Proceedings of the 10th Korea-Japan Joint Workshop on PSA

- For Asian PSA Network -

2009. 5. 18 ~ 20
Haevichi Hotel & Resort
Proceedings of the 10th Korea-Japan Joint Workshop on PSA
–For Asian PSA Network–
May 18-20, 2009, Haevichi Hotel & Resort, Jeju, Korea

(Eds.) Joon-Eon YANG* and Toshimitsu HOMMA

Nuclear Safety Research Center
Japan Atomic Energy Agency
Tokai-mura, Naka-gun, Ibaraki-ken

(Received October 13, 2009)

The tenth Korea-Japan Joint Workshop on Probabilistic Safety Assessment (PSA) was held in the Jeju island of Korea, on May 18-20, 2009 organized by Korea Atomic Energy Research Institute (KAERI). The purpose of the workshop was to provide a forum for presentation and discussions on experiences and technical achievements related to PSA, risk-informed and performance-based approach, and other relevant issues in both countries.

Since the first Korea-Japan Joint Workshop on PSA started in 1992, the workshops have provided an important and timely opportunity for exchange and discussion of the relevant information to all PSA practitioners and users of risk information in the industry, research, academia and regulatory arena. This was the tenth anniversary of the Joint Workshop with the main theme of “For Asian PSA Network” and participants included those from China, Taiwan and the United States of America besides Korea and Japan.

Two keynote speeches were presented by the former chairmen of this workshop, Prof. Chang-Sun Kang of Seoul National University and Prof. emeritus Shunsuke Kondo of Tokyo University. We had two special lectures, 70 papers presented by experts at 10 technical sessions related PSA, the special session on the status of PSA in Korea, Japan, China and Taiwan and panel discussion on their cooperation in PSA. This report provides the summary of each session, and all the presentation materials presented in the 10th Korea-Japan Joint Workshop on PSA.

Keywords: PSA, Risk Informed Regulation, Severe Accident, Reactor Safety

* Korea Atomic Energy Research Institute (KAERI)
第 10 回日韓 PSA ワークショップ報文集
－アジアに向けた PSA ネットワーク－
2009 年 5 月 18 日～20 日、ヘビチホテル＆リゾート、済州島、韓国

日本原子力研究開発機構
安全研究センター

（編） Joon-Eon YANG*，本間 俊充

（2009 年 10 月 13 日受理）

第 10 回日韓 PSA ワークショップは、韓国原子力研究所の主催で 2009 年 5 月 18 日から 20 日に韓国の済州島で開催された。ワークショップの目的は、PSA、リスク情報を考慮し性能に基づくアプローチ、及び他の関連課題に関する両国の経験や技術的到達点についての発表と議論のフォーラムを提供することにあった。

1992 年に第 1 回日韓 PSA ワークショップが開始され以降、本ワークショップは産業界、研究機関、大学及び規制機関の PSA 実務者やリスク情報の利用者に、その時々に PSA 関連の情報を提供してきた。今回は、“アジアに向けた PSA ネットワーク”をテーマに第 10 回の記念すべき共同ワークショップとなり、韓国、日本両国他、中国、台湾、米国からも参加があった。

本ワークショップの前共同議長である Chang-Sun Kang ソウル大学教授及び近藤聡介東京大学名誉教授による基調講演が企画された。2 つの特別講演の他、PSA に関連した 10 の技術セッションに70件の発表があり、両国の他、中国及び台湾における PSA の現状に関する特別セッションと PSA における協力についてパネルディスカッションが開催された。本報告書には、各セッションのまとめと共に、基調講演及び特別講演を含む全発表の資料を掲載した。

---

原子力科学研究所（駐在）〒319－1195 茨城県那珂郡東海村白方白根 2－4
* Korea Atomic Energy Research Institute (KAERI)
Contents

Opening Session
Summary of Opening Session ........................................................................................................................................... 1
   Un-Chul LEE (Seoul National Univ.), Toshimitsu HOMMA (JAEA),
   Mitsumasa HIRANO(Tokyo City Univ.), Jong-In LEE (Korea Nuclear Society)

Keynote Speech .......................................................................................................................................................... 3
Current Issues and Challenges on Nuclear Safety ........................................................................................................ 5
   Chang-Sun KANG (Seoul National Univ.)
Actions Necessary to Promote Nuclear Energy Utilization for Solving Global Problems We Face ............................................................................................................................................. 17
   Shunsuke KONDO (Tokyo Univ.)

Special Lectures ......................................................................................................................................................... 27
Reflection of lesson learned from Niigata-ken Chuetsu-oki Earthquake to Seismic PSA ............................................. 29
   Katsumi EBISAWA (JNES)
Aging Related Degradation Assessment of Structures and Passive Components for Use in Performing PSAs ............................................................................................................................................. 46
   Joseph BRAVERMAN (BNL)

Technical Session
Session I-A: Risk Informed Regulation .......................................................................................................................... 57
Session I-A Summary ................................................................................................................................................... 59
   Mitsuhiro KAJIMOTO (JNES), Chang-Ju LEE (KINS)
I-A-1. Recent Revision of Regulatory Guide on Classification of Safety Importance using Risk Information .......................................................................................................................................................... 61
   A. HIDAKA (JAEA)
   Do Sam KIM (KINS)
I-A-3. A Risk-Informed and Performance-Based Approach for Improving Regulatory Inspection Program in Korea .................................................................................................................................................... 78
   Dae-Wook CHUNG (KINS)
   Huichang YANG (ENESYS Co., Ltd.)
   Tsuyoshi UCHIDA (JNES)
   Yong Suk LEE (FNC Technology Co., Ltd.)

Session I-B: PSA Methodology .................................................................................................................................... 113
Session I-B Summary ................................................................................................................ 115
     Katsunori OGURA (JNES), Hak-Kyu LIM (KOPEC)
I-B-1. Truncation Error Evaluation for a PSA Model ...................................................... 117
     Jong-Soo CHOI (KINS)
I-B-2. Usage of Information Criterion for Reducing Modeling Uncertainty in Reactor Safety 126
     Akira YAMAGUCHI (Osaka Univ.)
I-B-3. The Development of a 3-D Risk Matrix for Qualitative Maintenance Risk Management ........................................................................................................................................... 136
     Pi-Lin HSU (INER)
I-B-4. Some approaches for quantification of important factors in PSA for digital I&C systems 144
     Man Cheol KIM (KAERI)
I-B-5. Effect Estimation of an Automatic Periodic Tests in NPP Digital I&C Systems by
     Fault Injections .................................................................................................................. 151
     Seung Jun LEE (KAERI)
I-B-6. An Approach for Accident Event Sequence Analysis by Different Phases in
     Nuclear Power Plant .......................................................................................................... 162
     Yu YU (INET)

Session I-C: Severe Accident Management & Level 3 PSA(I) ................................................ 169
Session I-C Summary .............................................................................................................. 171
     Toshimitsu HOMMA (JAEA), Kwang-II AHN (KAERI)
I-C-1. Development of Simplified Risk Measure Based on Dose (SiRD) ......................... 173
     Dong-ha KIM (KAERI)
I-C-2. The Development of a Relationship Framework between LERF and Level-3 PSA .... 182
     Kyungmin KANG (KINS)
     Toshimitsu HOMMA (JAEA)
I-C-4. Optimization of Relocation Decisions using the Method of Probabilistic Accident
     Consequence Assessment ................................................................................................. 203
     Shogo TAKAHARA (JAEA)
I-C-5. Development of an Off-site Risk Assessment Tool for the Risk-Informed Application .. 209
     Jongtae JEONG (KAERI)

Session II-A: Risk Informed Application (I) ........................................................................ 217
Session II-A Summary ............................................................................................................. 219
     Hidetaka IMAI (TEPCO), Dae-Wook CHUNG (KINS)
II-A-1. Analysis and Evaluation of Accident Sequence Precursor ..................................... 221
     Hiroaki SHIMOZAKI (JNES)
II-A-2. Analysis of Risk Change by adding Bypass Function into RPS/ESFAS ...................... 229
     Myung-Ki KIM (KEPRI)
II-A-3. Risk Management at the NPP .................................................................................. 236
     Young H. IN (ERIN)
II-A-4. Tech. Spec. Optimization Study for the RPS/ESFAS at Kori Unit 2 ....................... 246
     Bag Soon CHUNG (KEPRI)

Session II-B: PSA Softwares ................................................................................................. 253
Session II-B Summary ............................................................................................................. 255
Akira YAMAGUCHI (Osaka Univ.), Jongtae JEONG (KAERI)

II-B-1. Development of a PSA Tool for an Interim Storage Facility of Spent Fuels .......................... 257
   Jongtae JEONG (KAERI)

   Toshimitsu HOMMA (JAEA)

II-B-3. Development of Condition Monitoring and Diagnosis System for Standby Diesel Generator ................................................................. 272
   Kwang-Hee CHOI (KEPRI)

II-B-4. W-RIMS Development ........................................................................................................ 280
   Myungsu KIM (KHNP)

II-B-5. Pilot Application of ORION Program ............................................................................ 294
   Hae Cheol OH (KEPRI)

Session II-C: Severe Accident Management & Level -3 PSA (II) ......................................................... 303

Session II-C Summary .................................................................................................................. 305
   Mitsuhiro KAJIMOTO (JNES), JinHo SONG (KAERI)

II-C-1. Severe Accident and Accident Management Study at JNES ............................................. 307
   Mitsuhiro KAJIMOTO (JNES-Japan)

II-C-2. Recent Regulatory Activities Related to Severe Accident Management in Korea ............ 320
   Tae-hyeong KIM (KINS-Korea)

II-C-3. Consideration of Emergency Source Terms Pebble-bed for High Temperature
   Gas-Cooled Reactor .............................................................................................................. 334
   Tao LIU (INET-China)

II-C-4. Probabilistic Estimation of the Early and Cancer Fatality Risks at the Korean NPPs ...... 341
   Do Sam KIM (KINS-Korea)

II-C-5. Implementation of a Molten Core Cooling Strategy in a Severe Accident Management Guideline ............................................................................. 355
   JinHo SONG (KAERI-Korea)

Session III-A: PSA & Applications .......................................................................................... 365

Session III-A Summary ............................................................................................................. 367
   Jin-Hee PARK (KAERI), Jong-Soo CHOI (KINS)

III-A-1. Upgrade of Internal Events PSA Model using the AESJ Level-1 PSA Standard for
   Operating State ................................................................................................................... 369
   Eisuke SATO (TEPCO SYSTEMS Corp.-Japan)

   Application ......................................................................................................................... 381
   Seok Won HWANG (KHNP-Korea)

   Inn Seock KIM (ISSA Technology-USA)

III-A-4. Development of Regulatory PSA Model MPAS in Korea ............................................. 400
   Jin Hee Park (KAERI-Korea)

Session III-B: Fire PSA ........................................................................................................... 407

Session III-B Summary ............................................................................................................. 409
   Tsuyoshi UCHIDA (JNES), Moon-Hak JEE (KEPRI)
III-B-1. Development of fire PSA Database System ................................................................. 411
Dongkyu KIM (KOPEC-Korea)

System ........................................................................................................................................ 421
Katsunori OGURA (JNES-Japan)

III-B-3. A Comparative Study of Two Quantification Algorithms and Importance
Measures in a Fire PRA Model ...................................................................................................... 428
Kilyoo KIM (KAERI-Korea)

III-B-4. Improved Fire-PSA with Quantitative Fire Risk Assessment ......................................... 433
Moon-Hak JEE (KEPRI-Korea)

Session III-C: Severe Accident & Safety Analysis (I) ................................................................. 443

Session III-C Summary ............................................................................................................. 445
Tao LIU (INET), Han-Chul KIM (KINS)

III-C-1. Evaluation of MELCOR code for THAI-HM2 Test ..................................................... 447
Jung-Jae LEE (KINS-Korea)

III-C-2. MELCOR Improvement and Applications ................................................................. 459
Mitsuhiro KAJIMOTO (JNES-Japan)

III-C-3. Validation of MELCOR Iodine Chemistry Model with BIP Test Data ......................... 468
Dong Ju JANG (KINS-Korea)

III-C-4. Analysis of RCS Feed & Bleed Operation to Mitigate a Severe Accident for
OPR1000 .................................................................................................................................... 477
Rae-Joon Park (KAERI-Korea)

Session IV-A: GEN-IV & Non NPP Applications ................................................................. 487

Session IV-A Summary ............................................................................................................. 489
Yoshinori UEDA (JNES), Tae Woon KIM (KAERI)

Yoshinori UEDA (JNES-Japan)

Hitoshi TAMAKI (JAEA-Japan)

using the DRSM ......................................................................................................................... 509
Tae-Ho WOO (Seoul National Univ.-Korea)

IV-A-4. Continuous Markov Chain Monte Carlo Method for Level 2 PSA of Liquid Metal
Reactor ..................................................................................................................................... 522
Akira YAMAGUCHI (Osaka Univ.-Japan)

Tae Woon KIM (KAERI-Korea)

Session IV-B: Maintenance Rule ............................................................................................ 541

Session IV-B Summary ............................................................................................................. 543
Chung-Kung LO (INER), Dong Wook JERNG (KHNP)

IV-B-1. Risk Importance Determination Process of CANDU Maintenance Rule Function ...... 545
Mi-Ro SEO (KEPRI-Korea)

IV-B-2. Development of the Tool Infrastructure for Implementation of the Maintenance Rule . 550
Chung-Kung LO (INER-Taiwan)
IV-B-3. Insights on the Application of a Standard Program for Maintenance Effectiveness
  Monitoring ..................................................................................................................... 559
Dong Wook Jerng (KHNP-Korea)
IV-B-4. Implementation of Maintenance Rule for CANDU Units ............................................. 567
Myung-Ki KIM (KEPRI-Korea)
IV-B-5. Performance Criteria Development for PHWR Maintenance Rule ............................... 582
Mi-Ro SEO (KEPRI-Korea)

Session IV-C: Severe Accident & Safety Analysis (II) ............................................................ 589

Session IV-C Summary ............................................................................................................. 591
Mitsuhiro KAJIMOTO (JNES), Rae-Joon PARK (KAERI)
IV-C-1. A Preliminary Analysis of Safety Depressurization Capability for Shin-Ulchin
  Units 1&2 ....................................................................................................................... 593
  Han-Chul KIM (KINS-Korea)
IV-C-2. Analyses of Containment Source Term of BWR5 considering Iodine Chemistry in
  Suppression Pool with THALES-2 Code ........................................................................ 606
  Jun ISHIKAWA (JAEA-Japan)
IV-C-3. Comparison of ISAAC T/H Behavior with CATHENA for CANDU Large LOCAs .... 612
  Dong-ha KIM (KAERI-Korea)
IV-C-4. Development of T/H Uncertainty Analysis S/W MOSAIQUE ................................. 626
  Ho-gon LIM (KAERI-Korea)
IV-C-5. Thermal-Hydraulic Analysis for LBLOCA in OPR1000 and Application of RSM
  for PSA ........................................................................................................................... 640
  Tae-jin KIM (Seoul National Univ.-Korea)

Session V-A: Risk Informed Application (II) ......................................................................... 651

Session V-A Summary ............................................................................................................... 653
Akihide HIDAKA (JAEA), Myung-Ki KIM (KEPRI)
  Namduk SUH (KINS-Korea)
  Hidetaka IMAI (TEPCO-Japan)
V-A-3. Increase Safe and Operating Reliability by Development of Planned Outage
  Process Standardization for Nuclear Power Plant ......................................................... 670
  Jong-hyuck PARK (KEPRI-Korea)
V-A-4. Safety and Economic Results of RI-ISI Program at Ulchin Units 3&4 ......................... 677
  Bagsoon CHUNG (KEPRI-Korea)

Session V-B: Seismic PSA ...................................................................................................... 683
Session V-B Summary ............................................................................................................. 685
Katsunori OGURA (JNES), In-Kil CHOI (KAERI)
V-B-1. Study on Effects of Correlative Degree of Component Damages on Seismic PSA
  during Component Outage ......................................................................................... 687
  Yasuhiro KATAGIRI (NEL-Japan)
V-B-2. Recent R&D Activities on Seismic PSA in KAERI ..................................................... 693
In-Kil CHOI (KAERI-Korea)
V-B-3. Radiological Consequence Analysis for Seismic Events in BWR Plants ......................... 707
Kyoko FUNAYAMA (JNES-Japan)
V-B-4. A Study on the Uncertainty of Seismic Hazard in the PSHA for a Korean NPP Site .... 714
HyunMe Rhee (KAERI-Korea)

Appendix 1 Special Session: Status of PSA in each country or proposal for cooperation ..... 725
Appendix 2 Panel Discussion: How to Improve the Cooperation in PSA ................................. 759
Appendix 3 List of Participants .............................................................................................. 763
目次

開会式
開会式概要 ................................................................................................................................. 1

Un-Chul LEE (ソウル大学), 本間 俊充 (JAEA),
平野 光将 (東京都市大学), Jong-In LEE (韓国原子力学会)

基調講演
原子力安全の最新の課題と動向 ........................................................................................................ 5

Chang-Sun KANG (ソウル大学)
我々が直面するグローバルな問題の解決に向けて原子力エネルギー利用の促進に必要な活動 ................................................................................................................................. 17

近藤駿介 (東京大学)

特別講演
新潟県中越沖地震から得られた知見の地震 PSA への反映 .............................................................. 29

蛭沢勝三 (JNES)

PSA 実施における経年劣化評価の利用 .......................................................................................... 46

Joseph BRAVERMAN (ブロックヘブン国立研究所)

テクニカルセッション
セッション I-A: リスク情報を活用した規制 .............................................................................. 57

セッション I-A のまとめ .................................................................................................................. 59

議長：樋口光廣 (JNES), Chang-Ju LEE (KINS)

I-A-1. リスク情報を活用した「重要度分類指針」の最近の改訂 ........................................... 61

日高昭秀 (JAEA)

I-A-2. 韓国原子力発電所に対する性能目標の策定 ................................................................ 66

Do Sam KIM (KINS)

I-A-3. 勧告における検査プログラム及び規制体系改善のためのリスク情報の活用
及びパフォーマンスに基づく手法について .............................................................................. 78

Dae-Wook CHUNG (KINS)

I-A-4. 検査所見に対するリスク評価プログラムSEIFの開発 ........................................... 86

Huichang YANG (ENESYS Co., Ltd.)

