A Perspective on the Use of Risk Information

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Risk-Informed Decision Making (1)

- **Decision making must be based on the current state of knowledge of the decision maker (DM)**
  - **The current state of knowledge regarding design, operation, and regulation is key.**
  - **The current state of knowledge is informed by science, engineering, and operating experience, including past incidents.**
- **What we know about plant behavior is not easily available to the DM**
  - **Accident sequences, human performance, risk significance of systems, structures, and components, etc**
  - **Until the Reactor Safety Study, the significance of support systems and human errors had not been appreciated**

Risk-Informed Decision Making (2)

- **PRAs provide this information to the DM**
  - **PRAs do not predict the future**

- **As a consequence, the characterization "risk-informed" would appear to be superfluous**
  - **A fuzzy concept that may be abused**

- **However, it is useful as a communication tool among industry and regulatory staffs.**

The Problem with Low Frequencies (1)

- PRAs for advanced reactors (not yet built) report CDF and LRF estimates in the range of 10^{-6} to 10^{-9} per reactor year
- Return periods of 10^{6} to 10^{9} years
- Age of the earth: 4.6×10^{9} years
- Age of the earth's crust: 2×10^{9} years
- What do these numbers mean?

The Problem with Low Frequencies (2)

- The CDF and LERF estimates do not include digital I&C failures, management failures, safety culture, operating experience
- IEs of 10^{-8} per year. Do we know what happens?
- Events that have occurred have not been of incredibly low frequency (Chernobyl, Fukushima)
- Worth pursuing incredible IEs?

Terminology

Global statistical analysis:

- **Estimation of accident frequencies (CDF and LERF) based on core damage events and large early release events, i.e., events at the plant level**

PRA:

- **Estimation of accident frequencies (CDF and LERF) using identified accident scenarios and statistical evidence and models at the component level**

Assumption Underlying Global Statistical Estimates

- **A simple formula:**

$$F = \frac{N}{T}$$

F = frequency (events/reactor year)

N = number of events

T = total number of operating reactor years

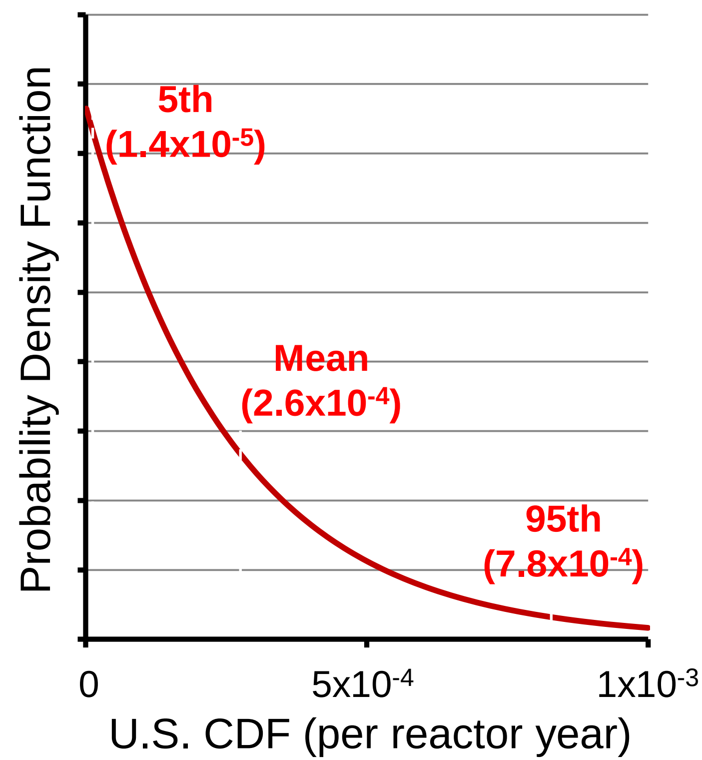
- **Important Assumption: All of the reactors in the population are nominally identical and are operating under the same regulatory system (*exchangeable* events).**

Global Statistical Estimates of CDF

- **U.S. Experience**
  - ➢ **1 core damage event (TMI-2)**
  - ➢ **3,839 LWR reactor years**

- **Exchangeability is assumed between TMI-2 and current reactors (PWRs, BWRs, all sites).**

