Initiatives on utilizing risk information

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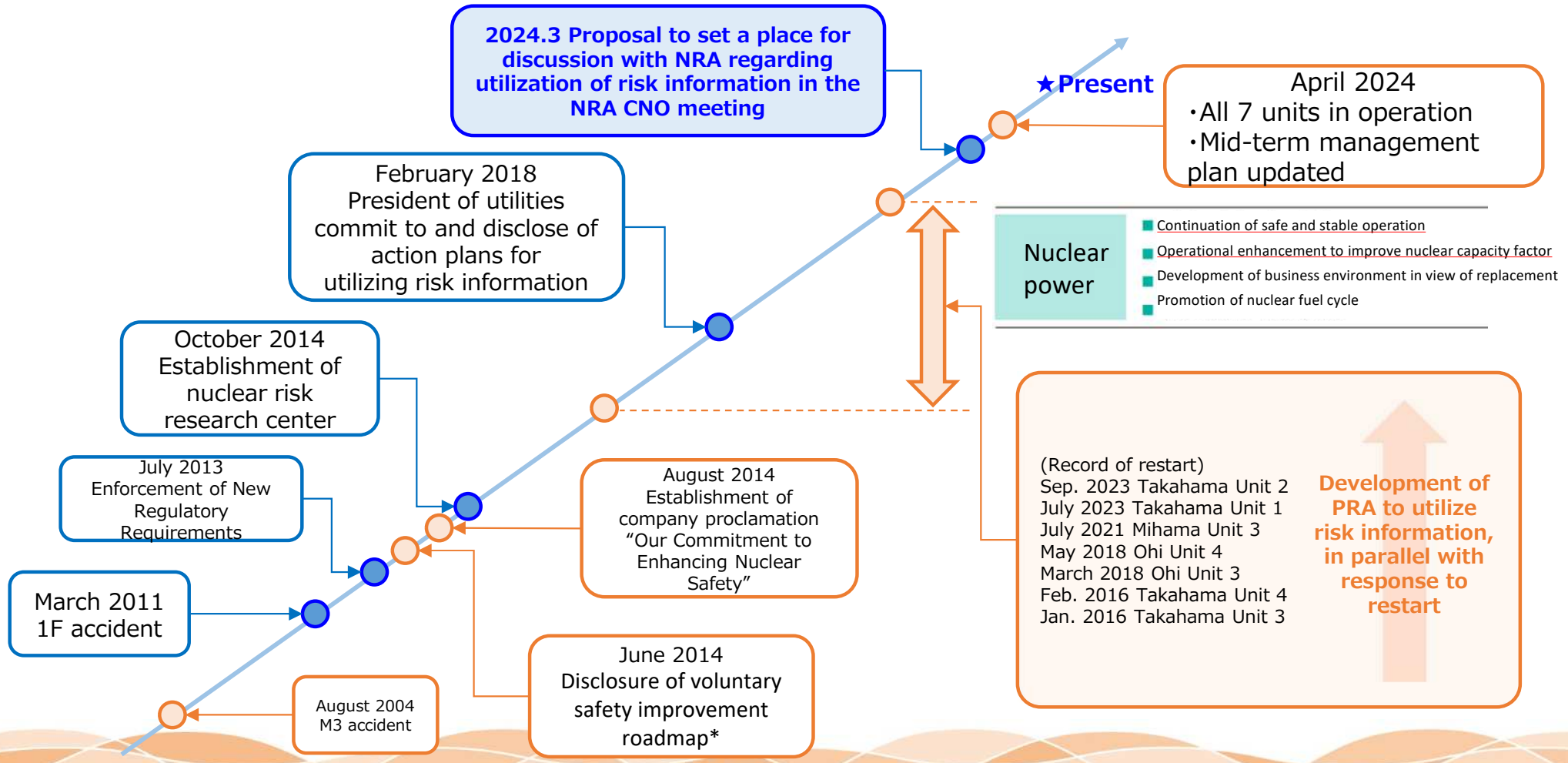
Current situation with RIDM

- ○ In parallel with response to restart, KEPCO is developing PRA for each plant to utilize risk information in both regulated and voluntary area.
- ○ Discussions have begun with the Nuclear Regulation Authority (NRA) regarding utilization of risk information. The direction is to discuss specific issues and improvement measures for subject items and link them to practical application, after identifying and sharing individual items that lead to “safety improvement” and “optimization of resources” with the regulatory authority and industry.



*Hereafter, compiled and disclosed every semester

Examples of past initiatives

○ We are proactively utilizing risk information not only for safety improvement measures in ‘Safety Assessment Reporting for Continuous Improvement’ (hereinafter simply referred to as ‘SAR’) and SDP evaluations (determination of the importance of trouble events) in nuclear regulatory inspections, but also for voluntary purposes.

Examples of past initiatives	Contents
① Safety improvement measures in SAR ⇒P4～7	• A systematic approach was introduced to identify the importance of accident sequences from the "absolute value" and "contribution ratio" of the CDF. • Effective risk reduction measures were then introduced for accident sequences with high importance.
② SDP evaluations in nuclear regulatory inspections ⇒P8～10	• As part of the SDP evaluation, PRA is used to determine the importance of trouble events. • We provide our PRA models to NRA, who confirmed the appropriateness of them (L1 internal events).
③ Voluntary risk management for changes in configuration ⇒P11～14	• When isolating equipment not subject to LCO, risk information is used to specify compensation measures and reduce CDF and ICCDP. • Furthermore, in preparation for earthquakes, etc., measures are implemented such as prohibiting cranes from passing directly above equipments during operation.

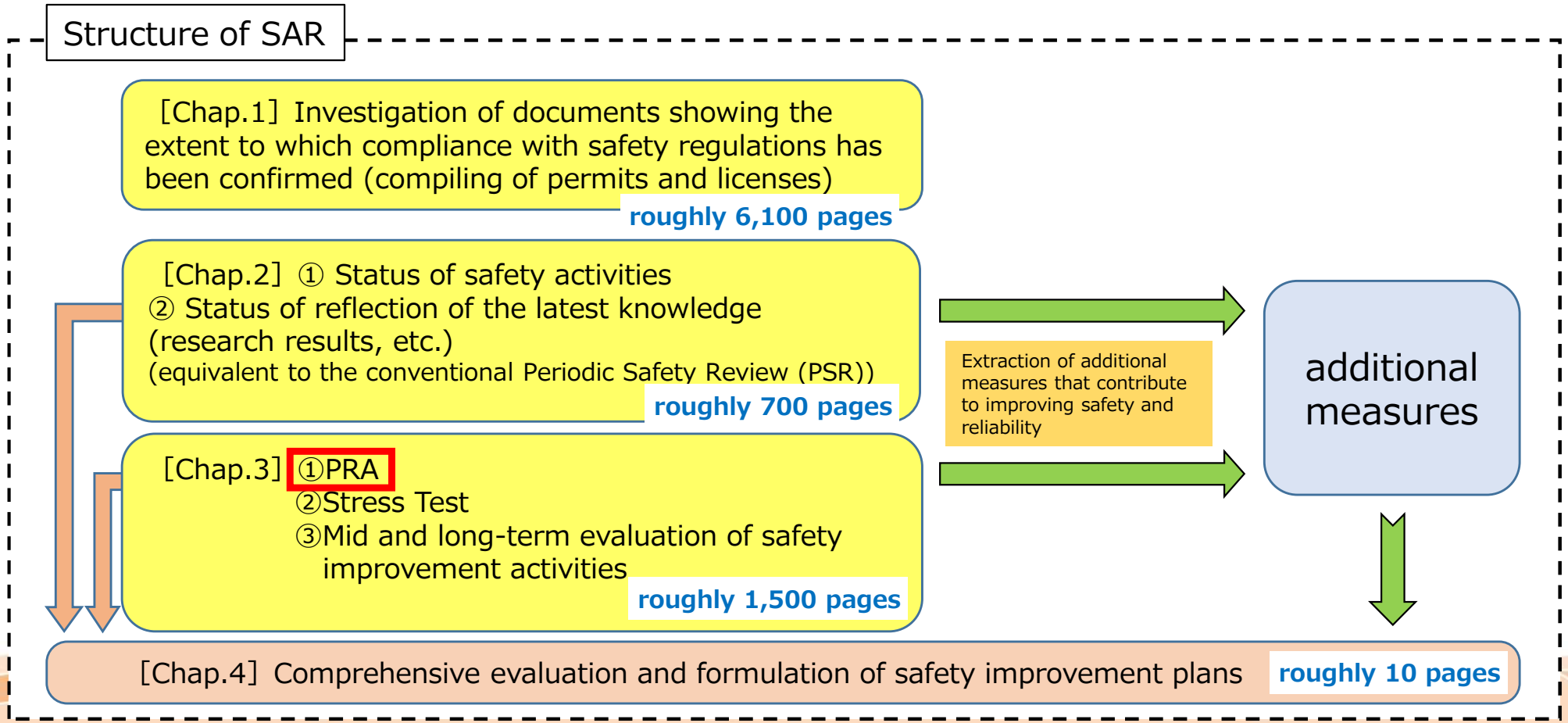
①Safety improvement measures (1/4)

○ The SAR for restarted plants has been institutionalized, and additional measures that contribute to improving safety and reliability are extracted based on the results of PRA, etc.

<Objectives>

- The NRA has institutionalized safety improvement evaluations with the objective of operators taking voluntary and continuous measures to improve safety.
- Evaluate the safety of plants at the end of regular inspections, identify improvement measures (additional measures), and specify plans.

