Colloquium at Dept. of Nuclear & Quantum Engineering, KAIST, Daejeon, Nov. 29, 2016

Beyond Design Basis Accidents in Design Process

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1. Basic Approach to Nuclear Safety

JNES The Accident at TEPCO's Fukushima Dai-ichi NPS

ENSREG EU "Stress tests" specifications

Report of Japanese Government
to the IAEA Ministerial Conference on Nuclear Safety
- The Accident at TEPCO's Fukushima Nuclear Power Stations -

Extreme Natural hazard

European Council "Stress Tests" for UK Nuclear Power Plants

Beyond Design Basis Accident

Strengthen emergency preparedness and response

Multiple Failure Events

Multi-unit Event

Fukushima Daiichi Status Report IAEA

OECD/NEA Working Group Operating Experience: Report on Fukushima NPP Precursor Events, 2014

Plant Robustness

Accident with core melt

Cliff-edge Effects

Accident without Core Melt

Hardened Safety Core

OECD/NEA International Workshop on Crisis Communication: facing the challenges, May 2012

Extended Design Basis

Convention on Nuclear Safety
National Report of Japan
for the Second Extraordinary Meeting

Design Robustness

Beyond Design Basis Enhancement

Accident Tolerance

Plant Robustness

IAEA Action Plan on Nuclear Safety

Beyond Design Basis External Events

WENRA/ENSREG, "Action Plan: Follow-up of the peer review of the stress tests", July 2012

OECD 2013 Report, 'Nuclear Safety Response and Lessons Learnt from the Fukushima accident'

Strengthening Defence in Depth

Design Basis Extension

OECD/NEA Forum on the Fukushima Accident: Insights and Approaches, June 2011

Design Extension Conditions

1. Basic Approach to Nuclear Safety

Safety Concern after Fukushima Accident

- Beyond Design Basis External Events
- Multiple Failure Events
- Design Extension Conditions

- Accident without Core Melt
- Accident with Core Melt
- Design Basis Extension
- Design Basis Extension Events

Extreme Natural External Hazard

→ **Multiple failures including loss of electric power**

→
- **Loss of core cooling**
- **Loss of containment function**
- **Radioactive release causing long term offsite contamination**

Multiple System Failures : Beyond Current Design Basis

1. Basic Approach to Nuclear Safety

Main Pillars for Nuclear Safety

Fundamental Safety Objectives

To protect people and the environment from harmful effect of ionizing radiation

prevention and mitigation of nuclear & radiation accidents

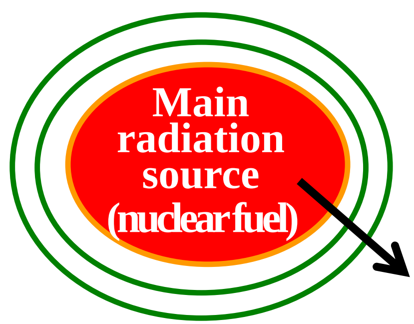
Defence in Depth

A number of consecutive and independent levels of protection

Fundamental Safety Function

- Control reactivity
- Removal of heat from core
- Confinement of radioactive material

Plant Robustness with Sufficient Safety Margin



Multipla Physical Barriers

1st : Fuel Pellet
2nd : Fuel Cladding
3rd : RCS Pressure Boundary
4th : Containment
Exclusion Area *(Virtual Barrier)*

1. Basic Approach to Nuclear Safety

▣ General Concept of DiD

Primary means of prevention of accidents and mitigation of the consequences of accidents, if they occurs.

Implemented through a number of consecutive & independent levels of protection
- to maintain multiple physical barriers
- to preserve three fundamental safety functions

✓ *If one level of protection or barrier were to fail, the subsequent level or barrier would be available.*

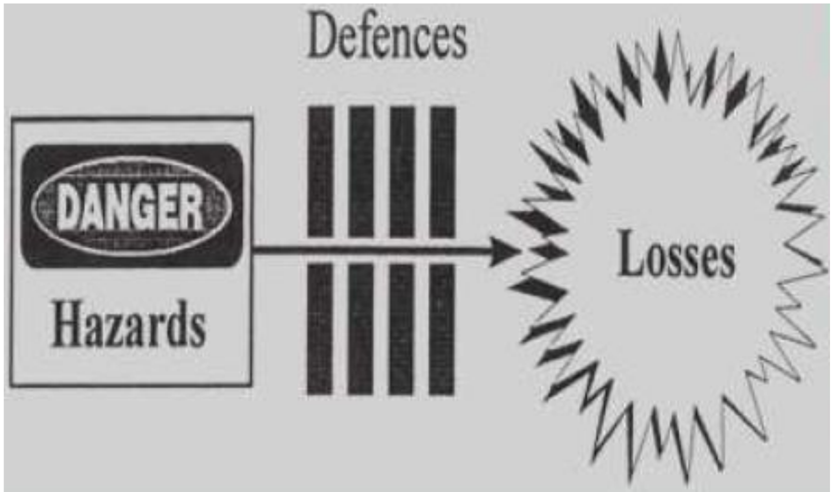
✓ *The independent effectiveness of the different levels of defence is a necessary element of DiD to avoid the failure of one level reducing the effectiveness of other levels.*

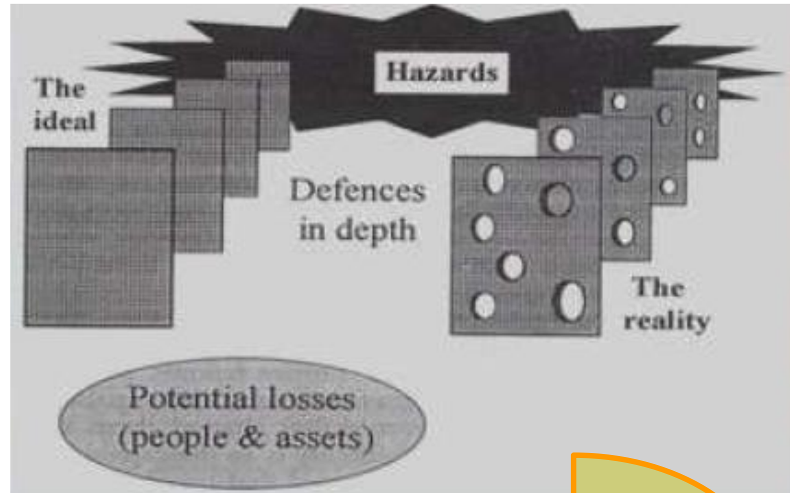
☞ A number of consecutive & independent levels of protection is not a basis for continued operation in the absence of one level of defence.