I-A-5. 日本におけるリスク情報を活用した規制検査精度の改善 ........................................ 94

Tsuyoshi UCHIDA (JNES)

I-A-6. 段階的規制におけるリスク情報を活用した性能指標及びしきい値に関する
検討 .......................................................................................................................................... 103

Yong Suk LEE (Future and Challenge)

Session I-B: PSA の方法論 ........................................................................................................ 113

Session I-B のまとめ .................................................................................................................. 115
議長：小倉克規 (JNES), Hak-Kyu LIM (KOPEC)
I-B-1. PSAモデルにおける打切り誤差の評価 ......................................................... 117
    Jong-SoO CHOI (KINS)
I-B-2. 原子炉安全におけるモデル化による不確実さ低減のための情報基準の利用 .......... 126
    山口彰 (Osaka Univ.)
I-B-3. 定性的保守リスク管理のための3次元リスクマトリックスの開発 .......................... 136
    Pi-Lin HSU (INER)
I-B-4. デジタル制御機器システムに関する重要な要因の定量化手法の検討 .................. 144
    Man Cheol KIM (KAERI)
I-B-5. 異常注入による原子力発電所のディジタル制御機器に対する自動定期検査
    の効果推定 ........................................................................................................ 151
    Seung Jun LEE (KAERI)
I-B-6. 原子力発電所の異なる段階における事故シーケンス解析の検討 .......................... 162
    Yu YU (INET)

Session I-C: シビアクシシメントマネジメントとレベル3PSA (I)....................... 169
Session I-C のまとめ ......................................................................................... 171
議長：本間俊充 (JAEA), Kwang-II AHN (KAERI)
I-C-1. 線量評価に基づく簡易リスク測定法(SiRD)の開発 ......................................... 173
    Dong-ha KIM (KAERI)
I-C-2. LERFとレベル3PSAの関係についてのフレームワークの構築 .......................... 182
    Kyungmin KANG (KINS)
    本間俊充 (JAEA)
I-C-4. レベル3PSAを用いた移転の導入と解除に関する最適化 .................................... 203
    高原省五 (JAEA)
I-C-5. リスク情報活用のためのオフサイトにおけるリスク評価ツールの開発 ........... 209
    Jongtae JEONG (KAERI)

Session II-A: リスク情報とその応用 (I) .............................................................. 217
Session II-A のまとめ ......................................................................................... 219
議長：今井英隆 (TEPCO), Dae-Wook CHUNG (KINS)
II-A-1. 事故シーケンス前兆事象の解析及び評価 ..................................................... 221
    下崎敬明(JINES)
II-A-2. 原子力防護系/工学的安全施設系にバイパス機能を付加した場合のリスク変化解
    析 ...................................................................................................................... 229
    Myung-Ki KIM (KEPRI)
II-A-3. 原子力発電所におけるリスク管理 .............................................................. 236
    Young H. IN (ERIN)
II-A-4. Kori2号機の原子力防護系/工学的安全施設系に対する技術仕様書最適化に関す
    る検討 ............................................................................................................ 246
    Bag Soon CHUNG (KEPRI)

Session II-B: PSA ソフトウェア ........................................................................ 253
Session II-B のまとめ ......................................................................................... 255
議長：山口彰 (大阪大学), Jongtae JEONG (KAERI)
II-B-1. 使用済み燃料中間貯蔵施設のための確率論的安全評価ツールの開発 .......................... 257
   Jongtae JEONG (KAERI)
II-B-2. グローバル感度解析コードGSALabの開発 ................................................................. 265
   本間俊充 (JAEA)
II-B-3. 予備ディーゼル発電機の状態モニタリング及び診断システムの開発 .......................... 272
   Kwang-Hee CHOI (KEPRI)
II-B-4. ウェブ基盤リスクモニタリングシステムの開発 ............................................................. 280
   Myungsu KIM (KHNP)
II-B-5. ORIONプログラムの試験利用 .................................................................................. 294
   Hae Cheol OH (KEPRI)

Session II-C: シビアアクシシデントマネジメントとレベル 3PSA (II) .............................................. 303

Session II-C のまとめ .................................................................................................................. 305

IIC-C-1. シビアアクシシデント及びアクシシデントマネジメントに関するJNESの研究 .......... 307
    桟本光宝（JNES-Japan）
IIC-C-2. 近年の韓国におけるシビアアクシシデントマネジメントに関する規制活動 .......... 320
    Tae-hyeong KIM (KINS-Korea)
IIC-C-3. 高温ガス炉の緊急時計画のためのソースタームの検討 .............................................. 334
    Tao LIU (INET-China)
IIC-C-4. 韓国の原子力発電所における急性死亡リスク及びがん死亡リスクの確率論的評価 .......... 341
    Do Sam KIM (KINS-Korea)
IIC-C-5. シビアアクシシデントのマネジメントガイドラインにおけるMolten Core
    Cooing Strategyの実施 .............................................................................................................. 355
    JinHo SONG (KAERI-Korea)

Session III-A: PSA とその応用 ........................................................................................................ 365

Session III-A のまとめ .................................................................................................................. 367

III-A-1. 運転状態に関する日本原子力学会レベル1PSA標準を用いた内部事象PSA
    モデルの改善 .......................................................................................................................... 369
    佐藤英介（TEPCO SYSTEMS Corp.-Japan）
III-A-2. ウェブベースのプラント信頼性データ情報システムの開発とその応用 .......... 381
    Seok Won HWANG (KHNP-Korea)
III-A-3. TMI-2事故の経験に基づく保守規則の安全上の意味 .............................................. 388
    Inn Seock KIM (ISSA Technology-USA)
III-A-4. 韓国における規制局用PSAモデルMPASの開発 .................................................. 400
    Jin-Hee Park (KAERI-Korea)

Session III-B: 火災 PSA .................................................................................................................. 407

Session III-B のまとめ .................................................................................................................. 409

III-B-1. 火災PSAデータベースの整備 ...................................................................................... 411
    Dongkyu KIM (KOPEC)
III-B-2. 火災 PSA 手法及び火災解析コードの開発 .............................................................. 421
   小倉克規 (JNES)
III-B-3. 火災 PRA モデルに関する 2つの定量化アルゴリズム及び重要度指標の包括的
   研究 ..................................................................................................................... 428
   Kiyoo KIM (KAERI)
III-B-4. 定量的火災リスク評価を用いた火災 PSA の改良 ........................................... 433
   Moon-Hak JEE (KEPRI)

Session III-C: シビアアクシデントと安全解析 (I) ......................................................... 443

Session III-C のまとめ .................................................................................................. 445
   議長：Tao LIU (INET), Han-Chul KIM (KINS)
III-C-1. MELCOR コードを用いた THAI-HM2 試験の評価 ....................................... 447
   Jung-Jae LEE (KINS-Korea)
III-C-2. MELCOR の改善と応用 .............................................................................. 459
   根本光宗 (JNES-Japan)
III-C-3. BIP 試験名データに基づく MELCOR よう素化学モデルの検証 ..................... 468
   Dong Ju JANG (KINS-Korea)
III-C-4. OPR1000 相におけるシビアアクシデント緩和のための RCS Feed & Bleed
   Operation の分析 .................................................................................................. 477
   Rae-Joon Park (KAERI-Korea)

Session IV-A: 次世代炉及び原子力関連施設 .................................................................. 487

Session IV-A のまとめ .................................................................................................. 489
   議長：上田喜徳 (JNES), Tae Woon KIM (KAERI)
IV-A-1. 核燃料サイクル施設を対象としたリスク評価研究の概要 ............................... 491
   上田喜徳 (JNES)
IV-A-2. 再処理施設 PSA のための機器故障率の整備 .................................................. 502
   玉置等史 (JAEA)
IV-A-3. DRSM を用いた超高温炉 (VHTR) の動的安全評価 ...................................... 509
   Tae-Ho WOO (ソウル大学)
IV-A-4. 連続マルコフ連鎖モンテカルロ法を用いた液体金属冷却原子炉のレベル
   2 PSA .................................................................................................................. 522
   山口彰 (大阪大学)
IV-A-5. リスク情報を活用した KAIMER-600 相に関する事故シナリオ解析 .................. 529
   Tae Woon KIM (KAERI)

Session IV-B: 保全規則 ................................................................................................. 541

Session IV-B のまとめ .................................................................................................. 543
   議長: Chung-Kung LO (INER), Dong Wook JERNG (KHNP)
IV-B-1. リスク重要度を考慮した CANDU 炉の保全規則機能の決定過程 .................. 545
   Mi-Ro SEO (KEPRI)
IV-B-2. 保全規則を実施するための手法基盤の開発 .................................................. 550
   Chung-Kung LO (INER-Taiwan)
IV-B-3. 保全有効性モニタリングのための標準プログラム適用に係わる知見 ................ 559
   Dong Wook Jerng (KHNP)
IV-B-4. CANDU炉用保全規則の実施 ................................................. 567
   Myung-Ki KIM (KEPRI)
IV-B-5. 加圧水型重水炉(PHWR)の保守規則要性能基準の策定 ................. 582
   Mi-Ro SEO (KEPRI)

Session IV-C: シビアアクシデントと安全解析 (II) ................................ 589
Session IV-C のまとめ ........................................................................... 591

議長：梶本光広 (JNES), Rae-Joon PARK (KAERI)
IV-C-1. Shin-Ulchin 1号機及び2号機の安全減圧能力に関する予備的解析 ... 593
   Han-Chul KIM (KINS)
IV-C-2. THALES-2による圧力抑制プールでのヨウ素化学を考慮したBWR5のソース
   タームの解析 ................................................................. 606
   石川淳(JAEA)
IV-C-3. CATHENAによるCADU炉での大破断LOCAに関するISAAC熱流体挙動の比較
   612
   Dong-ha KIM (KAERI)
IV-C-4. 熱流体不確実さ解析S/W MOSAIQUEの開発 ................................ 626
   Ho-gon LIM (KAERI)
IV-C-5. OPR1000におけるLBLOCAの熱流体解析とRSMのPSAへの適用 .......... 640
   Tae-jin KIM (ソウル大学)

Session V-A: リスク情報の適用状況 (II)................................................. 651
Session V-A のまとめ ........................................................................... 653

日高 昭秀 (JAEA), Myung-Ki KIM (KEPRI)
V-A-1. 韓国におけるリスク情報を活用した規制実施に係わる現状 ................ 655
   Namduk SUH (KINS)
V-A-2. 東京電力における停止時リスクモニタリング ................................ 665
   今井 英隆 (TEPCO)
V-A-3. 計画された待機除外手順の標準化による安全と運転信頼性の向上 .......... 670
   Jonghyuck PARK (KEPRI)
V-A-4. Ulchin3号炉及び4号炉においてリスク情報を活用した教養期間中検査を実
   施した場合の安全及び経済的効果 ...................................... 677
   Bag Soon CHUNG (KEPRI)

Session V-B: 地震 PSA ........................................................................... 683
Session V-B のまとめ........................................................................... 685

小倉克規 (JNES), In-Kil CHOI (KAERI)
V-B-1. 供用停止期間の地震PSAにおける機器損傷の相関度効果の研究 .................. 687
   Yasuhiro KATAGIRI (NEL)
V-B-2. 近年の韓国における地震PSAのR&D活動 ................................... 693
   In-Kil CHOI (KAERI)
V-B-3. BWRにおける地震時の放射線影響の分析 ..................................... 707
   舟山京子(JNES)
V-B-4. 韓国の原子力発電所を対象としたPSIAにおける地震ハザードの不確実さに関
   する研究 .......................................................................... 714
   HyunMe Rhee (KAERI)
| Appendix  1 | 特別セッション: 各国のPSAの状況と連携への提案 .................................. 725 |
| Appendix  2 | パネルディスカッション: PSA分野における連携の改善に向けて ........ 759 |
| Appendix  3 | 参加者名簿 ................................................................................. 763 |
Summary of Opening Session

Opening remarks (Un-Chul LEE)
Prof. Lee introduces participants from China and Taiwan. Prof. Lee emphasized the role of Korea-Japan Joint Workshop on PSA (KJPSA) in the cooperation between Korea and Japan in the PSA area. As the slogan “For Asian PSA Network” says, Prof. Lee suggests close collaboration among Asian countries to improve the PSA technology, especially in the newly expanded area of PSA such as risk-based technology and risk-informed regulation, application, and design. Prof. Lee wishes every participant has good time during the workshop.

Welcome Address (Toshimitsu HOMMA)
On behalf of all Japanese participants, Dr. Homma expresses gratitude to the organizers in Korean side (KAERI, KINS, KHNP, KEPRI, and KOPEC). Dr. Homma mentioned that the organizing committee has done wonderful work in the preparation of the workshop. Dr. Homma expressed special thanks to general secretary Dr. Yang and Mr. Han and Dr. Choi. Dr. Homma introduces Dr. Kajimoto of JNES for giving the message from Prof. Hirano, one of the co-chair of the workshop.

Welcome Address (Message from Prof. Mitsumasa HIRANO, read by Dr. Mitsuhiro KAJIMOTO)
Prof. Hirano says he is sorry for not being able to attend the workshop due to the new flu. Prof. Hirano introduces the PSA-related activities of Japanese Nuclear Safety Commission. Prof. Hirano also introduces the development of a standard for the PSA technology by Atomic Society of Japan. Prof. Hirano mentions the importance of the PSA technology for the introduction of risk-informed regulation. Prof. Hirano emphasizes the importance of continued effort in the advancement of the PSA technology and the sharing of information, and the contribution of the workshop for those purposes.

Congratulatory Address (Dr. Jong-In LEE)
Dr. Lee expresses his gratitude to the honorary chairs and the chairs of the workshop. Dr. Lee says that he is a member of the workshop and he also participated previous workshops several times. Dr. Lee mentions the importance the PSA technology in the practical application of risk-informed regulation. Dr. Lee says that he expects the workshop to provide important and timely discussions to all PSA and RIR-related people. Dr. Lee also says that he hope all the participants have constructive discussions on the establishment of the PSA network in Asia.
This is a blank page.
Keynote Speech
Chang-Sun KANG
Shunsuke KONDO
This is a blank page.
Current Issues and Challenges on Nuclear Safety

May 18, 2009
C-S. Kang
Professor Emeritus, Seoul National University
Member, INSAG-IAEA
Special Advisor to the President, KAERI

Contents of Presentation

- INSAG Activities
- Challenges to Current and Emerging Key Safety Issues
  - Global Nuclear Safety Regime
  - Operational Nuclear Safety
  - Relationship between Nuclear Safety and Security
  - PSA Update
What is INSAG?

**1985-2003:** International Nuclear Safety Advisory Group (INSAG)
- Advising DG of IAEA on Nuclear Safety

**2003-date:** International Nuclear Safety Group (INSAG)
- Independent from IAEA-DG
  - INSAG will provide authoritative advice and guidance on nuclear safety approaches, policies and principles.
  - INSAG will provide recommendations and opinions on current and emerging nuclear safety issues to the IAEA, the nuclear community and the public.


- Asmolov, Director, Kurchatov Institute, Russian Federation
- Alonso, Chair of Nuclear Technology, Politechnical University, Spain
- Echávarri, Director-General, OECD/NEA
- Kang, Nuclear Engineering Department, Seoul National University, Korea
- Laaksonen, Director General, Radiation and Nuclear Safety Authority (STUK), Finland
- Meserve, President, Carnegie Institute of Washington, U.S.A.
- Sharma, Chairman, Atomic Energy Regulatory Board, India
- Torgerson, President, AECL, Canada
- Birkhofer, ISAr Institute for Safety and Reliability GmbH, Germany
- Lauvergeon, Chairperson, AREVA, France
- Chang, General Manager, China Power Investment Corporation, China
- Drabea, President, State Office for Nuclear Safety (SUJB), Czech Republic
- Rising, Vice-President, Vattenfall Ab, Sweden
- Weightman, HM Chief Inspector, Nuclear Safety Directorate, HSE, UK
- Suzuki, Chairperson, Nuclear Safety Commission, Japan
- Couto, Head of Nuclear Regulation and Licensing, Argentine
- Slabber, PBMR Technical Director, South Africa
Documents Produced by INSAG

<table>
<thead>
<tr>
<th>Document</th>
<th>Title</th>
<th>Year</th>
</tr>
</thead>
<tbody>
<tr>
<td>75-INSAG-1</td>
<td>Summary report on the post-accident review meeting on the Chernobyl accident</td>
<td>1986</td>
</tr>
<tr>
<td>75-INSAG-2</td>
<td>Radionuclide source terms from severe accidents to nuclear power plants with light water reactors</td>
<td>1987</td>
</tr>
<tr>
<td>75-INSAG-3</td>
<td>Basic safety principles for nuclear power plants</td>
<td>1988</td>
</tr>
<tr>
<td>75-INSAG-4</td>
<td>Safety culture</td>
<td>1991</td>
</tr>
<tr>
<td>75-INSAG-5</td>
<td>The safety of nuclear power</td>
<td>1992</td>
</tr>
<tr>
<td>75-INSAG-6</td>
<td>Probabilistic safety assessment</td>
<td>1992</td>
</tr>
<tr>
<td>75-INSAG-7</td>
<td>The Chernobyl accident: Updating of INSAG-1</td>
<td>1993</td>
</tr>
<tr>
<td>INSAG-8</td>
<td>A common basis for judging the safety of nuclear power plants built to earlier standards</td>
<td>1995</td>
</tr>
<tr>
<td>INSAG-9</td>
<td>Potential exposure in nuclear safety</td>
<td>1995</td>
</tr>
<tr>
<td>INSAG-10</td>
<td>Defense in depth in nuclear safety</td>
<td>1996</td>
</tr>
<tr>
<td>INSAG-11</td>
<td>The safe management of sources of radiation: Principles and strategies</td>
<td>1999</td>
</tr>
<tr>
<td>INSAG-12</td>
<td>Basic safety principles for nuclear power plants INSAG-3 Rev. 1</td>
<td>1999</td>
</tr>
<tr>
<td>INSAG-13</td>
<td>Management of operational safety in nuclear power plants</td>
<td>1999</td>
</tr>
<tr>
<td>INSAG-14</td>
<td>Safe management of the operating lifetimes of nuclear power plants</td>
<td>1999</td>
</tr>
<tr>
<td>INSAG-15</td>
<td>Key practical issues in strengthening safety culture</td>
<td>2002</td>
</tr>
<tr>
<td>INSAG-16</td>
<td>Maintaining knowledge, training and infrastructure for research and development in nuclear safety</td>
<td>2003</td>
</tr>
<tr>
<td>INSAG-17</td>
<td>Independence in regulatory decision making</td>
<td>2003</td>
</tr>
<tr>
<td>INSAG-18</td>
<td>Managing change in the nuclear industry: The effects on safety</td>
<td>2003</td>
</tr>
<tr>
<td>INSAG-19</td>
<td>Maintaining the design integrity of nuclear installations throughout their operating life</td>
<td>2003</td>
</tr>
<tr>
<td>INSAG-20</td>
<td>Stakeholder involvement on nuclear issues</td>
<td>2006</td>
</tr>
<tr>
<td>INSAG-21</td>
<td>Strengthening the global nuclear safety regime</td>
<td>2006</td>
</tr>
<tr>
<td>INSAG-22</td>
<td>Nuclear safety infrastructure for a national nuclear power programme supported by the IAEA Fundamental Safety Principles</td>
<td>2008</td>
</tr>
<tr>
<td>INSAG-23</td>
<td>Improving the international system for operating experience feedback</td>
<td>2008</td>
</tr>
<tr>
<td>INSAG-24</td>
<td>Relationship between safety and security (in the final draft form)</td>
<td>2009</td>
</tr>
<tr>
<td>INSAG-25(?)</td>
<td>probably on PSA update</td>
<td>2009</td>
</tr>
</tbody>
</table>

Key Safety Issues under Discussion

- Global Nuclear Safety Regime: INSAG-21 and 22
- Operational Nuclear Safety: INSAG-23
- Relationship between Nuclear Safety and Security: INSAG-24 (Draft)
- PSA Update: Potential INSAG-25 (?)
Challenges to Key Safety Issue: “Strengthening Global Safety Regime”

- Global Harmonization of Nuclear Safety
- Establishment of Nuclear Safety Infrastructure for Newcomers
- Trade Practice between Exporter and Importer of Nuclear Systems
- More Legally Binding Mechanism for Global Safety
- International or Multinational Safety Review (ex. MDEP)

Challenges to Key Safety Issue: “Enhancing Operational Nuclear Safety”

- Operating Experience Feedback
  - Comprehensive analysis of events
  - Proper dissemination of results
- Share of knowledge
  - Up-to-date safety related R&D results and operating experiences
- Complacency
  - Continuing investments in staff, systems and equipment
- Life Extension and Power Ascension
  - Ageing and Safety Margins
- Reliance on Contractors
  - Operator’s responsibility of controlling contractors
- Safety Culture
  - Leadership and Management
  - Operators
  - Independent External Review
Security: Terrorists' Attack (9/11/01)
relates to the prevention, detection and response to malevolent acts, theft and sabotage which could lead to accidents or threats of causing accidents.