Structure of SAR

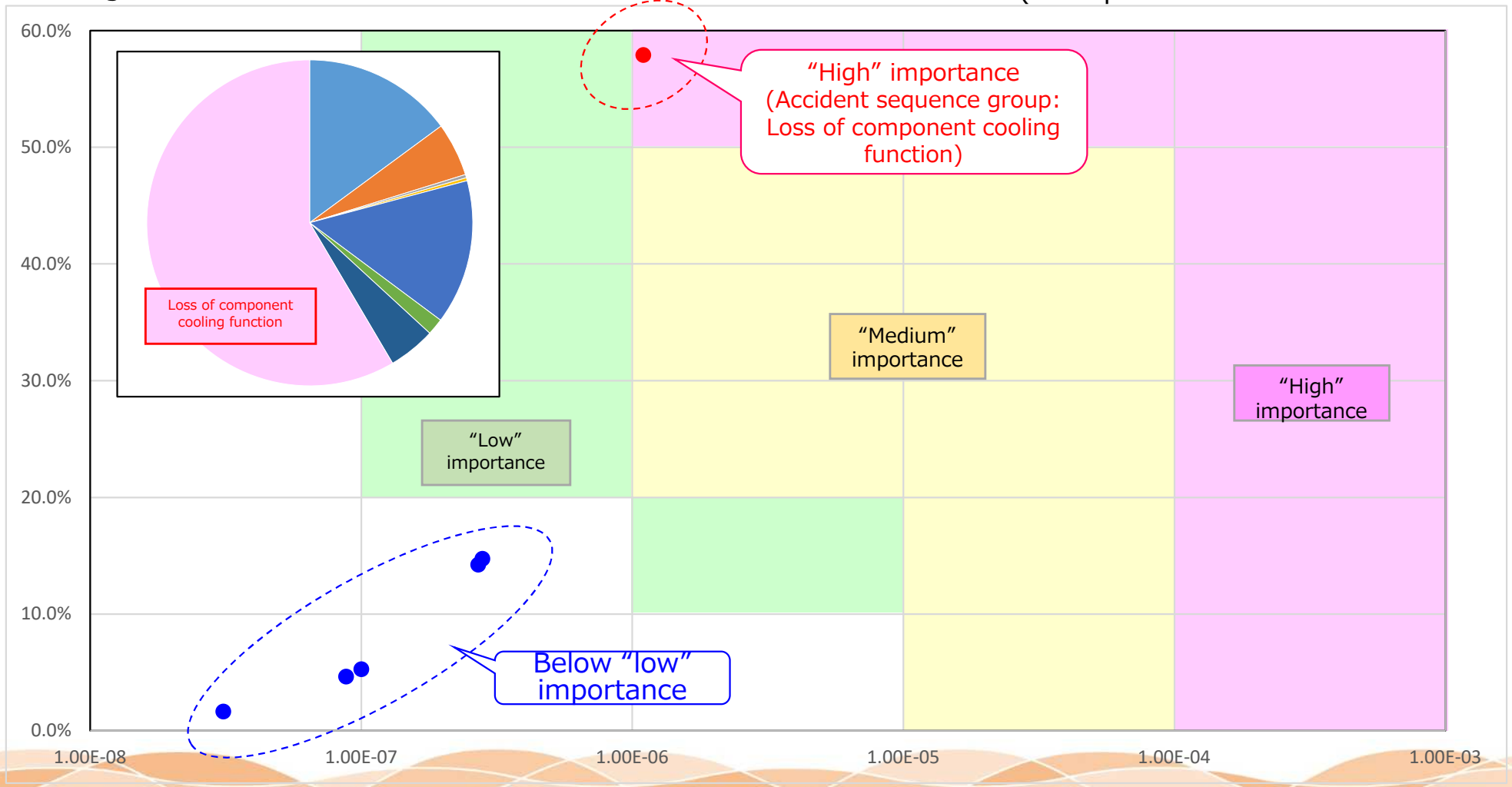


①Safety improvement measures (2/4)

- ○ Importance is mapped based on CDF by accident sequence group and percentage of contribution to total CDF.
- ○ Accident sequence groups with high importance were identified, and safety improvement measures that significantly contribute to risk reduction were implemented on a priority basis, and have already been publicized in SAR.

Percentage of contribution to total CDF

(Example of Ohi Power Station Unit 3)



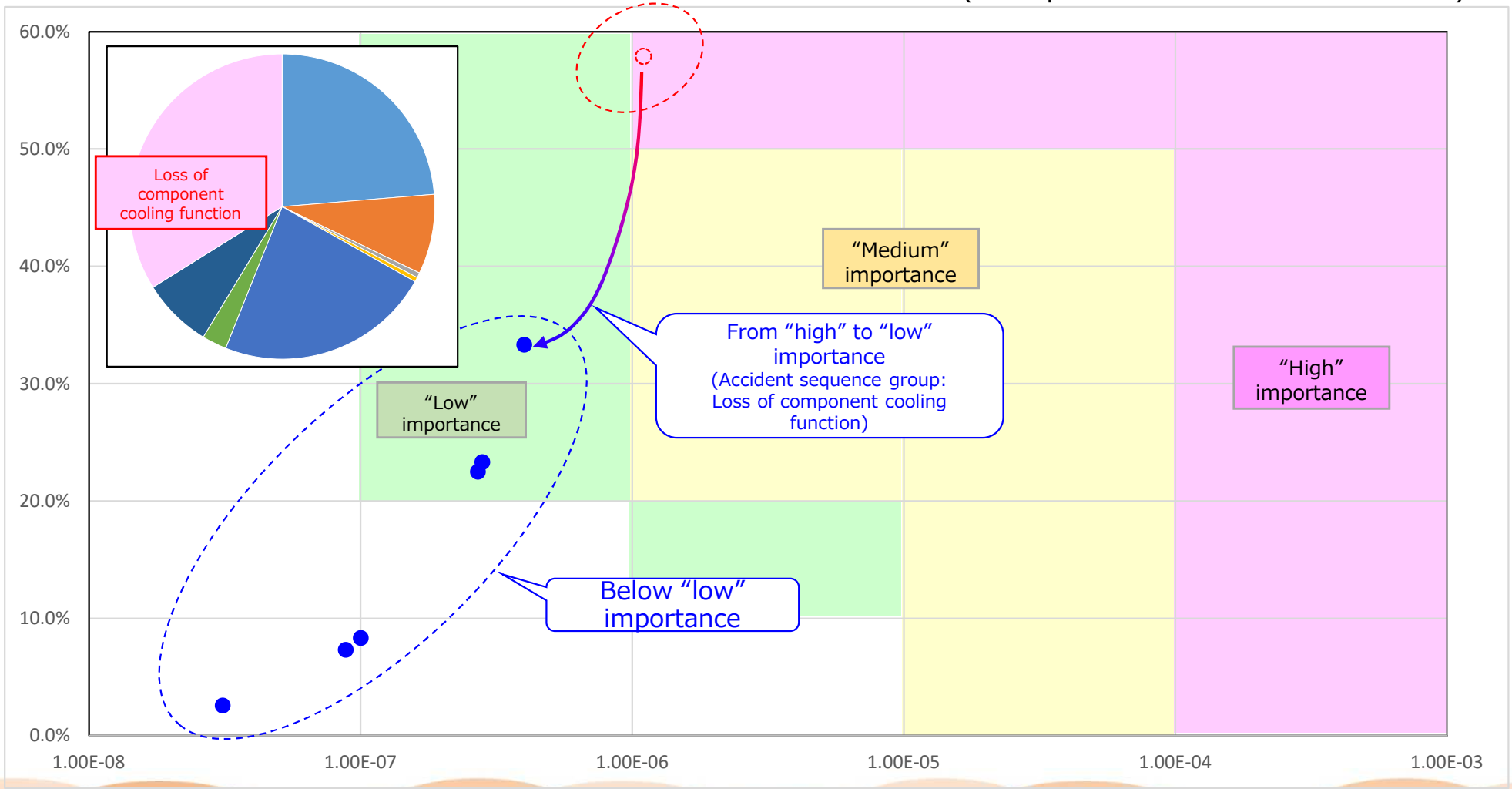
CDF by accident sequence group

①Safety improvement measures (3/4)

- ○ By introducing the “RCP shutdown seal”, risk of “loss of component cooling function” was reduced.
- ○ Importance of all accident sequence groups became below “low”, and total CDF decreased by approx. 40%.

Percentage of contribution to total CDF

(Example of Ohi Power Station Unit 3)



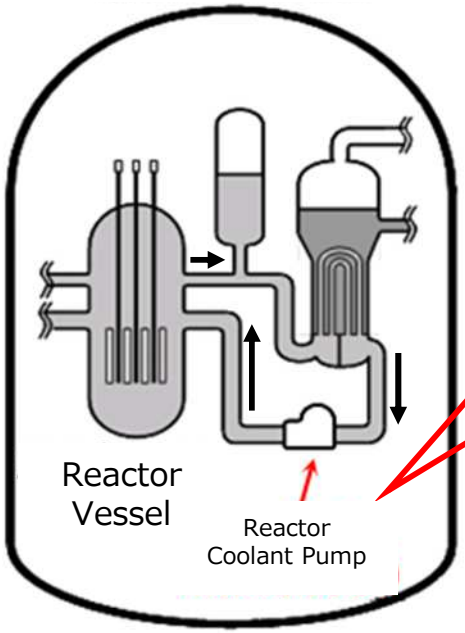
CDF by accident sequence group

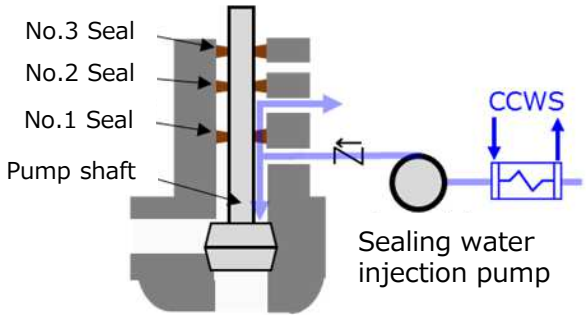
①Safety improvement measures (4/4)

- ○ Dominant risk is a scenario that leads to core damage due to leakage of primary coolant from the RCP seal (LOCA) resulting from loss of thermal barrier function of the reactor coolant pump (RCP) seal caused by loss of component cooling function (CCWS).
- ○ It was decided to introduce the “RCP shutdown seal” to reduce the risk of “loss of component cooling function”.