- **Exchangeability is invalid.**



From: ***Probabilistic Risk Assessment and Regulatory Decision making: Some Frequently Asked Questions***, **NUREG-2201, March 2016**

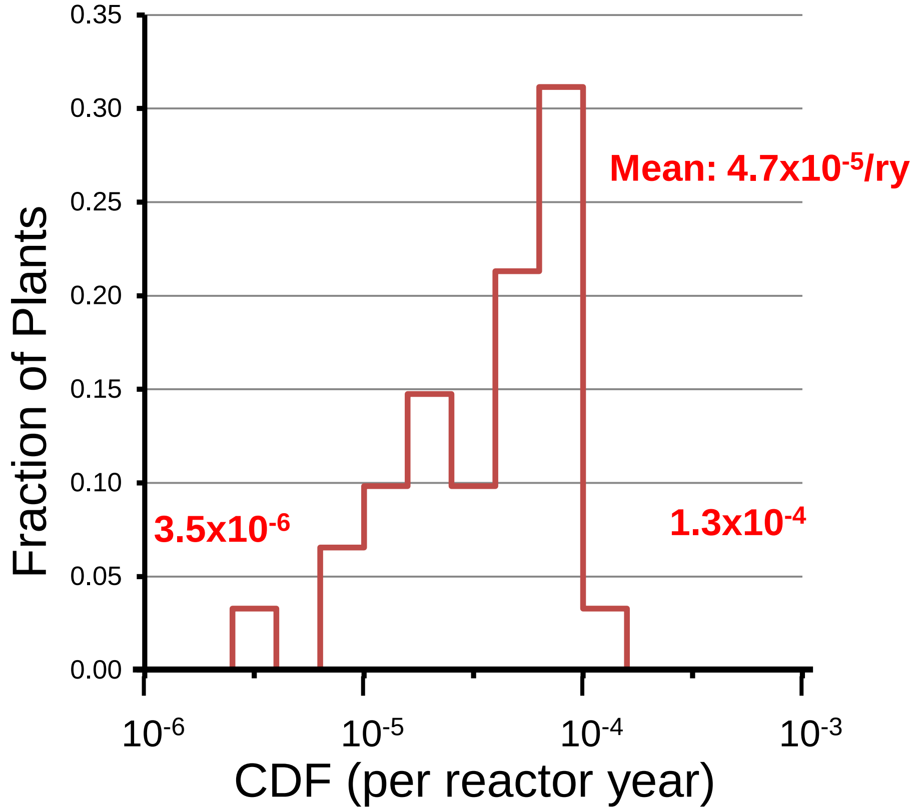
Why is Exchangeability Invalid?

Major changes are instituted after accidents:

- **Regulatory changes after TMI and Fukushima**
- **Establishment of INPO after TMI**
- **IPE and IPEEE programs after TMI**
- **FLEX after Fukushima**

PRA CDF Estimates for U.S. Plants

- **Current point estimates including internal and external events (61 units)**
  - **Post 2000 (90% after 2005)**
  - **Plant-to-plant variability reflects differences in designs and modeling**



From License Amendment Requests (LAR) and Severe Accident Management Alternative (SAMA) analyses

A Continuous Learning Process: TMI Accident

- **Upgraded requirements for auxiliary feedwater systems, containment building isolation, and reliability of pressure relief valves, among others**
- **Upgraded emergency planning regulations**
- **Added requirements related to hydrogen control**
- **Revamped operator training and staffing requirements**
- **Established fitness-for-duty programs**

A Continuous Learning Process: Fukushima Accident

- **Requiring mitigation strategies for beyond-design-basis external events**
- **Requiring consideration of multi-unit accidents**
- **Mandating severe accident capable containment vents for BWRs with Mark I and II containments**
- **Requiring integration of emergency operating procedures and procedures for coping with severe accidents**

A Continuous Learning Process: Analysis

- **Significance of small LOCA, human error, and support systems** (Reactor Safety Study)
- **Significance of seismic and fire risk** (Zion and Indian Point PRAs)
- **Significance of low power and shutdown operations** (French PRA)
- **Risk contributors are plant specific, even for sister units** (Indian Point PRAs)

What is the Message?