Challenges to Key Safety Issue:
“Developing the Relationship Between Nuclear Safety and Security”

What is Nuclear Safety?
What is Nuclear Security?

- **Nuclear Safety** - “the achievement of proper operating conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of workers, the public and the environment from undue radiation hazards”.

- **Nuclear Security** - “the prevention and detection of and response to theft, sabotage, unauthorized access, illegal transfer or other malicious acts involving nuclear material, other radioactive substances or associated facilities”.
Nuclear Security

- Nuclear security has been focused on:
  - nuclear weapons,
  - NPT regime and disarmament.
- New dimensions
  - Post cold war
  - Post 9/11
- No clear distinction among:
  - safeguards,
  - safety, and
  - security.

Security: Changes and Challenges
- Post Cold War, Post 9/11 -

Main Players
- Nation-states
- Bi-polar superpowers
- Gov-industry-military complex

Sub-national, Non-states, small states
Global network
Mass media and public

Threats
- High density, high intensity, centralized
- Lower Probability
- Physical overkill

Low density, low intensity, decentralized
Higher probability
Socio-psychological terror

Motives
- Geopolitical
- Predictable - calculable

Malevolent
Unpredictable and incalculable
**Security: World Inventories of Nuclear Facilities and Materials**

- Pu >1,670 tons civil, 155 tons military
- HEU >175 tons civil, >1,720 tons military
- 442 operating nuclear power plants in 31 States
  - 270,000 tons of spent fuel produced (1,800 tons Pu)
- 246 operating research reactors (>100 with HEU)
- 240 shut down research reactors (several 10 with HEU)
- 18 conversion plants
- 40 fuel fabrication plants
- 7 reprocessing plants
- 13 enrichment plants
- 89 storage facilities
- > 100,000 Category I and II radioactive sources
- > 1,000,000 Category III radioactive sources

---

**INSAG-24: “Relationship Between Nuclear Safety and Security”**

**Backgrounds:**

- Recent terrorist events catalyst for the development of international nuclear security legal instruments to address this increased threat.
- Nuclear safety and nuclear security have a common purpose *the protection of people, society and the environment* and many common principles, although their implementation may differ.
- Potential synergies, but also need to manage the impact of one discipline on the other to avoid potential conflicts.
- The purpose of the report is to provide a better understanding of nuclear safety and nuclear security interfaces and the ways to take them into proper account.
INSAG-24: 
Arrangements for Safety & Security

- Legislative and regulatory framework set up by the State

- Responsibility of the State:
  - designate competent authorities
  - define rules for confidentiality and information protection and carries out checks on individuals
  - continuously assess the threat
  - define the design basis threat

- Responsibility of operators
  - take prime responsibility for the installation and this responsibility cannot be delegated.

In case of emergencies, shared responsibilities between the operating organization (on-site) and the state/competent authorities (off-site).

Management of a crisis linked with malicious acts demands a greater number of State bodies than managing a crisis purely dependent on safety.

---

INSAG-24: Common Basic Principles & Potential Divergences

- Leadership and management
  - Similar principles for safety culture and security culture
  - Higher involvement of the State to define security measures
  - Differences in handling of information: transparency vs. confidentiality
  - Individuals of diverse backgrounds and experience for security

- Optimization of protection
  - Assessment of the risk, using a graded approach
  - Permanent safety and security systems
  - Improved dispositions of feedback (techniques, experience, threats)

- Prevention of event occurrences
  - Defense in depth for safety described in INSAG 10,
  - Defense in depth for security in the Amended CPPNM and INFCIRC/225
  - Close cooperation between security & safety specialists

- Emergency preparedness and response
  - Complementarity between contingency and emergency plans
  - Safety measures under the operator’s responsibility, while security measures under the State responsibility
  - Performance of joint exercises
Backgrounds:

- INSAG-6 ("Probabilistic Safety Assessment")
  - provides general views on the PSA performance and applications
  - discusses benefits and cautions for the potential users of the PSA
  - highlights areas of PSA that require deeper elaboration
- INSAG-12 ("Basic Safety Principles for Nuclear Power Plants")
  - probabilistic safety criteria were proposed, and suggests the core damage frequency and frequency of large off-site release for operating plants and plants under design
  - Both INSAG-6 and INSAG-12 encourage consistent performance and application of PSA as one of the safety assessment tools.

New Safety Guides on PSA:

- DS394 (Performance and Application of Level 1 PSA)
- DS393 (Performance and Application of Level 2 PSA)
  - Recommendations are based on current good practices.
  - Current trends, challenges and problematic areas of PSA are not included.
  - (ex. Risk-informed decision making, probabilistic safety goals, new methodological challenges, etc.)

Challenges to Key Safety Issue:

**"PSA Update"**

Scope:
- full scope Level-2 PSA.
Quality:
- extremely increased.
Data:
- combination of plant-specific and generic data
Role:
- wide applications in safety analysis, design, operation, licensing, etc.
Review:
- independent review as an integrated part of the PSA process
Safety Goals:
- numerical goals defined in many countries, but no international consensus.
Methodology:
- reasonable procedures, powerful codes for PSA, but problems in assessing low probable events and ageing phenomena
PSA: New Areas for Further Development

- Consideration of Ageing Effects
- Reliability of Passive Systems, Software and Digital Systems
- Internal and External Hazards PSA
- Consideration of Extreme Events and Global Warming
- Consideration of the Uncertainties and RIDM
- Probabilistic Safety Goals and Acceptance Criteria
- PSA Knowledge Management and Succession Planning
- International Cooperation and Communication on PSA

*Reference: New challenges and emerging issues in risk assessment for nuclear power plants, IAEA, NSNI/SAS/2009/1/March 18*

---

PSA: New Areas for Further Development (1/3)

- **Consideration of Ageing Effects**
  Danger of underestimation
  Need for comprehensive reliability PSA models including the realistic assessment of the risk
- **Internal and External Hazards PSA**
  Further maturity of PSA methodology with the emphasis on external hazards PSA
  - Hazards frequency assessment
  - Hazards impact to passive systems and computer-based digital systems
- **Consideration of Extreme Events and Global Warming**
  Natural phenomena: earthquake, tsunamis and seiches, hurricanes, cyclones and typhoons, floods, tornadoes
- **Consideration of the Uncertainties and RIDM**
  - Advantages of quantitative estimates for the uncertainties
  - Comprehensive uncertainties estimates in the decision making process
Probabilistic Safety Goals and Acceptance Criteria
No consensus in Member States on the use of Probabilistic Safety Goals (PSG)

- Probabilistic safety criteria and objectives:
  - Core damage frequency, large early release frequency, large release frequency, health effects, etc.?
- Other important related questions require consensus on:
  - What should be compared with the criteria/target? Mean, median, 95% bound of risk estimates, etc.?
  - Are numerical risk limits: Formal requirements or orientation values/targets?
  - Is the scope of the PSA required to meet the criteria/objectives: Internal initiating events, internal and external hazards, operation modes, etc.?
  - Is the generic PSG in terms of frequency (f) of doses (D) to an individual resident at a nuclear site boundary?
  - Is the trend to develop criteria for risks to people: consensus on the specific notions of the risk “tolerability” and “de minimis” limits?

Comprehensive PSGs need to be defined.

- What are the objectives of the Safety Goals?
- Should not only specify numbers?
- Should be formulated in a way encouraging further safety enhancement even when they are met?

PSA Knowledge Management and Succession Planning

- Training on PSA and succession
- Experienced PSA analysts approaching the retirement age
- Young nuclear engineers to get on-job training.
- Development of recommendations for knowledge management and succession planning.
- PSA documentation for appropriate knowledge management and smooth succession

International Cooperation and Communication on PSA

- Need for improved communication of PSA results and risk insights
- Not limited to nuclear engineers and scientists: Scientists in other fields, and the general public
- Wider international co-operation and co-ordination
Summary

Current Issues and Challenges on Nuclear Safety
- Strengthen Global Nuclear Safety Regime
- Operational Nuclear Safety
- Relationship between Nuclear Safety and Security
- PSA Update

PSA: New Areas for Further Development
- Consideration of Ageing Effects
- Reliability of Passive Systems, Software and Digital Systems
- Internal and External Hazards PSA
- Consideration of Extreme Events and Global Warming
- Consideration of the Uncertainties and RIDM
- Probabilistic Safety Goals and Acceptance Criteria
- PSA Knowledge Management and Succession Planning
- International Cooperation and Communication on PSA
Global Energy Problems

- Global demand for energy will continue to grow at considerable pace as there are 1.6 billion people who have not access to electricity and the 2.4 billion who have no access to modern energy systems.
- It is necessary to solve the problems of climate change, energy security and the volatility of the price of oil and gas.
- Japan and other major developed countries are committing to reduce greenhouse-gas (GHG) emissions to 50% of the current level in the first half of this century.
- This means that the global GHG emissions should be 13 GtC/yr below that of business-as-usual case in 2050.
Global Energy Solutions

Achieving this target while increasing the supply of energy requires the global community to make utmost effort to deploy not only energy conservation and high-efficiency energy technologies but also non-GHG emitting energy production technologies such as renewable, nuclear, and carbon-sequestration technologies, on a gigantic scale.

Example: to avoid even 1/10 of the target or 1.3 GtC/yr, it is necessary to replace 900 GWe coal fired plants with nuclear power plants.

However, there is a large uncertainty even in the future of nuclear energy: the OECD Nuclear Energy Agency has projected that global nuclear power capacity in 2050 will be between 580 and 1400 GWe.

Nuclear Energy Vision

Nuclear energy will contribute as one of the mainstay technologies for electricity and heat generation to the fostering of economic growth/poverty eradication, energy security and low-carbon economy in many parts of the world.
Objectives for Global Nuclear Community

I. Sustain safe and efficient operation of nuclear power plants, installing new plants that are necessary to satisfy the need for electricity/GHG emission reduction and managing used fuel in appropriate ways:

II. Shape environment for facilitating the peaceful uses of nuclear energy in every part of the world:

III. Realize competitive and more sustainable nuclear energy technology through unremitting R&D activities.

To sustain stable operation of nuclear power plants and install new capacity:

- Maintain the public trust in both plant operator’s safety management and government’s regulatory activities for nuclear safety, security and nonproliferation through the promotion of
  - Open and transparent risk communication with the public unremittingly.
  - Steady business risk management activities, carefully considering lessons learned from operating experiences worldwide, new developments in science and technology, and changes in organizational culture and business environment that can have negative influences on the safe operation of the plant.

- Make it possible to deliver safe disposal of radioactive wastes.

- Prepare and execute plant ageing management activities to ensure their high capacity factor and superior safety and economic performance throughout their life of 60 years at least;

- Assure market force to continue to drive the construction of nuclear power plants that are necessary for satisfying the anticipated need for electricity/GHG emission reduction.
The 16 July 2007 Earthquake at Kashiwazaki-Kariwa NPP of TEPCO

- The seismic input to the plant significantly exceeded the level of design-basis seismic input of the plant. Nevertheless, the operating units were automatically shutdown and all plants behaved in a safe manner, during and after the earthquake.
- No significant damage of safety-related structures, systems and components (SSCs) of the plant has been reported, whereas non-safety related SSCs were affected mainly due to significant soil deformation as they were not connected to the bedrock.
- The public confidence in both nuclear safety regulation and operators’ safety management was shaken by the intense media attention to the fire of a non-safety-related transformer and the inadvertent release of radioactivity, though the amount was extremely minor.

Lessons Learned from The Seismic Events at the KK NPPs

- Need for considering the inclination of a nearby fault toward the plant and the geological structure of the underground of the plant in the evaluation of seismic input from the fault and the flexibility of floor in the evaluation of the response of plant structures to the input.
- Need for reviewing the appropriateness of seismic design of seismic class C structures and components from the viewpoint of business continuity planning.
- Need for paying close attention to the appropriateness of emergency planning and fire-fighting capability in extreme seismic conditions.
Lessons Learned for Risk Management

Due attention should be paid not only to the sources of knowledge risk but also to the sources of relationship risk and process-engagement risk.

- Knowledge risk materializes when knowledge base is deficient due to neglect of lessons learned from experiences anywhere and new knowledge in science and technology that have impact upon the risk.
- Relationship risk appears when ineffective collaboration and communication among functions and/or organizations exist and insufficient knowledge is applied to risk assessment.
- Process engagement risk arises when faulty operational procedures exist and distorted knowledge is applied in risk assessment.

It is the task of leadership in enterprise and government administration to name these risks and seek to rise to them.

NUMO: Nuclear Waste Management Organization

Activities of JAEA

- Start R&D of geological disposal
- 1st R&D progress report
- 2nd R&D progress report
- R&D for safety regulation
- Scientific research at URL
- Enhancing reliability of disposal technique

Activities of NUMO:
- Encourage communities to apply for suitability review as an area for siting a geological disposal facility of high-level radioactive waste (HLW).
- 2012 Selection of areas for detailed investigation
- 2027 Selection of a site for repository
- Promote technology development for the improvement of safety, economy and efficiency of the geological disposal of HLW.
The Siting of a HLW Disposal Facility

- In 2000, the AEC decided that the activity to decide the site for a HLW disposal facility should be promoted in an open and transparent way and the site should be determined after detailed suitability review of the area of municipalities that apply for invitation.
- In 2004, the NUMO, an organization authorized to promote the disposal activity, started to invite mayors of municipalities to apply for site suitability review.
- However, no mayor has successfully applied so far: even the announcement of a mayor to study the merit and demerit of the application has paralyzed the administrative affairs of the municipal office due to the intense media attention and rallies and demonstrations to protest the announcement.
- The Government as well as the NUMO have started to strengthen public information activities on the possible public support for the sustainable development of the municipalities that locate the site from the view point of equity of benefit, as well as safety and the importance of the disposal facility.

To shape the environment for facilitating the peaceful uses of nuclear energy in every part of the world;

- Build a global consensus that nuclear energy is an essential measure against global warming/poverty eradication;
- Support countries considering the introduction of nuclear power internationally;
- Strengthen the international system for ensuring nuclear safety, security and nonproliferation;
To shape environment for facilitating the peaceful uses of nuclear energy in every part of the world;

- Build a global consensus that nuclear energy is an essential measure against global warming/poverty eradication;
  - Induce to consider the construction of NPP as a clean development mechanism (CDM) project activity of post-Kyoto Protocol framework to be determined;
  - Induce the World Bank to set up innovative financing schemes for NPP construction and that for sea-water desalination in particular.

- Support countries considering the introduction of nuclear power internationally; and

- Strengthen the international system for ensuring nuclear safety, security and nonproliferation;

To shape environment for facilitating the peaceful uses of nuclear energy in every part of the world;

- Build a global consensus that nuclear energy is an essential measure against global warming;

- Support countries considering the introduction of nuclear power internationally;
  - Support the IAEA and strengthen its human and financial resources as it is developing international references and standards, providing guidance, organizing workshops and offering a service to review the progress in infrastructure development.
  - Promote dialogues, provide consultation and services through bilateral and multilateral frameworks such as GNEP and FNCA, recognizing that human resource development and stakeholder engagement are central issues that need urgent attention.

- Strengthen the international system for ensuring nuclear safety, security and nonproliferation;
To shape environment for facilitating the peaceful uses of nuclear energy in every part of the world;

- Build a global consensus that nuclear energy is an essential measure against global warming;
- Support countries considering the introduction of nuclear power internationally;
- Strengthen the international system for ensuring nuclear safety, security and nonproliferation;
  - Make sure to adhere to international conventions related to nuclear safety, nonproliferation and security.
  - Reinforce the IAEA’s legal authority in nuclear verification, safety and security, universalizing the Additional Protocol and accepting various IAEA review missions for mutual learning.
  - Actively promote the establishment of credible multilateral nuclear fuel supply assurances, as a complement to the market, with a view to reducing incentive to have a national nuclear fuel cycle facility.

Evolution of Nuclear Energy Technology

We are introducing Generation III+ technologies and looking for Generation IV technologies.

http://www.gen-4.org/Technology/evolution.htm
Pursue Technology Innovation

Aiming for innovation of LWR technology;
- Pursue high performance of reactor plant materials, instrumentation and controls, and fuel, and better management of used fuel and waste.
- Develop innovative LWRs incorporating advances in science and technology: high burn-up fuel, seismic isolation technology, advanced construction technology, advanced information technology and so on.

Aiming for realization of sustainable nuclear energy technology from the long term perspective, promote R&D of Generation IV nuclear energy systems that have potential to make significant contributions in the future to sustain low-carbon society.
- Fast reactor and its fuel cycle technology that satisfies the request for enhanced safety, reliability and utilization of fuel, increased proliferation resistance, friendliness to the neighbor, low heat generation rate of radioactive waste;
- Promising nuclear energy technologies, such as high temperature water-splitting technology and grid appropriate reactors, that contribute to new missions and markets such as sea-water desalination, hydrogen production, district heating etc.

Conclusion

Nuclear energy is one of the key energy supply sources of the future. It can make a major contribution to the fostering of economic growth/poverty eradication, energy security and low-carbon economy in many parts of the world.

Global nuclear community should rise to contribute to sustain safe and efficient operation of nuclear power plants, install new plants that are necessary to satisfy the need for electricity/GHG emission reduction and manage used fuel in appropriate ways.

Safety, security and nuclear safeguards should be ensured at any plant in any country. The community should support states willing to develop a nuclear power program in their efforts to establish required infrastructure, in close cooperation with the IAEA.

It is also vitally important for the community to pursue to realize sustainable nuclear energy technologies.