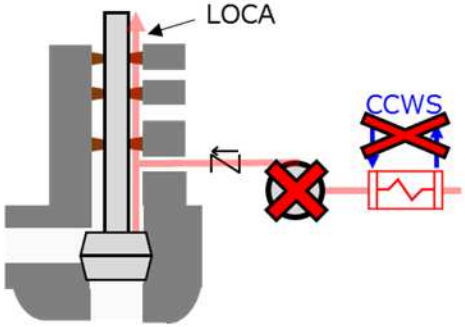
(Example of Ohi Power Station Unit 3)

Containment vessel

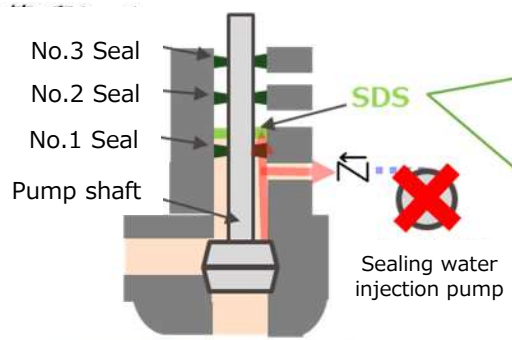




RCP is thermally protected with seal water

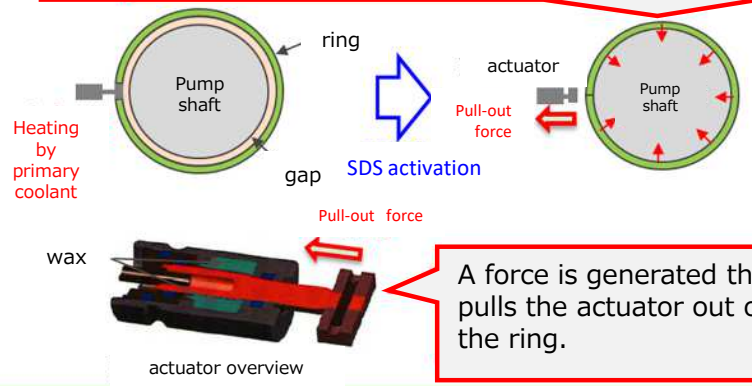


Risk of LOCA occurring due to RCP seal being damaged during loss of CCWS



RCP overview (location of SDS)

The ring with its reduced diameter fits tightly onto the axis of the RCP.

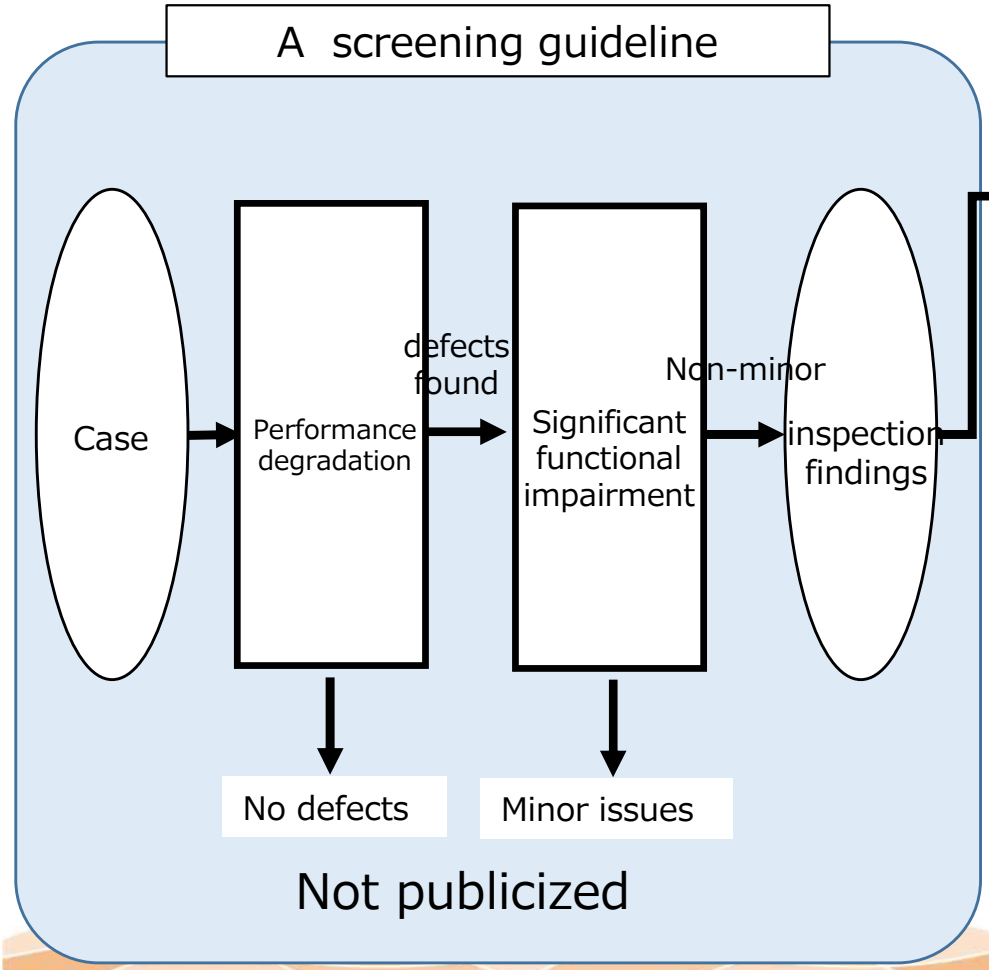


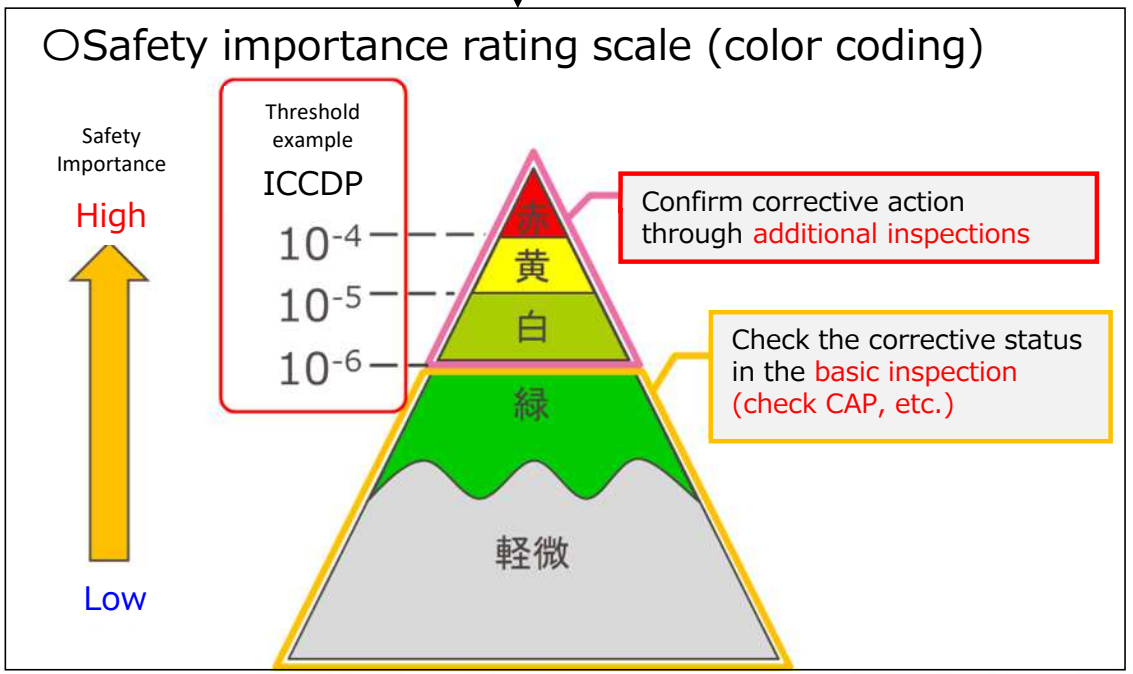
A force is generated that pulls the actuator out of the ring.

②Determining the importance of inspection findings in ROP（SDP） （1/3）

- ○ The Nuclear Regulatory Inspection (ROP), which began operation in April 2020, determines the degree of regulatory involvement according to the importance of inspection findings regarding unsafe conduct by operators.
- ○ The importance of inspection findings is determined through qualitative or quantitative evaluation, with PRA being used when conducting quantitative evaluation.

