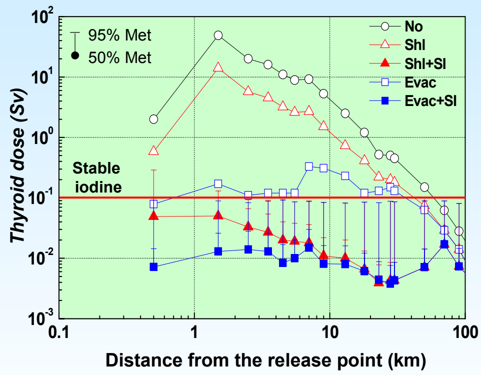
Large Late Release





- For large late release, evacuation area is unlikely to occur beyond 10 - 20 km.
- For the sheltering area, stable iodine intake can be very effective to reduce the thyroid dose.

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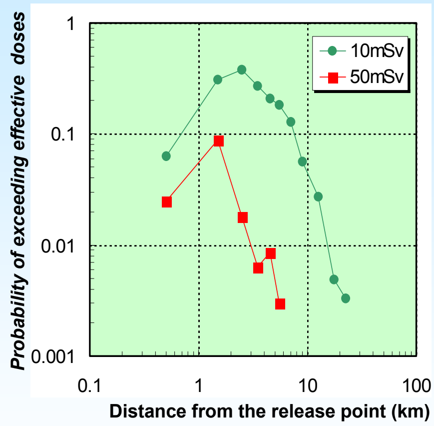
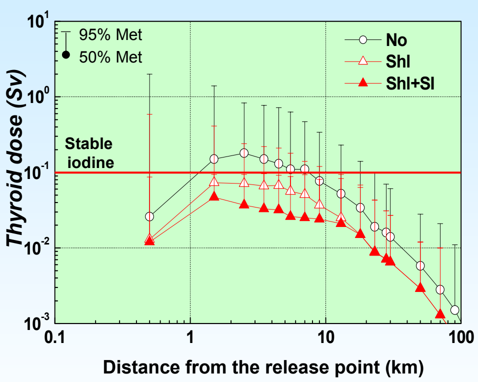
Control Release





- For control release, evacuation area is unlikely to occur beyond a few kilometers and sheltering area is unlikely to occur beyond about 10 km.
- For very severe weather conditions, sheltering with stable iodine intake is needed only close to the site.

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Conclusions

- For the representative source terms, the analysis has been performed using the Level 3 PSA approach to evaluate the effectiveness of protective action strategy involving a combination of evacuation, sheltering and administration of stable iodine.
- The study indicates that pre-distribution of stable iodine might be considered for the people close to the site in planning.
- The study also indicates that administration of stable iodine should be considered as a supplement to sheltering at greater distances from the site in planning.
- The results of this study will be expected to form the basis for future technical guidance for protective action strategies.

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I -C-4

The 10th Korea-Japan Joint Workshop on PSA, Haevichi Hotel &Resort, Jeju, Korea, 2009.

Optimization of Relocation Decisions using the Method of Probabilistic Accident Consequence Assessment

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Nuclear Safety Research Center
Japan Atomic Energy Agency

JAEA

1

Background

Optimization of radiological protection

The source related process to keep the likelihood of incurring exposures, the number of people exposed, and the magnitude of individual doses as low as reasonably achievable, taking economic and societal factors into account.

(ICRP, Publ. 103)

Objectives

- The application of a Level 3 PSA code to the optimization of relocation.
  Relocation is one of the long-term protective action in case of radiological emergency.
  - ✓ To describe an appropriate framework for the optimization process in the basis of ICRP Publ.103.
  - ✓ To develop the optimum dose criteria for introducing and terminating relocation.

2

Optimization of Relocation in Publ. 63

◆ Each protective action should be balanced against the cost in such a way that the net benefit achieved by protective action is maximized.

$$B = (Y_0 - Y) - (X + R + A_i + A_s - B_c)$$

(IAEA Safety Series 109)

- B : The net Benefit
- Y_0 : The detriments due to radiation with taking no reaction
- Y : The residual detriments when the action is taken
- R : The physical risks of any protective measure itself
- X : The resources needed to implement the measure
- A_i : Individual anxiety caused by the protective measure
- A_s : Social disruption caused by the protective measure
- B_c : The reassurance benefit from the protective measure

$$B = (Y_0 - Y) - X$$
$$= \alpha \cdot \Delta S - k \cdot t \cdot N$$

(ICRP Publ.63)

- α : The cost of unit collective dose (person·Sv)
- ΔS : The averted dose due to protective action (Sv)
- N : The number of relocated people (person)
- t : period of relocation
- k : The ongoing cost per individual per unit time

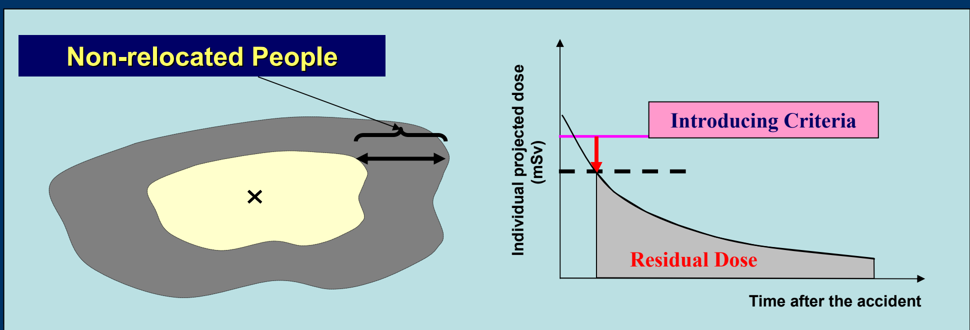
➢ Average averted effective dose of about 1 Sv may serve as an almost always justified level for relocation.
➢ Relocation is optimised is about 10 mSv per month continuing and prolonged exposure

Optimization of protective action in Publ. 103

◆ The new recommendation focuses on optimization with respect to the overall strategy, rather than the individual measures.

◆ It is necessary to evaluate the total residual doses that result from implementation of the protective strategy, and compare with the appropriate reference level.

➢ The residual dose to the relocated groups after returning to their home
➢ The received dose to the people in the non-relocated group



4

Level 3 PSA

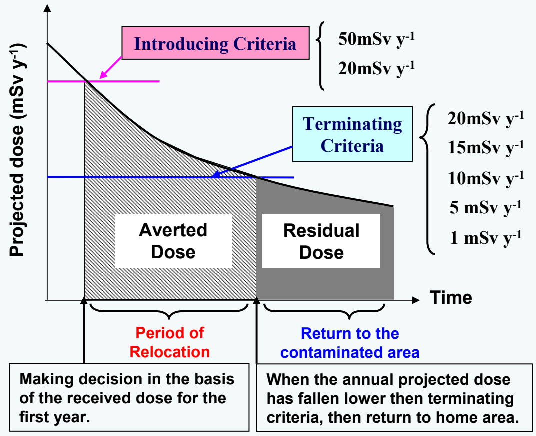
◆ **Level 3 PSA code**
- ✓ OSCAAR code

◆ **Site date**
- ✓ Tokai Site
- ✓ Meteorological condition: 248 sequence
- ✓ Population distribution: 1997 census

◆ **Accident condition**
- ✓ Inventory: 1,100 MWe BWR
- ✓ Release time: 27 hours
- ✓ Release height: 40 m
- ✓ Release fraction

Xe-Kr	I	Cs-Rb	Te-Sb	Sr-Ba	Ru	La
9.5×10^{-1}	3.3×10^{-2}	2.8×10^{-2}	2.8×10^{-4}	1.2×10^{-8}	2.4×10^{-11}	5.2×10^{-12}

◆ **Protective measures**
- ✓ Sheltering : <30km, 1 day, 10mSv
- ✓ Evacuation : <10km, 1 week, 50mSv
- ✓ Relocation :



➢ The results of Level 3 PSA are given as the expectation values from the probabilistic analysis, which represent a weighted average over a spectrum of meteorological conditions.

Classification of inhabitants

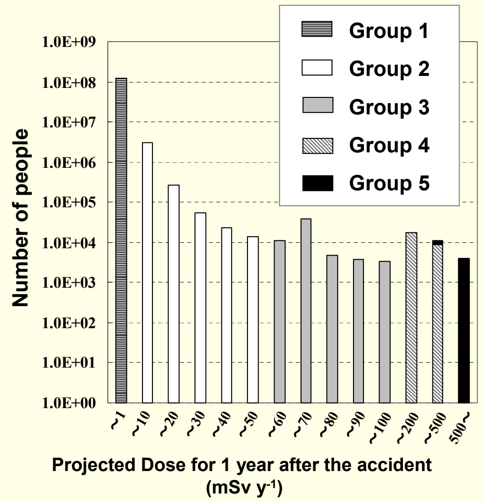
◆ The total collective effective dose is not a useful tool for making decisions (ICRP Publ.103)

Averted Dose	Projected Dose for 1 year	Relocated or Non-Relocated	Group name
Justified	Deterministic health effects Significant risk of cancer	Relocated	Group 5
1 Sv			Group 4
	100 mSv		Group 3
	Risk of cancer	Introducing Criteria	
		Non-relocated	Group 2
	1 mSv		
	Non-afflicted people		Group 1

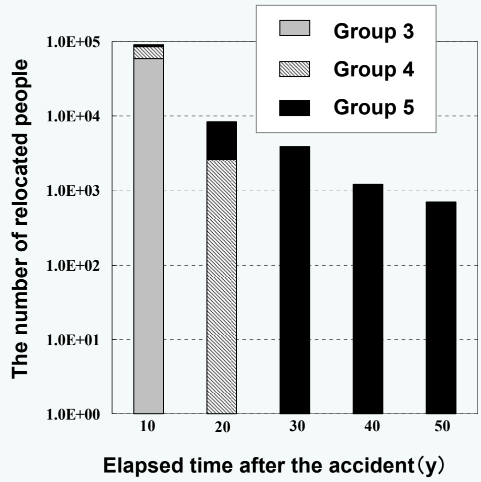
➢ Inhabitants whose averted dose due to relocation is greater equal 1 Sv are classified into a group in terms of Justification criteria.
➢ At dose higher than 100 mSv, there is an increased likelihood of deterministic effects and a significant risk of cancer.
➢ Inhabitants whose received dose for first year is less than 1 mSv are not regarded as afflicted people.