- Global statistical analysis requires the assumption that TMI-2 and Fukushima are exchangeable with current reactors. They are not.
- It is the qualitative insights from operational experience that are useful in regulatory decision making, not the frequencies of core damage and release derived from this experience.
- PRA results represent current design, operation, and regulation.

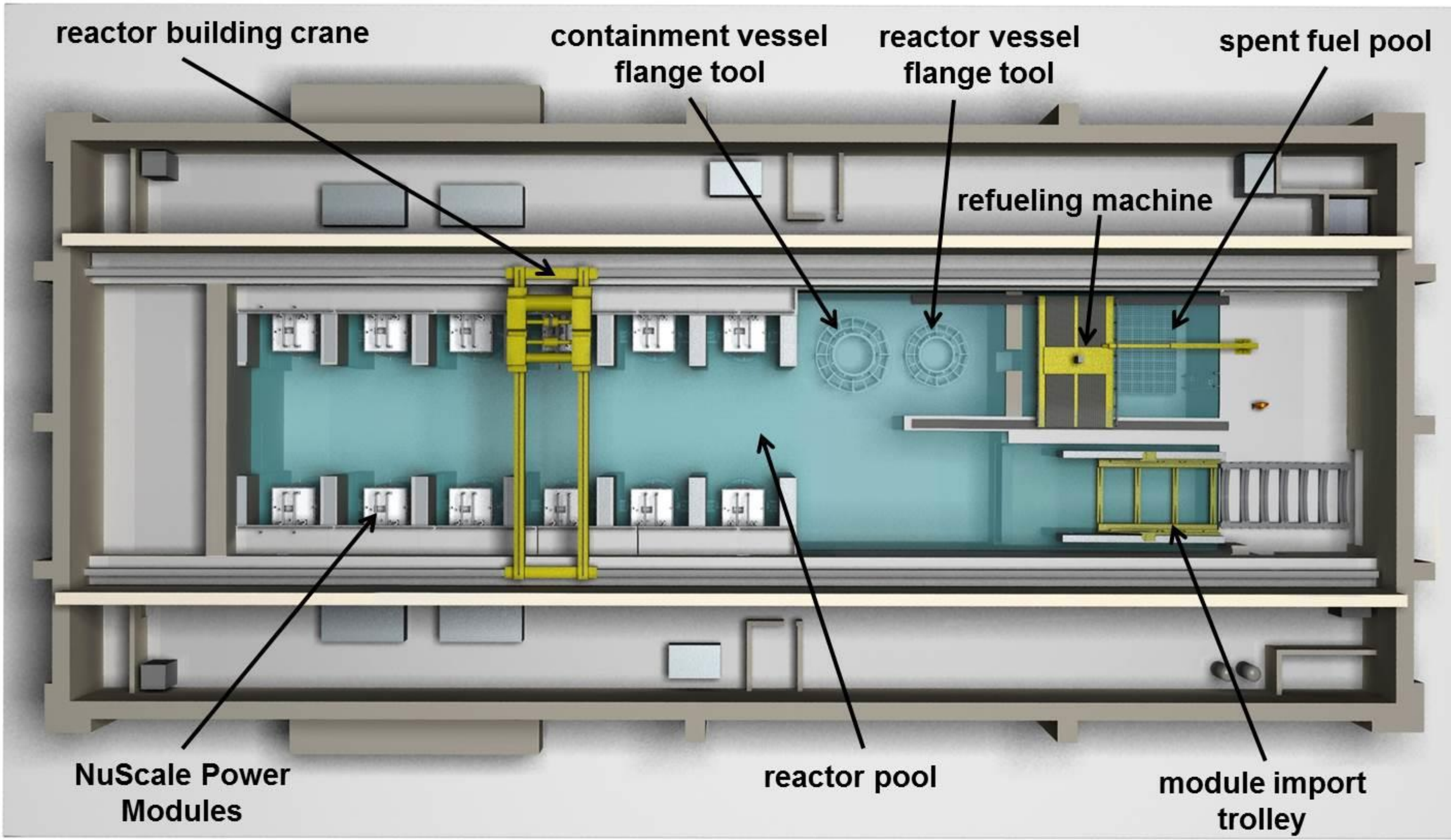
(Early) Concluding Remarks

- **Regulatory decision making must be based on the current state of knowledge.**
  - **The current state of knowledge regarding design, operation, and regulation (as reflected in the PRAs) is key.**
  - **The current state of knowledge is informed by science, engineering, and operating experience, including past incidents.**
  - **The need for the assumption of *exchangeability* between past, present, and future reactors makes global statistical estimates of little value in regulatory decision making.**
- **PRAs do not "predict" the future; they evaluate and assess potential accident scenarios to inform the decision makers' current state of knowledge.**