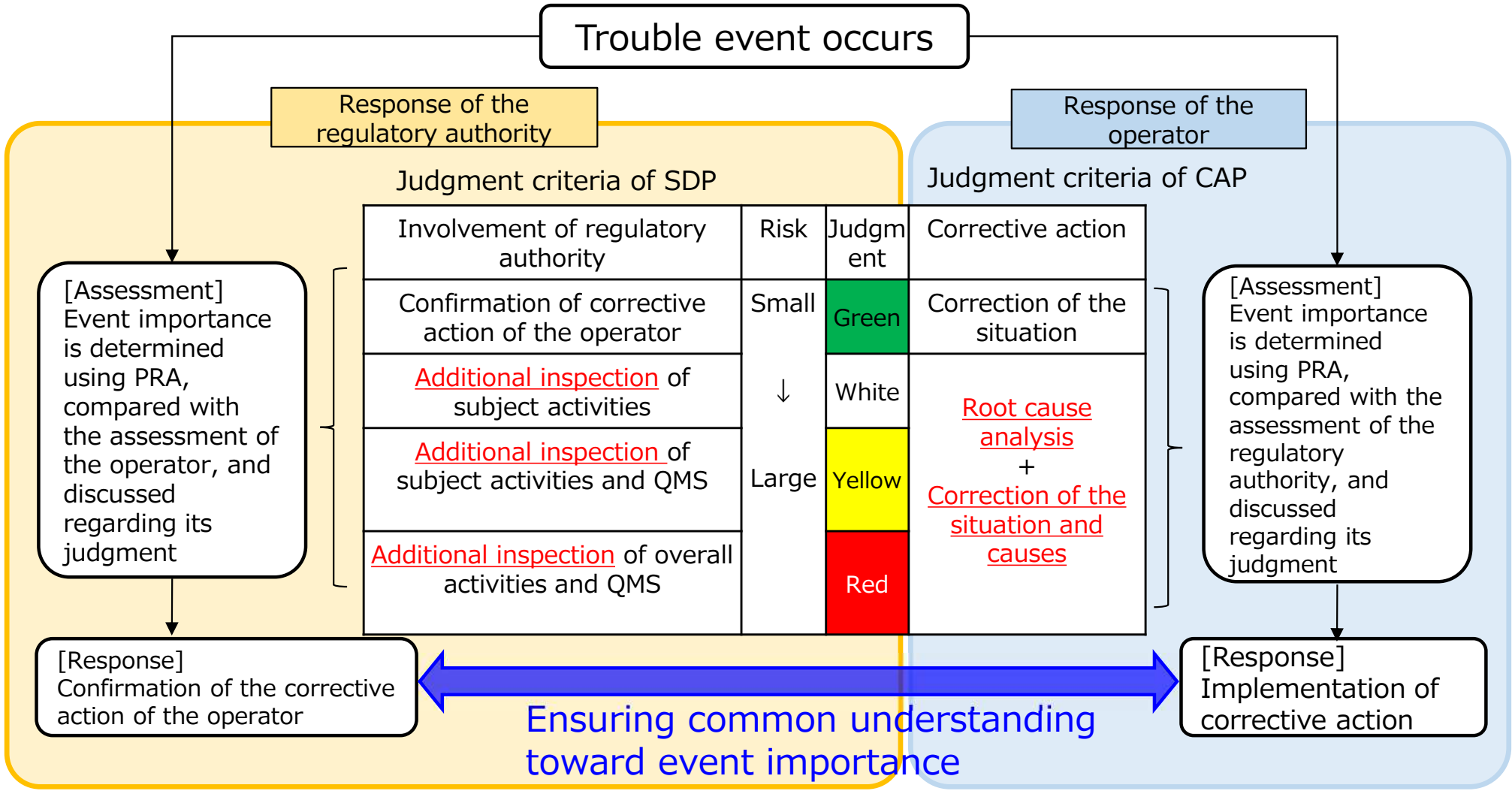
SDP : Significant Determination Process





②Determining the importance of inspection findings in ROP（SDP） (2/3)

- ○ When a trouble event occurs, the operator and the regulatory authority assess event importance using PRA, mutually compare them and discuss the judgment.
- ○ This ensures a common understanding of event importance.



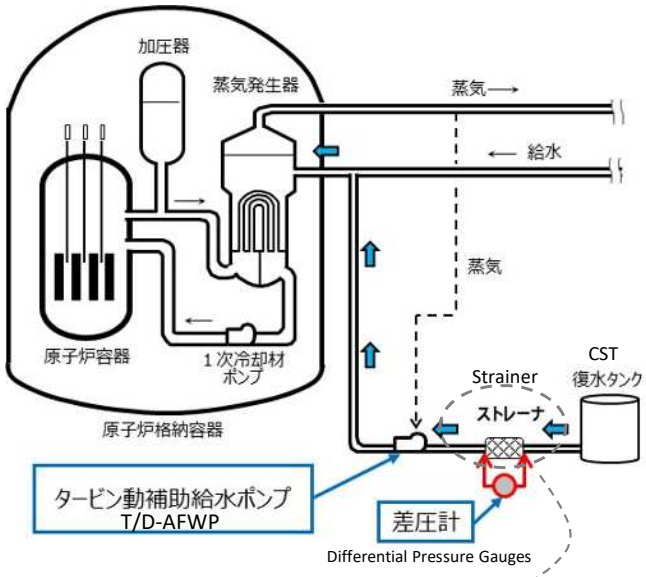
Regulatory response: Significance Determination Process (SDP), In-housei response: Corrective Action Process (CAP) *1

*1: In CAP activities, PRA information (ΔCDF, ΔCFF) is being used as an index for investigating the causes of problems in power plant operation and classifying the scope and depth of corrective actions.

②Determining the importance of inspection findings in ROP（SDP） (3/3)

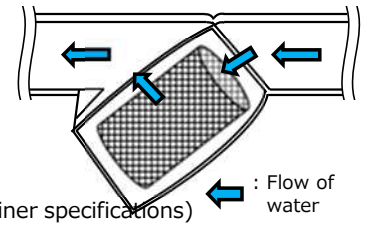
(Event: Increase in indicated value on the differential pressure gauge of turbine driven auxiliary feed water pump inlet strainer at Mihama Unit 3)

- ○ Since PRA of entry into LCO of the pump confirmed that it falls below the threshold of “large” impact, the operator determined the event as “medium” impact and explained it to NRA.
- ○ NRA assessed the event as an inspection finding of “green” importance.



[Strainer image]

Cylindrical mesh basket is installed to the container inside the pipe.



(Strainer specifications)
Pipe diameter: Approx. 250mm
Material: Stainless steel
Mesh: Dual structure of approx. 0.4mm and approx. 1.8mm

[Outline of the event]

- ➢ Indicated value on the differential pressure gauge of turbine driven auxiliary feed water pump inlet strainer increased, which was determined as entry into LCO.
- ➢ As overhaul of the strainer found sludge consisting of iron attached, the strainer and pump inlet pipe were cleaned (no abnormality on the differential pressure gauge body).
- ➢ Subsequent test confirmed that there is no problem with continuing operation of the pump, and thus, returned from entry into LCO.

[PRA based on CAP (Kansai Electric Power)]

- ➢ It was confirmed that it falls below the threshold of “high” impact, and was explained so to NRA.

	Assessment result	Threshold of “large” impact	Judgment
ΔCDF	1.42×10^{-7}	Greater than or equal to 1.0×10^{-6}	OK
ΔCFF	9.91×10^{-8}	Greater than or equal to 1.0×10^{-7}	OK

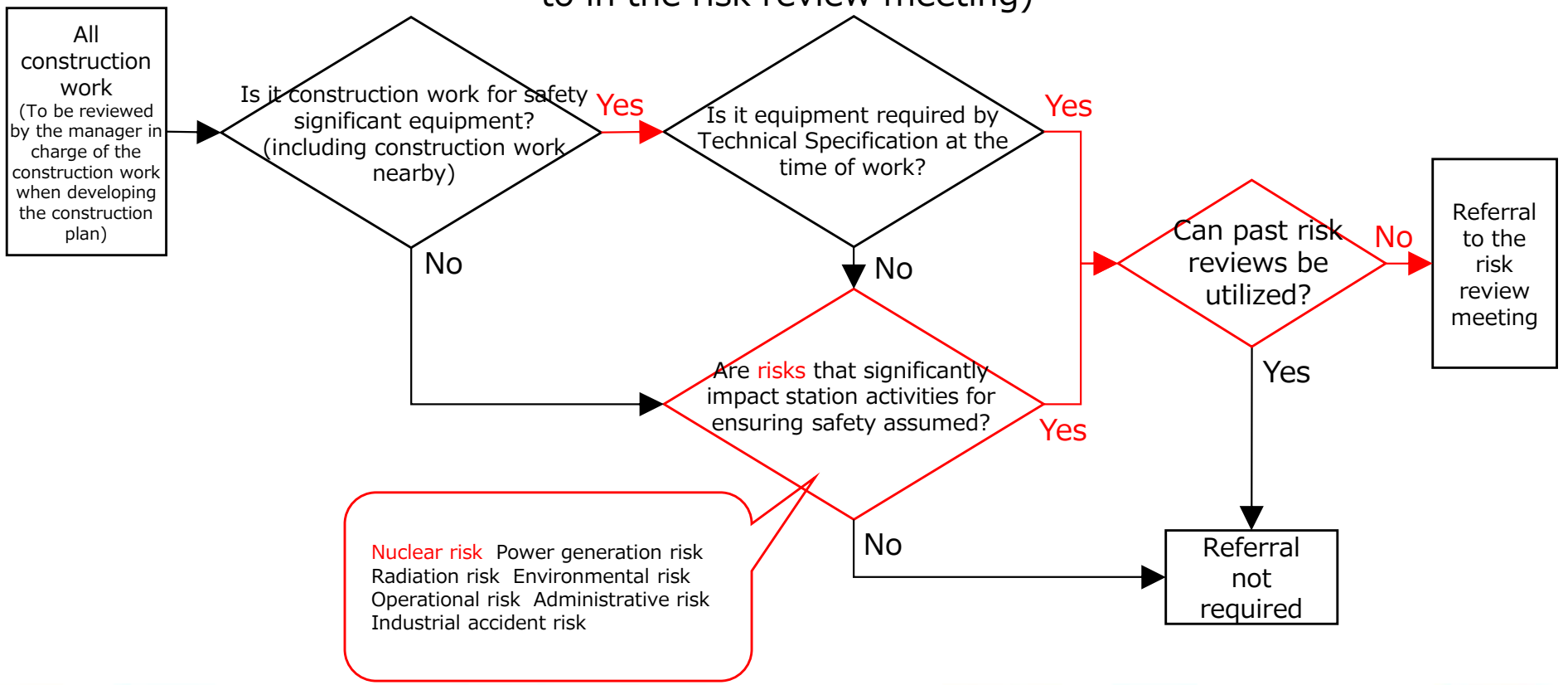
[Assessment of importance (NRA)]