- All DiD levels shall be kept available at all times and any relaxations shall be justified for specific modes of operation.

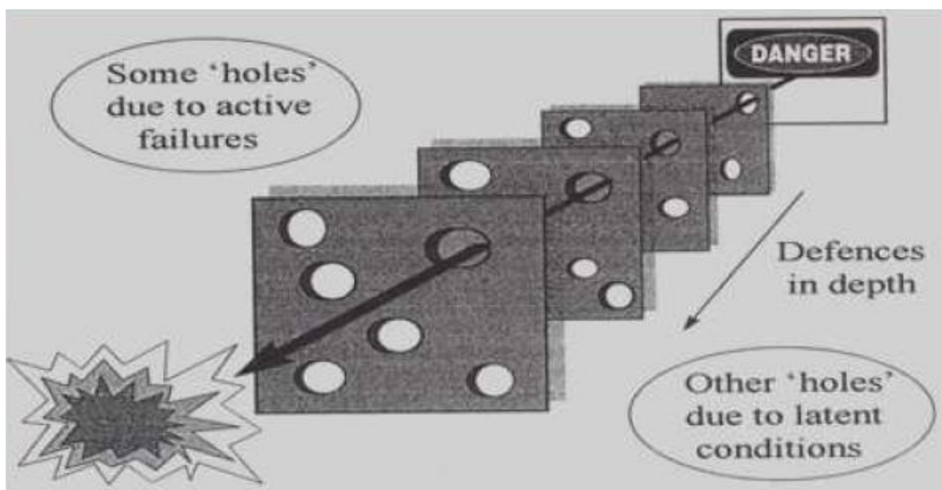
1. Basic Approach to Nuclear Safety

Defence-in-Depth against Accident





Defences are never perfect !



Source : 제8차 PSA 워크숍 (Status of the Linkage between DSA and PSA in the United States by Inn Seock Kim), 2013.3.31~4.1, 곤지암리조트

1. Basic Approach to Nuclear Safety

Objective of Each Level of Protection & Essential Means

Level (events)	Objective	Essential Means
Level 1 (NO)	Prevention of abnormal operation and failures	Conservative design, and high quality in construction & operation
Level 2 (AOO)	Control of abnormal operation & detection of failures	Control, limiting & protection systems and other surveillance features
Level 3 (DBA)	Control of accident within the design basis	Engineered safety features, Accident procedures
Level 4 (SA)	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management
Level 5 (post-SA situation)	Mitigation of radiological consequences of significant releases of radioactive material	Off-site emergency response

NO: normal operation, AOO: anticipated operational occurrence,
DBA: design basis accident, SA: severe accident

Source : "BSPs for NPP", 75-INSAG-3 Rev.1 (INSAG-12), 1999

1. Basic Approach to Nuclear Safety

Overview of DiD : Relation with Other Safety Measures

Strategy	Accident prevention			Accident mitigation	
Operational state of the plant	Normal operation	Anticipated operational occurrences	Design basis and complex operating states	Severe accidents beyond the design basis	Post-severe accident situation
Level of defence in depth	Level 1	Level 2	Level 3	Level 4	Level 5
Control	Normal operating activities		Control of accidents in design basis	Accident management	
Procedures	Normal operating procedures		Emergency operating procedures	Ultimate part of emergency operating procedures	
Response	Normal operating systems		Engineered safety features	Special design features	Off-site emergency preparations
Condition of barriers	Area of specified acceptable fuel design limit		Fuel failure; Severe fuel damage	Fuel melt; Uncontrolled fuel melt	Loss of confinement
Colour code	NORMAL		POSTULATED ACCIDENTS		EMERGENCY

Source : "BSPs for NPP", 75-INSAG-3 Rev.1 (INSAG-12), 1999

2. IAEA Approach to beyond Design Basis Accidents

2. IAEA Approach to beyond Design Basis Accidents

Plant States

<table>
<tr><th colspan="2" rowspan="2">Operational states</th><th colspan="4">Accident conditions</th></tr>
<tr><th colspan="2"></th><th colspan="2">Beyond design basis accident</th></tr>
<tr><td>Normal operation</td><td>Anticipated operational occurrences</td><td>(a)</td><td>Design basis accident</td><td>(b)</td><td>Severe accident</td></tr>
</table>

(a) : accident conditions which are not explicitly considered design basis accident but are encompassed by them.

(b) : beyond design basis accident without significant core degradation

IAEA NS-R-1 (2000), Safety of Nuclear Power Plants : Design

IAEA SSR-2/1Rev.1(2016), Safety of Nuclear Power Plants : Design

<table>
<tr><th colspan="2">Operational states</th><th colspan="3">Accident conditions</th></tr>
<tr><td rowspan="2">Normal operation</td><td rowspan="2">Anticipated operational occurrences</td><td rowspan="2">Design basis accident</td><td colspan="2">Design Extension Conditions</td></tr>
<tr><td>Without significant fuel degradation</td><td>With core melt</td></tr>
</table>

Beyond design basis accident → Design Extension Conditions

2. IAEA Approach to beyond Design Basis Accidents

Expected Frequencies of Occurrence of Different Plant States

Plant States	Frequency of Occurrence
Normal operation	-
Anticipated operational occurrences	> 10^{-2} events per year
Design basis accidents	$10^{-2} \sim 10^{-4}$ events per year
Design extension conditions without significant fuel degradation	$10^{-4} \sim 10^{-6}$ events per year
Design extension conditions with core melt	> 10^{-6} events per year

Source : IAEA-TECDOC-1791, May 2016

Definition

Anticipated operational occurrences (AOO) : An operational process deviating from normal operation, expected to occur at least once during the plant lifetime.

Design basis accident (DBA) : Accident conditions against which a NPP is designed according to established design criteria, and for which damage to fuel and the release of radioactive material are kept within authorized limit

2. IAEA Approach to beyond Design Basis Accidents

Design Extension Conditions (DECs)

- Postulated accident conditions that are not considered for DBAs,
- but considered in the design process for the facility in accordance with best estimate methodology, and
- for which releases of radioactive material are kept within acceptable limits.