International collaboration is essential to the success in rising to these challenges, not only because the collaboration could reduce the duplication of efforts but also because it could produce better and brighter solutions in our pathway to the goals.
This is a blank page.
Special Lectures

Katsumi EBISAWA
Joseph BRAVERMAN
This is a blank page.
The 10th Korea-Japan Joint Workshop on PSA

Reflection of Lesson learned from Niigata-ken Chuetsu-oki (NCO) Earthquake to Seismic PSA

May 18-20, 2009
Jeju Island, Korea

Incorporated Administrative Agency
Japan Nuclear Energy Safety Organization
Katsumi EBISAWA
ebisawa-katsumi@jnes.go.jp

Contents

I. Background
II. Lesson learned from NCO Earthquake
   II-1. Amplification of Earthquake Motion
   II-2. Flexibility of Building Floor
   II-3. Integrity of Component
III. Reflection to Methodology of Seismic PSA
   III-1. Seismic Hazard Evaluation
   III-2. Fragility Evaluation
   III-3. Accident Sequence Evaluation
IV. Conclusions
I. Background

The lessons learned from Niigata-ken Chuetsu-oki (NCO) earthquake occurred July, 2007 extremely affected to the nuclear seismic safety. The main lessons are as follows.

1. Why did the observed seismic ground motions far exceed those designed?
2. Why were the response spectra derived from the observed seismic motions different from those of the vibration model of the conventional seismic design?
3. Why did the safety related functions of shutdown, cooling, and containment maintained effectively?

The overview of cause investigation will be introduced.


The above lessons reflected to each evaluation methods of seismic hazard, fragility and accident sequence in seismic PSA.

The overview of reflection will be also introduced.

II. Cause Investigation on NCO EQ
(1) Why did 3 pulses happen?
(2) Why did the observed seismic ground motions exceed those designed and the Unit 1 shows the highest values which are nearly double of the design response?
(3) Why did the acceleration values of observed seismic motions of the Unit 1 become nearly double of the Unit 5? (Those two Units locate in 2.5 km apart each other in the site).

(i) Sequential rupture of 3 asperities which broke out strong seismic motion is one of causes of amplifying pulse wave.
(ii) Asperity 3 is very close to the site, and radiates strong seismic motion.
Analysis on Effects of Deep Ground Structure Generation of 3-D Underground Structure Model

Used data: Boring surveys, reflection surveys, geological maps, etc., performed by the former Japan National Oil Corporation.

(i) Earthquake bedrock near the site is deep as about 5~8 km.
(ii) The deep underground structure has irregularity in propagation path of seismic motion from epicenter.

Analysis of Amplification Characteristics of Pulse Waves at Unit 1

(i) Irregularity in deep underground structure concentrate and storage seismic ground motion energy, and tend to lead seismic motion to the site.
(ii) Pulse wave at the layer near free bedrock are amplified largely.
(iii) Amplifying factor is 3~4 times.
Analysis of difference of amplification characteristic among Units

1. Amplifying factor between seismic bedrock and free base stratum at Unit 1 side is 1.5 times larger than Unit 5 side. Cause of the difference might be irregularity in deep underground structure, ground parameter such as Vs and distance from asperity 3.
2. Ratio of amplifying factor between through irregular structure in deep underground and regular structure was 1.5 times. Amplifying factor at Unit 5 side is almost 1 and ground structure this side may be relatively regular.
3. Amplifying factor difference between Unit 1 side and Unit 5 side may be due to difference of irregularity in deep underground structure.

Snap shot of propagation of Seismic motion from source

1. Generation of pulse wave due to failure of ASP 3
2. Propagation of pulse wave due to failure of ASP 1 and 2
3. Propagation and Amplification of pulse wave
4. Amplification of pulse wave at Shiiya and Nishiyama layers
5. Arrival of pulse wave To the site
II-2 Flexibility of Building Floor

Why were the response spectra of buildings derived from the observed seismic motions different from those of the vibration model of the conventional seismic design?

- The profile of the analyzed response spectrum is largely different from that of the observed.
  - A rigid floor was assumed and analyzed, however, the real floor is flexible and a floor deformation is feasible.

- Policy of the investigation into the model
  - A floor deformation is considered.
  - The interaction reflecting nonlinearity of ground around the building is considered.
  - The interaction between Reactor Buildings and Turbine Building is considered.

⇒ Investigation using the 3-D FEM Model

3-D FEM Model (Units 1 and 4 Reactor Buildings)

Seismometer (Ground floor)

Concrete damping: 3%

3-D FEM Model (Nuclear Reactor Buildings of the Units 1 and 4)

(Seismometer: Concrete on the Foundation)

Ground around the Reactor Building

Building ground interacting section

Horizontal 440m

133m
The results on the analysis of the FEM model simulate the observed records well. The consideration of the floor flexibility is essential.

The values of the analyzed response spectra are very large. The floor rigidity would be over-estimated.

Model-setting policy
- Considering a foundation deformation.
- Considering the soil-structure interaction reflecting the decrease in rigidity of the ground around the building.

⇒ Investigation using a A-axis-symmetrical-axis FEM Model

Together with the improvement of the 3-D FEM model considering building floor flexibility and upgrading of the standardization procedure, the reproducibility in the lumped mass model will be studied. The achievement of the study will be applied for the JAEA bench-mark study.
II-3 Integrity of safety related SSCs and about 3500 troubles

Observed ground motion far exceeded the design of KK NPP. However Safety related SSCs (Shut down, Cooling and Containment ) function maintained. TEPCO reported the damages or nonconformities of about 3500 that were not important accidents. They are the precious information to identify accident scenarios on seismic PSA.

1. Damage on wheel driving shaft connection of over head crane (K-6)
2. Crack on connection valve junction between main condenser water boxes and sea water leak (K-4)
3. Degradation of water tightness of watertight doors for RCIC and RHR (K-7)
4. Dislocation of blowout panel of reactor building (K-3)
5. None
6. Dislocation of air duct connected to Stack
7. Slope failure of a part of east side slope of the switch yard
8. None
9. None, In-leak of water due to fire fighting pipe failure and flooding on lowest basement floor
10. Detection of insolite at main stack (K-3)
11. Fire of in-house transformer 38 (K-3)
12. Oil leak of start-up transformers 38 (K-3, 4, 6)
13. Subsidence of yard
14. Falling down of the secondary structures of ceiling
15. None

---

Functional failure of ceiling crane at unit 6

- Part: joint part
- Failure mode: shearing and bending by putdown earthquake motion
Fire of transformer at unit 3

- Part: joint part
- Failure mode: shearing and bending by uptown earthquake motion

Flooding and release of water with FP at Unit 6

Leak course in building (3F)

Leak of water and release course to sea in Reactor building

Reactors:
- Controlled area
- Noncontrolled area
- 4th floor
- 3rd floor
- Mid 3rd floor
- Nonradioactive drain tank
- Operating floor and cable of fuel handling machine which became the leak course

Spent fuel pool

Flow point

Power feeding box of fuel handling machine

Dripped down from the vicinity of upper duct

Dripped down into duct below and flew outside

Wash port

Sea
III. Reflection to Seismic PSA Methodology

III-1. Seismic Hazard Evaluation
  ① Treatment of Seismic Source Model at near Site:
     Utilization of Fault Model
  ② Treatment of Uncertainty for Earthquake Motion:
     Management of Logic Tree
  ③ Treatment of Stress concentration Zone

III-2. Fragility Evaluation
  ① Floor Flexibility of Building
  ② Integrity of Component
  ③ Functional Failure of Ceiling Crain under Vertical motion
  ④ Utilization to Seismic Margin Evaluation

III-3. Accident Sequence Evaluation
  ① Reimprovement of Accident Scenarios
  ② Improvement of Frequency of Fire and Flooding caused by Earthquake
  ③ Improvement of Evaluation Method of Core Damage Frequency for Multi-NPPs

III-1 (1) Handling of near site seismic source
- Effectiveness of source model -

Lesson learned from NCO earthquake:
cause analysis on amplification factor of NCO earthquake is achieved by source model (fault model) method and 3-D underground structure model.
In other word, cause analysis is difficult if source model method is not available

Situation of seismic hazard evaluation:
In many case, seismic hazard of nuclear site is dominated by near sources inside 60km radius area, never the less by specified source or by average hazard of the region. So, detail evaluation of near sources is essential and setting of upper limit is very important. Seismic hazard evaluation of JAEA seismic PSA standard procedure describe both prediction by attenuation relationship and by source model.

Direction of improvement:
(1) Recognize effectiveness of source model prediction in viewpoint of:
   • detail evaluation of seismic source and ground motion
   • resolution on mechanism of seismic source and ground motion
and outgrow from evaluation by attenuation equation only
(2) Detail evaluation is achieved by source model method, setting 16 parameters. However, at the site where less information on these parameters, caution should be paid on large uncertainty. Hereafter, to improve source model method, quantitative evaluation on uncertainty of each parameters and interaction between parameters.

[Note]: Consultancy meeting of IAEA Seismic Hazard Evaluation guide DS422 was held in Tokyo at February 2009. Based on opinion of Japanese specialist of this area (Takada, Ebisawa, Irikura, Okamura, Kameda), source model method is discussed and adopted to the guide.

---

**III-1 (2) Treatment of Uncertainty in Earthquake and Ground Motion - Utilization of Logic Tree -**

- New Seismic Design Review Guide
  - New Guide requires consideration of uncertainty concerned with the evaluation process of the Design Basis Ground Motion (DBGM) Ss and referring of its exceedance probability.
- Lessons learned from NCO Earthquake
  - Latest findings from the NCO Earthquake also show the necessity of consideration of uncertainty and referring of exceedance probability in the determination of DBGM Ss.

In the open committees of regulatory body, deliberation on Ss of utilities and evaluation of exceedance probability in the probabilistic seismic hazard has been carried out but it is not proceeding efficiently. The reasons are 1) there are few committees knowing probabilistic seismic hazard evaluation fully enough, 2) there is no rule of its deliberation, and 3) utilities’ evaluation does not necessarily meet to the AESJ Seismic PSA Implementation Standards.

- Direction of Improvement
  - There are many opinions from regulatory body and also utilities that seismic hazard evaluation for each site should be performed along the open deliberation rule by the public organization such as JNES.
  - JNES proposed a draft framework of the open deliberation rule, which was made so as to be able to utilize logic tree practically, referring to the implementation procedure of logic tree in AESJ Seismic PSA Implementation Standards.
1. Uncertainty factors in determination of ground motion Ss are roughly classified into those in earthquake parameters and those in ground motion, and both of them are studied. It is a common understanding that the final target is determination of the design basis ground motion necessary to structure design, not setting earthquake parameters.

2. It is confirmed that there are two kinds of uncertainty factors; one is those which can be treated by probability, and the other is those which can not be expressed by probability such as difference of expert opinions. It is re-confirmed that it is especially quite important to form consensus between experts regarding the latter factors.

3. It is re-confirmed whether there are any effective methods other than logic tree method in order to quantify uncertainty or not. If there are none, then it should not be cloud but make clear whether to apply logic tree method.

4. It is first priority to secure explanation-ability and transparency through whole deliberation process. It is prohibited to make discussions only for the sake of discussion such as staying and hesitating on the way of the course from setting earthquake parameters to determination of basic ground motions.

5. Sensitivity analysis on the factors proposed as expert opinions is performed timely on the way of logic tree formation and their contributions to basic ground motions are shown quantitatively. Factors with less contributions are left late without persistence and progress of deliberation should be promoted. Technical issues are clarified concerning these factors with less contributions with leaving evidence, and commended to academic society and/or association.

6. It is a common understanding that recent scientific knowledge and findings are utilized usefully, but that there is a possibility of facing a situation in which there is no way other than engineering judgment finally in the determination of basic ground motions.

7. In a case that there occurs discussion how ground motion Ss influences the function of structures, reference information will be given. But emphasis is put persistently on the discussion of determination of ground motion.

---

### III-1 (3) Handling of stress concentrating

- Lessons learned from NCO earthquake
  - Seismic activity around NCO epicenter area is much high and so called "Stress concentrating zone". And there also exists "Earthquake blank region"

- Direction of improvement
  - Handling on seismic hazard evaluation:
    1. "Stress concentrating zone": Revise b value in Gutenberg-Richter equation of seismic source in that zone
    2. "Earthquake blank region": Adopt Non-Poisson Process (Renewal Process) in the earthquake frequency evaluation at that region

---
III-2 (1) Consideration of Building Floor Flexibility

Lesson learned of building floor flexibility:
1) Building model considering floor flexibility, instead of current rigid floor model, should be adopted for fragility evaluation of building.
2) Floor response spectrum or time history acquired using building model considering floor flexibility, instead of current rigid floor model, should be adopted for fragility evaluation of facilities.
3) In case detail evaluation of fragility is not required, flexible floor model is not essential requirement.

<table>
<thead>
<tr>
<th>Building floor model</th>
<th>Floor response spectrum</th>
<th>Fragility</th>
</tr>
</thead>
<tbody>
<tr>
<td>Current handling</td>
<td>rigid</td>
<td>Floor response value by rigid floor model is smaller than that of by flexible floor model at 0.1~0.3 Sec. region. So, facility at the region like piping is non-safety side.</td>
</tr>
<tr>
<td>From now</td>
<td>flexible</td>
<td>In case of piping, fragility by rigid floor is smaller than that by flexible floor</td>
</tr>
</tbody>
</table>

III-2 (2) Consideration of Component Integrity

Lessons learned from the Integrity of S-class Components:
1) These lessons are used to verify the reliability of fragility evaluation by the following manner:
   ① Confirmation of maintenance of component function at the observed ground motion $\alpha$ means that CFP(α) [Conditional Failure Probability] is extremely close to 0.
   ② CFP(α) is calculated using median mean value and logarithmic standard deviation $\beta$ of realistic response and capacity. Important parameter $\beta$ is to be confirmed whether the value is appropriate or not.
   ③ A probability extremely close to $\theta$ is regarded around $10^{-4}$
2) Analysis examples confirm that $\beta$ used in Japan is appropriate.
III-2 (3) Consideration of Effectiveness of Vertical motion

- Lessons learned from the damage of R/B over head crane
  1) The cause of the damage of the over head crane was identified as the effect of vertical response.
  2) Implementation of review for fragility evaluation method considering vertical response.
  3) Review policy for fragility evaluation is:
     Firstly, to review identification of facilities to end to be affected by vertical response,
     Then, for each facility identified, to review failure mode and mechanism.
     Adding the above, to propose countermeasures to prevent failure and to verify their effectiveness.
  4) Implementation of shaking table test to verify the policy of 3)
     ⇒ Over head crane.

Typical facilities tend to be affected by up-down motion
- Vertical shaft pump
- Piping
- Over head crane
- Refueling machine

In the revision of the Regulatory Guide for Reviewing Seismic Design of Nuclear Power Reactor Facilities, vertical motions are considered for design basis ground motion and improvement of the assessment method of the vertical nonlinear response for structure and equipment is required. At the Chubu-oki Earthquake, a overhead crane in the Kashiwazaki-Kariwa power plant was damaged and the study on the influence of the vertical motions is thought to be important.

Target Equipment
- Overhead crane (consisting of a garter, a trolley, a hanging load, lugs, etc.)

Contents of the Test
- Component Tests (performed in FY 2007)
  ① Factor analyses of the functional limit
  ② The mutual uplift and the collision assessment of garter/trolley/hanging load
  ③ The assessment of the restitution coefficient of wheels
- Reduced Scale Model Test (performed in Oct. 2006)
  ① Additional investigation point from NCE
     ⇒ Effect confirmation of the fall-prevention work (lug), etc.
     ② Analysis in detail is ongoing. Preliminary reports are available on the JNES homepage.
     - Effectiveness of lugs was confirmed.
     - The uplift behavior was understood.
     http://www.jnes.go.jp/katsuden/topics2008.html#081128

⇒ The uplift mechanism for the vertical motions will be clarified and the nonlinear analysis method will be improved.
The effectiveness of the fall-prevention work will be confirmed and the results will be applied to the integrity criteria in the seismic back-check.
Lessons learned from Plant Seismic Margin
- "Shutdown", "Cooling" and "Confinement" function were kept and plant safety was maintained. The reason is to be said due to adequate plant safety margin but not yet explained quantitatively.
- JNES is planning to estimate the margin quantitatively by utilizing seismic PSA.

- Fragility evaluation method of AESJ Seismic PSA Implementation Standard include JNES method (detailed direct method), JAERI method (less detailed, partially factor method) and Zion method (simplified, factor method).
  - JAERI method: Useful for seismic margin estimation because;
    1. Realistic response and capacity of components are treated separately. So deviation between them can be indicated clearly.
    2. Conservativity factors of design response in realistic response evaluation are separately treated in ground motion setting, ground response, building response and component response. So, each contribution is indicated individually. Zion method is hard to explain the seismic margin rationally.
  - For fragility evaluation, at least JAERI method or JNES method is preferable, considering application of the fragility evaluation method to seismic margin estimation hereafter.
  - Although the development of capacity data base is an task, an idea is that the difference in seismic design of Korea and Japan is expressed by coefficient and Japanese capacity data are corrected as to meet in Korea

- The average seismic margin can be evaluated by the comparison with the median of the "functional limit strength" and the realistic response.
- The seismic margin including variation can be understood quantitatively by the consideration of each "logarithmic standard deviations".
III-3 (1) Reconstruction of Accident Scenario

A. Influence of main shock on its core damage

A1. Direct effect
   1. Building/Structure/component with safety function (Damage of As/A equipment)

A2. Indirect effect
   A2.1. Those except for building/structure/component with safety function
          (Secondary effect affecting As/A equipment and leading to core damage)
   - Indoor equipment
     1. Effect of failure of overhead crane to containment vessel / pressure vessel
     2. Effect of damage of B/C equipment to As/A component
     3. Effect of turbine missile to adjoining building
   - Outdoor equipment
     4. Effect of collapse of stack to building
     5. Effect of land slope collapse to building / surrounding facility
     6. Loss of off-site power due to damage of electric grid tower, etc.
     7. Loss of cooling function due to stop of water supply

A2.2. Effect due to human error
   - Operation error of plant operators/workers due to high stress during and after earthquake
   - Failure of back-up operation by blackout due to damage of insulator transformer, etc.
   - Obstacle of passage in the power due to soil liquefaction / damage of landslide-preventing wall
   - Effect of damage of secondary material on ceiling, etc. to operators, etc.
   - Mistake or error through planning / design of plant, selection of materials, production and assembling

B. Influence of after shock to core damage

1. Evaluation of increment of core damage frequency due to after shock.

III-3 (2) Evaluation of Seismically caused fire and flooding

Current situation: A fire of C class transformer caused by the Chuetsu-oki earthquake called attention to possibility of fire of A class equipment

- No fire
  - Accident scenario only for earthquake
    - Accident scenario for seismically caused fire

- Fire
  - Accident scenario for fire propagation
  - Development of seismic cause fire probability evaluation method
    - Grouping of fire sources (including transformer)
    - Study of generating mechanism
    - Development of prob. evaluation method
      \[ p_f(x) = p_a(x) \cdot p_o(x) \]

- No flooding
  - Accident scenario only for earthquake
    - Accident scenario for seismically caused flooding