- ➢ NRA assessed the event as an inspection finding of “green” importance.
  (“Green” importance is equivalent to ΔCDF being less than 1.0×10^{-6} and ΔCFF being less than 1.0×10^{-7})

③Voluntary Risk management (1/4)

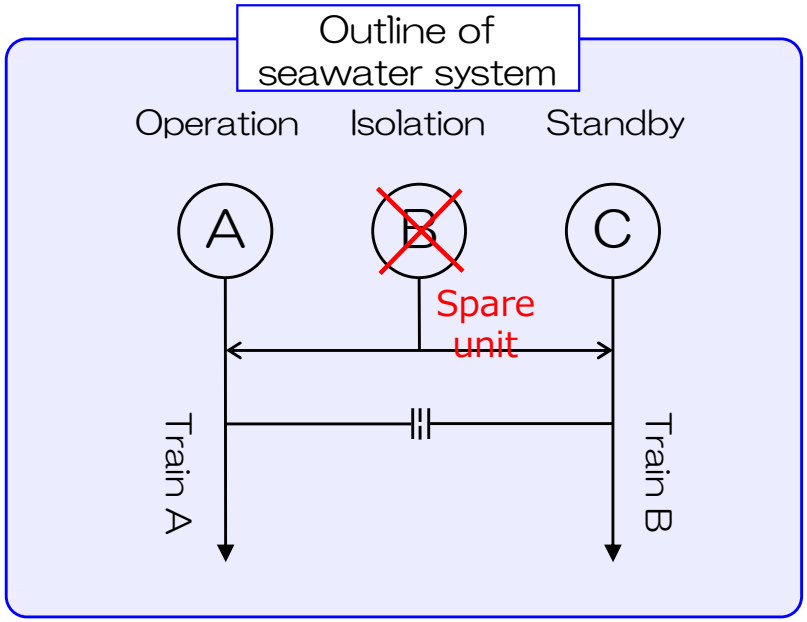
- ○ The section manager assesses risks and reviews compensatory measures for the risks, for construction work that could have a significant impact on station operations.
- ○ Risk review meetings are held for persons concerned at the power station, including site leadership, to verify the risk assessment results and validity of compensatory measures.

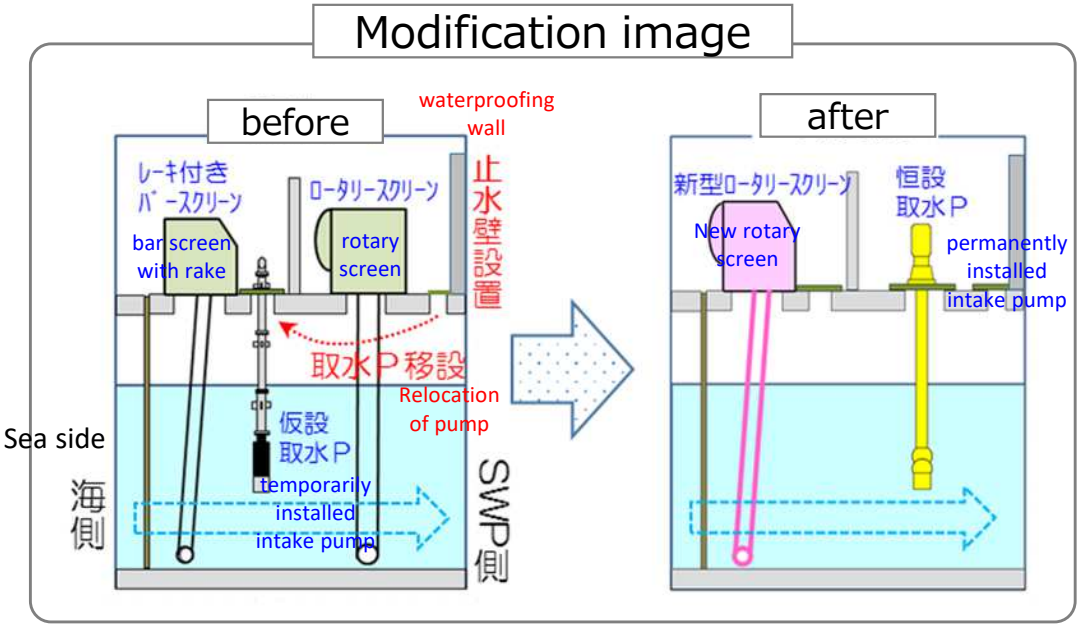
(Flow of identifying construction work to be referred to in the risk review meeting)



③Voluntary Risk management (2/4)

- ○ While implementing a measure to improve the reliability of facilities, the spare B-seawater pump that are not subject to LCO was isolated during operation.
- ○ In order to make the necessary preparations and quickly introduce OLM, we are accumulating experience in utilizing risk information on a voluntary basis.





○At Ohi Unit 3&4, frequency of intake pump failure increased due to temporarily installing the intake pump when setting the waterproofing wall in the seawater pump area as a safety improvement measure (no impact on CDF as it is intake pump of the flash evaporator).

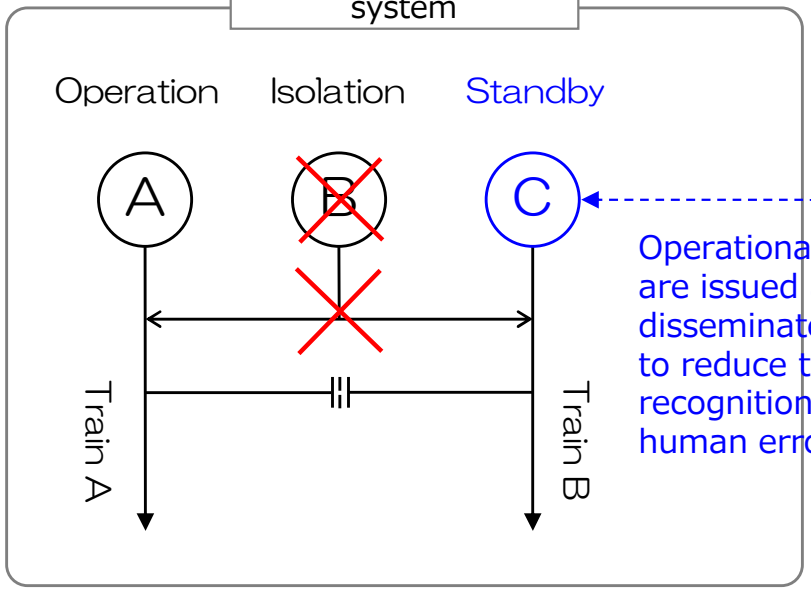
○Thus, it was planned to permanently install the intake pump by integrating the two screens and securing space.

○Risk assessment was conducted since it became necessary to isolate the seawater pump (1 spare unit) during reactor operation, from the standpoint of avoiding construction congestion during outage and ensuring safety.

③Voluntary Risk management (3/4)

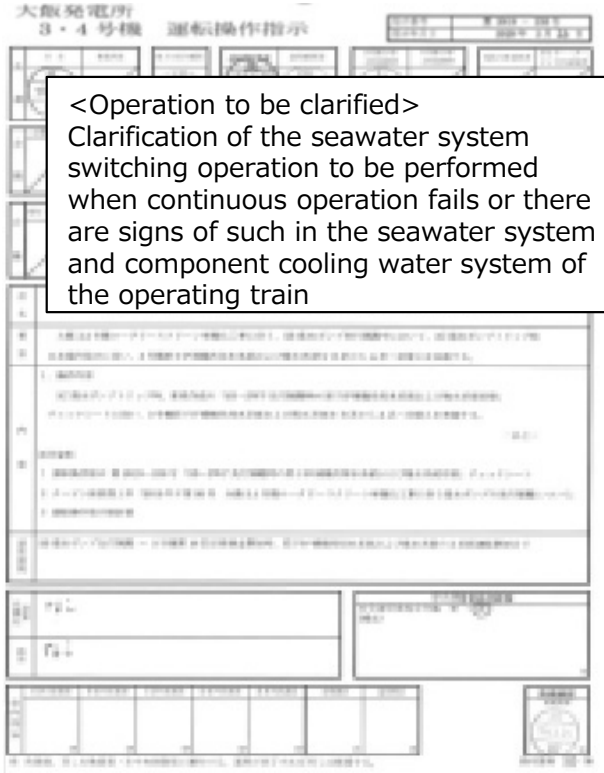
- ○ As a result of identifying reference events with high FV importance in materializing compensatory measures for isolation work of B seawater pump, it was determined that it is especially important to ensure switching to C seawater pump which is on standby.
- ○ Therefore, operational instructions that clarify the timing and procedures of the operation to switch to C seawater pump was issued ahead of isolation work of B seawater pump, and thoroughly disseminated to operators to reduce the risk of recognition failure and human errors.
- ○ Regarding external events, measures were taken in preparation for earthquakes and the like, such as prohibiting crane operations from passing over seawater pumps, based on site leadership.