Multiple failures without core damage
\+ Multiple failures with core damage (Severe Accident)

※ *Terminology first introduced in European Utility Requirement(2012) to introduce some accident sequences selected on deterministic and probabilistic basis that go beyond design basis conditions, including complex sequences and severe accidents.*

- *Purpose was to improve the plant safety by extending the design basis.*

※ *A similar concept was adopted by WENRA, term 'DEC' was initially not explicitly used.*

- *To consider AOOs and single initiating event DBAs connected with complete loss of a safety function (designed to respond to that event) and to postulate core melt sequences that challenge containment.*

2. IAEA Approach to beyond Design Basis Accidents

Multiple System Failure Events in Current Regulation

Event	Countermeasure
Station Blackout (SBO)	Alternative alternating current power
Anticipated Transient Without Cram (ATWS)	Diverse protection system
Loss of Total Feedwater (LOFT)	Safety depressurization system

2. IAEA Approach to beyond Design Basis Accidents

▣ Design Consideration of DECs

Technical Objective

To further improve the NPP safety by enhancing the plant's capabilities to withstand accident conditions more severe than DBAs, without unacceptable radiological consequences,

- A set of DECs to be derived on the basis of engineering judgement, deterministic assessments and probabilistic assessments.
- DECs to be used to identify the additional accident scenarios to be addressed in the design,
- DECs to be used to plan practicable provisions for the prevention of such accidents or mitigation of their consequences if they do occur.
- Require additional safety features for DECs or extension of the capability of safety systems.
- Each unit of a multiple unit NPP with its own safety systems and its own safety features for DECs.
    - Potential for specific hazards to give rise to impacts on several or even all units on the site simultaneously.

2. IAEA Approach to beyond Design Basis Accidents

- Protective actions that are limited in terms of lengths of time and areas of application shall be sufficient for the protection of the public, and sufficient time shall be available to take such measures.
  - Limited protective action : no permanent relocation, no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restrictions in food consumption
- The possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is ‘practically eliminated’.

※ Definitions

- *An early radioactive release : a radioactive release for which off-site protective actions would be necessary but would be unlikely to be fully effective in due time.*
- *A large radioactive release : a radioactive release for which off-site protective actions that are limited in terms of lengths of time and areas of application would be insufficient for the protection of people and of the environment.*
- *The possibility of certain conditions arising may be considered to have been ‘practically eliminated’ if it would be physically impossible for the conditions to arise or if these conditions could be considered with a high level of confidence to be extremely unlikely to arise.*

2. IAEA Approach to beyond Design Basis Accidents

▣ Elaboration on Defence in Depth for DECs

Level		Objective	Essential Design Means	Essential Operational Means
Level 1		Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation, normal operation systems including monitoring & control systems	Operational rules & normal operating procedures
Level 2		Control of abnormal operation & detection of failures	Control, limitation & protection systems and other surveillance features	Abnormal operating procedures/emergency operating procedures
Level 3	3a	Control of DBAs	Engineered safety features (safety systems)	Emergency operating procedures
	3b	Control of DECs to prevent core melt	Safety features for DECs without core melt	Emergency operating procedures
Level 4		Control of DECs to mitigate the consequences of severe accidents	Safety features for DECs with core melt, Technical support center	Complementary emergency operating procedures/ severe accident management guidelines
Level 5		Mitigation of radiological consequences of significant releases of radioactive material	On-site and off-site emergency response facilities	On-site and off-site emergency plans

Source : IAEA-TECDOC-1791, 1999

2. IAEA Approach to beyond Design Basis Accidents

▣ Main Elements of Design Basis of SSCs for Different Plant States

Design Basis (Extended) | Beyond Design Basis

<table>
<tr><th colspan="2">Operational States</th><th colspan="3">Accident Conditions</th><th rowspan="6">No cliff edge effect</th><th>Conditions practically eliminated</th></tr>
<tr><td>NO</td><td>AOO</td><td>DBAs</td><td colspan="2">Design Extension Conditions</td><td rowspan="5"></td></tr>
<tr><td></td><td></td><td></td><td>Without significant fuel degradation</td><td>With core melt (Severe Accidents)</td></tr>
<tr><td rowspan="2">DiD Level 1</td><td rowspan="2">Level 2</td><td colspan="2">Level 3</td><td rowspan="2">Level 4</td></tr>
<tr><td>3a</td><td>3b</td></tr>
<tr><td colspan="5">Loads & Conditions generated by external & internal hazards (for each plant states)</td></tr>
<tr><td colspan="5">Criteria for functionality, capability, margins, layout and reliability (for each plant states)</td><td colspan="2" rowspan="3">No plant equipment designed for these conditions</td></tr>
<tr><td rowspan="2">Design basis of equipment for operational state</td><td colspan="2" rowspan="2">Design basis of safety systems including SSCs necessary to control DBAs & some AOOs for operational state</td><td colspan="2">Design basis of safety features for DECs including SSCs necessary to control DECs</td></tr>
<tr><td>Features to prevent core melt</td><td>Features to mitigate core melt (containment systems)</td></tr>
</table>

2. IAEA Approach to beyond Design Basis Accidents

▣ Cliff Edge Effects (CEEs)

- CEEs imply high consequences following a small deviation in a plant parameter.
  - Worst case : Large release as the consequence, or
  - Failure of a physical barrier.
- CEEs more likely to occur when the parameter has the potential to affect the functionality of many SSCs at once.
  - Failure of containment due to hydrogen detonation
  - Earthquake causing a LOCA
  - External hazards (e.g. flooding exceeding the design value)

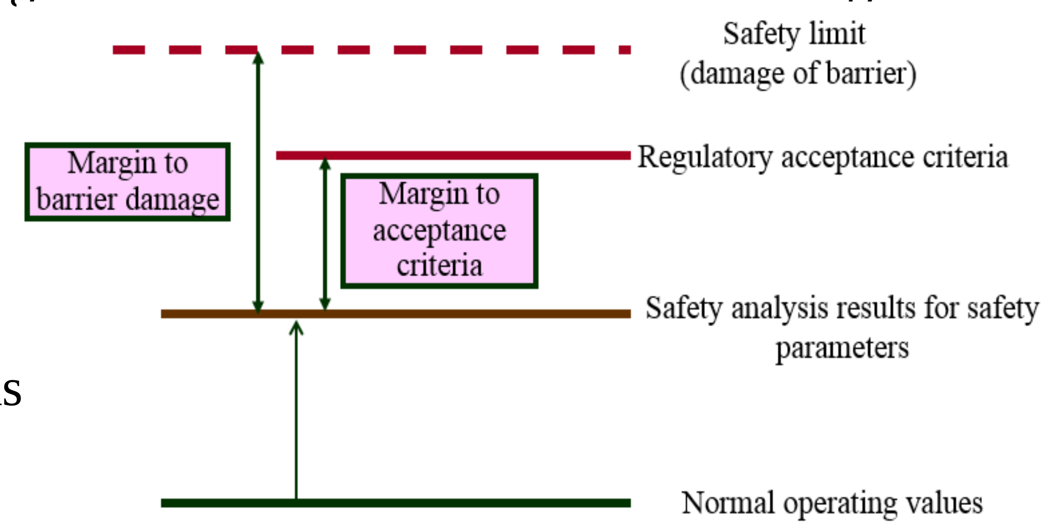
*※ **Definitions** : An instance of severely abnormal plant behaviour caused by an abrupt transition from one plant status to another following a small deviation in a plant parameter, and thus a sudden large variation in plant conditions in response to a small variation in an input.*