- Flooding
  - Accident scenario for flooding due to earthquake
    - Development of seismic cause flooding probability evaluation method
      - Grouping of flooding sources (bunk, piping etc.)
      - Study of generating mechanism
      - Development of prob. evaluation method
      \[ p_f(x) = p_a(x) \cdot p_o(x) \]

Screening by probability

(1) Flooding probability depend on the existence and capacity of drain equipment
III-3 (3) Evaluation of Multi-Unit Plants

**Recognition:** multi-unit on same site / possibility of simultaneous loss of functions / essential in regard to risk-based assessment

**Task:** Enhancement of CDF assessment methodology by considering response and fragility correlations and seismic motion correlation in terms of intensity and time

**Example of influence on CDF of correlation of damage**

- **CDF of multi-units under correlation conditions**
- **Equipments:** correlation in response and resistance
- **Input E-Q. motions:** correlation of input motions

**Evaluation method of correlation of damage between plants**

<table>
<thead>
<tr>
<th>Correlation coefficient of damage between facilities in plant</th>
<th>Correlation of capacity</th>
</tr>
</thead>
<tbody>
<tr>
<td>Calculation of response</td>
<td>Calculation of input motion</td>
</tr>
</tbody>
</table>

**CDF:** CDF of Plant 1, CDF: Overlap area of CDF of Plant 1 and CDF of Plant 2

**IV. Conclusions**

- The lessons learned from NCO EQ extremely affected to the seismic safety. The overview of cause investigation was introduced.
- The above lessons reflected to each evaluation methods of seismic hazard, fragility and accident sequence in seismic PSA. The overview of reflection was also introduced.
- JNES is ready to share the lessons learned from NCO EQ with international nuclear community e.g. through various chances like IAEA, OECD/NEA, NRC and individual countries.
- In order to actively contribute to further improvement of seismic safety, JNES will contribute to the IAEA’s International Seismic Safety Center.
Aging Related Degradation Assessment of Structures and Passive Components for use in Performing PSAs

Joseph Braverman
EM&I Group / NEIS Division
Energy Sciences and Technology Department
Brookhaven National Laboratory

Presented to
The 10th Korea-Japan Joint Workshop on PSA
May 18-20, 2009

Presentation Outline

- Need for Consideration of Aging Degradation in Nuclear Power Plants (NPPs)
- NRC Regulations and Regulatory Guidance Related to Aging
- Component Aging Degradation Assessment Process for Use in PSA
- Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSAs
- BNL/KAERI Collaboration Program on Aging
Need for Consideration of Aging Degradation in Nuclear Power Plants (NPPs)

- U.S. Nuclear Power Plants (NPPs) are aging. Many of the plants are approaching their 40-year design life.
- Management of age-related degradation is important for the current safe operation of nuclear power plants and for licensing renewal.
- Past studies and inspections have identified aging degradation of structures and passive components.
- Little is known about how degradation could affect the response and resistance of structures and passive components under various design loads.
- Lack of reliable inspection techniques for inaccessible areas.

NRC Regulations and Regulatory Guidance Related to Aging

Maintenance Rule
- 10 CFR 50.65 - Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Maintenance Rule).
- NRC Regulatory Guide 1.160, Rev. 2, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants.

License Renewal Rule
- Regulatory Guide 1.188 “Standard Format and Content for Applications To Renew Nuclear Power Plant Operating Licenses.”
NRC Regulations and Regulatory Guidance Related to Aging (Cont’d)

Other
- 10 CFR 50.55a Codes and Standards - Imposes the inservice inspection (ISI) requirements of the ASME Boiler and Pressure Vessel (B&PV) Code
- 10 CFR 50, Appendix J - Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors
- RG 1.127, Revision 1 - Inspection of Water-Control Structures Associated with Nuclear Power Plants
- RG 1.35 Inservice Inspection of Ungrooved Tendons in Prestressed Concrete Containments Rev. 3
- NRC Regulatory Guide 1.35.1, Determining Prestressing Forces for Inspection of Prestressed Concrete Containments, July 1990
- RG 1.147, Rev. 15 – Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1
- RG 1.54, Rev. 1 – Service Level I, II, and III Protective Coatings applied to Nuclear Power Plants
- RG 1.192, Operation and Maintenance Code Case Acceptability

Component Aging Degradation Assessment Process for Use in PSA

- Selection of Critical Components
- Identify/Improve Analytical or Test Methods
- Perform Analyses/Tests
- Develop Fragility Curves for Use in PSA
Selection of Critical Components

- Higher Risk Significant Components
- Structures and Passive Components
- Those Most Affected by Aging
  - Past Experience
- Adequacy of Existing Programs

Structures and Passive Components to Consider

- Anchorages
- Cable Tray Systems
- Concrete
- Conduit Systems
- Containment
- Cooling Tower
- Electrical Conductors
- Exchangers
- Filters
- HVAC Duct
- Insulation/seal
- Piping System
- RPV
- Structural Seismic Gap
- Structural Steel
- Tanks
- Vessels
- Water-Control Structures
Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Past US Studies Related to Age-Related Degradation of Structures and Passive Components:

1. Assessment of Age-Related Degradation of Structures and Passive Components for U.S. Nuclear Power Plants; NUREG/CR-6679 by BNL
2. Probability-Based Evaluation of Degraded Reinforced Concrete Components in Nuclear Power Plants; NUREG/CR-6715 by BNL

Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSAs

Reinforced Concrete Shear Walls

Development of Seismic Fragility Curves:
- Identify analytical method
- Benchmark analytical method
- Design representative member
- Define limit state/capacity
- Develop structural statistics for member
- Perform fragility analysis
  - Undegraded
  - Degraded – various levels
Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls – Identify Analytical Method:

- Computerized Solution
  - Finite Element Method (FEM)

- Empirical Equations
  - Barda et al. Methodology

- Testing

Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls - Benchmark Analytical Method:

- Testing – Done in Japan

- Computerized Solution
  - Finite Element Method (FEM)
Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls – Design representative member

Representative Shear Wall Design

Finite Element Model of Representative Shear Wall Design

Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls - Define limit state/capacity

Load Deflection Curve

Limit State Equals
4 x Elastic Limit
Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls - Develop structural statistics for member:

<table>
<thead>
<tr>
<th>Property</th>
<th>Mean</th>
<th>( V_C )</th>
<th>CDF</th>
</tr>
</thead>
<tbody>
<tr>
<td>Concrete (4,000 psi)</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Comp. Strength</td>
<td>4,400</td>
<td>0.16 N</td>
<td>N</td>
</tr>
<tr>
<td>Splitting strength</td>
<td>475 psi</td>
<td>0.18 N</td>
<td>N</td>
</tr>
<tr>
<td>Initial tangent modulus</td>
<td>3,834 ksl</td>
<td>0.18 N</td>
<td>N</td>
</tr>
<tr>
<td>Max comp. strain</td>
<td>0.004</td>
<td>0.20 N</td>
<td>N</td>
</tr>
<tr>
<td>Grade 60 reinforcement</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Yield strength</td>
<td>71 ksi</td>
<td>0.10 LN</td>
<td>N</td>
</tr>
<tr>
<td>Modulus of Elasticity</td>
<td>29,000 ksl</td>
<td>NA</td>
<td>NA</td>
</tr>
<tr>
<td>Placement of reinforcement</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Effective depth, ( d )</td>
<td>( d ) (in)</td>
<td>0.90/d N</td>
<td>N</td>
</tr>
<tr>
<td>Analysis Shear (( B_{sh} ))</td>
<td>1.00</td>
<td>0.14 N</td>
<td>N</td>
</tr>
</tbody>
</table>

Note: 1 in. = 25.4 mm; 1 psi = 6.895 kPa; 1 ksi = 6.895 MPa

\( V_C \) = Coefficient of Variation

CDF = cumulative distribution function

N = normal distribution; LN = lognormal distribution; NA = not applicable

Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA

Reinforced Concrete Shear Walls – Fragility Analysis:

- Fragility Curve is the conditional probability of failure for a given value of demand (e.g., pga in g’s)

\[
F_R(x) = \Phi\left[ \frac{\ln(x / m_R)}{\beta_R} \right]
\]

\( \Phi\left[ \right] \) = standard normal probability integral

\( m_R \) = median capacity

\( x \) = demand parameter

\( \beta_R \) = logarithmic standard deviation
Example of Past Aging Degradation Analytical Assessment for Use in Seismic PSA
Reinforced Concrete Shear Walls – Perform Fragility Analysis:

BNL/KAERI Collaboration Program on Aging

- **Objective:** Development of Seismic Capability Evaluation Technology for Degraded Structures and Components

- **Scope:** Specific tasks over a 5 year period to develop seismic fragility methodology of structures and passive components considering aging degradation effects

- **Results:** To be used in support of periodic safety reviews, license renewal applications, and for upgrade of the seismic safety of NPPs in Korea
BNL/KAERI Collaboration Program on Aging

Scope of Research

- **Year 1: Study of Degradation Occurrences – Completed**
  - Collect and review degradation occurrences in US nuclear power plants
  - Identify important aging characteristics needed for seismic capability evaluations
  - Results documented in BNL Report-81741-2008, KAERI/RR-2931/2008

- **Year 2: Review Time-Dependent Material Degradation Models – Completed**
  - Identify modeling methodologies for the long-term behavior of material degradation in NPPs
  - Focus on the most common time-dependent changes in material properties (e.g., loss of material and cracking)

- **Year 3: Seismic Fragility Analysis – Just Initiated**
  - Intent to demonstrate seismic fragility calculation methodology
  - Select representative structure/passive component and perform fragility analysis – undegraded & degraded conditions
  - Computerized finite element analysis method or closed form solution

BNL/KAERI Collaboration Program on Aging (Cont’d)

Scope of Research

- **Year 4: Technical Assistance to KAERI for Fragility Analysis of Other Structures/Components – Future**
  - Identify important aging characteristics for other structures/components
  - Identify suitable analytical/test methods for determining seismic fragility
  - Perform seismic fragility analyses / process available test data

- **Year 5: Technical Assistance to KAERI for Degradation Acceptance Criteria - Future**
  - Similar to BNL’s recent approach for NRC aging research project (see BNL NUREG/CRs)
  - Should consider effects on core damage frequency (CDF) and large early release frequency (LERF)
Issues to Consider for Further Research

- Develop an Operating Experience database for each operating NPP and for each class of operating NPPs
- Initiate and maintain an Operating Experience database for each new-generation NPP and for each class of new-generation NPPs
- Further improve analytical methods to assess fragility and impact on safety that take into account effects of aging
- Incorporate test data on fragility capacity into PSA assessments
- Develop improved and more specific acceptance criteria for degradation on both a deterministic and probabilistic basis
- Improve condition assessment methods and inspection tools to assess potential degradation of structures and passive components - especially in inaccessible areas
This is a blank page.
Session I-A Summary

Chair: Mitsuhiro KAJIMOTO (JNES), Chang-Ju LEE (KINS)

Mr. Hidaka pointed out Japan's current guideline on the maintenance program using PSA information. Relating with Japan's new inspection program, he explained recent revision of regulatory guide in terms of operating consideration for SSCs with safety functions, as well as the policy for utilization of risk information.

Since "the policy on the severe accident" in 2001, Korea had to set performance goals considering results of each levels of PSA in NPPs. Mr. Kim presented about the outline & flowchart of the performance goal which has been recently developed for Korean nuclear reactors. Some application principles for utilizing performance goals are suggested.

Mr. Chung presented about the R&D status for developing integrated safety performance assessment program, as well as suggested overall framework for graded periodic inspection program.

Mr. Yang presented about current status and methodology for developing KINS SEIF (significance evaluation inspection finding) program, as a supporting tool of integrated safety performance assessment program, as previously explained by Mr. Chung.

Mr. Uchida presented about the direction of current improvement of regulatory inspection system in Japan, on the viewpoint of the utilization of risk information. He explained many applying areas in terms of the utilization of risk information, such as maintenance program, performance criteria, regulatory inspection, etc.

Mr. Lee presented about the current status and methodology for developing risk-informed performance indicators, as a supporting tool of integrated safety performance assessment program, as explained by Mr. Chung.
This is a blank page.
Introduction

  - To provide fundamental criteria to the relative importance of safety functions in applying various requirements for safety design in the process of licensing review.

- The guide, prepared originally for design phase, had been also referred to construction and operation phases because the reliability of safety function should be maintained at all phases according to their importance.

- At operation phase, type or interval of maintenance for SSC in NPPs had been defined based on the guide.
Use of risks for inspection

- **New inspection program** was initiated by NISA in January 2009.
  - Linkage with plant life management (PLM)
  - Elongation of time interval of periodical inspection
  - Enhancement of effectiveness in inspection
    - Endorsement of the *maintenance program* prepared by utilities based on *JEAC-4209*
- In the maintenance program, type or interval of the maintenance is defined considering the PSA results and the safety importance classification guide.
- In present framework, if there are deviations between the PSA results and the deterministic based safety importance classification guide then the most conservative safety classification (higher safety class) is applied.
- Possible reason for this conservatism was that the former guide prepared originally for design phase does not describe clearly what concept is kept during operation phase and the policy for utilization of risk information as well as insufficient experiences in this field.

Safety Importance Classification Guide (1/2)

**Table 1. Classification of Safety Function Importance**

<table>
<thead>
<tr>
<th>Classification by safety importance</th>
<th>Categorization by function</th>
<th>Abnormality prevention functions</th>
<th>Abnormality mitigation functions</th>
</tr>
</thead>
<tbody>
<tr>
<td>SSC related to safety</td>
<td>Class 1</td>
<td>PS-1</td>
<td>MS-1</td>
</tr>
<tr>
<td>SSC related to safety</td>
<td>Class 2</td>
<td>PS-2</td>
<td>MS-2</td>
</tr>
<tr>
<td>SSC related to safety</td>
<td>Class 3</td>
<td>PS-3</td>
<td>MS-3</td>
</tr>
<tr>
<td>SSC not related to safety</td>
<td>No class</td>
<td>No safety functions</td>
<td></td>
</tr>
</tbody>
</table>

**Class 1:** Maintain the highest reliability reasonably achievable  
**Class 2:** Maintain high reliability  
**Class 3:** Reliability equivalent to or higher than ordinary industrial facilities  
**PS:** SSCs that loss of their functions may cause abnormal conditions, leading to undue radiation exposure of the public.  
**MS:** SSCs that have the functions to prevent escalation of abnormal conditions or to mitigate undue radiation exposure of the public.
Safety Importance Classification Guide (2/2)

Table Attached to Commentary: Examples of Importance Classification of Safety Function in PWR and BWR

<table>
<thead>
<tr>
<th>Class</th>
<th>Function</th>
<th>SSC (PWR)</th>
<th>SSC (BWR)</th>
</tr>
</thead>
<tbody>
<tr>
<td>PS-1</td>
<td>1) Reactor Coolant Pressure Boundary</td>
<td>Components and pipelines comprising the reactor coolant pressure boundary</td>
<td>Same as on the left</td>
</tr>
<tr>
<td>MS-1</td>
<td>3) Prevention of overpressure in reactor coolant pressure boundary</td>
<td>Pressurizer safety valve (Opening function)</td>
<td>Safety relief valve (Opening function as safety valve)</td>
</tr>
<tr>
<td>PS-2</td>
<td>1) Reseating of safety valves and relief valves</td>
<td>Pressurizer safety valve (Related with reseating function)</td>
<td>Safety relief valve (Related with reseating function)</td>
</tr>
</tbody>
</table>

The rest is omitted.

Outlines of Revision of Safety Importance Classification Guide

- On March 9, 2009, the guide was partly revised to show the policy for use of risks when the guide is referred to operation phase.
- According to NSC’s policy on RIR introduction, present revision should treat use of risks not for design phase but for operation phase.
- Method of revision
  - Main text should not be changed and instead, some statements are added to the commentary of the guide.
  - Preparation of relevant document which complements the added statements to show detailed policy for use of risks during operation phase.
- Points of issue
  - What requirements are demanded by the guide for operation phase?
    - No change of safety function class from design to operation phase
    - Table of SSC classification attached to commentary is example.
  - When the maintenance level is determined separately from the SSC classification attached to commentary under above conditions, are there any contradictions to the guide?
Statements added to Commentary

Contents of commentary of the guide
- I. Objective
- II. – IV. (An omission)
- V. Design Consideration for SSCs with Safety Functions
  - (An omission) Specific measures to be taken for ensuring the required level of reliability in the operation depend on the characteristics, etc. of individual SSCs. Therefore, specific measures to meet the individual reliability requirements shall be adequately determined in the light of the fundamental objectives of this guide.

Added the following in present revision
- For example, when the concrete measures or requirements for maintenance of SSC are determined for operation phase, it is adequate to refer to risks such as operational experience and/or PSA results maintaining the safety function specified in this guide. This reflects recent progress in PSA technology as well as the viewpoint of enhancement of scientific rationality, consistency and transparency in nuclear safety and appropriate allocation of limited resources.

Highlight of Relevant Document
- It is expected that activities of regulatory body and utilities be more detailed and effective by utilization of risks in operation and higher reliability be maintained by appropriate allocation of limited resources.
- The guide prescribes that classification of safety function should be kept from design to operation phase while table of SSC classification attached to the commentary is an example (beyond the guide).
- Although the level of maintenance for SSC with safety function has been conventionally determined based on examples in the guide, it is more appropriate to be realized considering also risk significance.
- By doing so, even though the SSC is assigned to high safety importance in the examples, the level of maintenance can be changed to appropriate method in case of low risk significance as far as the safety function is kept.
- It is preferable for utilities to use PSA that reflects actual design and operation management considering voluntarily performed AM.
- Adequacy of PSA results and level of maintenance determined by reference to risks should be sufficiently confirmed by both utilities and regulatory body.
Example of Use of risks for inspection

<table>
<thead>
<tr>
<th>System</th>
<th>Components</th>
<th>Risk significance</th>
<th>Safety importance classification guide</th>
<th>Maintenance significance</th>
</tr>
</thead>
<tbody>
<tr>
<td>High press. core spray</td>
<td>Motor valve, Strainer, etc.</td>
<td>High, Low</td>
<td>MS-1</td>
<td>High</td>
</tr>
<tr>
<td>Low press. core spray</td>
<td>Motor valve, Strainer, etc.</td>
<td>Low</td>
<td>MS-1</td>
<td>High (present), Low (future)</td>
</tr>
</tbody>
</table>

Criteria for risk significance
High: F-V ≥ 0.005 or RAW ≥ 2, Low: F-V < 0.005 and RAW < 2

Maintenance Significance for SSC
High: Preventive maintenance - Time or condition based maintenance
Low: Breakdown maintenance partially including preventive maintenance

Utilization of risk information in the maintenance program would enable further effective and individually-targeted maintenance / inspection for SSC.