Outline of seawater system



Operational instructions are issued and thoroughly disseminated to operators to reduce the risk of recognition failure and human errors

Operational instructions



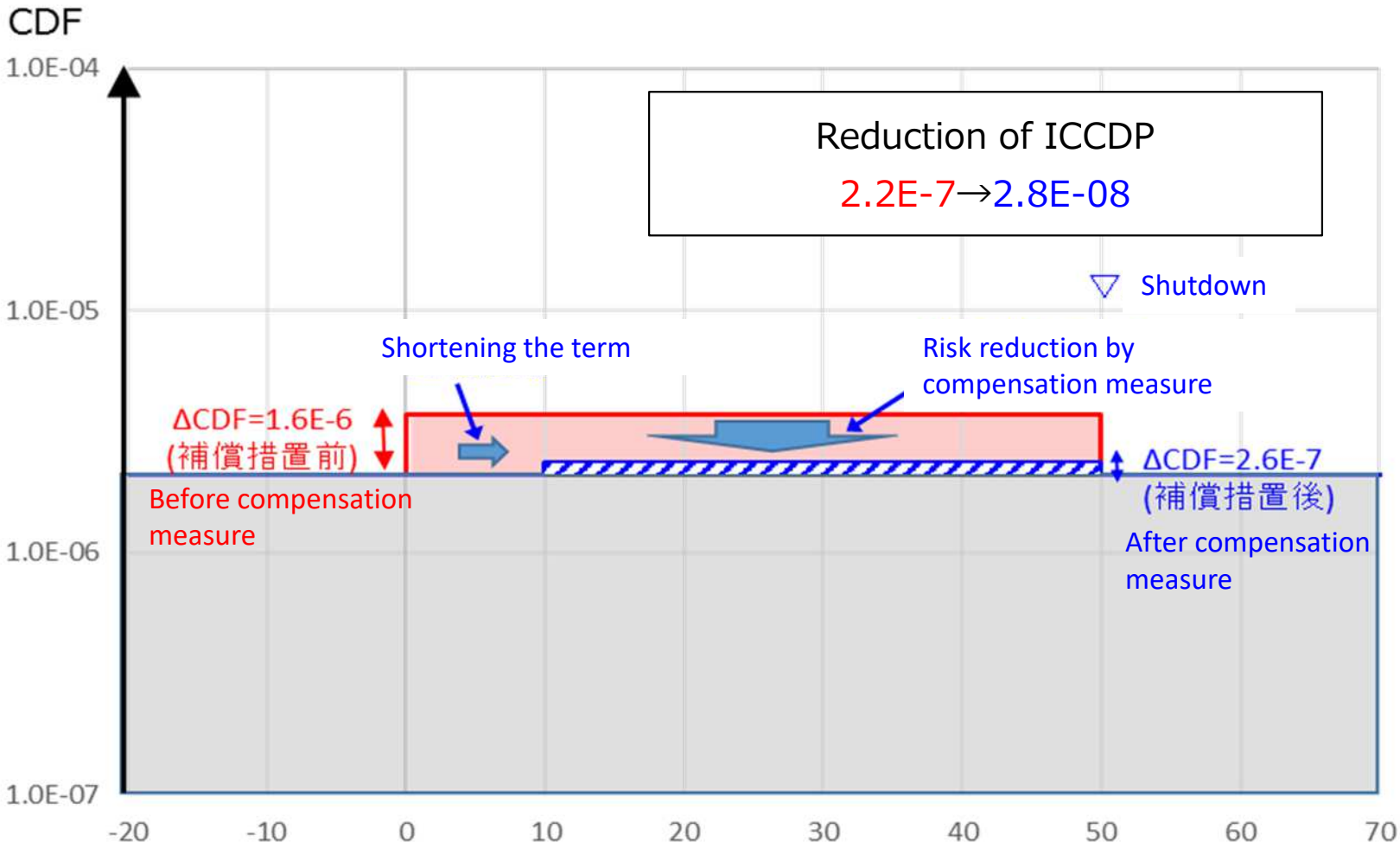
<Operation to be clarified>
Clarification of the seawater system switching operation to be performed when continuous operation fails or there are signs of such in the seawater system and component cooling water system of the operating train

(Example of reference event with high FV importance)

Rank	FV	Reference event	Probability of recognition failure	Probability of human error
2	0.36	C seawater pump startup (at A,B seawater pump failure)	6.1E-3 →1.4E-4	7.9E-3 →2.0E-3

③Voluntary Risk management (4/4)

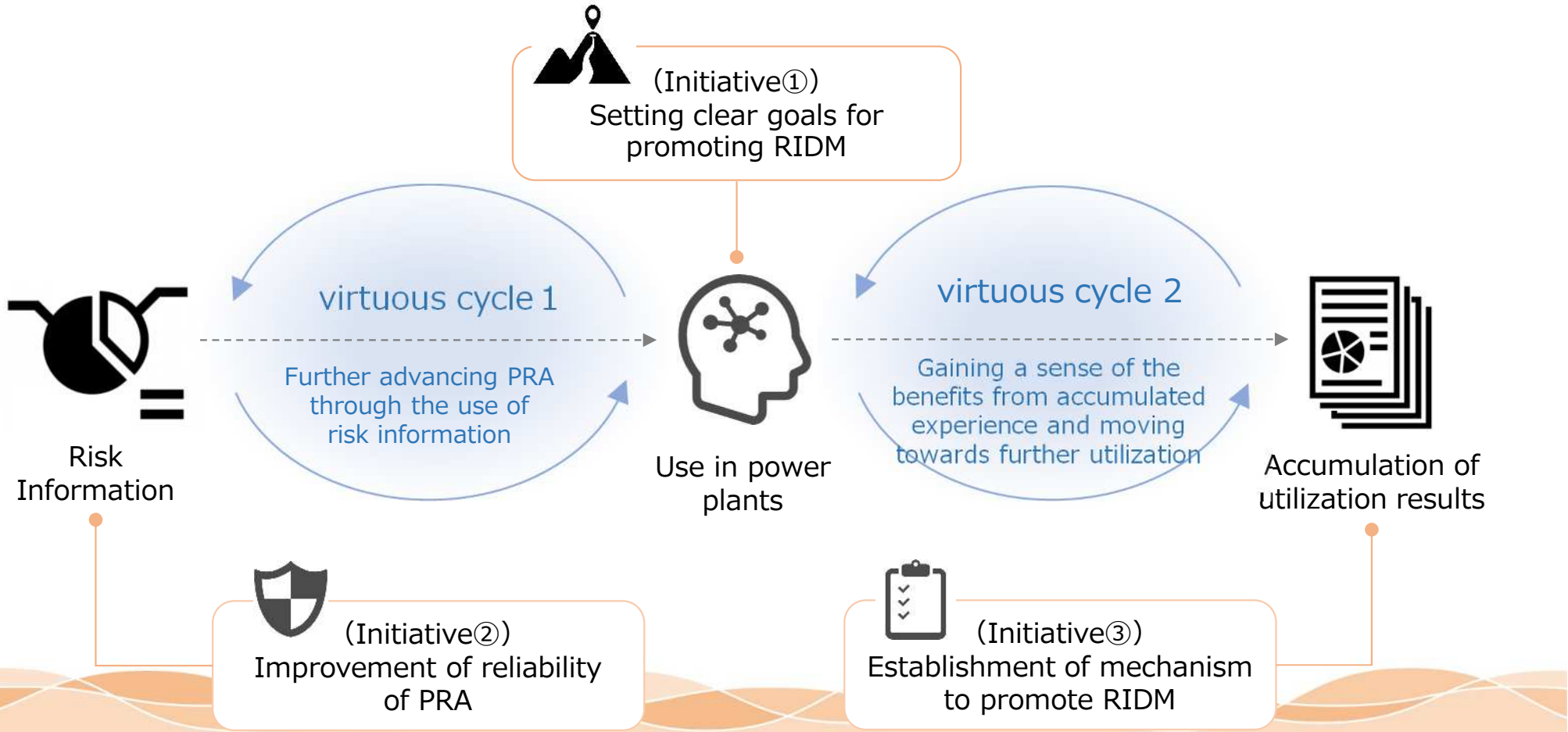
- ○ As a result of taking compensation measures, ΔCDF was reduced to approximately 1/6.
  In addition, as a result of shortening the term with safety as the top priority, ICCDP was reduced to approximately 1/8.
- ○ Although it is believed that the risk has been sufficiently reduced, if specific target values existed for CDF and ICCDP, a more objective judgment would be possible.



ICCDP : incremental conditional core damage probability
Here, the calculation is made by multiplying the ΔCDF associated with the isolation of the standby seawater pump by the actual isolation period (50 days before compensation measures).

Future initiatives

- ○ Risk information is already being used for individual projects, but its use has remained ad hoc, so we have taken recent steps with the idea of first setting performance goals for our company and then improving through systematic approach at each power station.
- ○ As we accumulate experience in using risk information, we hope that each staff will truly feel "improved safety and efficiency," which will lead to further use of the information. We also expect that this will provide insights that should be reflected in the PRA model, leading to further advances in the PRA. We are now in the phase of creating this kind of virtuous cycle.



(Initiative①) Performance goals of Kansai Electric Power

- ○ In December 2003, the Nuclear Safety Commission compiled an interim set of safety goals and performance goals, and we have made the performance goal our internal rules in June 2024.
- ○ We will continue to take steps to establish a mindset that further promotes RIDM and improves safety by managing resources effectively and efficiently, and we will continue to communicate this message both inside and outside the company.

(Performance goals)

They are the target level of risk control at reactor facilities, stipulated using the occurrence frequency of accidents such as core damage, and are supplementary goals to determine conformity to safety goals.