- *WENRA definition : A cliff edge effect happens where a small change in a parameter leads to a disproportionate increase in consequences.*

2. IAEA Approach to beyond Design Basis Accidents

Cliff Edge Effects & Safety Margins

- When design basis of an SSC is exceeded, failure is prevented by available margin.
  - Margins are particularly important if exceeding them leads to a CEE
  - The goal of design to ensure adequate margin to avoid CEEs
- Adopting margins in the design of a NPP is a common practice
  - to improve the robustness of the design
  - to provide an effective mean to deal with uncertainties.
- Larger design margins for items ultimately necessary to prevent large or early radioactive releases and specifically against external hazards to avoid cliff edge effects.
  - For DECS without core melt, the uncertainties in deterministic analysis similar to DBAs
  - For DECS with core melt, the uncertainties are larger than DBAs



3. Safety Demonstration of DECs

3. Safety Demonstration of DECs

▣ Identification & Grouping of DECs

Types of DECs without Significant Fuel Degradation

Very unlikely events that could lead to situations beyond the capability of safety systems for DBAs	▪ Demonstration based on best estimate analyses that the safety systems are capable of and qualified for mitigating the event under consideration. ▪ Inclusion of specific safety features for DEC is necessary.
Multiple failures (e.g. CCFs in redundant trains) that prevent the safety systems from performing their intended function to control the PIE	▪ An example is LOCA without actuation of a safety injection system. ▪ The failures of supporting systems implicitly included
Multiple failures that cause the loss of a safety system while this system is used to fulfill the fundamental safety functions in NO	▪ Designs that use, for example, the same system for the heat removal in accident conditions and during shutdown

3. Safety Demonstration of DECs

List of DECs without Significant Fuel Degradation

Based on DSA not design specific

- ATWS
- SBO
- Loss of core cooling in the residual heat removal mode
- Extended loss of cooling of fuel pool and inventory
- Loss of normal access to the ultimate heat sink

Based on PSA design or plant specific (Technology Dependent)

- Total loss of feed water
- LOCA plus loss of one ECCS (high or lower)
- Loss of the component cooling water system or the essential service water system (ESWS)
- Uncontrolled boron dilution
- Multiple steam generator tube ruptures
- Steam generator tube ruptures induced by main steam line break
- Uncontrolled level drop during mid-loop operation or during refuelling
- AOO or DBA combined with the failure of the protection system (I&C)

3. Safety Demonstration of DECs

Types of DECs with Core Melt

Select a representative group of severe accident conditions to be used for defining the design basis of the mitigatory safety features for these conditions

- Have sufficient knowledge about the phenomena associated with different severe accidents
- Identify accident sequences that lead to core melt and the plant conditions at the onset of the core melt

Main objective is to maintain the integrity of the containment

- For severe accident scenarios, identify systems protecting containment integrity against i) dynamic loads, ii) high pressure and temperature, iii) molten fuel impact
- Identify safety features to prevent that severe accident phenomena, such as hydrogen detonation, cause the loss of containment integrity
- In the long term, achieve cooling and stabilization of the molten fuel, the removal of heat from the containment and the relief of the containment pressure

3. Safety Demonstration of DECs

▣ Deterministic Analysis of DECs : Additional Requirements

- Best-estimate method with realistic assumptions
- Single failure criterion not required for the safety features for DEC
- Safety demonstration by the automatic actuation of safety systems and the use of safety features in combination with expected actions by the operator.

Acceptance Criteria for DECs

Radiological Consequences

- Discharges or releases of radioactive material,
- Whole body effective doses, equivalent doses for selected organs or tissues, and
- Radioactivity or contamination levels of ground, water, crops and food items
- Acceptability of radiological consequences related to off-site emergency response actions

Degree of Integrity of Physical Barriers

Surrogate variables determining integrity of barriers, such as pressures, temperatures, stresses, strains, etc.

3. Safety Demonstration of DECs

<u>Acceptability of radiological consequences related to off-site emergency response actions</u>

- Criteria with emergency action levels (EALs)
- Acceptance criteria for design need to be significantly lower than the EALs adopted for emergency measures
- The target would be to minimize the need for emergency measures

❖ International Basic Safety Standards

- Generic criterion for sheltering and evacuation : 100 mSv of projected effective dose in the first 7 days. (lower based on reference level 20~100 mSv)
- Generic criterion for initiating temporary relocation : 100 mSv of projected effective dose in the first year
- Generic criterion for iodine thyroid blocking : 50 mSv of projected equivalent dose to the thyroid only due to exposure to radioiodine.

3. Safety Demonstration of DECs

Examples of Acceptance Criteria for Different Plant States

Level		Objective (Plant State)	Criteria for maintaining integrity of barriers	Criteria for limitation of radiological consequences
1		Prevention of abnormal operation and failures (Normal operation)	No failure of any of the physical barriers except minor operational leakages	Negligible radiological impact beyond immediate vicinity of the plant. Acceptable effective dose limits are bounded by the general radiation protection limit for the public (1 mSv /year commensurate with typical doses due to natural background), typically in the order of 0.1 mSv/year.
2		Control of abnormal operation & detection of failures (AOO)		Negligible radiological impact beyond immediate vicinity of the plant. Acceptable effective dose limits are similar as for normal operation, limiting the impact per event and for the period of 1 year following the event (0.1 mSv/y)
3	3a	Control of DBAs (DBA)	No consequential damage of the reactor coolant system, maintaining containment integrity, limited damage of the fuel	No or only minor radiological impact beyond immediate vicinity of the plant, **without the need for any off-site emergency actions**. Acceptable effective dose limits are typically in the order of few mSv.
	3b	Control of DECs to prevent core melt (DEC without significant fuel degradation)		The **same or similar radiological acceptance criteria** as for the most unlikely design basis accidents
4		Control of DECs to mitigate the consequences of severe accidents (DEC with core melt)	Maintaining containment integrity	**Only emergency countermeasures that are of limited scope in terms of area and time** are necessary
5		Mitigation of radiological consequences of significant releases of radioactive material	Containment integrity severely impacted, or containment disabled or bypassed	Off-site radiological impact necessitating emergency countermeasures

Source : IAEA-TECDOC-1791, 2016

3. Safety Demonstration of DECs

▣ Concept & Design Provisions of Practical Elimination

Possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is 'practically eliminated'.