Summary

- The present revision of the safety importance classification guide clarified that what the guide requires for operation phase is the preservation of safety functions determined at design phase.
- As long as the safety functions are maintained, the maintenance level for SSCs can be determined (including downgrade) using risks separately from the classification examples in the guide and it does not contradict the fundamental policy of the guide.
- Present revision showed the way to utilization of risk information which allows that the less conservative safety classification between the guide and PSA results can be applied to the maintenance significance for SSCs if the trouble information data and experiences are accumulated in future.
Table of Contents

I. Background
II. Selection of the performance measures
III. Development of the performance goal criteria
IV. Application principles
V. Summary
I. Background

I.1 Policy on the Severe Accident (2001.8)

- **Safety Goals** (Quantitative Health Objectives: QHOs)
  - The risk of prompt fatalities to an average individual in the vicinity of the nuclear power plant (NPP) should not exceed 0.1% of the risks resulting from other accident.
  - The risk of cancer fatalities to the population in the area of nuclear power plant should not exceed 0.1% of the sum of cancer fatality risks from all other causes.

- **To facilitate the achievement**, 
  - The performance goals will be set considering prevention of core damage and reduction of radioactive material release.

I.2 Roles of the performance goals

- **Needs to decide whether a NPP satisfy the safety goals**
  - Risk surrogates that could be directly compared to the QHOs
  - Satisfaction of the performance goals means
    - achievement of the safety goals.
    - meeting the engineering objectives for the operation and design of NPPs

- **Criteria for RIR/RIA**

- **Requirements for the performance goals**
  - "Prevention of core damage and reduction of radioactive material release" (The policy on severe accident)
  - Results at each levels of PSA can be the performance measures.

- PSA of NPPs
  - Level 1 PSA : Core damage frequency, etc.
  - Level 2 PSA : Determination of release frequency, etc.
  - Level 3 PSA : Risks to the public, etc.
II. Selection of the performance measures

II.1 Selected Measures

- **Must conform to the policy on severe accident**
  - Prevention of core damage and reduction of radioactive material release

- **Considerations**
  - Representative parameters of facilities related to the core integrity (Level 1 PSA) and containment isolation function (Level 2 PSA)
  - Should be clearly defined to easily quantify.
  - Sufficient international application experience and following domestic practice in the similar application.

⇒ **Selected Measures for the performance goals**
  - First measure: Core Damage Frequency (CDF)
  - Second Measure: Large Early Release Frequency (LERF)

II.2 Core Damage Frequency (CDF)

- **Definition**
  : the frequency of an accident which can cause the fuel in the reactor to be damaged

  - **PWR**
    - Core damage corresponds to the case where a peak clad temperature goes above a threshold criteria
    - In Korea, the threshold is 1204 °C (Notice of MEST 2008-16: Performance of ECC system)

  - **CANDU**
    - Severe core damage: failure of two or more fuel channels
II.3 Large Early Release Frequency (LERF)

Definition

- the frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects (NUREG/CR-6595)

Definitions in the Application

Method 1:

- NUREG/CR-6595, Appendix A.2 First definition
  - Early containment failure
  - Containment bypass
  - Containment isolation failure

※ KHNP used this definition in the PSA of domestic NPPs.

Method 2: LERF definition 2

- Accidents with short (<6 hours) delay time from core uncovery to atmospheric release of radioactive materials based on PSA results.
  - Case 1 + Containment failure before reactor vessel break

- This case obtained by investigating the available evacuation time from the data reported by the licensee’s Level 2 PSA. They are as follows.

  1) Early containment failure
     - core uncovery ~ containment failure : 1.75 ~ 6.6h
  2) Containment bypass
     - core uncovery ~ atmospheric release : 0.1 ~ 6.4h
  3) Containment isolation failure
     - core uncovery ~ atmospheric release : 0.3 ~ 0.77h
  4) Containment failure before reactor vessel break
     - core uncovery ~ atmospheric release : 0.5 ~ 1.0 h
Method 3: LERF definition 3
- NUREG/CR-6595, Appendix A.2 Second definition
  : LERF consists of the frequency of release classes associated with the early failure and bypass containment failure modes which have release fractions of the volatile/semi-volatile fission products (Iodine, Cesium, Tellurium) equal to or greater than about 2.5% to 3%.

Method 4: LERF definition 4
- NUREG/CR-6595, Appendix A.2 third definition
  : LERF is the frequency of early failure and bypass containment failure modes that have a release fraction of iodine equal to or greater than about 10%.

III. Development of the Performance goal criteria

III.1 Procedures

1. Evaluation of the risks corresponding to safety goals
   ✓ The risk of accident and the risk of cancer fatality from other causes (obtained based on the statistical data: 1983–2006)

2. Evaluation of the conditional probability of prompt and cancer fatality

3. Determination of the Performance goals
   ✓ LERF criteria to satisfy the goal of early fatality
   ✓ CDF criteria to satisfy the goal of cancer fatality
III.2 Evaluation of the risks corresponding to safety goals

**Risk of early and cancer fatalities in Korea**
(From KNSO, 1983~2006)

- **Accident fatality**
  - 69.35 per 100,000 annually (Fig. 2)
    : Average risk = 6.935×10⁻⁴/y, 6.0×10⁻⁵/y after the year 2000
  - Risk of early fatality corresponding to the safety goal
    : 6.0×10⁻⁵/y×0.1% = 6×10⁻⁷/y
    : Conservatively, 5×10⁻⁷/y selected as a goal

- **Cancer fatality**
  - 111.49 per 100,000 annually (Fig. 2)
    : Average risk = 1.115×10⁻³/y
    : It is continuously increasing: 7.2×10⁻⁴/y→ 1.37×10⁻³/y
  - Risk of cancer fatality corresponding to the safety goal
    : 1.37×10⁻³/y×0.1% = 1.37×10⁻⁶/y
    : Conservatively, 1×10⁻⁶/y selected as a goal
III.3 LERF criteria to satisfy the goal of early fatality

- Individual Early Risk (IER)

\[
IER = \sum_{i} \text{LERF}_i \times \text{CPEF}_i = \text{LERF} \times \text{CPEF}_{\text{AVG}}
\]  

where,

- \( \text{LERF}_i \) : frequency of the release capable of causing early fatalities for LERF accident sequence \( "n" \)
- \( \text{CPEF}_i \) : conditional probability of early fatality

\[
\text{LERF} = \sum_{i} \text{LERF}_i
\]

\[
\text{CPEF}_{\text{AVG}} = \frac{\sum_{i} \text{LERF}_i \times \text{CPEF}_i}{\text{LERF}} : \text{average of CPEFs}
\]
### Conditional probability of early fatality (CPEF$_n$)

- Conditional probability of an individual becoming a prompt (or early) fatality for an accident sequence “n”

\[
CPEF_n = \frac{EF_n}{TP(1.6km)}
\]  

(2)

where,

- \(EF_n\) : number of early fatalities within 1.6km conditional on the occurrence of accident sequence “n”
- \(TP(1.6km)\) : total population to 1.6 km

- Evaluation of CPEF$_n$ using MACCS2 code based on
  - Source term information derived from Level 2 PSA, and
  - Site-specific meteorological data, population distribution with emergency response scenario

### Derivation of LERF criteria

- CPEF for the determination of performance goal :
  - For conservatism, we selected the CPEF of the case only with dose dependent relocation without evacuation for internal events
  - Various definitions of LERF does not make significant differences in CPEF, thus, in this study, we used the LERF definition 2 (Available time for evacuation < 6h).

- Performance goal for the early fatality

\[
IER = LERF \times CPEF < 5.0 \times 10^{-7} / y
\]

- CPEF : 1.63x10$^{-7}$ (Wolsong) ~ 7.0x10$^{-5}$ (Ulchin3,4)

\[\Rightarrow\text{LERF} : 7.0 \times 10^{-6}/y \sim 3.0 \times 10^{-4}/y\]
III.4 CDF criteria to satisfy the goal of cancer fatality

**Individual Latent Risk (ILR)**

\[
ILR = \sum_{n} F_n \times CPLF_n = CFF \times CPLF_{AVG}
\]  \hspace{1cm} (3)

where,

- \( F_n \) : frequency of occurrence of accident sequence “n”
- \( CPLF_n \) : Conditional probability of cancer fatality
- \( CFF = \sum_{n} F_n \) : Containment failure frequency
- \( CPLF_{AVG} = \frac{\sum_{n} F_n \times CPLF_n}{CFF} \) : Average value of CPLF \( n \)

**Conditional probability of cancer fatality (CPLF\( _n \))**

- Conditional probability of cancer fatality for an accident sequence “n”

\[
CPLF_n = \frac{LF_n}{TP(8.0km)}
\]  \hspace{1cm} (4)

where,

- \( LF_n \) : number of cancer fatality within 8.0 km
- \( TP(8.0km) \) : total population to 8.0 km

- Evaluation of CPLF\( _n \) using MACCS2 code based on
  - Source term information derived from Level 2 PSA, and
  - Site-specific meteorological data, population distribution with emergency response scenario
Derivation of CDF criteria

- Selection of CPLF for the performance goal
  - The selected criteria must be used for all Korean NPPs
    - For conservatism, the case with no protective action was selected.
    - CPLF for Kori 2 \( (6.25 \times 10^{-3}) \) were selected
  - Performance for the cancer fatality
    \[
    ILR = CFF \times 6.25 \times 10^{-3} < 1.0 \times 10^{-6} / y
    \]
    - if we set the conditional containment failure probability (CCFP) equal 1, then
    \[
    \Rightarrow CFF = CCFP \times CDF = CDF
    \]

Core Damage Frequency (CDF) \( < 1.0 \times 10^{-4} \text{/년} \)

IV. Application Principles

IV.1 Application level

Safety targets or objectives rather than strict limits

- Due to
  - the Uncertainty and insufficient analysis range of PSA
  - Consideration of the plant designed before PSA application
  - Not sufficient experience of the risk application

Application area of the performance goals

- Risk criteria for the NPP design and operation
- Criteria for RIR/RIA
- Parameters representing the plant safety level
IV.2 Application principles

**CANDU plants**
- Although the design concepts are different,
- Same performance goals to the PWR should be used because the defense in-depth concept (prevention of core damage and mitigation of the atmospheric release) applies also.

**Definition of core damage for the performance goal**
- PWR: Peak clad temperature >1204°C
- CANDU: Failure of two or more fuel channels

IV.2 Application principles (continued)

**New plants**
- The increase of risk due to the addition of new nuclear power plants should be low as much as possible.
- Goals: One tenth of that of operating nuclear power plant

**Initiating events**
- All initiating (internal, external) events must be considered. (except security and physical protection)
V. Summary

- **Development of the performance goals**
  - Draft:
    - CDF : $1.0 \times 10^{-4}/y$ (same as the IAEA criteria)
    - LERF : $7.0 \times 10^{-6}/y$~$3.0 \times 10^{-4}/y$ (Need further research)

- **Future works**
  - Need to clarify the following topics
    - CDF definition in CANDU reactor
    - Uncertainties (especially related to the Level 2 and 3 PSA)
    - Consideration on the application of initiating events
    - Application strategy of performance goals to actual plants.
A Risk-Informed and Performance-Based Approach for Improving Regulatory Inspection Program in Korea

2009. 5. 18

Dae-Wook Chung, Nam-Chul Cho
dwchung@kins.re.kr

Korea Institute of Nuclear Safety

CONTENTS

1. Introduction
   - Current Status and Perspective of Nuclear Power in Korea
   - Summary of PSA and Risk-Informed Activities

2. Approach to Improving Regulatory Inspection Program
   - Overall Framework for Graded Regulation
   - Developments of Risk-Informed Periodic Inspection (RIPI) Program
   - Integrated Safety Performance Assessment (ISPA) Program

3. Graded Periodic Inspection (GPI) Program

4. Development of Individual ISPA Sub-Programs
   - KINS-SEIF Program
   - Risk-Informing KINS Safety Performance Indicator (SPI) Program

5. Future Works
1. Introduction

- Current Status and Perspectives of Nuclear Power
  - 20 NPPs are in operation
    - Nuclear power accounts for around 40% of national electricity supply
  - 4 more NPPs are under construction (APR-1400, advanced type)
  - Furthermore, national energy strategy is announced recently
    - Number of NPPs will be almost doubled by 2030
    - Nuclear power will account for more than 60% of national electricity supply
  - As regulator, the KINS needs more effective and efficient way in nuclear safety regulation through;
    - Managing and improving safety more effective way
    - Distributing regulatory resources more efficient way
    - Encouraging the licensees to voluntarily improve safety performance

- Summary of PSA and Risk-Informed Activities
  - Since 1989, PSA has been one of licensing submittals for new NPP
  - By “Severe Accident Policy Statement”, at least level 2 PSA for all operating nuclear units had been completed in 2006, and subject to periodic update
  - In parallel, extensive R&D programs have been underway by both KINS and industry for more than 10 years
  - Since 2006, the KINS has been working on the development and implementation of risk-informed regulation as appropriate, mostly for
    - Review of licensee application to RI-ISI and RI-STI/AOT changes and
    - Improving regulatory inspection program
2. Approach to Improving Regulatory Inspection Program

- Developments of Risk-Informed Periodic Inspection (RIPI) Program

  - Basic Ideas
    - Incorporating risk-significant failure events into related inspection item, and
    - Adjusting inspection resources commensurate with risk significance and performance records

  - Key improvements are focused on those inspection items related to the prevention (or minimization) of highly risk significant
    - Common cause failure events,
    - Post-accident operator errors (i.e., errors during EOP performance) events, and
    - Root causes of independent failure events

  - RIPI program has been developed and incorporated into the regulatory inspection program for all 20 operating NPPs since 2006
2. Approach to Improving Regulatory Inspection Program

Development of RIPI Program for Each NPP

- Plant-specific RIPI program
- Improving inspection items of concern
- Identify risk-significant failure events and relevant inspection items
- Existing inspection items by Rx type
- Risk-Based approach and Op. Experiences
- Plant-specific PSA

Basic Concept of Improving Inspection Items

- Risk Significance

Integrated Safety Performance Assessment (ISPA) Program

- Quantify Safety Performance Results and Determine Safety Performance Grade (SPG)

Regulatory Inspection Program

Risk Assessment of Operational Accident/Event

Risk-Informed Performance Indicator

Maintenance Effectiveness Monitoring

Assessment of Licensee Capability

Operational Risk Management

Risk Monitoring Program

MPAS Regulatory PSA Model

Significance Evaluation of Inspection Findings (KINS-SEP)

Risk-Informing KINS-SEP

Pilot Implementation of MR

Human Factor

Inspections on Human Factor

Safety Culture

General Status of Inspection Results

Conduct of Periodic Inspection

Etc.
3. Graded Periodic Inspection (GPI) Program

Basic Ideas

- Determine the "safety performance grade (SPG)" of each NPP based on the result of ISPA program

<table>
<thead>
<tr>
<th>SPG</th>
<th>Excellent</th>
<th>Average</th>
<th>Poor</th>
</tr>
</thead>
<tbody>
<tr>
<td>&lt;20%</td>
<td>70-100%</td>
<td>&lt;10%</td>
<td></td>
</tr>
</tbody>
</table>

- The inspection program is differentiated by the SPG of each NPP

  | Excellent | Grade: Conduct relaxed IPIP* (benefit) |
  | Average   | Grade: Conduct IPIP                     |
  | Poor      | Grade: Conduct Enhanced IPIP**          |

  * Inspection items are relaxed and inspection resources are decreased (~20%)
  ** Inspection items remain the same and inspection resources are increased (~30%)

- With GPI program, it is expected the KINS is able to
  - Managing and improving safety more effective way
  - Distributing regulatory resources more efficient way
  - Encouraging the licensees to voluntarily improve safety performance

---

3. Graded Periodic Inspection (GPI) Program

Estimated Savings in Inspection Man-days (Example)

<table>
<thead>
<tr>
<th>SPG (No. of Units)</th>
<th>Minimum Man-days</th>
<th>Average Man-days</th>
<th>Maximum Man-days</th>
</tr>
</thead>
<tbody>
<tr>
<td>Excellent (4)</td>
<td>256</td>
<td>280</td>
<td>304</td>
</tr>
<tr>
<td>Average (14)</td>
<td>1064</td>
<td>1148</td>
<td>1232</td>
</tr>
<tr>
<td>Poor (2)</td>
<td>176</td>
<td>188</td>
<td>200</td>
</tr>
<tr>
<td>Total (20)</td>
<td>1496 (74.8%)</td>
<td>1516 (80.3%)</td>
<td>1736 (86.8%)</td>
</tr>
</tbody>
</table>

- 100 man-days are assumed to be needed per each unit with Existing Inspection Program

---

Estimated Savings in Inspection Man-days (Example)
4. Development of Individual ISPA Sub-Programs

- **KINS-SEIF Program**
  - KINS SEIF stands for “KINS Significance Evaluation of Inspection Findings”
  - PC-based fast-running, easy-to-handle computer program for the inspectors to evaluate risk significance of inspection findings
    - The inspector himself/herself can perform the evaluation at the site
    - For every inspection finding, both qualitative and quantitative evaluations are available in the program provided the affected component is modeled in PSA
    - Multiple components/systems affected by the inspection finding can be selected for calculating $\Delta$CDF of inspection finding
    - The inspector can be aware of the risk insights of inspection finding he identifies
    - All evaluations are reported and managed in KINS headquarter via internet
  - Two types of risk information are produced
    - Absolute $\Delta$CDF (and color) of inspection finding regardless of base CDF for providing input to ISPA process (subject to detailed evaluation by risk analyst)
    - Relative portion of $\Delta$CDF of inspection finding to base CDF for providing insight to inspectors
  - The development of KINS-SEIF program is completed and beta version will be released to selected inspectors in May 2009 for final comments

---

**Evaluation Process of KINS-SEIF Program**

1. **Inspection Finding**
2. **Mapping down process**
3. **Identify and select affected system(s)/component(s)**
   - **Modeled in PSA?**
     - YES
     - NO
6. **Qualitative Evaluation Process**
7. **Risk Significance Evaluation**
   - **$\Delta$CDF/CDF**
     - <10\(^{-6}\)
     - 10\(^{-6}\) - 10\(^{-5}\)
     - 10\(^{-5}\) - 10\(^{-4}\)
     - >10\(^{-4}\)
9. **Normal**
10. **Caution**
11. **Important**
12. **Inspectors**
13. **ISPA**
4. Development of Individual ISPA Sub-Programs

Risk-Informing KINS Safety Performance Indicators (SPI) Program

- KINS already operates safety performance indicators (SPI) program to measure the safety performance status of each licensee and it is open to the public.
- The SPI program is being risk-informed using MSPI and USwC approaches for the use in the ISPA program as follows:

<table>
<thead>
<tr>
<th>Risk-Informed SPI</th>
<th>KINS SPI (Existing)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Category</td>
<td>Performance Indicator</td>
</tr>
<tr>
<td>Initiating Events</td>
<td>Simple Reactor Trip</td>
</tr>
<tr>
<td>Unplanned Scram with Complication (USwC)</td>
<td>Power Change</td>
</tr>
<tr>
<td>Power Change</td>
<td></td>
</tr>
<tr>
<td>Mitigating System</td>
<td>MSPI – EDG System</td>
</tr>
<tr>
<td></td>
<td>MSPI – HPSIS</td>
</tr>
<tr>
<td></td>
<td>MSPI – AFWS</td>
</tr>
<tr>
<td></td>
<td>MSPI – RHRS</td>
</tr>
<tr>
<td></td>
<td>MSPI – CWS</td>
</tr>
</tbody>
</table>
5. Future Works

Future works needed
- Quantification process for each individual ISPA sub-program
- Determination of weighting factors among sub-programs
- Finalization of Rx type specific RIPI program
- Finalization of categorization criteria for safety performance grade (SPG)
- Others, if necessary
DEVELOPMENT OF RISK EVALUATION PROGRAM SEIF FOR INSPECTION FINDINGS

Huichang Yang
ENESYS Co., Ltd.
Nam Chul Cho, Dae Wook Chung, Chang Joo Lee
Korea Institute of Nuclear Safety
2009. 5. 18


CONTENTS

❖ BACKGROUND AND OBJECTIVES
❖ METHODOLOGY
❖ FUNCTION AND STRUCTURE OF SEIF
❖ EXAMPLE APPLICATION
❖ FUTURE WORKS
BACKGROUND AND OBJECTIVES

Background

- USNRC has been implemented Significance Determination Process (SDP) under Reactor Oversight Process (ROP) for years.
- In Korea, Integrated Safety and Performance Assessment (ISPA) Program, in which the processes to evaluate and assess the safety and performance from quantitative and qualitative viewpoints, are included, is being developed as a part of graded regulation framework.