6

Dose distribution and the number of relocated people



Projected Dose for 1 year after the accident
(Introducing Criteria 50mSv y^{-1})



The number of relocated people as a function of time after the accident
(Introducing Criteria 50mSv y^{-1},
Terminating Criteria 20mSv y^{-1})

Relocation cost and residual detriments

Relocation Cost: X (US$)

$X = k \cdot t \cdot N$

- k : 6,000 US$ $person^{-1}$ y^{-1}
- N : person
- t : year

Residual Detriment: Y (US$)

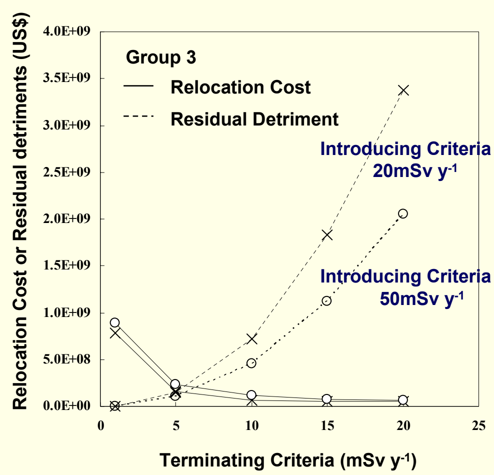
$Y = \alpha \cdot S$

- S : person Sv
- α : US$ $person^{-1}$ Sv^{-1}

$$\alpha = \alpha_0 \left(\frac{d}{d_0}\right)^a \left(\frac{1}{1+r}\right)^t$$

(Lochard et al.,1996)

- α_0 : US$ $person^{-1}$ Sv^{-1}
- a : Risk aversion factor
- d_0 : Base line =1mSv
- d : mSv
- r : Discount Rate = 3%
- t : year



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Efficiency of Relocation

◆ Minimization of the Cost needed to implement a relocation and the residual detriment when a relocation is taken.

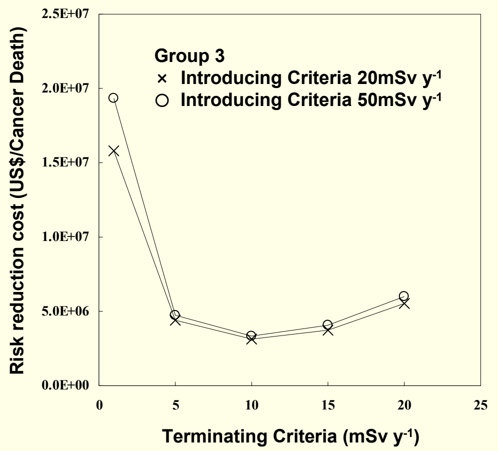
Risk Reduction Cost

$$\text{Risk Reduction Cost} = \frac{X+Y}{\Delta R} \left(\frac{\text{US\$}}{\text{Cancer Death}}\right)$$

X : Relocation Cost (US$)
Y : Residual Detriments (US$)
ΔR : Avertable risk of cancer death

$$\Delta R = \sum \Delta r_{lifetime} \cdot pop(\Delta r_{lifetime})$$

Δr : Individual lifetime probability of cancer death
$pop(r)$: the number of people who is exposed to a residual risk, r



➢ Risk Reduction Cost strongly depend on Terminating Criteria, rather than Introducing Criteria.

SUMMARY

◆ Level 3 PSA provide a good tool for establishing the numerical guidance of relocation in the basis of the comprehensive risk assessment that takes into account the full range of postulated events.

◆ Efficiency of relocation strategy strongly depend on terminating criteria, rather than introducing criteria.

◆ We should develop an appropriate framework in which the residues of radiation exposure for the entire affected population over the whole period should be taken into account for optimization of relocation

Issues

◆ The results of optimization of the protective actions is strongly influenced by the economic value of a unit collective dose.
  ➢ The α value is needed in Japan.
  ➢ It is necessary to consider the differences between WTP(an individual would accept to pay for avoiding the risk) and WTA (an individual would accept in compensation for avoiding the risk).

◆ More consideration is needed for the combination other long-term protective actions such as the food ban.

10


Thank you for your attention
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I -C-5

Development of an Off-site Risk Assessment Tool for the Risk-Informed Application

May. 18 2009

Jongtae Jeong, Joon-Eon Yang

Radioactive Waste Technology Development Division
Korea Atomic Energy Research Institute

Contents

KAERI 2

Introduction

- The area of Risk-Informed Application in Korea:
  - Extension of RI-STI
  - Extension of RI-AOT
  - RI-ISI
  - Extension of RI-ILRT interval
    - Based on an estimation of the population dose using the level-3 PSA tool, sensitivity analysis was performed.
- There are too many uncertainties in the estimation of the exposure dose and the associated risk.
- There is a growing need to reduce uncertainties in the estimation of off-site risk using the Level-3 PSA tool.
- The purpose of this research is to make a development plan for the off-site risk assessment, which can be used in the RIA or Level-3 PSA.
- We will focus on the reduction of uncertainties through the change of atmospheric dispersion model and the use of meteorological data.

KAERI 3

Introduction

Table 1. Meteorological data and deposition: Uncertainties and Sensitivities

Parameter or modeling assumption	Quantities most sensitive	Contribution to uncertainties in CCDFs		
		Early fatalities, injuries	Latent-cancer fatalities	Contaminated area, property damage
Sampling of meteorological data	Selection of weather sequences	Major	Low to moderate	Low
Definition of stability categories	Frequency of occurrence of stability categories	Low to moderate	Low	Low to moderate
Choice of dispersion parameters	Airborne concentrations	Moderate to major	Moderate	Moderate
Changing vs. constant weather	Weather sequences	Low to moderate	Moderate	Moderate
Gaussian theory vs K-theory (modeling)	Airborne concentrations	Low	Low	Low
Low wind speeds	Airborne concentrations	Low	Low	Low
Surface roughness	Dispersion parameters Deposition velocity	Low to moderate	Low	Low to moderate
Dry deposition velocity	Quantity of radioactive material on ground	Moderate	Moderate	Major
Rainfall model	Quantity and location of deposited material	Major	Low	Moderate
Terrain (modeling)	Plume trajectory	Moderate	Low	Moderate

Source: PRA Procedure Guide, NUREG/CR-2300)

KAERI 4

Atmospheric Dispersion Modeling

- The Gaussian plume model is widely used in the off-site risk estimation (Level-3 PSA);
  - Economical use of computing time.
  - The availability of meteorological parameters.
  - In some circumstances, the results do not differ sufficiently from those of more complicated models.
  - The observed maximum downwind concentration is within 10~20 % of the calculated value for a surface level source and within 20~40% for an elevated source.
  - The Gaussian plume model is valid for release heights of up to about 200m.
- The terrain effect on the dispersion of the radioactive material cannot be modeled by using the Gaussian plume model.
- We will consider the puff model in order to reduce the uncertainties in the atmospheric dispersion by taking into consideration of the terrain effect on the dispersion.
- We will consider the CALPUFF model used as a regulatory models by US EPA.