Definition

- *if it would be physically impossible for the conditions to arise, or*
- *if these conditions could be considered with a high level of confidence to be extremely unlikely to arise.*

- *Deterministic nature on the consideration of the physical impossibility*
  - Inherent safety characteristics of the system or reactor type to demonstrate that the event cannot occur by the laws of nature
- *Probabilistic nature and the use of probabilistic methods*
  - *On the very low (extremely unlikely) probability of a condition*
  - *Degree of confidence of the probability estimate is very high*

Identify conditions to be practically eliminated → Specify the design provisions for it → Assess the adequacy of provisions on the basis of DSA, PSA and engineering judgement

3. Safety Demonstration of DECs

▣ Categorization of Hypothetical Conditions to be Practically Eliminated

Events leading to prompt reactor core damage and early containment failure

1.1 Failure of large components in the reactor coolant system (Rx. Vessel, S/G shell, PZR)
1.2 Uncontrolled reactivity accidents

Very energetic phenomena in severe accident conditions for which technical solutions for maintaining containment integrity must be provided

Severe accident phenomena which could lead to early containment failure:

- 2.1 Core meltdown at high pressure (Direct Containment heating)
- 2.2 Steam explosion (Fuel coolant interaction)
- 2.3 Hydrogen detonation (explosion)

Severe accident phenomena which could lead to late containment failure:

- 2.4 Containment failure due to fast over-pressurization (Loss of containment heat removal)
- 2.5 Containment boundary melt-through (Molten core concrete interaction)

Non confined severe fuel damage

3.1 Severe accident with containment bypass

- Severe accident conditions with an open containment (in shutdown states)
- Conditions relating to SG tube rupture or interfacing system LOCA

3.2 Significant fuel failure in a storage pool

3. Safety Demonstration of DECs

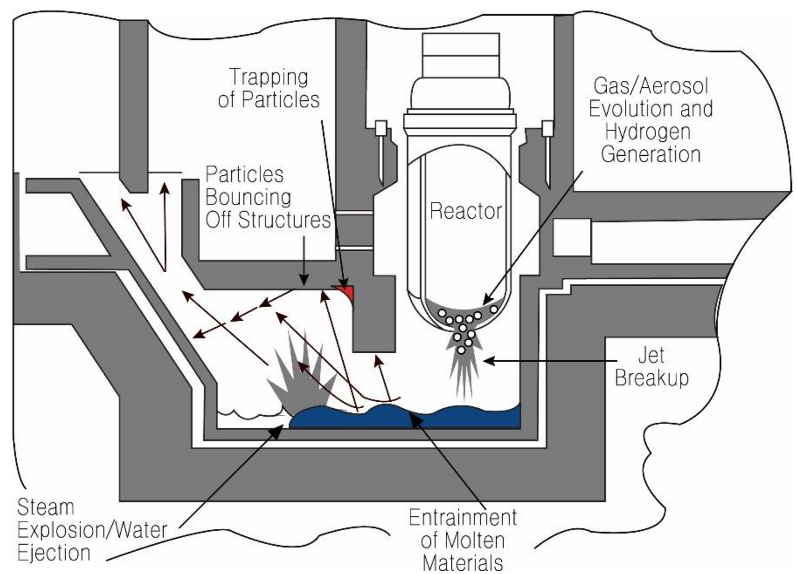
※ Core meltdown at high pressure (Direct Containment heating)

Sequence : Violent discharge of molten corium material into containment resulting in direct containment heating by chemical reaction

- ❖ High pressure core melt situations to be eliminated by design provisions to depressurize the RCS when a meltdown is found unavoidable.
  - ▪ Convert the high pressure core melt to a low pressure core melt sequence with a high reliability before a discharge of molten core.
- ☞ Reliable dedicated depressurization systems (Safety feature for DECs)

Safety Demonstration

- ❖ Reliable system to depressurize RCS, and knowledge & means to ensure correct timing of depressurization.
- ❖ Deterministic analysis to demonstrate the effectiveness of the depressurization system in preventing direct containment heating.
- ❖ Traditional PSA techniques to demonstrate a high reliability of the depressurization systems including the operator initiation.



4. Country Specific Approach

4. Country Specific Approach - Japan

New Regulatory Requirements Based on Lessons Learned

(effective on July 2018)

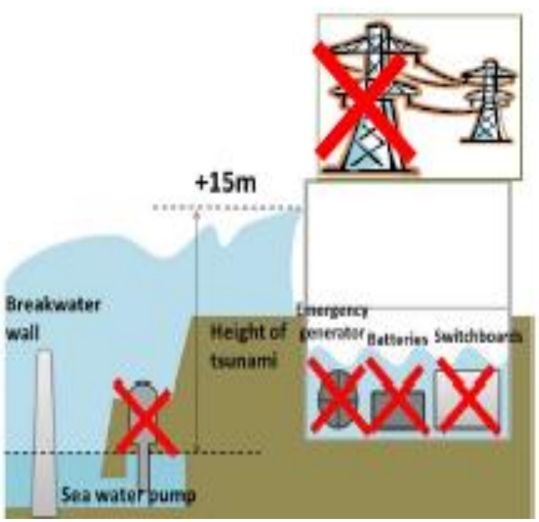
Large scale common cause failures due to extreme natural hazards led to long lasting SBO/LUHS, resulting in severe accidents.

Enhanced measures against natural hazards
- Not only earthquakes and tsunamis but also volcanic activities, tornadoes, forest fires, etc.