Objectives

- Significance Evaluation of Inspection Findings (SEIF) has been developed to:
  - Provide risk significance of inspection findings,
  - Provide information which can be used in grading regulatory activity, and
  - Provide supporting tool for ISPA as an implementation plan of risk and performance-based graded regulation.
METHODOLOGY

Significance Evaluation Methodology

- Identification of Information in Inspection Findings
  - Contents of inspection findings/SSCs
  - Major safety functions of identified SSCs
  - PSA-scoped SSCs
- Qualitative and/or Quantitative Evaluation
  - for PSA-scoped SSCs: quantitative evaluation using MPAS/AIMS/FTREX
  - for non PSA-scoped SSCs: qualitative evaluation
  - significance is categorized as GREEN/WHITE/YELLOW/RED
  - significance of inspection fields is categorized as NORMAL/CAUTION/SIGNIFICANT

METHODOLOGY (continued)

Significance Evaluation Methodology (continued)

- Qualitative and/or Quantitative Evaluation (continued)
  - Significance Determination Criteria

Different regulatory action will be imposed depending on the significance category
- Framework of graded regulation is being developed
METHODOLOGY

**Significance Evaluation Methodology**

- Qualitative and/or Quantitative Evaluation
  - Qualitative evaluation
    - Evaluate the significance in terms of
      - Initiating events
      - Mitigating systems
      - Integrity of defense-in-depth barriers
  - Phase 1 evaluation of SDP was referred

**SEIF Evaluation Process**

- Inspection Findings
- Identification of SSCs
- Identification of PSA-scoped SSCs
- Quantitative Evaluation
- Risk Significance
- SSCs Mapping to PSA Events
- Detailed Analysis By Expert Panel
- ISPA

- CDF/CDF
  - <10%
  - 10^-6 - 10^-5
  - 10^-5 - 10^-4
  - >10^-4
  - NORMAL
  - <10%
  - 10% - 100%
  - >100%
  - CAUTION
  - >100%
  - SIGNIFICANT

- CDF
  - <10^4
  - 10^-6 - 10^-4
  - 10^-4 - 10^-3
  - >10^-3

- No Match to Qualitative Evaluation Criteria
FUNCTION AND STRUCTURE OF SEIF

- Identification of Inspection Finding
  - Plant/unit
  - O/H Info., inspection fields and relevant SSCs
  - SSCs status (unavailability) and other assumptions
  - Evaluation method selection

FUNCTION AND STRUCTURE OF SEIF

- Quantitative Evaluation
  - SSCs mapping to events in PSA model
  - CDF calculation using regulatory PSA model and AIMS*/FTREX*
  * Developed by KAERI
FUNCTION AND STRUCTURE OF SEIF

Quantitative Evaluation
Results
- CDF, ACDF, evaluated category
- Risk significance of relevant inspection fields

Qualitative Evaluation
- Initiating events
- Mitigating systems
- DID barrier integrity
- For inspection findings regarded to be significant, those will be transferred to detailed analysis process and/or expert panel
FUNCTION AND STRUCTURE OF SEIF

- Qualitative Evaluation
  - Detailed inquiries for
    - Initiating events
    - Mitigating systems
    - DID Barrier integrity
    - USNRC SDP phase 1 process was referred

EXAMPLE APPLICATION

- Application of SEIF to Example Inspection Findings

<table>
<thead>
<tr>
<th>No.</th>
<th>Description</th>
<th>Year</th>
<th>CDF</th>
<th>Evaluation Result</th>
<th>ADIP</th>
</tr>
</thead>
<tbody>
<tr>
<td>1</td>
<td>Inappropriate in-service test for safety class valves</td>
<td>2004</td>
<td>7.6E-03</td>
<td>RED</td>
<td>7.6E-03</td>
</tr>
<tr>
<td>2</td>
<td>Inappropriate test of minimum flow differential pressure for CS pump</td>
<td>2006</td>
<td>6.2E-05</td>
<td>YELLOW</td>
<td>6.2E-05</td>
</tr>
<tr>
<td>3</td>
<td>Inappropriate temperature control for primary-secondary component cooling water</td>
<td>2004</td>
<td>9.7E-03</td>
<td>RED</td>
<td>2.6E-03</td>
</tr>
<tr>
<td>4</td>
<td>Shutdown cooling system sampling valve leakage</td>
<td>2006</td>
<td>GREEN</td>
<td>-</td>
<td>-</td>
</tr>
<tr>
<td>5</td>
<td>Lack of procedures for EDG and related equipment surveillance</td>
<td>2007</td>
<td>1.3E-04</td>
<td>RED</td>
<td>1.3E-04</td>
</tr>
<tr>
<td>6</td>
<td>Inappropriate activity on the exceed of performance parameters of EDG</td>
<td>2008</td>
<td>1.2E-06</td>
<td>WHITE</td>
<td>7.0E-06</td>
</tr>
<tr>
<td>7</td>
<td>Lack of periodic test procedures and lack of periodic tests for condensate vacuum system</td>
<td>2005</td>
<td>1.6E-05</td>
<td>YELLOW</td>
<td>1.0E-05</td>
</tr>
</tbody>
</table>

Note: Reference CDF is 5.49e-06/Ry.
FUTURE WORKS

- **Methodology Enhancement and Modification**
  - Evaluation process and methodology modification through pilot application
  - Qualitative Significance Evaluation Method Enhancement and Refinement

- **Development of Interface with KINS Information System and Web-based SEIF**

- **Development of Accident/Event Significance Evaluation (ASP) Methodology and Evaluation Module**
The Improvement of Regulatory Inspection System Utilizing “Risk Information” in Japan

18th – 20th May, 2009
The 10th Korea-Japan Joint Workshop on PSA

T. Uchida¹, T. Miyazaki², Y. Kasagawa², M. Sugawara¹, S. Miura³ and M. Yamashita¹

1: Probabilistic Safety Assessment Group
   Nuclear Safety Analysis and Evaluation Office
   Nuclear Energy System Safety Division
2: Inspection Engineering Group
   Inspection Affairs Division
3: Policy Planning and Coordination Division
Japan Nuclear Energy Safety Organization (JNES)

Contents

1. Direction of the improvement of the regulatory inspection system
2. Utilization of “Risk Information” to review Maintenance Program
   2.1 Utilization of “Risk Information” to identify “Importance of Systems / Functions for Maintenance”
   2.2 Utilization of “Risk Information” to Performance Criteria (PC)
3. Utilization of “Risk Information” to identify the safety significant activities for safety preservation regulatory inspection
4. Significance Determination Process (SDP)
1. Direction of the improvement of the regulatory inspection system in Japan

Background
- NISA, JNES and utilities have extensively discussed the way of regulatory inspection system for NPPs.
- Based on the discussion, NISA issued the interim report showing the direction of the improvement of regulatory inspection system for NPPs including the utilization of “Risk Information” to the system. (See the next page)
- NISA, JNES and utilities have been extensively preparing the “New Inspection System for NPPs”, which started since January 2009.
  - NISA improved the ordinances of METI in order to introduce the new regulatory inspection system. These ordinances were issued on January 2009.
  - NISA and JNES have developed the requirements on the maintenance program, regulatory review procedures, methodologies for the comprehensive plant evaluation of the safety performance of NPPs and so on.
  - Utilities have improved industry association level codes / guides and safety preservation rules.

Direction of the improvement of regulatory inspection system of NPPs on the viewpoint of the utilization of “Risk Information”

1. Enhancement of the inspection system for maintenance activities based on the maintenance program

   Application of “Risk Information” to review maintenance program
   - NISA/JNES uses the above “Risk Information” to……
     - The review of the importance of SSCs for maintenance, which utilities established
     - The review of performance criteria

2. Introduction of intensive inspection activities focusing upon safety significant preservation activities

   Application of “Risk Information” to identify the safety significant activities of utilities, which safety preservation regulatory inspection should be focused on.
   - Identification of utilities’ activities, which significantly affect risk of NPPs
   - Assessment of risk impacts of system configuration controls (e.g. management of outage schedules for maintenance of safety related systems)

3. Ensuring the plant safety intensively through comprehensive evaluation of individual plant features

   Application of “Risk Information” to the comprehensive regulatory assessment on safety performance of NPPs
   - Development of methods and criteria for the safety performance assessment with “Risk Information”
     - Significance Determination Process (SDP)
     - Performance Indicators (PIs)
2. Utilization of “Risk Information” to review Maintenance Program

Utilities improve their maintenance activities with “Risk Information”:
- Establish the Importance of Systems & Functions for Maintenance with “Importance of Safety Functions by NSC” as well as “Risk Information”
- Establish Performance Criteria for Important Systems and Functions
- Establish Maintenance Plan based on Importance of Systems & Functions for Maintenance

NSC: Nuclear Safety Commission

2.1 Utilization of “Risk Information” to identify “Importance of Systems / Functions for Maintenance”

JEAC4209-2007* requires to establish the importance of Systems / Functions for maintenance according to “Importance of safety functions issued by NSC” as well as “Risk Information”

NISA/JNES review on the following viewpoint:

1. “Importance of Systems / Functions for Maintenance” should be established based on “Risk Information” as well as “Importance of Safety Functions” issued by NSC

2. The following systems / functions should be identified as important for maintenance:
   - Systems / Functions, which are classified to Class-1 & 2 of “Importance of Safety Functions”
   - Systems / Functions of with “High” risk importance

2.2 Utilization of “Risk Information” to Performance Criteria (PC)

- JEAC4209-2007 requires to establish the Performance Criteria (PC), which shows the performance of plant level and system level, based on “Importance of Systems / Functions for Maintenance”.

- JEAC4209-2007 requires to establish the quantitative objectives for PC considering operation experience (e.g. system unavailability), the importance for maintenance and Allowable Outage Time (AOT).

**NISA/JNES reviews on the following viewpoint:**

1. PCs should consist of “Plant Level” & “System Level” criteria
2. Followings should be included in “Plant Level” criteria
   - Number of “Unplanned SCRAMs”
   - Number of “Unplanned Power Changes (>5% of rated power)”
   - Number of “Unplanned ESF Actuations”
3. “System Level” criteria should consist of MPFF & UA-H
   - Criteria should be assigned to the systems / functions of “High Importance for Maintenance”
   - Objectives of criteria should be assigned in consideration with “Importance of Maintenance”, AOT & ICCDP corresponding to objectives UA-H
An example of Performance Criteria

<table>
<thead>
<tr>
<th>Level</th>
<th>Description</th>
<th>Examples</th>
<th>Example</th>
</tr>
</thead>
<tbody>
<tr>
<td>Plant Level</td>
<td>Generic criteria of plant safety performance provided by maintenance activities regardless of “Importance for Maintenance”.</td>
<td>Number of unplanned SCRAMs/7000hrs</td>
<td>Unavailability hours (UA-H)</td>
</tr>
<tr>
<td>System Level</td>
<td>Individual indicators of the performance of safety important systems / functions, showing the effectiveness of maintenance activities for safety.</td>
<td>Number of unplanned power changes/7000hrs</td>
<td>Outage Time of Mitigation Systems (stand-by-system only)</td>
</tr>
<tr>
<td>Component Level</td>
<td></td>
<td>Number of unplanned actuation of ESF</td>
<td>Maintenance Preventive Functional Failures (MPFF)</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Number of functional failure of components which can cause system functional failures, which could be caused by such as poor maintenance.</td>
<td></td>
</tr>
</tbody>
</table>


An Example of quantitative objectives of PC (MPFF)

<table>
<thead>
<tr>
<th>Importance of Safety Functions</th>
<th>Class-1</th>
<th>Class-2</th>
<th>Class-3</th>
<th>Others (e.g. AM candidates)</th>
</tr>
</thead>
<tbody>
<tr>
<td>(High)</td>
<td>As Reliable as Reasonably Achieved</td>
<td>Highly Reliable</td>
<td>As Reliable as General Industries</td>
<td>No Objective</td>
</tr>
<tr>
<td>(Low)</td>
<td>MPFF &lt;&lt; 1/Cycle</td>
<td>MPFF &lt;&lt; 2/Cycle</td>
<td>MPFF ≤ 3/Cycle</td>
<td>No Objective</td>
</tr>
</tbody>
</table>

Consideration of Risk Importance

Final Quantitative Objectives

- **High** Risk Importance: MPFF < 1/Cycle
- **Low** Risk Importance: No Objective
3. Utilization of “Risk Information” to identify the safety significant activities for safety preservation regulatory inspection

Utilities’ activities, which significantly affect risk
- Quantitatively assessment of the changes of risk considering the reactor water level, configurations of mitigation systems, human errors and failures of SSCs
- Identification of hold points referring to change of risk.

Identification of items, which should be verified on the viewpoints of safety in the next plant condition, at hold points:
- e.g., prevention functions and mitigation systems in the next plant operating condition are prepared

Utilities’ safety significant activities identified can be utilized to improve “Rules for the Installation, Operation, etc. of Commercial Power Reactors”
- Witness inspection for utilities’ safety significant activities
- on the spot entry and inspection in case of derogation from LCO

4. Significance Determination Process (SDP)

Image of Comprehensive Plant Evaluation using PI and SDP

First stage screening of inspection findings

Evaluation of SDP for plant safety
- safety function
- plant risk etc.

Evaluation of SDP for QMS

Determination of safety significance of selected inspection findings

Inspection Findings

PI

Comprehensive Individual Plant Evaluation and Feedback of the output to the Next Inspection Program

Data acquisition and categorization

Evaluation of each PI

Determination of level of plant performance

Maintenance Activities by the Safety Preservation Rule

Quality Management System

Operation management, E&CO, surveillance, etc.

Periodic Safety Management Inspection (INES)

Nuclear Safety Inspection (NISA)

Licensee’s Periodic inspection

Inspection

Inspection by Regulatory Body

Periodic Inspection (NISA / JNES)

Licenses

Introduction of Significance Determination Process (SDP)

Introduction of Performance Indicator (PI)

Evalutation of each PI

Status of plant performance

Plant performance based on the maintenance activities specified by the technical specification

Incident Reports required by law (including cause analysis, countermeasure, etc.)
Image of SDP Flow for Plant Safety

- Inspection Findings (SIFs)
  - Minor SIFs ?
    - No
    - Yes
  - Categorization of SIFs
  - Evaluation of SDP for SIFs
    - Method - 1 SDP for Safety Function
    - Method - 2 SDP for Plant Risk
    - Method - 3 SDP for Public Radiation Effects
    - Method - 4 SDP for Occupational Radiation Effects
  - Integration of SDP results for Cross-Cutting SIFs
  - Determination of Safety Significance for SIFs (level - I, II, III, IV)

Risk-Informed Application

- Safety Significance level
  - High
  - Low

Image of Selection Flow of Minor SIFs for Safety Function and Plant Risk

- Selected Inspection Findings (SIFs)
  - Is the SIF corresponding to the following items?
    - Yes
    - No
    - (Example)
      - Inspection related
        - Violation of technical specification
        - Loss of safety function
      - Incident report related
        - Incident reported by law
        - Violation of limiting condition for operation (LCO)
  - Is the SIF corresponding to the degradation?
    - Yes
    - No
  - Is the SIF corresponding to the potential loss of function due to poor manual?
    - Yes
    - No
  - Is the SIF corresponding to the other loss of safety function?
    - Yes
    - No
  - Evaluation of SDP for SIFs (Safety Significance Level - I to IV)
  - Screen-out of less safety important inspection findings
  - Minor SIFs
Image of The SDP Evaluation Flow for Safety Function due to SIFs

Method – 1  SDP for Safety Function

<table>
<thead>
<tr>
<th>Safety grade</th>
<th>Consequence grade</th>
<th>Plant Risk</th>
</tr>
</thead>
<tbody>
<tr>
<td>Class 1 – SSCs</td>
<td>Loss of Safety Function</td>
<td>H, M</td>
</tr>
<tr>
<td></td>
<td>Influence to Safety Function</td>
<td>L, LL</td>
</tr>
<tr>
<td>Class 2 – SSCs</td>
<td>Loss of Safety Function</td>
<td>H, M</td>
</tr>
<tr>
<td></td>
<td>Influence to Safety Function</td>
<td>L, LL</td>
</tr>
<tr>
<td>Class 3 – SSCs</td>
<td>Loss of Safety Function</td>
<td>H</td>
</tr>
<tr>
<td></td>
<td>Influence to Safety Function</td>
<td>M, L, LL</td>
</tr>
<tr>
<td></td>
<td>Others (Violation of the safety preservation rule)</td>
<td></td>
</tr>
</tbody>
</table>

Image of The SDP Evaluation Flow for Plant Risk

Method – 2  SDP for Plant Risk

Selected Inspection Findings (SIFs)

- Is it affecting Plant Risk during rated power operation due to SIFs?
- No
  - Risk evaluation during shutdown operation
  - Risk evaluation due to external events
- Yes
  - Identification of initiating events to be evaluated due to SIFs.
  - Identification of the following:
    - system lost of function
    - fault duration
    - influenced systems due to SIFs

Simplified evaluation of the risk impact with ΔCDF, utilizing:
- PSA model of the reference plant,
- Check sheet designed for the reference plant.

Evaluation of the impact to the prevention system (i.e., contribution to the initiating event frequency)
Evaluation of the impact to the mitigation system (i.e., contribution to the unavailability of the system)

Comparison with risk criteria (ΔCDF) and assignment plant risk level for SIFs

- High
- Medium
- Low
- Low + Low

The unavailability of the mitigation system is estimated, reflecting the inspection finding.
(1) If the inspection finding affected cross-cutting areas (for example, the safety-function and the radiation effects), the safety significance should be determined using a higher level of each safety significance.

(2) For the inspection finding affecting both the safety-function and plant risk, the safety significance should be represented by that of safety-function which included the result of risk evaluation.