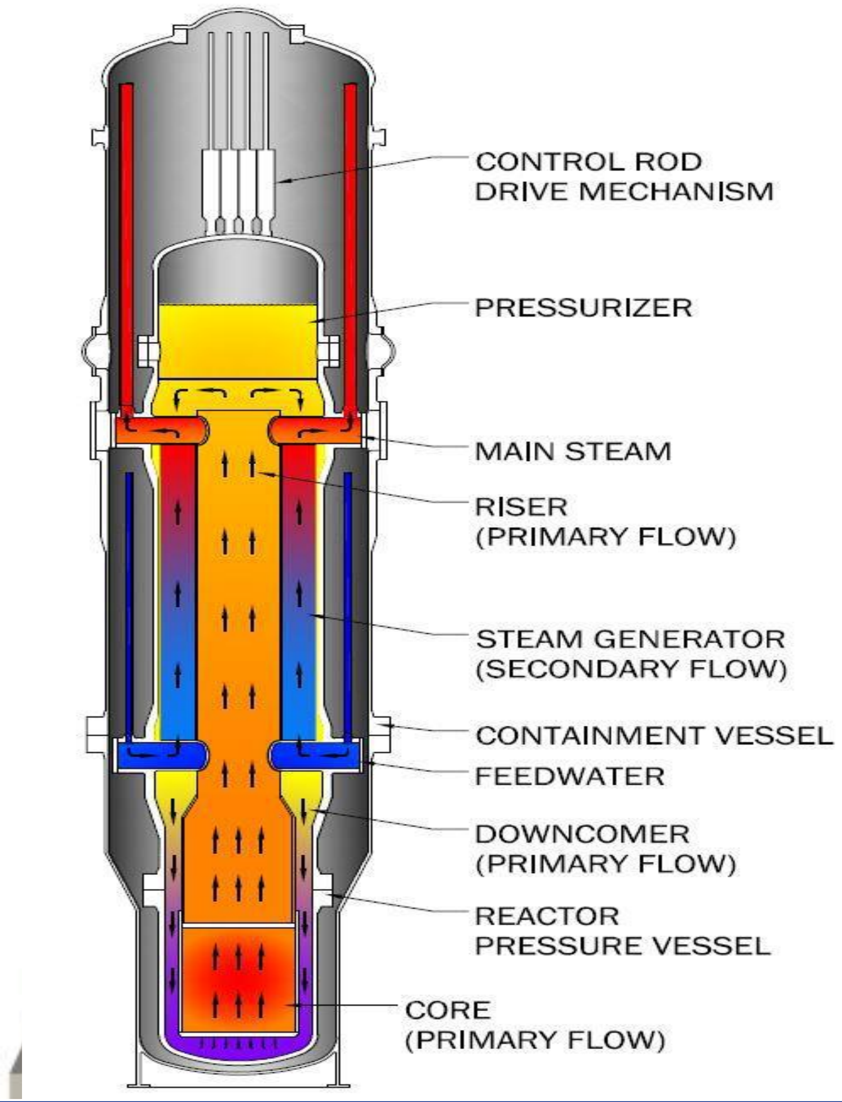
Multi-Unit and Adjacent Sites

- **U.S.A.**
  - Currently at most 3 units
  - Plant Vogtle will have 4
  - Geographically adjacent sites: Salem 1&2 (PWRs;) and Hope Creek (BWR; 3 total, PSEG); Nine Mile Point 1&2 (BWR; Constellation Energy) and FitzPatrick (BWR; Entergy; 3 total)
- **Canada**
  - Bruce Power: 8
- **Japan**
  - Kashiwazaki-Kariwa: 7