KAERI 5

Atmospheric Dispersion Modeling

- The CALPUFF Model:
  - a multi-layer, multi-species non-steady stae puff dispersion model.
  - can simulate the effects of time- and space-varying meteorological conditions on pollutant transport, transformation, and removal.
  - Can use the three dimensional meteorological fields developed by the CALMET
  - Or can use simple, single station winds
  - Contains algorithms for near-source effects such as
    - building downwash
    - transitional plume rise
    - subgrid scale complex terrain interactions
    - longer range effects such as wet scavenging and dry deposition
    - Chemical transformation
    - Vertical wind shear
    - Overwater transport and coastal interaction effects
  - Does not model the radioactive decay

KAERI 6

Atmospheric Dispersion Modeling

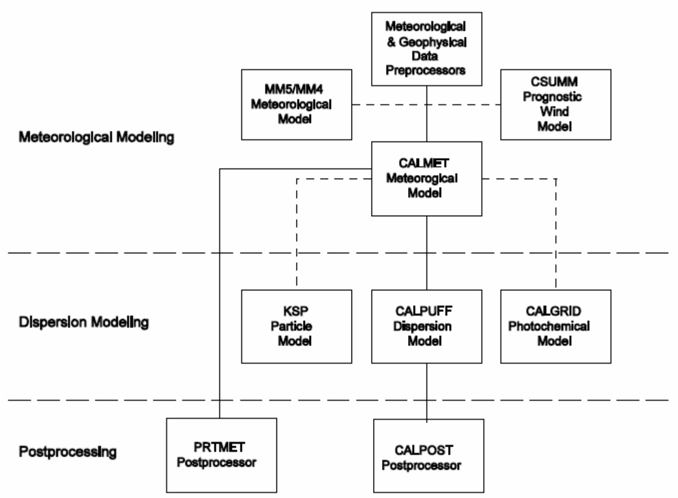

Fig. 1 Overview of the program elements in the CALMET/CALPUFF modeling system

Source: A User's Guide for the CALPUFF Dispersion Model, Version 5)

KAERI 7

Use of Meteorological Data

- There are big climate changes around the world.
- It is necessary to consider a large amount of data and to modify the sampling scheme of meteorological data.
- The seasonal variation of meteorological data will also have an impact on the risk estimation.
- In order to obtain insights into the use of meteorological data, we performed risk estimation based on the different sampling intervals of the rainfall rate for different years and seasons;
  - We analyze the meteorological data of the last 3 years
  - We suggest different sampling intervals of the rainfall rate
  - We suggest the necessity of modifying the sampling intervals of meteorological data

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Use of Meteorological Data

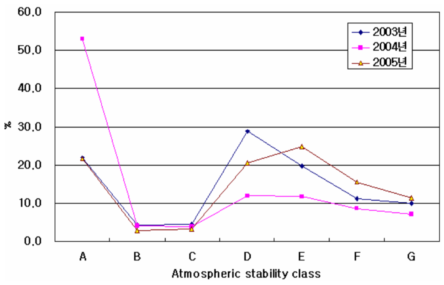

Figure 2. Distribution of the wind direction

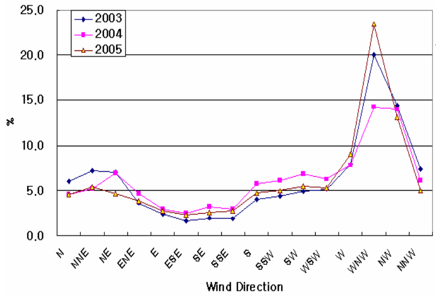

Figure 3. Distribution of the stability class

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Use of Meteorological Data

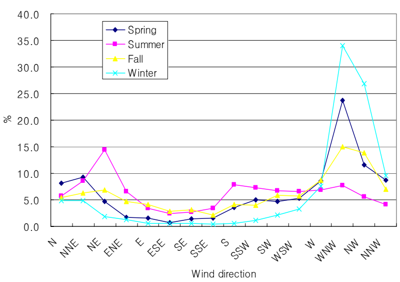

Figure 4. Seasonal variation of the wind direction

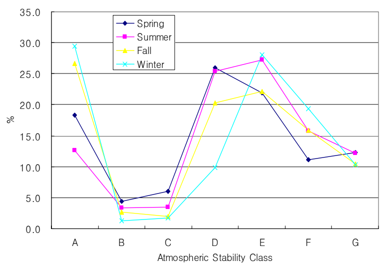

Figure 5. Seasonal variation of the stability class

KAERI 10

Use of Meteorological Data

Table 2. Analysis of the rainfall rate for each year

Year	2003	2004	2005
Total rainfall (mm)	2347	1423	1260
Rainfall hours	760	518	532
Percentage of rainfall (%)	8.7	5.9	6.1
Maximum rainfall (mm)	36.6	26.4	31
Hourly average rainfall (mm/hr)	2.9	2.7	2.4

Table 3. Analysis of the rainfall rate for each season

	Spring	Summer	Fall	Winter
Rainfall time (hr)	99	211	155	67
Total rainfall (mm)	140.7	547.9	481.8	89.4
Hourly average rainfall (mm/hr)	1.4	2.6	3.1	1.3
Maximum rainfall (mm)	7.1	31.0	18.0	7.6

KAERI 11

Use of Meteorological Data

Table 4. Initial condition weather bins

Weather Bin	Stability	Wind Speed (m/sec)
1	A/B	$0 < u \leq 3$
2	A/B	$3 < u$
3	C/D	$0 < u \leq 1$
4	C/D	$1 < u \leq 2$
5	C/D	$2 < u \leq 3$
6	C/D	$3 < u \leq 5$
7	C/D	$5 < u \leq 7$
8	C/D	$7 < u$
9	E	$0 < u \leq 1$
10	E	$1 < u \leq 2$
11	E	$2 < u \leq 3$
12	E	$3 < u$
13	F	$0 < u \leq 1$
14	F	$1 < u \leq 2$
15	F	$2 < u \leq 3$
16	F	$3 < u$

Source: MACCS User's Guide, NUREG/CR-4691)

Table 5. Sampling interval of the rainfall rate for each season

Spring	Summer	Fall	Winter
$0.0<x\leq1.0$	$0.0<x\leq1.0$	$0.0<x\leq1.0$	$0.0<x\leq1.0$
$1.0<x\leq2.0$	$1.0<x\leq3.0$	$1.0<x\leq3.0$	$1.0<x\leq2.0$
$2.0<x\leq4.0$	$3.0<x\leq7.0$	$3.0<x\leq5.0$	$2.0<x\leq3.0$
$4.0<x$	$7.0<x$	$5.0<x$	$3.0<x$

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Use of Meteorological Data

Table 6. The variation of the early fatality risks.

Season / Year	Spring	Summer	Fall	Winter
2003	3.20E-08	3.40E-08	3.26E-08	2.96E-08
2004	1.93E-08	1.97E-08	1.93E-08	1.87E-08
2005	3.30E-08	3.19E-08	3.25E-08	3.23E-08

Table 7. The variation of the cancer fatality risks.

Season / Year	Spring	Summer	Fall	Winter
2003	6.96E-08	6.81E-08	6.94E-08	6.68E-08
2004	4.65E-08	4.53E-08	5.53E-08	4.79E-08
2005	8.32E-08	7.44E-08	7.45E-08	7.71E-08

KAERI 13

Concluding Remarks

- We suggested a development plan for the offsite risk estimation for use in the risk-informed application.
- The basic principle in its development is to reduce the uncertainty.
- We will consider the puff trajectory model in order to assess the terrain effect on the dispersion of the radioactive material.
- The CALPUFF model, chosen as a reference dispersion model, must be modified to be able to determine the radioactive decay.
- We will consider the use of meteorological data at least for 3 years to determine the climate change.
- Also, we will develop and apply an appropriate sampling technique for the meteorological data in order to determine seasonal variations.
- The off-site risk assessment tool will be developed by considering the suggestions in this study.
- It will be very useful in the risk-informed application in Korea.

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국가 미래 에너지를 책임지는 연구원
감 사 합 니 다 !
THANK YOU!
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한국원자력연구원
Korea Atomic Energy Research Institute

Session II-A

Risk Informed Application (I)

Session II-A Summary

Chair: Hidetaka IMAI (TEPCO), Dae-Wook CHUNG (KINS)

II-A-1. Tsuyoshi UCHIDA (JNES): Analysis and Evaluation of Accident Sequence Precursor
Mr. Uchida from JNES presented the current status of development and implementation of Accident Sequence Precursor (ASP) in Japan. JNES developed the framework for analysis and evaluation of ASPs. Using the framework, analysis and evaluation of ASP were performed for 202 events which occurred in CY2008. 2 of 202 events were identified to be safety significant through the qualitative screening and the quantitative screening.

II-A-2. Myung-Ki KIM (KEPRI): Analysis of Risk Change by adding Bypass Function into RPS/ESFAS
In this study, Authors have performed the evaluation of CDF along with Surveillance Test Interval (STIs) to find out the effect of bypass function during test and maintenance on the safety. Through the study, Authors get insights that bypass function during test and maintenance gives the positive effects on the safety because the effect of preventing the reactor trip on the safety is more dominant over other negative effects such increasing the unavailability of “No trip signal” and “No ESFAS signal.”

II-A-3. Young H. IN (ERIN): Risk Management in the NPP
A famous Korean-American expert, Dr. Young Ho In from EPRIN-USA explained the comprehensive implementation status of risk-informed applications in Exelon Power Company.

II-A-4. Bag Soon CHUNG (KEPRI): Tech. Spec. Optimization Study for the RPS/ESFAS of Kori Unit 2
This paper described that the improvement of the evaluation for Surveillance Test Interval (STIs) and Allowed Outage Time (AOT) for the analog channels, logic cabinets and slave relays by a Probabilistic safety assessment approach which includes the fault tree models, signals, component reliability database, and most of the test and maintenance assumptions.

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Analysis and Evaluation of Accident Sequence Precursor (ASP)

18th – 20th May, 2009

The 10th Korea-Japan Joint Workshop on PSA

Hiroaki SHIMOZAKI,
Tomomichi ITO, Yoshikane HAMAGUCHI, Katsunori OGURA

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Nuclear Energy System Safety Division
Japan Nuclear Energy Safety Organization (JNES)

1. Introduction

Back Ground

- ✓ Recently the nuclear regulatory body in Japan recognizes the importance of risk-informed application to the nuclear safety regulation.
- ✓ Accident Sequence Precursor (ASP) program provides safety perspective of nuclear power plant operating experiences.