Enhanced reliability of safety functions such as power supply
- Use of mobile equipment, strengthened “diversity”

Mandatory measures against severe accidents
- Prevention and mitigation of core damage
- Suppression of radioactive materials dispersion

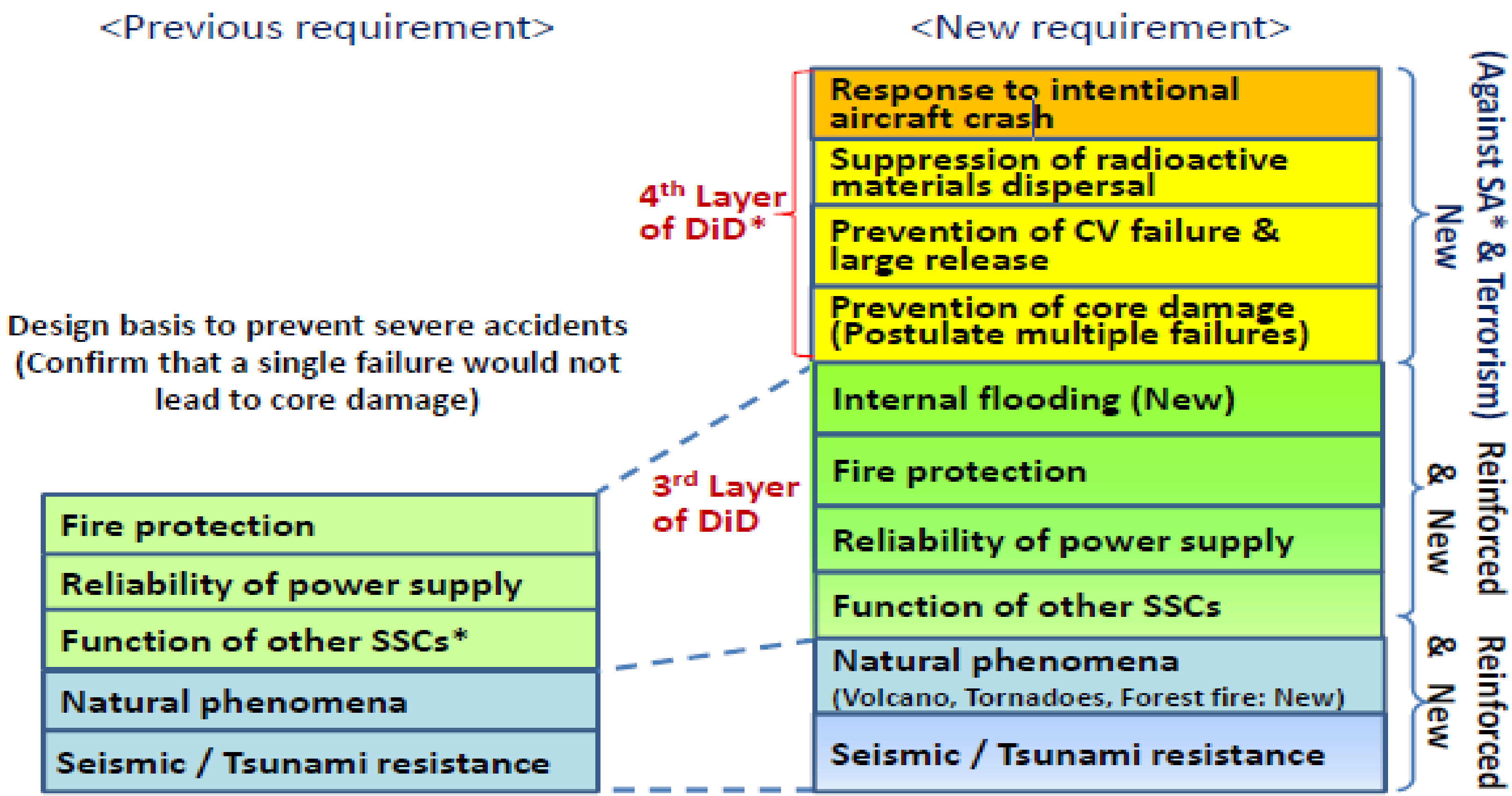
Back-fitted to existing plants



4. Country Specific Approach - Japan

Comparison between Previous and New Regulatory Requirements

Tighten measures to prevent or deal with severe accidents and acts of terrorism



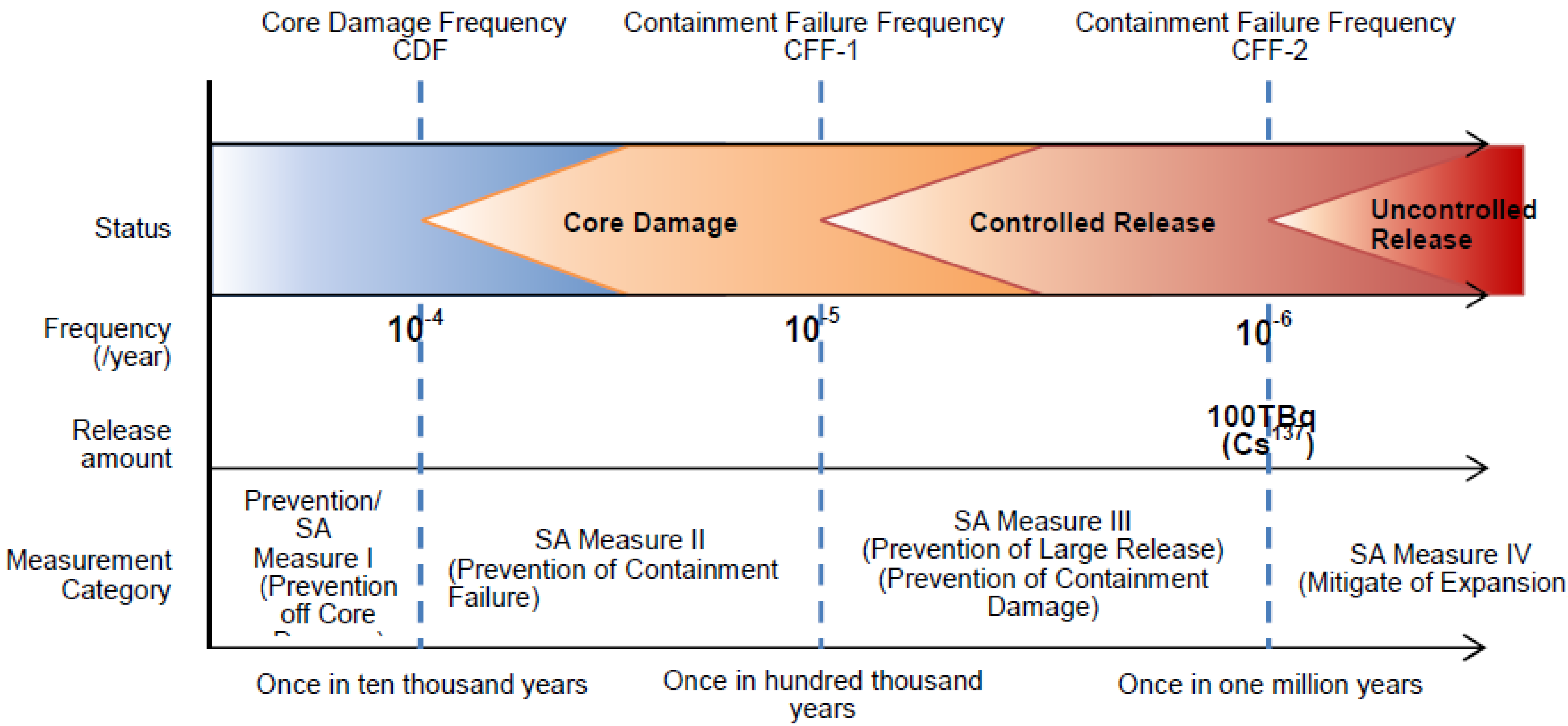
DiD* : Defense in Depth
SSCs* : Structure, Systems and Components
SA* : Severe Accident