NuScale Design of a Small Modular Reactor



NuScale Power Module



Current Situation

- ***General Design Criterion 5:*** **Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.**
- **U.S. Safety Goals are applied to single units.**
- **PRAs, with few exceptions, are performed for single units.**

Whole-Site Risk: Early Consideration

- In the early 1980s, the NRC staff proposed that Safety Goals be applied on a per-site basis
- Commission decided not to impose a "bias" against multi-unit sites
- Quantitative Health Objectives are now interpreted on a per-reactor basis

NUREG-0800, SRP Chapter 19

For small, modular integral pressurized water reactor designs, the staff reviews the results and description of the applicant's risk assessment for a single reactor module; and, if the applicant is seeking approval of an application for a plant containing multiple modules, the staff reviews the applicant's assessment of risk from accidents that could affect multiple modules to ensure appropriate treatment of important insights related to multi-module design and operation.

CDF for Multiunit/Multimodule Sites

- **Is CDF still applicable?**
- **For Small Modular Reactors, the radioactive inventory is much smaller than current large LWRs**
- **CDF proposed definitions and goals:**
  - **Site SCDF: The aggregate of frequencies of all event sequences that can lead to significant core degradation in any of one or more reactors on the site should be less than ZSCD per site year** (Candu Owners Group)
  - **Frequency of severe core degradation (SCDF) < 10^{-5} per reactor year. The effects of adjacent units at multi-unit stations are considered and accounted for when calculating the Safety Goals for internal events sequences at the representative unit (generally, the lead unit)."** (Canadian Nuclear Safety Commission)

CDF (IAEA)

- Single-unit CDF (SUCDF) : frequency per site-year of core damage involving one and only one reactor unit on the site
- Multi-unit CDF (MUCDF) : frequency per site-year of core damage involving two or more reactor units concurrently on the site
- Site CDF (SCDF) : frequency per site-year of core damage involving one or more reactor units on the site
- For n units: SCDF = nSUCDF

ACRS Letter, April 2004

- **Option 1**
  - The site goal (e.g., 10^{-4} per ry) is divided by the number of units at the site.
  - The risk from and the likelihood of a core damage accident at all sites cannot be precisely equal. However, there is the expectation that they be comparable.
- **Option 2**
  - CDF is an accident prevention goal and its value should be the same for each reactor at every site.
  - Requiring each module to have a CDF value given by the overall CDF goal divided by the number of modules introduces a new Safety Goal concept, a site CDF. Such a concept was never intended to be part of the Safety Goals.

MUPSA Challenges (CNSC Workshop)

- **Accident Progression and Source term Characterization**
  - Need to define new release categories that adequately describe the releases from multi-unit accidents. This includes release magnitudes, energies, and timing from reactor units, spent fuel storage, and other radiological sources
- **Evaluation of Radiological Consequences**
  - Includes consideration of different points of release from the plant, possible differences in time of release, and release energies for plume rise considerations.
- **Site-Based Safety Goals**
  - Need to define multi-unit site based acceptance criteria for evaluating the integrated risks from a multi-unit site PSA