Objectives

- ✓ The objective is to construct the framework of analysis and to evaluate ASP systematically. Evaluation results in CY2008 are discussed along with the following items.
  - (1) Identification of Safety Significant Events (ASP)
  - (2) Potential of Escalating into Accidents
  - (3) Possible Corrective Actions

The risk information could be utilized to the regulatory judgment (e.g. corrective actions to other plants).

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2. Framework for Analysis and Evaluation of ASP (1/4)

Evaluation Steps

- ✓ 1st step the qualitative screening of the events including incidents and accidents
- ✓ 2nd step the quantitative screening using a generic PSA model
- ✓ 3rd step the detailed evaluation using a realistic PSA model
- ✓ 4th step the assessment of corrective actions
  - a) Potential of Escalating into Accidents
  - b) Possible Corrective Actions

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2. Framework for Analysis and Evaluation of ASP (2/4)

PSA Models

① Quantitative screening (2nd Step)
The PSA models of Japanese typical plants are applied.
BWR : BWR3, BWR4, BWR5 and ABWR
PWR : 2Loop, 3loop, 4loop(dry containment type) and
4-loop PWR (ice condenser type)

② Detailed evaluation (3rd Step)
The modified PSA model is applied to reflect the configuration of the real target plant into the PSA model of the typical plant.

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2. Framework for Analysis and Evaluation of ASP (3/4)

- ASP Criterion
  The importance of an event is defined by the conditional core damage probability (CCDP).
  The event of which the CCDP is greater than 10^{-7}, is recognized as an accident sequence precursor (ASP). The value of 10^{-7} is a tentatively used as an ASP criterion.

Table.1 ASP Criterion Utilizing CCDP

Screening Out	ASP		
	Precursor	Important Precursor	Significant Precursor
CCDP< 10^{-7}	**CCDP:10^{-7}~10^{-5}**	**CCDP:10^{-5}~10^{-4}**	**CCDP> 10^{-4}**

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2. Framework for Analysis and Evaluation of ASP (4/4)

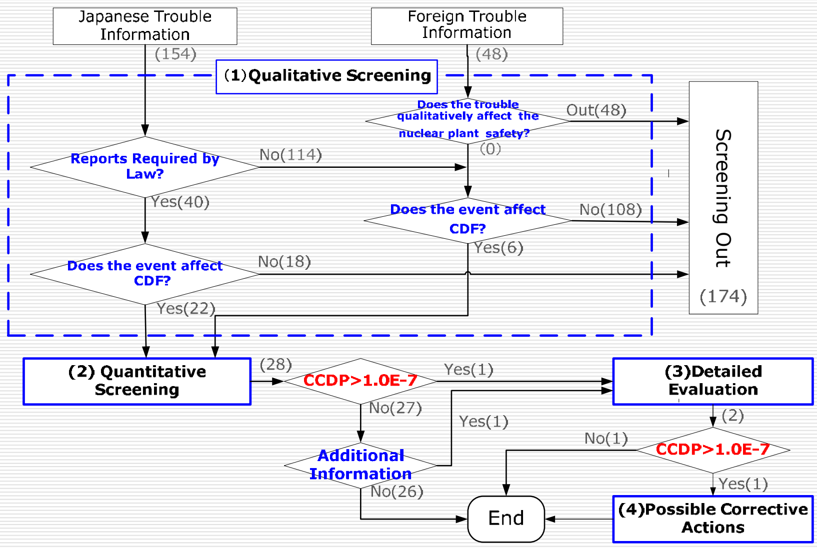

Fig.1 Flow for Analysis and Evaluation of ASP

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3. Example of Detailed Evaluation (1/3)

(1) Event Description

During the start-up of the plant (BWR4 type), the operator conducted the functional test of HPCI system. An alarm of HPCI turbine trip was occurred, and steam was detected near the HPCI governor valve. Then, HPCI operation was manually tripped by the operator.

According to the maintenance code, the operator checked the operability of RCIC. However, the RCIC pump stopped automatically during adjusting a test bypass valve.

Based on the situation, the operator judged that HPCI and RCIC were both not operable, and manually shutdown the reactor.

(2) Analyses Condition

- ✓ The modified PSA model is applied to reflect the real plant configuration (system configuration only) into the PSA model of the typical BWR4 plant.
- ✓ Japanese component failure rate database are used.
- ✓ According to the cause analysis, the failure of both systems was judged not to be a common cause failure.

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3. Example of Detailed Evaluation (2/3)

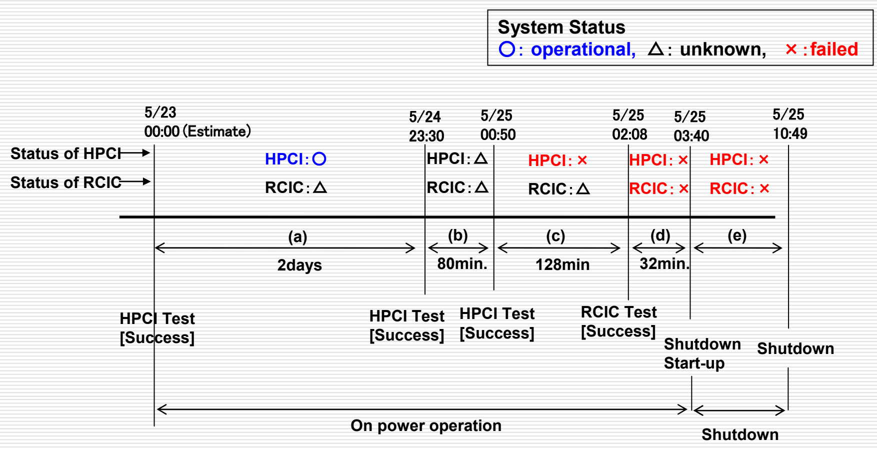

Fig.2 System Configuration during the Event

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3. Example of Detailed Evaluation (3/3)

The CCDP of the event was evaluated to exceed the ASP criterion (3.0×10-7).
The corresponding event was judged to be an safety significant event or an ASP.

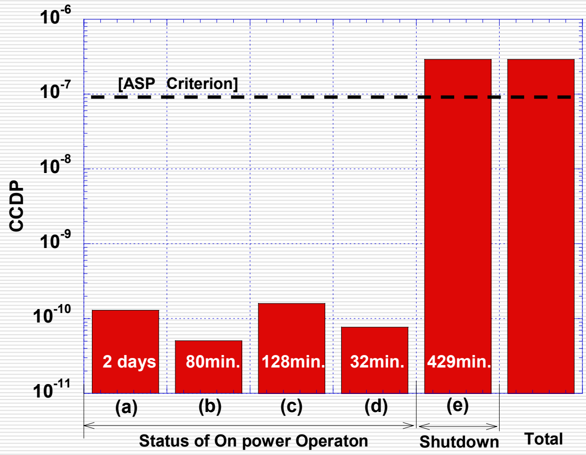

Fig.3 CCDP Result of Detailed Evaluation

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4. Potential of Escalating into Accidents (1/3)

Sensitivity analyses could reveal the event characteristics.
(1) Plant risk during the power operation is dominated by transients.
(2) The risk increase is relatively low when either of HPCI or RCIC is operable.
(3) The risk during the reactor shutdown increases when both HPCI and RCIC are at failed state.

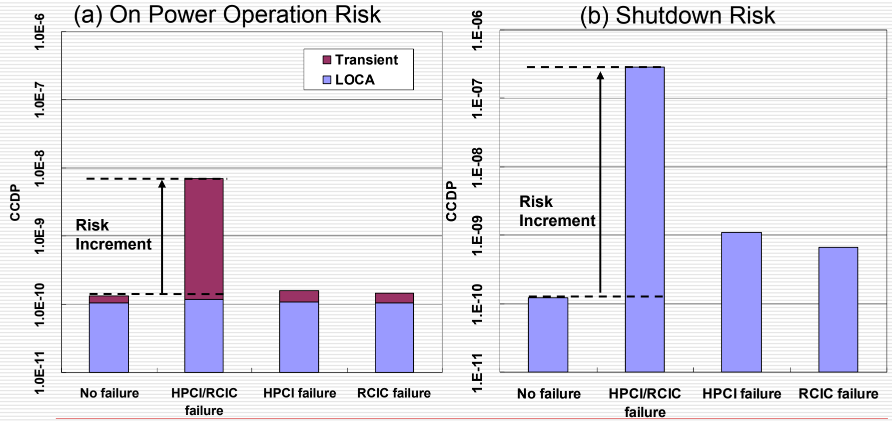

Fig.4 CCDP Results from Sensitive Analyses

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4. Potential of Escalating into Accidents (2/3)

- The LOSP is the dominant initiating event under the condition of simultaneous failure of HPCI and RCIC.
- The operator should take into account of external events (e.g. typhoon, thunderbolt, fallen snow etc.) which increase the likelihood of LOSP.