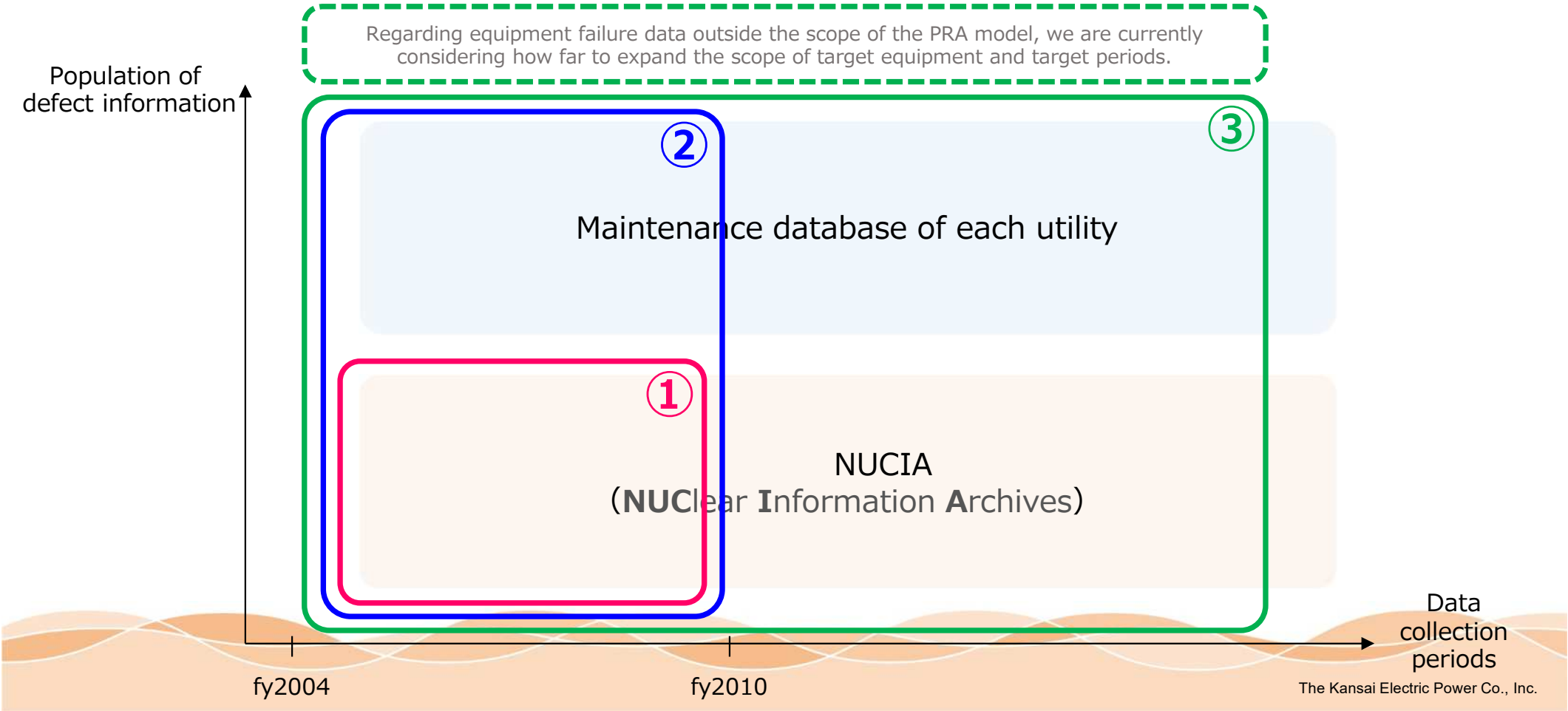
In the “Performance Goals for Light Water Power Reactor Facilities - Performance Goals Addressing Draft Safety Goals - (March 2006)”, the Nuclear Safety Commission of Japan has compiled the goals of core damage frequency (hereinafter “CDF”) and containment failure frequency (hereinafter “CFF”), which are indicators of PRA, as indicated in a. and b. Thus, they are set as performance goals in the Safety Management Guideline.

As safety goals from the perspective of environmental impact of (1)c. are goals addressing uncontrolled release of radioactive materials from the containment vessel, numerical goals equivalent to safety goals are set as performance goals.

a. CDF: Around 10^{-4} per year
b. CFF: Around 10^{-5} per year
c. Occurrence frequency of accidents with accidental release of Cs137 exceeding 100 TBq: Around 10^{-6} per year

(Initiative②) Improvement of reliability of PRA (Equipment failure data)

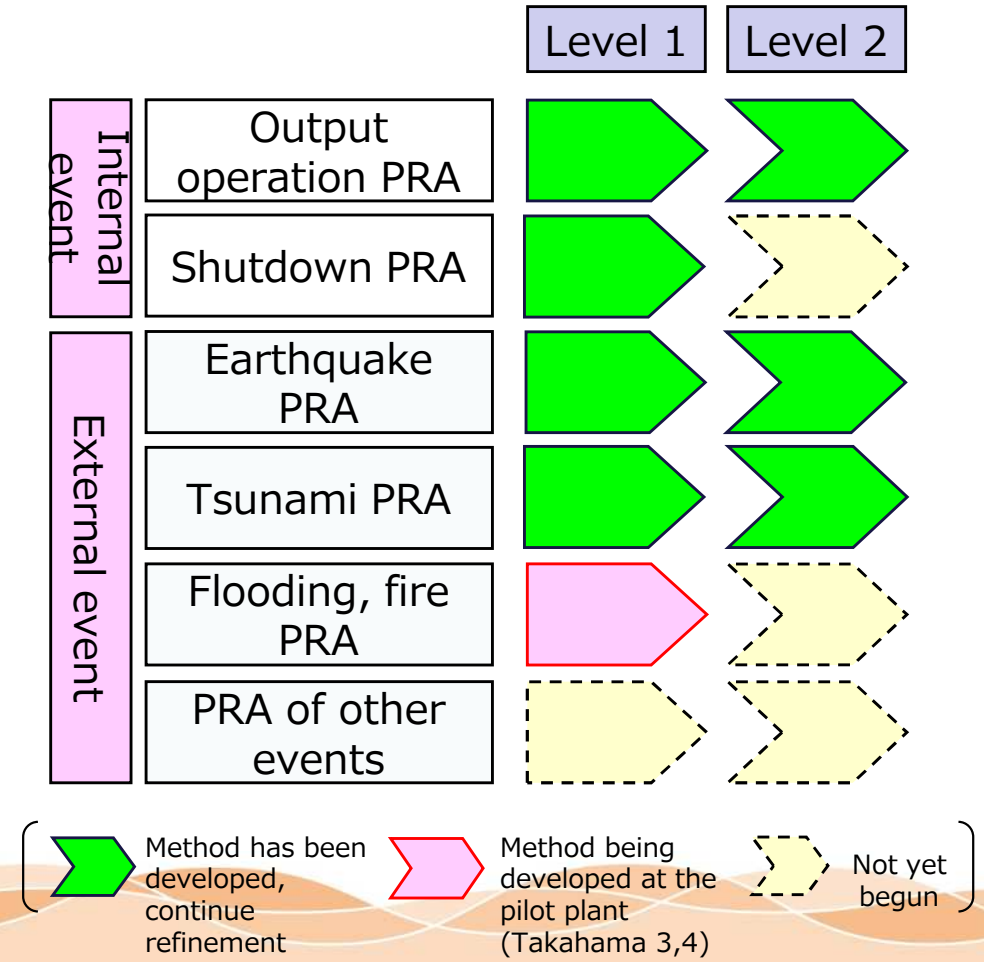
- ○ Regarding the collection and evaluation of equipment failure data, initially only NUCIA was used as the population (①), and regulatory authorities expressed concern that "the equipment failure data settings were more lenient than in the United States."
- ○ Currently, an NRRC guide has been published that sets out a data collection and evaluation process similar to that in the United States, and based on this, plant-specific malfunction information has been added to the population (②) and has been reflected in the PRA.
- ○ While accumulating RIDM results using equipment failure data that have undergone such improvements, we will continue to work on further improving the reliability of equipment failure data (③), such as by extending the period for collecting malfunction information and conducting U.S. expert reviews.

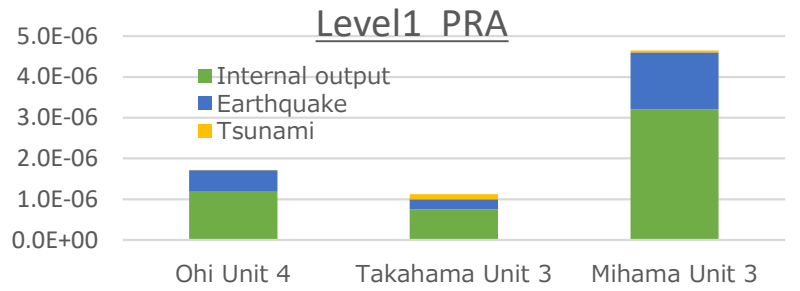


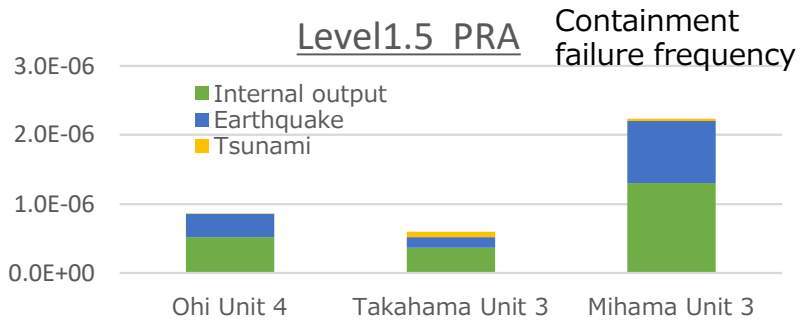
○ The latest knowledge are being reflected in the PRA assessment methodology, and the assessment methodology will continue to be improved.