(3) If the SIFs affected only the plant risk, the plant risk level should be translated into the safety significance level as follows:

<table>
<thead>
<tr>
<th>Plant Risk Level</th>
<th>Safety Significance Level</th>
</tr>
</thead>
<tbody>
<tr>
<td>High</td>
<td>I</td>
</tr>
<tr>
<td>Medium</td>
<td>II</td>
</tr>
<tr>
<td>Low</td>
<td>III</td>
</tr>
<tr>
<td>Low - Low</td>
<td>IV</td>
</tr>
</tbody>
</table>

4. Summary

- The new regulatory inspection system for maintenance program started since January 2009. JNES has been reviewing the maintenance programs for individual NPPs under the request of NISA.
  - The request of NISA includes the review of “Importance of Systems / Functions” & Performance Criteria, those are established by utilities utilizing of “Risk Importance”.
  - In order to review these items in maintenance program, JNES has developed
    - “Risk Importance” for 15 types / 55 plants
    - Detail requirements for utilities
    - Review points & procedures

- The new regulatory inspection system for safety significant activities in safety preservation inspection started since January 2009. In order to support NISA, JNES
  - Identified the candidates of safety significant activities using “Risk Information”
  - Developed the information on the characteristics of system configurations and the timing of safety significant activities for safety preservation inspection manual.

- JNES has developed the methods and “Risk Information” for Comprehensive Plant Evaluation (CPE) including PI & SDP.
  - JNES has prepared the manuals for these evaluations via trial evaluation experiences
  - Brushing up the methods and manuals via trial use of CPE including PI & SDP in FY2009
  - After the trial use, NISA/JNES intend essential use of CPE including PI & SDP
A Study on the Risk-Informed Performance Indicators and Thresholds for Graded Regulation

Yong Suk Lee
FNC Technology Co., Ltd.
Nam Chul Cho, Dae Wook Chung, Chang Joo Lee
Korea Institute of Nuclear Safety

2009.5.18

CONTENTS

I. Introduction

II. New Indicator Feasibility Study

III. Threshold Re-Evaluation

IV. Summary and Future Work
1. Introduction

- The research for the development of risk-informed and performance-based regulatory oversight (Integrated Safety and Performance Assessment (ISPA) Program) is ongoing in KINS.

- The risk-informed PI (Performance Indicator) is one of the main components for risk-informed and performance-based regulatory oversight.

- In this study, feasibility of some risk-informed PIs to be used in ISPA program has been evaluated and threshold re-evaluation has been performed for currently used KINS PIs.
  - New Indicator feasibility study
    - For BFRI, USwC, MSPI
  - Threshold re-evaluation
    - For currently used KINS PIs in Reactor Safety Area
  - The results in this study requires further refinement, and have not been formally approved by the KINS

---

### Table: Grade & Color Coding (qualitative)

<table>
<thead>
<tr>
<th>Area</th>
<th>Category</th>
<th>Specific Performance Indicator</th>
<th>Grade</th>
<th>Color Coding (qualitatively)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Safety</td>
<td>Operational Safety</td>
<td>Unplanned Reactor Scram</td>
<td>Good</td>
<td>Green</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Unplanned Power Reduction</td>
<td>Normal</td>
<td>Yellow</td>
</tr>
<tr>
<td></td>
<td>Multiple Barrier</td>
<td>Post-Meltdown Reaction</td>
<td>Warning</td>
<td>Orange</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Reactor Coolant Level</td>
<td>Warning</td>
<td>Orange</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Contaminant Regeneration</td>
<td>Warning</td>
<td>Orange</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Emergency Preparedness</td>
<td>Warning</td>
<td>Orange</td>
</tr>
<tr>
<td></td>
<td>Safety System</td>
<td>SI System Unavailability</td>
<td>Good</td>
<td>Green</td>
</tr>
<tr>
<td></td>
<td></td>
<td>ES System Unavailability</td>
<td>Good</td>
<td>Green</td>
</tr>
<tr>
<td></td>
<td></td>
<td>SEW System Unavailability</td>
<td>Good</td>
<td>Green</td>
</tr>
<tr>
<td></td>
<td>Regulatory Safety</td>
<td>Radiation Collective Dose</td>
<td>Good</td>
<td>Green</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Pupil Dose (Administrative)</td>
<td>Good</td>
<td>Green</td>
</tr>
</tbody>
</table>

---

TS: Technical Specification
1. Introduction

- **Comparison of NRC and KINS PI (Reactor Safety)**

<table>
<thead>
<tr>
<th>NRC Performance Indicator</th>
<th>KINS Performance Indicator</th>
<th>Data Performance of PI in USA</th>
</tr>
</thead>
<tbody>
<tr>
<td>SFI (Safety Function Inventory)</td>
<td>Unplanned Reactor Scrub Unplanned Power Changes Unplanned Systems with Control System Failure</td>
<td>New KINS Event Feasibility Study</td>
</tr>
<tr>
<td>Initiating Event (Reactor Overload Process)</td>
<td>Safety System Functional Failure</td>
<td>Not Required</td>
</tr>
<tr>
<td></td>
<td>FSFI: Emergency AC Power System</td>
<td>FSFI: Emergency AC Power System</td>
</tr>
<tr>
<td></td>
<td>MPFS: Low Pressure Feedwater System</td>
<td>MPFS: Low Pressure Feedwater System</td>
</tr>
<tr>
<td></td>
<td>MPFS: High Pressure Feedwater System</td>
<td>MPFS: High Pressure Feedwater System</td>
</tr>
<tr>
<td></td>
<td>MPFS: Steam Removal Systems</td>
<td>MPFS: Steam Removal Systems</td>
</tr>
<tr>
<td></td>
<td>MPFS: Emergency Core Cooling Systems</td>
<td>MPFS: Emergency Core Cooling Systems</td>
</tr>
<tr>
<td></td>
<td>NRC Regulatory</td>
<td>NRC Regulatory</td>
</tr>
<tr>
<td></td>
<td>Regulatory Control System</td>
<td>Regulatory Control System</td>
</tr>
<tr>
<td></td>
<td>Core Containment</td>
<td>Core Containment</td>
</tr>
<tr>
<td></td>
<td>Incorporation</td>
<td>Incorporation</td>
</tr>
<tr>
<td></td>
<td>Unidentified</td>
<td>Unidentified</td>
</tr>
</tbody>
</table>

II. New Indicator Feasibility Study

- **Initiating Event - BRIIE (Baseline Index for Initiating Events)**
  - "Industry level" initiating event PI in USA
  - Scope (PWR)
    - TRAN, LOCHS, LOMFW, LOOP, LOAC, LDCC, SORV, LOIA, VSLOC, SGTR
    - Based on NUREG-5750 initiating event data (1987-1995), and risk significance (CDF contribution > 1%)
  - BRIIE expanded CDF risk coverage of initiating event in NRC from less than 20% (ROP) to approximately 60% by covering risk significant events,
  - BRIIE feasibility for Korean NPPs (scope)
    - BRIIE also covered approximately 60% of CDF for majority of Korean NPPs
II. New Indicator Feasibility Study

- **Initiating Event - BR1IE (Baseline Index for Initiating Events)**
  - BR1IE monitors individual initiating events at the industry level against performance-based prediction limit.
  - BR1IE feasibility for Korean NPPs (performance threshold):
    - Almost 1/5~1/2 of the prediction limits compared to US PWRs

<table>
<thead>
<tr>
<th>Initiating Event</th>
<th>Mean Frequency Year</th>
<th>Baseline Year</th>
<th>Expected Occurrence</th>
<th>90% Prediction Limit</th>
<th>90% Prediction 10 Year Limit</th>
</tr>
</thead>
<tbody>
<tr>
<td>GPRN</td>
<td>9.4E-01</td>
<td>229.82</td>
<td>195.31</td>
<td>16.38</td>
<td>21</td>
</tr>
<tr>
<td>LDCV</td>
<td>2.0E-01</td>
<td>229.82</td>
<td>195.31</td>
<td>3.87</td>
<td>6</td>
</tr>
<tr>
<td>LOFI</td>
<td>3.3E-02</td>
<td>229.82</td>
<td>195.31</td>
<td>0.36</td>
<td>3</td>
</tr>
<tr>
<td>LOP</td>
<td>2.0E-02</td>
<td>229.82</td>
<td>195.31</td>
<td>0.71</td>
<td>2</td>
</tr>
<tr>
<td>LOAC</td>
<td>1.4E-03</td>
<td>229.82</td>
<td>195.31</td>
<td>0.28</td>
<td>1</td>
</tr>
<tr>
<td>USDR</td>
<td>1.0E-02</td>
<td>229.82</td>
<td>195.31</td>
<td>0.19</td>
<td>1</td>
</tr>
<tr>
<td>LDCA</td>
<td>1.0E-03</td>
<td>459.74</td>
<td>391.20</td>
<td>0.03</td>
<td>0</td>
</tr>
<tr>
<td>SDRF</td>
<td>1.0E-02</td>
<td>229.82</td>
<td>195.31</td>
<td>0.19</td>
<td>1</td>
</tr>
</tbody>
</table>

- **Initiating Event - USwC (Unplanned Scrams with Complications)**
  - "Plant level" initiating event P1 in USA
    - Complements "Unplanned Scrams" P1
    - Monitors potentially risk significant scrams than normal scram

<table>
<thead>
<tr>
<th>Category</th>
<th>USwC Categorization</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Trip</td>
<td>Did two or more control rods trip to fully insert?</td>
</tr>
<tr>
<td>Reactor Trip</td>
<td>Did the turbine fail to trip?</td>
</tr>
<tr>
<td>Power supply to Emergency Busses</td>
<td>Was power lost to any BPU?</td>
</tr>
<tr>
<td>Need to Activate Inhibit Sources</td>
<td>Was a Safety Injection Signal received?</td>
</tr>
<tr>
<td>Availability of Makeup Water</td>
<td>Was makeup water or other means of heat removal and pressure suppression unavailable?</td>
</tr>
<tr>
<td>Utilization of Breach Recovery ESPs</td>
<td>Was scram response procedure unable to be completed without re-entering another ESP?</td>
</tr>
</tbody>
</table>

- USwC feasibility for Korean NPPs (performance threshold):
  - Analyzed the scrams of Korean NPPs during the recent 5 years (2002-10, 1-2007, 3-30) to determine the USwC threshold.
  - It will be able to use the USNRC threshold (>1 per year) if USwC is selected as initiating P1 for Korean NPPs, (5% NPPs exceeded threshold).
II. New Indicator Feasibility Study

- Mitigating System – MSPI (Mitigating System Performance Index)
  - "Plant level" mitigating system PI in USA
    - Replacement of SSU (Safety System Unavailability) indicator in NRC
    - SSU is still used in KINS
  - MSPI Definition ⇒ MSPI = URI + UAI (Δ CDF Linear Approximation)
    (URI: Unreliability Index, UAI: Unavailability Index)
    - Reflects plant-specific design and operation
    - Allows trade-offs between Unreliability and Unavailability to optimize system performance
    - No penalty for on-line preventive maintenance hours up to pre-planned baseline
    - Performance Threshold: Δ CDF (RG 1.174)
      - Green/White 1.0E-6, White/Yellow 1.0E-5, Yellow/Red 1.0E-4

II. New Indicator Feasibility Study

- Mitigating System – MSPI (Mitigating System Performance Index)
  - MSPI feasibility for Korean NPPs
    - A main precondition for implementation of MSPI
      - Level 1 PSA for [ ] events for all domestic NPPs – Available
      - PSA DB system for all domestic NPPs – Will be Available (2010)
    - Can be calculated using Excel spreadsheet
III. Threshold Re-Evaluation

- Basic Philosophy for Setting Performance Thresholds (SECY 99-007)
  - Must be clearly defined
  - Risk-Informed to the extent practical
  - Accommodate defense in depth and indications based on existing regulatory requirements and safety analyses
  - Consistent with other NRC risk applications (e.g., Regulatory Guide 1.174) and existing regulatory requirements and safety analyses
  - Thresholds should provide sufficient differentiation to allow meaningful differentiation in performance and limit false positives
  - Sufficient margin should exist between nominal performance bands to allow for licensee initiatives to correct performance problems before reaching escalated regulatory involvement thresholds
  - Where appropriate plant-specific design differences should be accommodated
  - …

III. Threshold Re-Evaluation

- NRC Conceptual Framework for Setting Performance Thresholds

  - Performance Threshold (SECY 99-007)
    - "GREEN" (Acceptable)
      - Correctness objectives fully met
      - Nominal Risk/Nominal Demand from
      - Specified Performance
      - $\text{CDF} < 10^{-6}$
    - "YELLOW" (Acceptable)
      - Correctness objectives fully met with minimal reduction in safety margin
      - Outside bounds of nominal performance
      - $\text{CDF} < 10^{-6}$
    - "RED" (Unacceptable)
      - Plant performance significantly above design basis
      - Loss of confidence in safety of plant as demonstrated assurance of public health and safety with certified operation
      - Unacceptable margin to safety
III. Threshold Re-Evaluation

- Korean Safety Subsidiary Goal and Acceptance Guideline for CDF
  - Safety Subsidiary Goal: CDF < $1 \times 10^{-4}$/yr
    - Same as NRC
  - Acceptance Guideline for CDF (KINS/GT-N24)
    - Same as NRC, except for very small baseline CDF (below $1.0 \times 10^{-7}$/yr) NPP
    - All operating reactors in Korea have CDF higher than $1.0 \times 10^{-7}$/yr

- NRC performance threshold values can be applied in Korea. However, color coding scheme need to be modified.

III. Threshold Re-Evaluation

- Color coding scheme of NRC and KINS PI
  - KINS used similar PI threshold values with NRC
  - However, KINS color coding scheme which represents degree of safety level was different.

(NRC Performance Color Coding Scheme)
- Green: Acceptable (Objectives fully met)
- White: Acceptable (Minimal reduction in safety margin)
- Yellow: Acceptable (Significant reduction in safety margin)
- Red: Unacceptable

(KINS Performance Color Coding Scheme)
- Green: Excellent
- Cyan: Good
- Yellow: Normal
- Orange: Warning
III. Threshold Re-Evaluation

- **Initiating Event - Threshold of Unplanned Reactor Scram**
  - **Performance Indicator**: Green, Cyan, Yellow, Orange
  - **Unplanned Reactor Scram**: <0.26 (3 yr), >0.26 (3 yr), >1.2 (8 yr), >8 (20 yr)

- **Green/Cyan threshold (3/yr)**
  - Unplanned Scrams (1996-2006) ranged from 0.36/yr to 2.25/yr

- **Cyan/Yellow threshold (6/yr)**, Yellow/Orange threshold (20/yr)

<table>
<thead>
<tr>
<th>PSA Model</th>
<th>Cyan/Yellow (MCDF=1E-3/yr)</th>
<th>Yellow/Orange (MCDF=1E-6/yr)</th>
<th>Event Frequency in PSA</th>
</tr>
</thead>
<tbody>
<tr>
<td>GE</td>
<td>19,18</td>
<td>107,94</td>
<td>1,78</td>
</tr>
<tr>
<td>Westinghouse</td>
<td>6.75</td>
<td>52,54</td>
<td>1.27</td>
</tr>
<tr>
<td>Framatome</td>
<td>7.36</td>
<td>61,43</td>
<td>3.37</td>
</tr>
<tr>
<td>ANL</td>
<td>25,23</td>
<td>297,76</td>
<td>1.03</td>
</tr>
</tbody>
</table>

- Threshold values can remain unchanged. However, color coding need to be modified.
  - ΔCDF 1.0E-4 must be interpreted as RED (unacceptable), not ORANGE (warning).

---

- **Initiating Event - Threshold of Unplanned Power Reduction**
  - **Performance Indicator**: Green, Cyan, Yellow, Orange
  - **Unplanned Reactor Reduction**: <1.2 (6 yr), >1.2 (8 yr), >2.3 (10 yr), >8 (20 yr)

- No threshold for White/Yellow, Yellow/Red because the indicators could not be directly tied to risk data.
III. Threshold Re-Evaluation

- **Barrier Integrity** - Threshold of **Fuel Reliability**, Reactor Coolant Leakage, Containment Leakage, Emergency Preparedness

<table>
<thead>
<tr>
<th>Performance Indicator</th>
<th>Green</th>
<th>Cyan</th>
<th>Yellow</th>
<th>Orange</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Reliability</td>
<td>&lt;50%</td>
<td>50%</td>
<td>75%</td>
<td>100%</td>
</tr>
<tr>
<td>Reactor Coolant Leakage</td>
<td>&lt;50%</td>
<td>50%</td>
<td>75%</td>
<td>100%</td>
</tr>
<tr>
<td>Containment Reliability</td>
<td>&lt;50%</td>
<td>50%</td>
<td>75%</td>
<td>100%</td>
</tr>
<tr>
<td>Emergency Preparedness</td>
<td>&lt;50%</td>
<td>50%</td>
<td>75%</td>
<td>100%</td>
</tr>
</tbody>
</table>

(Proposed PI)

<table>
<thead>
<tr>
<th>Performance Indicator</th>
<th>Green</th>
<th>White</th>
<th>Yellow</th>
<th>Red</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel Reliability</td>
<td>&lt;50%</td>
<td>50%</td>
<td>75%</td>
<td>N/A</td>
</tr>
<tr>
<td>Reactor Coolant Leakage</td>
<td>&lt;50%</td>
<td>50%</td>
<td>75%</td>
<td>N/A</td>
</tr>
<tr>
<td>Containment Reliability</td>
<td>&lt;50%</td>
<td>50%</td>
<td>75%</td>
<td>N/A</td>
</tr>
<tr>
<td>Emergency Preparedness</td>
<td>&lt;50%</td>
<td>50%</td>
<td>75%</td>
<td>N/A</td>
</tr>
</tbody>
</table>

- Green = Expected performance
- White = Within Technical Specification limits
- Yellow = Technical Specification limits exceeded

IV. Summary and Future Work

- "Reactor Safety Area" PIs (Performance Indicators) to be used in Korean ISPA Program have been proposed in this study (draft version)

<table>
<thead>
<tr>
<th>Area</th>
<th>Category</th>
<th>Specific Performance Indicator</th>
<th>Grade &amp; Color Coding (energy)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Safety</td>
<td>Multiple Barrier</td>
<td>Unpiloted Reactor Steam</td>
<td>Green</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Unpiloted Reactor Radiation</td>
<td>&lt;20</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Uranium Cycles</td>
<td>&lt;20</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Reactor Coolant Leakage</td>
<td>&lt;50%</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Containment Reliability</td>
<td>&lt;50%</td>
</tr>
<tr>
<td></td>
<td></td>
<td>Emergency Preparedness</td>
<td>&lt;50%</td>
</tr>
</tbody>
</table>

- MPS = Performance Unit
- MPSI = Modified Performance Unit

- Review of "Radiation Safety Area" PIs needed in the future
- Consultation with KINS Operational Safety Analysis Department (which is practical PI management team) is needed to finalize proposed PIs.
This is a blank page.