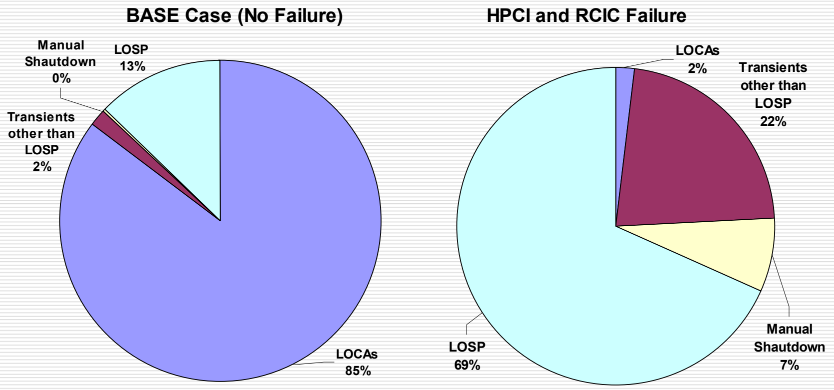

Fig.5 Contribution of each initiating event to the local CDF

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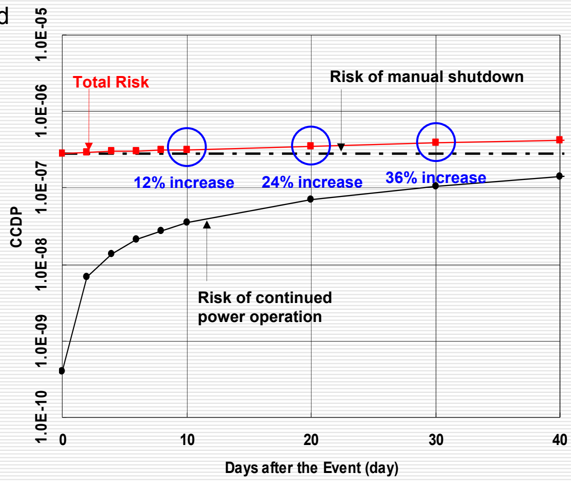
4. Potential of Escalating into Accidents (3/3)

- The total risk is dominated by the risk of the manual shutdown.
- The risk of the continued operation is relatively low (i.e. 12%, 24%, 36%).



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5. Possible Corrective Action (1/3)

Based on basic events with the large FV value, potential corrective actions to reduce the plant risk are identified.

- Calibration error of pressure process switch (a,b,c,d) FV : 8×10^{-1}

  Possible corrective action
  Confirmation of procedure for forced start-up of water injection.
  Check of instrument calibration test results and confirmation process.

- Failure to start of emergency diesel generator（EDG) FV : 2×10^{-2}

  Possible corrective action
  Confirmation of an other EDG operability (Surveillance of the EDG)

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5. Possible Corrective Action (2/3)

- Diagnostic failure to manually start up alternative injection FV : 2×10^{-3}

  Possible corrective action
  Confirmation of operating condition for alternative injection start up

- Failure of manually operation of ADS FV : 1×10^{-3}

  Possible corrective action
  Confirmation of operating procedure for manual ADS backup

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5. Possible Corrective Action (3/3)

Table.2 Effect of Corrective Action to Reduce the Risk

Case	Corrective Action	CCDP (*1)	Decreasing Risk(*2)
1	Forced start-up of LPCS Check of instrument calibration test results	6.8×10^{-8}	77 %
2	Confirmation of the another EDG availability	3.0×10^{-7}	0.2 %
3	Confirmation of operating condition of alternative injection start up	3.0×10^{-7}	0.1 %
4	Confirmation of procedure for manual ADS backup	3.0×10^{-7}	0 %
	(*1) CCDP = CCDP of On power operation + CCDP of Shutdown (*2) Decreasing Risk = 1-(CCDP of each case / CCDP of base case)		

✓Countermeasure for instrument calibration error prevention is availableness.

✓The evaluation results will be applied to the regulation judgment for the reference information.

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6. Summary

- JNES developed the framework for analysis and evaluation of ASPs. Using the framework, analysis and evaluation of ASP were performed for 202 events which occurred in CY2008.
- 2 of 202 events were identified to be safety significant through the qualitative screening and the quantitative screening.
- 2 events were analyzed and evaluated in the detailed evaluation step and one of two was identified as an ASP.
- The ASP event of "simultaneous failure of HPCI and RCIC in BWR4 plant" is important at loss of off-site power, and is identified to be safety significant. A potential corrective action to reduce the plant risk was discussed and identified.
- JNES has been conducting ongoingly the ASP evaluation, and has a plan to start a risk trend analysis.

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Analysis of Risk Change by adding Bypass Function into RPS/ESFAS

Korea Electric Power Corporation,
Korea Electric Power Research Institute
Kim, Myung-Ki

Contents

Introduction

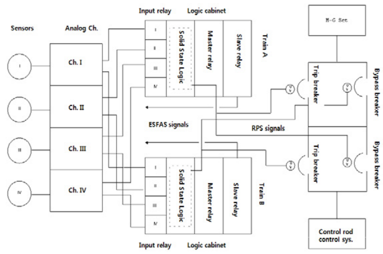
- The reactor protection system(RPS) and engineered safety actuation feature system(ESAFS) of a nuclear power plant is one of the critical safety systems to protect a reactor when the plant is out of normal situations.
- It is designed to have redundant structures and periodically tested to maintain high reliability and good performance according to surveillance test interval(STI) and allowed outage time (AOT) requirements, determined by engineering judgment without analytical analysis.
- Recently, however, these are rearranged such as 3 month STI instead of I month STI through the risk analysis using PSA model providing core damage frequency(CDF) and large early release frequency(LERF).
- Even though these relief the operation burden of plant staff, the test work of RPS/ESFAS that is performed under trip condition of the tested channel still needs more attention of the staff because of a little mistake or failure of a component causing reactor trip.

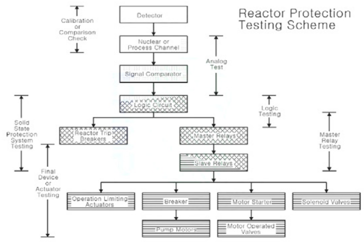
Introduction

- In order to get more flexible test environment, the bypass function card could be installed in the RPS/ESFAS. Where it is installed, the tested channel is under success condition, that is, no trip condition, so that the failure of components and its related human errors do not cause the reactor trip.
- Utilizing, however, the bypass function card gives a negative effect on the safety, in that under the reactor trip being required, the trip signal is defeated by the bypass function card, resulting threatens the safety.
- The adequateness of the introduction of bypass function into RPS/ESFAS requires the analysis of reliability and risk of the plant considering both positive and negative effects.
- We perform the feasibility study for Kori Unit 2, which is a 2 loops Westinghouse PWR 600 MWe.

System Descriptions

- RPS/ESFAS consists of sensors, analog channels, logic cabinets, master/slave relays, and reactor trip breakers.
- In order to verify the function of RPS/ESAFS, the technical specifications require performing surveillance test on analog channels, SSPS, master/slave relays, and trip breakers.
- In our study, OTDT signal in the reactor trip signals and MSI-CNMT Pr Hi signal in the ESFAS signals are selected as target signals for the study.





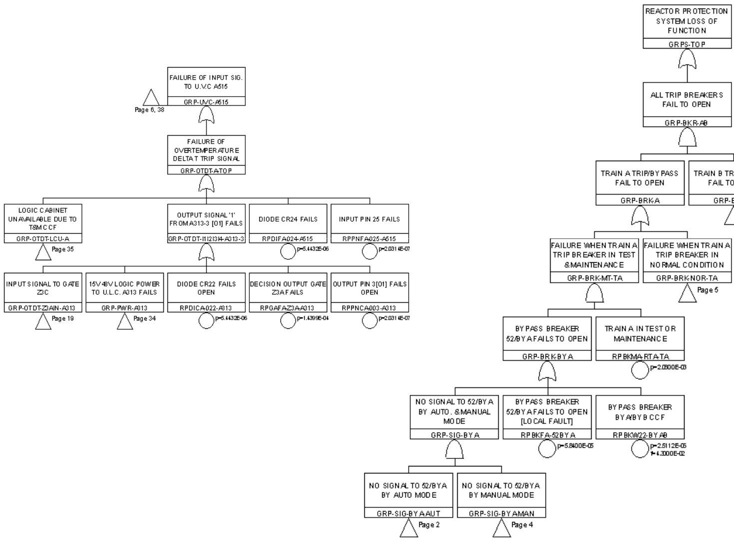
Analysis Methods

- To evaluate reliability and risk for RPS/ESFAS, four fault trees are constructed and then merged into PSA model.

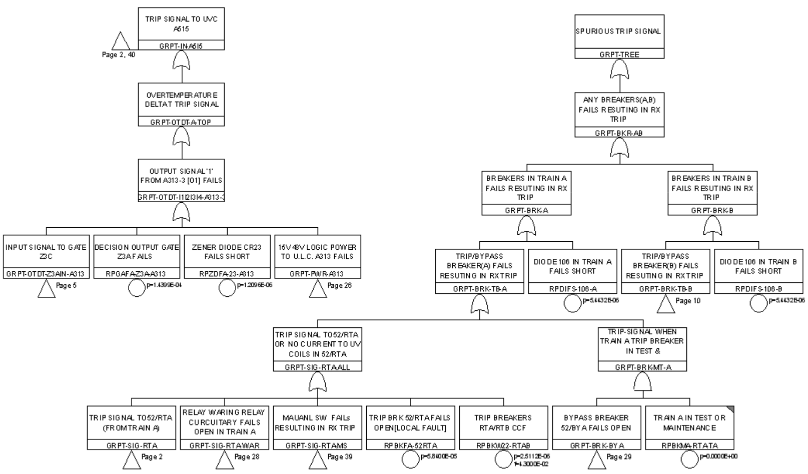
Heading of Fault Trees for OTDT in RPS	Heading of Fault Trees for MSI-CNMT Pr Hi in ESFAS
No OTDT trip signal due to failure of components and T/M 1) Without Bypass function 2) With Bypass function	No MSI-CNMT Pr Hi signal due to failure of components and T/M 1) Without Bypass function 2) With Bypass function
Spurious OTDT trip signal due to failure of components and T/M 1) Without Bypass function 2) With Bypass function	Spurious MSI-CNMT Pr Hi signal due to failure of components and T/M 1) Without Bypass function 2) With Bypass function

- Signal-specific FTs developed have particular characteristics in the level of detail of basic events. The FT for the analog channels are constructed with card-based basic events, whereas the fault tree for the logic cabinets are constructed with component-based basic events, such as transistor, IC, and so on.
- In final CDF analysis, test caused risk is considered based on operating experiences.