Source : NRA "New Regulatory Requirements for Light-Water NPPs"

4. Country Specific Approach - Japan

Safety Goals

- Core damage frequency : 10^{-4}/year, Containment failure frequency : 10^{-5}/year
- Environmental contamination : "The frequency of the release of Cs-137 larger than 100 TBq during nuclear emergency should be less than once in one million years (excluding those due to security events)



4. Country Specific Approach – EU(WENRA)

Statement of Safety Objectives for New NPPs (Nov. 2010)

Category	Objective
O1. Normal operation, abnormal events & prevention of accidents	• reducing the frequencies of abnormal events • reducing the potential for escalation to accident situations
O2. Accidents without core melt (Multiple Failure Events, MFEs)	• reducing the core damage frequency • no or only minor off-site radiological impact
O3. Accidents with core melt (CMAs)	• CMAs leading to early or large releases have to be practically eliminated • Limited protective measures in area and time and sufficient time available
O4. Independence between all levels of defence-in-depth : particular through diversity provisions	
O5. Safety and security interfaces : synergies between safety and security enhancements	
O6. Radiation protection & waste management : individual/collective doses, radioactive discharges, quantity and activity of radioactive waste	
O7. Leadership and management for safety	

4. Country Specific Approach – EU(WENRA)

Refined Structure of the Levels of DiD

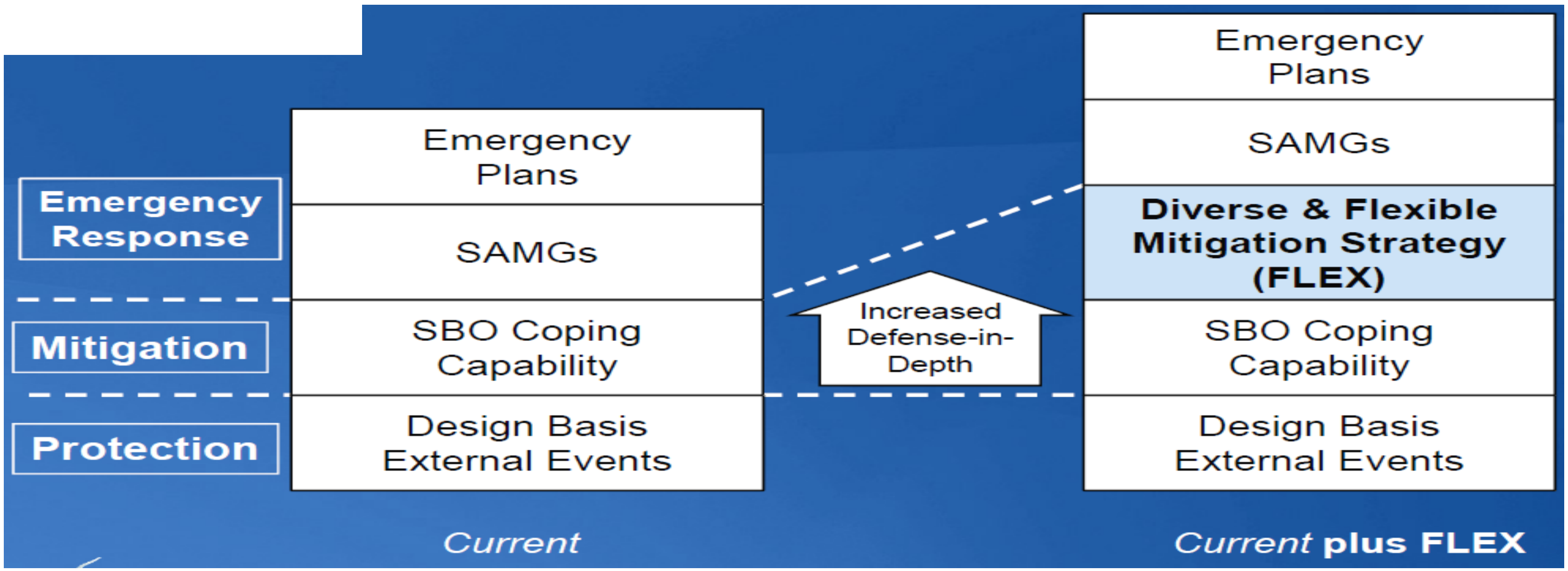
Source : WENRA Safety of new NPP designs (Oct. 2012)

Level	Objective	Essential Means	Radiological Consequences
Level 1 Normal operation	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation, control of main plant parameters inside defined limits	No off-site radiological impact (bounded by regulatory operating limits for discharge)
Level 2 AOO	Control of abnormal operation & failures	Control and limiting systems and other surveillance features	
Level 3 — **3a** Single initiating events	Control of accidents to limit radiological releases and prevent escalation to core melt conditions.	Reactor protection system, safety systems, accident procedures	No off-site radiological impact or only minor radiological impact
Level 3 — **3b** multiple failure events		Additional safety features(3), accident procedures	
Level 4 core melt accidents (short and long term)	Control of accidents with core melt to limit off-site releases	Complementary safety features to mitigate core melt, Management of accidents with core melt (severe accidents)	Off-site radiological impact may imply limited protective measures in area and time
Level 5	Mitigation of radiological consequences of significant releases of radioactive material	Off-site emergency response, Intervention levels	Off site radiological impact necessitating protective measures

4. Country Specific Approach - USA

▣ Mitigation Strategies

- EDMG (Extensive Damage Mitigation Guidelines) : Mitigation Strategies for beyond DBCs such as large explosion or fire : 10CFR50.54(hh)(2)
- FLEX (FLEX Support Guidelines) : Diverse & Flexible mitigation Strategies for bDBEE : NEI 12-06, Draft RG-1301