Licensing Non-Light Water Reactors USNRC Strategies

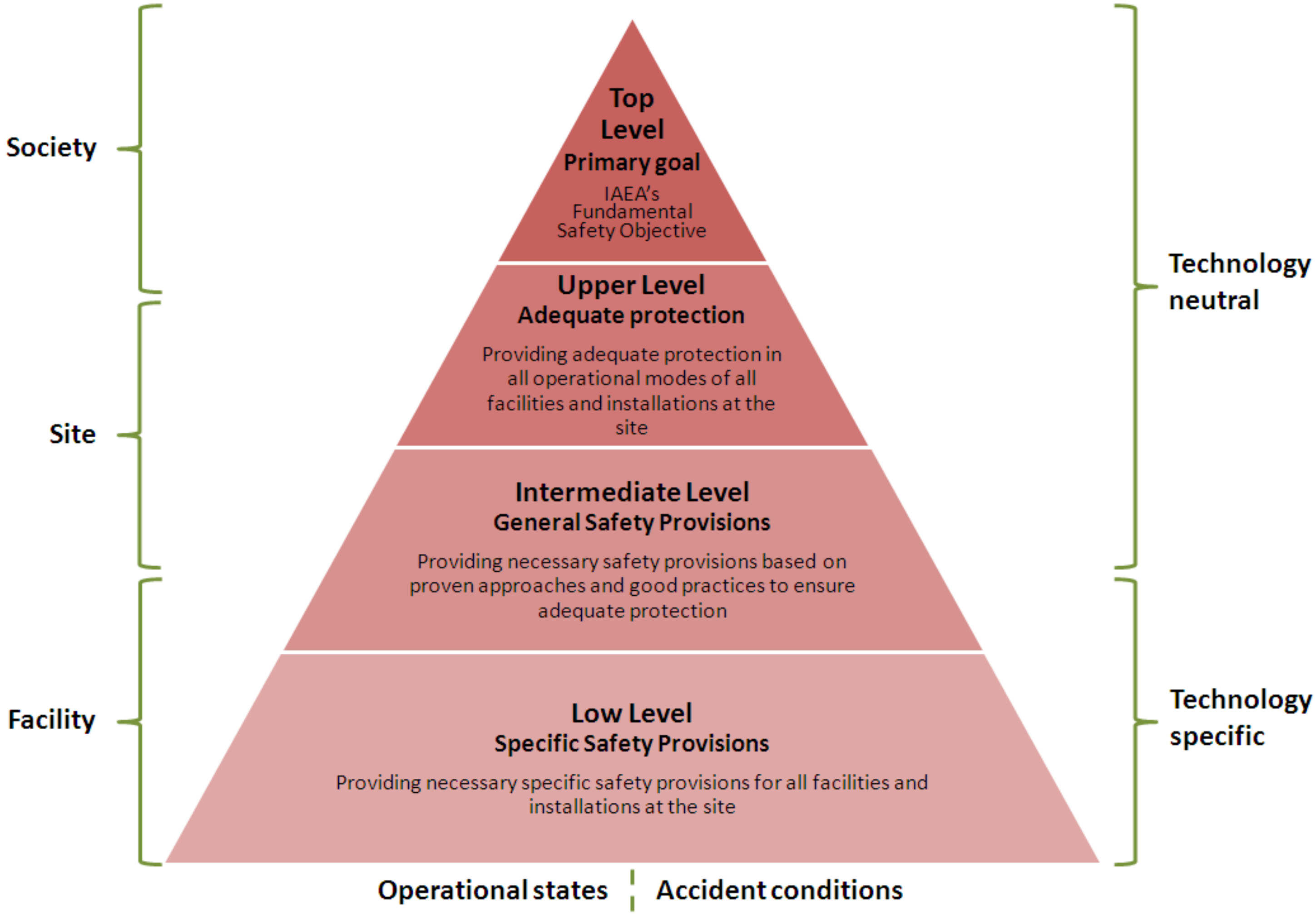
- Establish a more flexible, risk-informed, performance-based, non-LWR regulatory review process within the bounds of existing regulations, including the use of conceptual design reviews and staged-review processes. This flexibility will accommodate potential applicants having a range of financial, technical, and regulatory maturity, and a range of application readiness.

- Identify and resolve technology-neutral policy issues that impact the regulatory reviews, siting, permitting, and/or licensing of non-LWR nuclear power plants (NPPs).

Comments

- Advanced reactors use a variety of fuels and coolants
- The only common metrics are those related to risk.
- If the regulatory framework is to be technology-neutral or technology-inclusive, it must use risk metrics, e.g., frequency of radioactive releases.
- Other parts of the regulations should be risk-informed also, e.g., the General Design Criteria, the Defense-in-Depth Principle (NUREG-2150)