NO TRIP SIGNAL FAULT TREE: OTDT



SPURIOUS TRIP SIGNAL FAULT TREE: OTDT



Reliability and Risk Analysis Results

- Through the reliability analysis for RPS with and without bypass function according to STI 1, 3, 6 month
  - unavailability of "No OTDT trip signal due to failure of components and T/M"
  - probability of "Spurious OTDT trip signal due to failure of components and T/M"

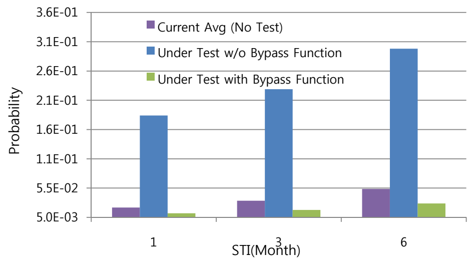
Fault Tree for RPS signals	Unavailability (1 M)	Unavailability (3 M)	Unavailability (6 M)
Without Bypass function			
No OTDT Trip signal	3.38E-06	3.79E-06	4.89E-06
Spurious OTDT Trip signal	2.23E-02	3.36E-02	5.37E-02
With Bypass function			
No OTDT Trip signal	3.41E-06	3.81E-06	4.90E-06
Spurious OTDT Trip signal	2.19E-02	3.34E-02	5.36E-02

Reliability and Risk Analysis Results

- By installing bypass function card, unavailability of "No OTDT trip signal" is slightly increased, which means that under reactor tip situation, the probability of the reactor not being trip increases, giving negative effects on the safety.
- Whereas, the probability of "Spurious OTDT trip signal" is slightly decreased, which means that unnecessary reactor trip due to failure of components and T/M decreases, giving positive effects on the safety.

Reliability and Risk Analysis Results

- During the test and maintenance, the probability of "Spurious OTDT trip signal" in bypass function is quite less than that of without bypass function.



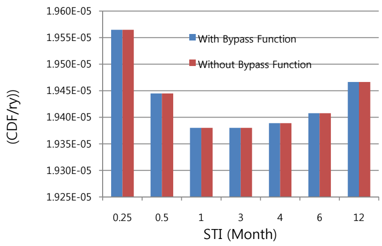
Reliability and Risk Analysis Results

- Through the reliability analysis for ESFAS with and without bypass function according to STI 1, 3, 6 month
  - unavailability of "No MSI-CNMT Pr Hi signal due to failure of components and T/M"
  - probability of "Spurious MSI-CNMT Pr Hi signal due to failure of components and T/M"

Fault Tree for ESFAS signals	Unavailability (1 M)	Unavailability (3 M)	Unavailability (6 M)
Without Bypass function			
No MSI-CNMT Pr Hi signal	1.0338E-03	1.1617E-03	1.2990E-03
Spurious MSI-CNMT Pr Hi signal	+ 1.1795E-02	1.1893E-02	1.2075E-02
With Bypass function	-		
No MSI-CNMT Pr Hi signal	1.0885E-03	1.2303E-03	1.3747E-03
Spurious MSI-CNMT Pr Hi signal	1.1739E-02	1.1829E-02	1.1999E-02

Reliability and Risk Analysis Results

- The results of CDF analysis considering these conflicting effects show that bypass function reduces CDF by 6.92E-10, that is, the safety effect of "Spurious OTDT trip signal" is dominant than that of "No OTDT trip signal."
- However, in the risk aspects, the CDFs for both cases are almost the same and the CDFs along with STIs are as follows:



Conclusions

- In the study, we perform the evaluation of CDF along with STIs to find out the effect of bypass function during test and maintenance on the safety.
  - Bypass function makes no trip condition at tested channel during test and maintenance.
- Through the study, we get insights that bypass function during test and maintenance gives the positive effects on the safety because the effect of preventing the reactor trip on the safety is more dominant over other negative effects such increasing the unavailability of "No trip signal" and "No ESFAS signal."
- In the risk aspect, however, the difference between CDFs for with and without bypass function is in the negligible range.
- We get insight that bypass function gives the flexible test environment resulting in removing the hidden trip elements, without degrading safety of the plant.

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The 10th Korea-Japan Joint Workshop on PSA (KJPSA)
2009. 5. 18-20, Jeju, Korea

Risk Management at the NPP

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Contents

- Background
- What we do
  - a(4), MSPI, Missed Surveillance, Mode Change, NOED, SDP, CDBI, MD 8.3
  - PARAGON, Workweek Analysis
- MR and Risk Management
- Impact on the station
- Conclusions

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Background

- Exelon Nuclear Fleet
  - 10 sites and 17 units
  - 12 BWRs and 5 PWRs
  - Mainly located in Illinois & Pennsylvania
  - Planning to build 2 ABWR units in Texas
- Risk Management Group
  - Outsourced 6 years ago to ERIN Engineering & Research
  - 9 PSA - Byron/Braidwood is one model
  - 10 Site Risk Management Engineers
  - ~30 additional PSA support engineers

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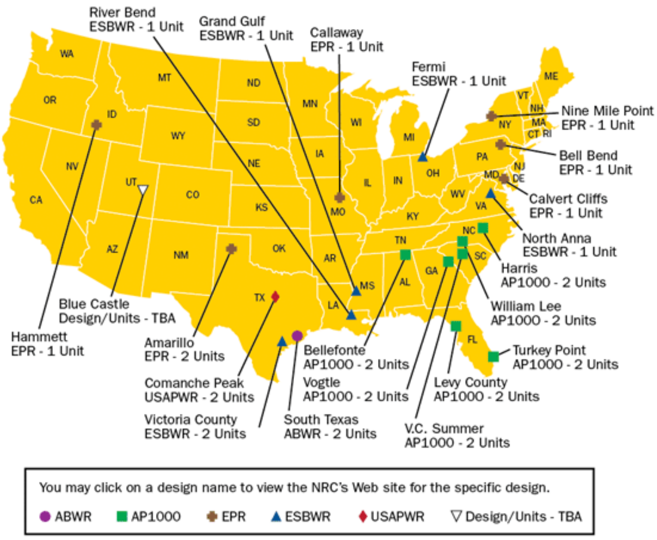

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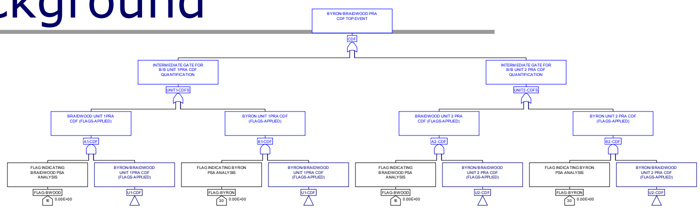
Background



- Byron and Braidwood PSA Model
  - Original IPE
  - Modified IPE
  - 1999 R0 Conversion from IPE
  - 2000 R1 Model enhancements
  - 2001 R2 Interim version
  - 2001 R3 CV pump mods incorporated
  - 2002 R4 Model enhancements
  - 2003 R5 1st Periodic update
  - 2008 R6 2nd Periodic update
  - 2011 -- 3rd Periodic update

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Operational Applications of Risk

- Maintenance Rule a(4) – Work Management/Operations
- Operator Training
  - Long Range Planning
  - Risk Insights/Operation Action Information Provided for Simulator Exercises.
- Significance Determination Process (SDP)
- NRC Management Directive (MD-8.3)
- Tech Spec 3.0.4.b – Mode Changes
- Tech Spec 4.0.3 – Missed Surveillances
- Notice of Enforcement Decision (NOED)

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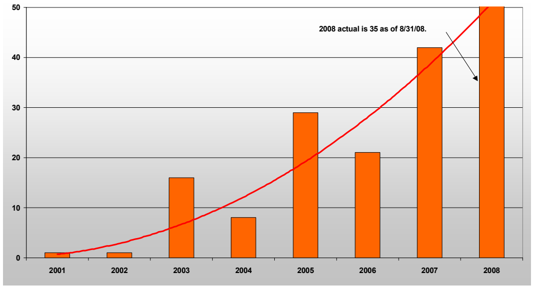
Operational Applications of Risk

- Technical Speciation Changes
  - CV, RH, SI, CS, CC, SX – Completion Time Changed from 72 hours to 7 days
  - Added 72 hour CT for SX Crosstie
- DG – CT Changed from 72 hours to 14 Days
- 120V AC Inverters – CT Changed from 24 hours to 7 days
- ESFAS/RPS – Multiple Completion Time and Surveillance Interval Changes