▣ Proposed Rulemaking : Mitigation of beyond Design Basis Events

- 10CFR50.155 : Integrated Response Capability, Equipment, Training, Drills and Exercises, Change Control

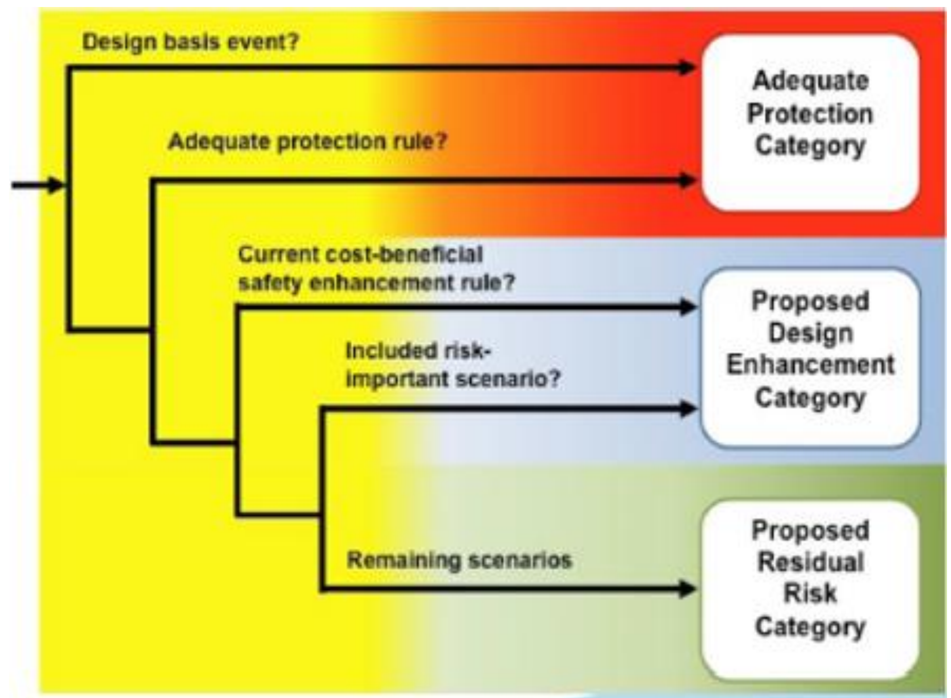
4. Country Specific Approach - USA

▣ Near-Term Recommendations (SECY-11-0093, Priority : SECY-11-0137)

- Improvement Activity 1 of Recommendation 1 : Establish a Design-Basis Extension Category of Events and Associated Regulatory Requirements
- Premature to establish a formal “design-basis-extension” category of events with corresponding regulatory requirements

▣ Risk Management Regulatory Framework (RMRF)

- 10 to 15 Years Future Regulation Policy of USNRC
- Extension of design basis requirements
- Re-Categorizing Events
  - APC : Design basis events or events with adequate protection rule
  - DEC : Design enhancement events or events by risk-important scenarios
  - RRC : Remaining scenarios



4. Country Specific Approach – Korea

▣ Severe Accident Management within Regulatory Framework

- Amendment to Nuclear Safety Act : effective on June 2016
  - Submission of accident management planning (including severe accident) as licensing documents for operating license
  - Submitted as a operator's voluntary initiative.
- Amendment to Enforce Regulation Concerning the Technical Standards of Reactor Facilities : (under way)
- New NSSC Notices : (under way)
  - "Provisions on Detail Criteria for the Scope and Capability Assessment of Accident Management"
  - " Provisions on Documentation of Accident Management Planning"
- Miscellaneous Amendment to NSSC Notices : (under way)

4. Country Specific Approach – Korea

Scope of Accidents Subject to Accident Management

Category	Accidents or Conditions
Design Basis Accidents	-
Multiple Failure Accidents	▪ ATWS, station blackout, multiple s/g tube rupture, loss of total feedwater, interfacing LOCA, loss of shutdown cooling capabilities, loss of ultimate heat sink, loss of safety injection or recirculation injection concurrent with small break LOCA, loss of cooling in spent fuel pool. ▪ Additional accidents with similar frequency and consequence to the above accidents identified based on PSA
Natural and Human Induced Hazards beyond Design Basis	▪ Earthquake, meteorology, hydrology, oceanology ▪ Intentional aircraft crash ▪ Combination of hazards
Accidents with Significant Fuel Degradation (Severe Accidents)	Phenomena or conditions threatening containment integrity ▪ Burning or detonation due to combustible gas, containment overpressurization, interaction of molten corium with concrete, violent discharge of molten corium to containment, Direct containment heating, interaction of molten corium with coolant, containment bypass due S/G tube failure. ▪ Additional phenomena or conditions with similar frequency and consequence to the above identified based on PSA

4. Country Specific Approach – Korea

Assessment of Accident Management Capability

Assessment by Deterministic Approach		
Category	Criteria for Integrity of Barriers	Criteria for Radiological Consequences
Multiple Failure Accidents	▪ No significant fuel damage	Siting Criteria (EAB, LPZ, PCD) • Whole Body Dose : 250 mSv • Thyroid Dose : 3000 mSv
Natural and Human Induced Hazards beyond Design Basis	▪ Recover & maintain the fundamental safety functions	
Accidents with Significant Fuel Degradation	▪ Maintain containment integrity to prevent large radioactive release	

Assessment by Probabilistic Approach (Risk)

- Prompt fatality and cancer fatality risks of individuals in vicinity of NPP sites less than 0.1 % of total risks respectively.
- Sum of the frequency of the accidents with the radioactive release of Cs-137 exceeding 100 TBq less than 10E-6/year

5. Summary

5. Summary

<u>Typical Trend in Safety Demonstration</u>

Postulated Single Failure Events in Design	→	Postulated Multiple Failure Events in Design
Severe Accidents in Probabilistic Approach	→	Severe Accidents in Deterministic Approach
Severe Accident Management in Operator’s Voluntary Initiative	→	Severe Accident Management in Regulatory Framework
Safety & Emergency Measures for a Single Units in a Site	→	Safety & Emergency Measures for Multiple Units in a Site

Area of Interest

- ☞ To what extent is the scope of beyond design basis accident to be addressed in design process?
- ☞ How to define design basis extension category of events?
- ☞ How to identify the additional accident scenarios to be addressed in the design process ?
- ☞ How to establish analysis methodology & acceptance criteria ?
- ☞ How to strengthen the defence in concept ?


Thank you
Question & Comment