IAEA Goal Hierarchy



NRRC Mission and Vision

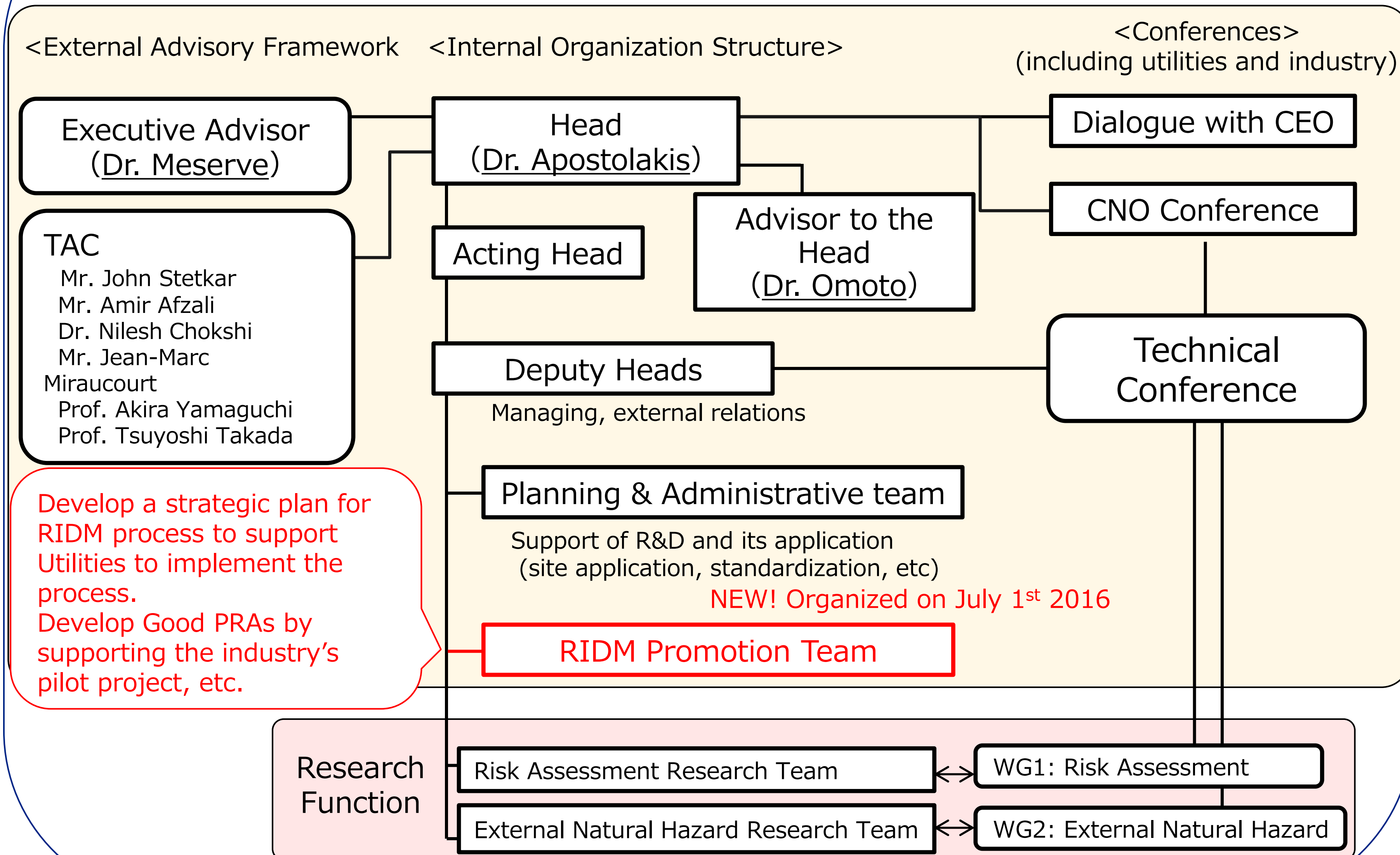
Mission Statement

To assist nuclear operators and nuclear industry to continually improve the safety of nuclear facilities by developing and employing modern methods of Probabilistic Risk Assessment (PRA), risk-informed decision making and risk communication.

Vision Statement

To become an international center of excellence in PRA methodology and risk management methods, thereby gaining the trust of all the stakeholders.

NRRC Organization



NRRC Activities

- Position paper for proper application of RIDM in Japan
  - Establishment of RIDM Promotion Team
  - Pilot projects for establishing “Good” PRAs: Ikata Unit 3, Kashiwazaki-Kariwa Units 6 and 7
- White paper on RIDM applications in the U.S.A.
  - What was the motivation?
  - How can Japan benefit from the U.S. experience?
- Research projects
  - Human Reliability Analysis
  - Seismic PRA
    - ✓ SSHAC process for Ikata Unit 3 (Senior Seismic Hazard Analysis Committee)
  - Fire PRA
  - Volcano PRA