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Risk Insights

- Risk Reduction Improvements:
  - Auxiliary Building Flooding
    - Extended SX Discharge Above Lake Level – Limits Size of Flood Source
  - Added alternate cooling capability (FP) for CV Pump Lube oil coolers – allows use of CV for RCP Seal Cooling following loss of SX (due to floods or other causes)
  - Created Aux Building Flooding Abnormal Operating Procedures (0/1/2BwOA PRI-8). Includes determination of SX isolation capability and impact of isolation actions.
  - AF Crosstie Modification In Progress – Allow use of opposite unit's Motor Driven AF pump

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Risk Communications

- PSA model changes or updates
  - Driven by need to reflect changes to the plant and procedures in risk-informed decision-making and insights
  - Impact the calculated benefit (CDF/LERF impact) of plant and procedure changes
  - Impact MSPI margin in both the positive and negative directions
  - Insights driving plant changes should be considered at a functional level and not at a component level because changes or updates can impact component importance
- Communication of Insights
  - A corporate alignment initiative has been developed to drive consideration of CDF insights

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2009 Impacts

- Workload continues to increase due to Burden Reduction Initiatives
- Burden Reduction is actually Burden Shift to PRA
  - EDG CT – Implemented
  - Inverter CT – Implemented
  - RI-ISI – Implemented
  - MSPI - Implemented
  - Missed Surveillances – Implemented
  - Mode Change – Implemented
  - RPS/ESFAS CT and STI - Implemented
  - Relocation of TS Surveillance to Owner Control Document – Planned 2009
  - Barriers – Planned 2009

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Near term activities

	2008 Activities	2009 Activities	2010 Activities	2011 Activities
1	RPS / ESFAS Implementation	CDBI - Byron	PRA Peer Review	PRA Periodic Update
2	B/B RM FASA	R6D Roll-out	Fire PRA Peer Review	Seismic PRA - Update
3	R6C Roll-out (Partial)	R6E (AF Xtie)	Credit for B.5.b Strategies	LPSD PRA
4	R6D Development	B/B Internal Flood Analysis	Initiative 4B	License Renewals
5	Fire PRA Work (Braidwood)	Fire PRA Work (Byron)	One-Time SX AOT for Buried Piping Replacement	Extended Power Uprates
6	~~CIV LAR~~	PRA Model Documentation	Address Peer Review Findings	10CFR50.69 (Option 2)
7	~~SI Accummulator LAR~~	Initiative 5B	2009 LAR RAIs	PARAGON Changes for 6x
8	~~RCP Flywheel Surv Interval LAR~~	TSTF-427 Barrier Assessments	RI ATWS Submittal	
9	RG 1.200 Self-Assessment	Heavy Loads in (a)(4)	CIV LAR	
10	Braidwood RI-ISI 10-Year Update	Fire & External Events in (a)(4)	RCP Flywheel Surv Interval LAR	
11	PRA Model Documentation	Byron RI-ISI Periodic Update	PARAGON Changes for 6x	
12	Update WOG Database	Security Target Sets	PARAGON 1.3 Implementation	
13	EOP Changes - HRA Notebook Revision	RI-ISI Review and RAIs		
14		SI Accummulator LAR		
15		PARAGON Changes for 6x		
16		RITS Tier 3 Decision Trees		

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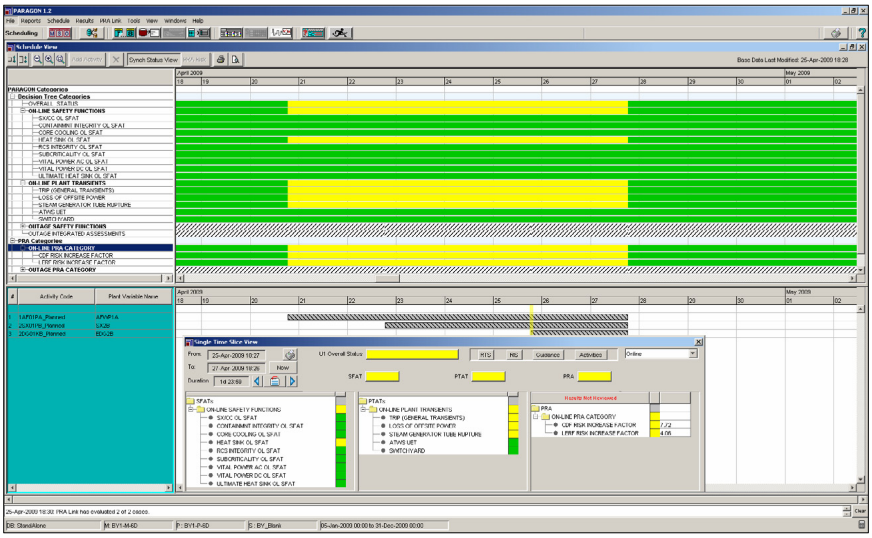
Online Maintenance Schedule



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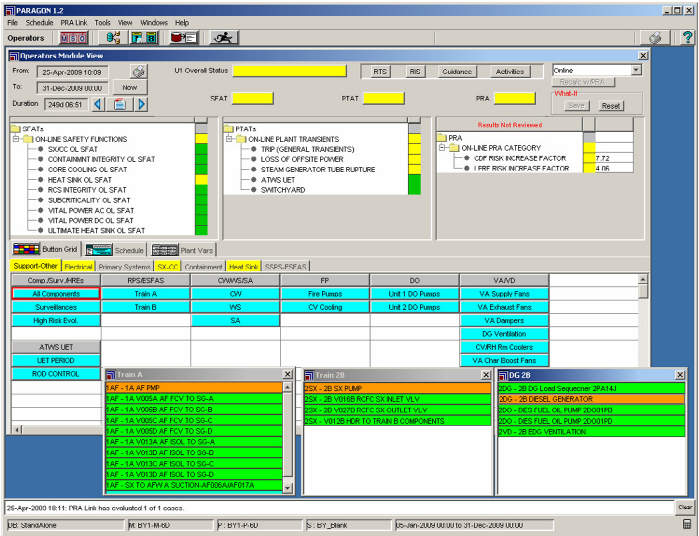
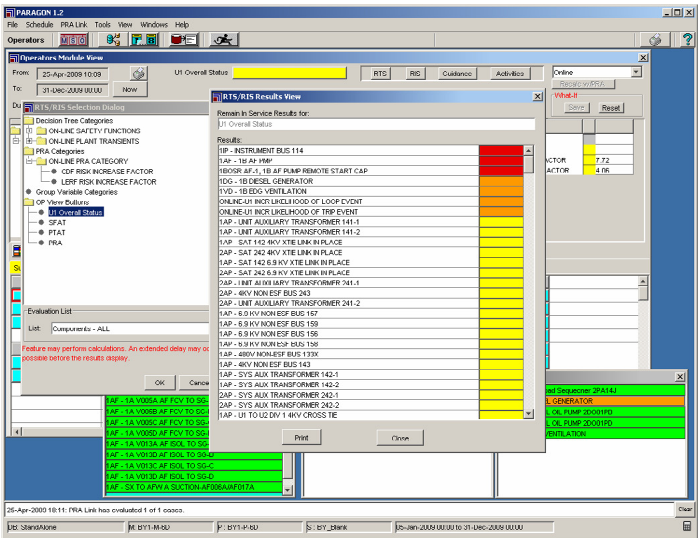
PARAGON Software



Remain-in-service (RIS)



Protected Equipment

- Methods of Posting:
  - Protective Covers (clamshells, plastic cylinders or rings)
  - Curtains (examples are tarps, paper, screens, fire resistant blankets, etc.)
  - Placement of highly visible reminders such as "little men," orange cones, easels, or reusable laminated signs
  - Magnetic placards that are placed on breaker doors, room doors, or panels to mark the protected equipment.
  - In addition, SSCs are protected by the online schedule (i.e., Train A may not be allowed to be worked on, if this is a 'B' train week).

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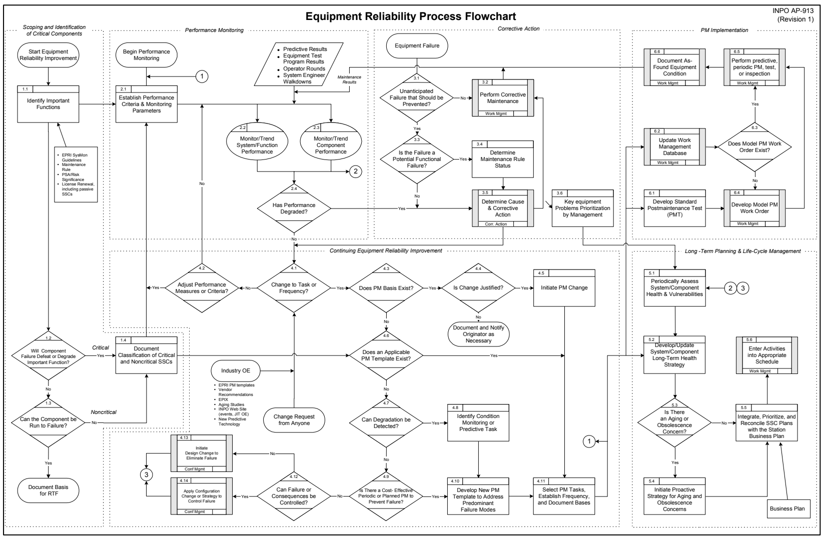
Equipment Reliability (AP-913)



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Related Programs

- Reliability Centered Maintenance (RCM)
- Maintenance Rule (MR)
- Equipment Reliability (ER)
- Nuclear Asset Management (NAM)
- Life Cycle Management (LCM)

- Risk Management (RM)
  - Defense-in-depth (DID) management
- Margin Management
- Safety Culture

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What are most critical elements?

- From overall technological perspective
  - Decision making process
  - Deterministic and probabilistic
- From the station perspective
  - Changing the mindset or perspective
  - Changing the values and culture
- For the long term consequences
  - Safety assurance perspective
  - O&M perspective
  - Design perspective

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Conclusion

- 'Why should the risk management be considered, if at all?'
- The realism: 'as-built and as-operated' is important
  - Perform the work at the station and train the station staff
- Perform the work to change the station toward:
  - The safer operation and
  - Higher confidence in operations, maintenance and planning

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THANK YOU!

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II-A-4

Tech. Spec. Optimization Study for the RPS/ESFAS at Kori Unit 2

Korea Electric Power Corporation
Korea Electric Power Research Institute

Bagsoon Chung

May 18-19, 2009 10th Korea-Japan PSA Workshop

Contents

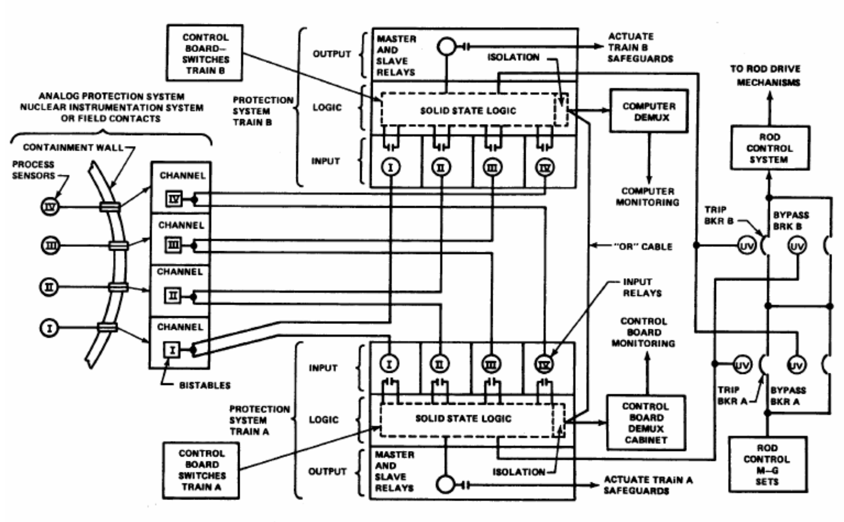
Ⅰ. Introduction & Background

- Evaluated changes to **STI and AOT** for **the analog channels, logic cabinets and slave relays** by a **Probabilistic safety assessment approach** which includes the fault tree models, signals, component reliability database, and most of the test and maintenance assumptions

  ⇒ NRC approved increasing the STI, test times, and AOTs for the analog channels as well as the AOTs for the logic cabinets, master relays, and slave relays.

- **Inadequately Frequent Testing and Short AOT**
  - **Significant Manpower and Human Factor Considerations**
  - **Adverse Effect**
    - Increasing Components Wear
    - Increasing Unavailability by Testing
    - Reactor Trip by Human Error or irrelevant
    - Inadequate Equipments Arrangement after Testing

Simplified Diagram of the Protection System



Ⅱ. Proposed Changes

Component	Current	Proposed
◦ Analog Channels		
- STI (Months)	**1**	**3**
- Test AOT (Hours)	**2**	**4**
- Maintenance AOT (Hours)	**1**	**6**
◦ Logic Cabinet		
- STI (Months)	2	2
- Test AOT (Hours)	**2**	**4**
- Maintenance AOT (Hours)	**6**	**12**
◦ Trip Breaker (No Change)		
- STI (Months)	2	2
- Test AOT (Hours)	**2**	**2**
- Maintenance AOT (Hours)	**6**	**6**
◦ Master Relay		
- STI (Months)	2	2
- Test AOT (Hours)	**2**	**4**
- Maintenance AOT (Hours)	**6**	**12**

Ⅲ. Risk Assessment

- Kori Unit 2 Plant-Specific Reliability Data (1996. 1 ~ 2005. 12)

Components	No. of Failure
NIS Power Supply (NQ_PWR)	3
NIS Summing Amplifier (NM310_SUM)	3
NIS Bistable Relay Driver (NC_BIS)	1
Channel Test Card (NCT)	2
Summing Amplifier (NSA)	1
Loop Power Supply (NLP)	1
Signal Comparator Card (NAL)	8
Lead/Lag Card (NLL)	4
Multiplier/Divider Card (NMD)	2
Flow Transmitter	2
Level Transmitter	1
Pressure Transmitter	1
Total	29

Ⅲ. Risk Assessment

- Reactor Trip Signal Unavailability

Reactor Trip Signal	Unavailability (Current)	Unavailability (Proposed)	Rate (%)
Low Feedwater Flow	4.10E−06	4.40E−06	7.2
Power Range, Neutron Flux (High)	2.66E−06	2.68E−06	0.9
Power Range, Neutron Flux (Low)	2.78E−06	2.81E−06	0.9
Neutron Flux, High Positive Rate	2.73E−06	2.75E−06	0.9
Source Range, Neutron Flux	3.52E−06	3.60E−06	2.1
Overpower ΔT (OPΔT)	3.14E−06	3.48E−06	10.8
Overtemperature ΔT(OTΔT)	3.44E−06	3.89E−06	13.0
Pressurizer Pressure − High	2.68E−06	2.68E−06	0.3
Pressurizer Pressure − Low	2.72E−06	2.73E−06	0.4
Pressurizer Water Level − High	3.32E−06	3.39E−06	2.2
Underfrequency – Reactor Coolants Pump	2.71E−06	2.72E−06	0.4
Undervoltage – Reactor Coolants Pump	2.71E−06	2.72E−06	0.4
RCS Loss of Flow	2.66E−06	2.67E−06	0.3
Steam Generator water Level – Low − Low	2.62E−06	2.63E−06	0.3
Turbine Trip	2.96E−06	2.99E−06	1.1

* Cutoff Value : 1.0E−12

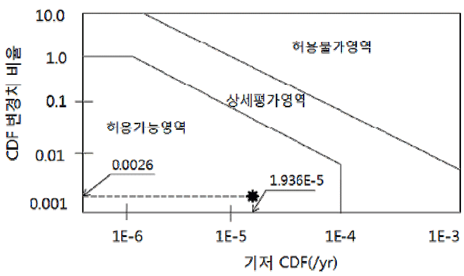
Ⅲ. Risk Assessment

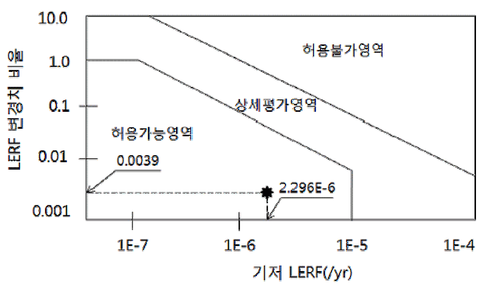
- Risk Increment by Changing STI/AOT

RISK	Current	Proposed	Increasing Rate (%)
CDF	1.936E-5	1.941E-5	0.26
LERF	2.296E-6	2.305E-6	0.39

*Truncation : 1.0E−11

- KINS/GT−N24 Acceptance Criteria (STI/AOT Change)





Ⅳ. Performance Monitoring

- **Monitored Under Maintenance Rule Program**
  - In general, the reliability of the components is monitored under Maintenance Rule Program
  - Kori Unit 2 Maintenance Rule Program will have been developed by June 2009
  - New provisions for unavailability monitoring is required Until Maintenance Rule Program working Properly.

- **Provide Reliability Goals in Tech. Spec.**
  - Develop the reliability goals and specify them in Tech. Spec.
  - If the pre-established reliability criteria is exceeded, back to a month interval

Ⅳ. Performance Monitoring

- **Developing Reliability Goals**

(for 36 months)

No. of Test Failure caused by the same component	Test Interval
Up to 1	Sustain test interval for 3 months
More than 2	Back test interval to 1 month

Ⅳ. Performance Monitoring

Unavailability Increase Rate Adding 2 failures to Base Value

RPS Signal	Analogue Channel (Bistable)	Failure No.	Unavailability (Current)	Unavailability (Proposed)	Rate (Increase; %)
Low Feedwater	FB510A	Base Value	4.10E-06	4.40E-06	7.2
		Base Value + 2	4.11E-06	4.43E-06	7.8
OPDT	TB411B	Base Value	3.14E-06	3.48E-06	10.8
		Base Value + 2	3.14E-06	3.49E-06	11.0
OTDT	TB411A	Base Value	3.44E-06	3.89E-06	13.0
		Base Value + 2	3.44E-06	3.90E-06	13.3

Ⅳ. Performance Monitoring

Performance Criteria by Related Components

Components	Performance Criteria (No. of Test failures for 36 months caused by the same components)
Bistable	1
Loop Power Supply	1
Summing Amplifier	1
Lead/Lag Card	1
Potentiometer	1
Function Generator Card	1
Multiplier/Divider	1

V. Conclusions

- The risk averted by eliminating a potential plant shutdown, can almost offset the increase in risk of the proposed changes due to increased signal unavailability while at power
- The proposed changes being considered having a minor impact on the availability of the RT and ESF actuation signals and the plant risk.
- The performance will have been monitored by provisions specified in Tech. Spec. until Maintenance Rule Program works effectively.