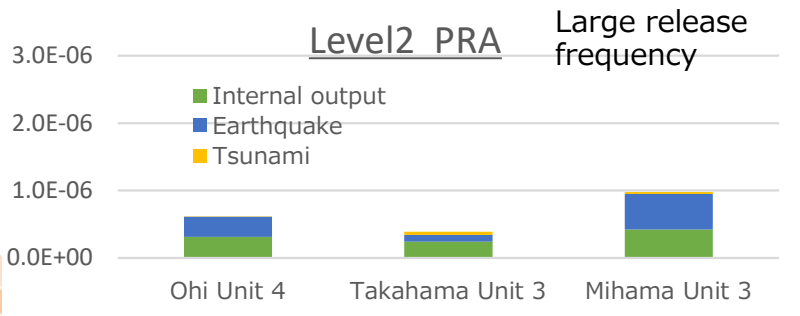
○ PRA has unincorporated risks (explicit risks of external events, and unknown risks), but in RIDM, it is important to evolve PRA while using it with the recognition of its incompleteness.

(Internal/external) Refinement and development of assessment methods





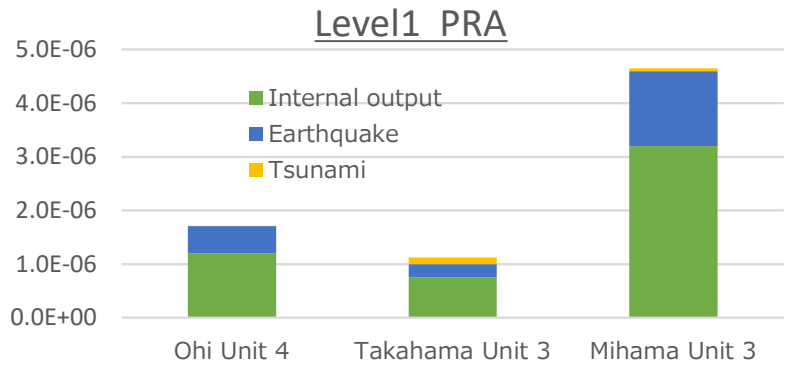




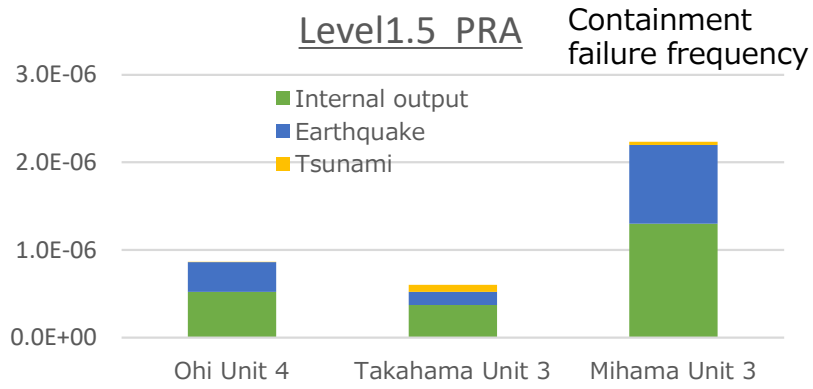
(Reference) PRA of Takahama Unit 4 / Takahama Unit 3 / Mihama Unit 3

Level 1 PRA (Core damage frequency: CDF)			
Plant	Ohi Unit 4	Takahama Unit 3	Mihama Unit 3
Internal output	1.2E-06	7.5E-07	3.2E-06
Internal shutdown	1.0E-06	6.9E-07	1.4E-06
Earthquake	5.1E-07	2.5E-07	1.4E-06
Tsunami	3.7E-09	1.2E-07	5.0E-08
Total*	1.7E-06	1.1E-06	4.7E-06

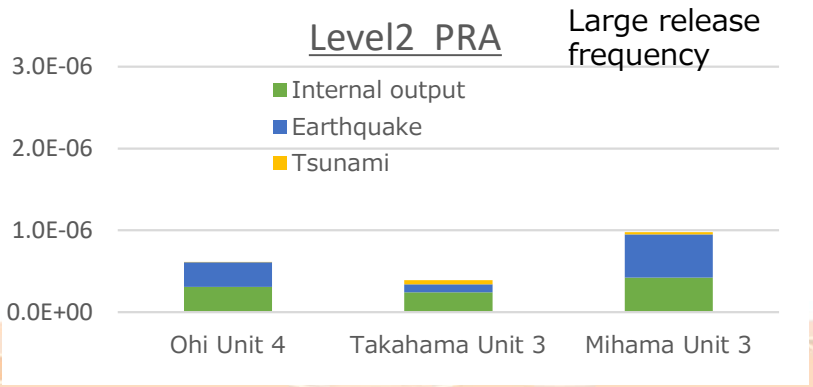
*Total value for internal output, earthquake and tsunami



Level 1.5 PRA (Containment failure frequency: CFF)			
Plant	Ohi Unit 4	Takahama Unit 3	Mihama Unit 3
Internal output	5.2E-07	3.7E-07	1.3E-06
Earthquake	3.4E-07	1.5E-07	9.0E-07
Tsunami	3.2E-09	8.1E-08	3.4E-08
Total	8.6E-07	6.0E-07	2.2E-06

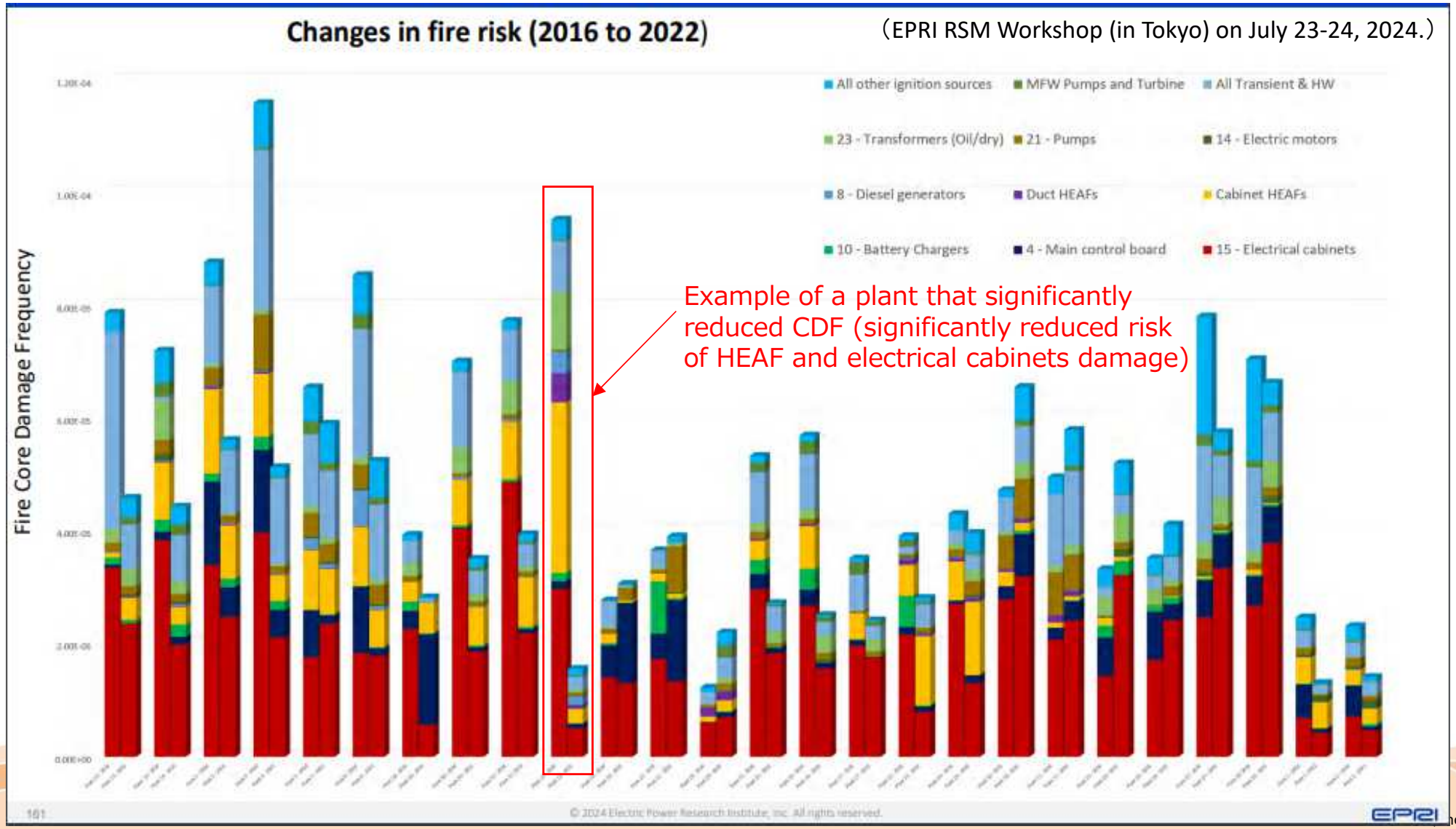


Level 2 PRA (Large release frequency: LRF)			
Plant	Ohi Unit 4	Takahama Unit 3	Mihama Unit 3
Internal output	3.1E-07	2.4E-07	4.2E-07
Earthquake	3.0E-07	1.0E-07	5.3E-07
Tsunami	3.1E-09	5.0E-08	2.6E-08
Total	6.1E-07	3.9E-07	9.8E-07

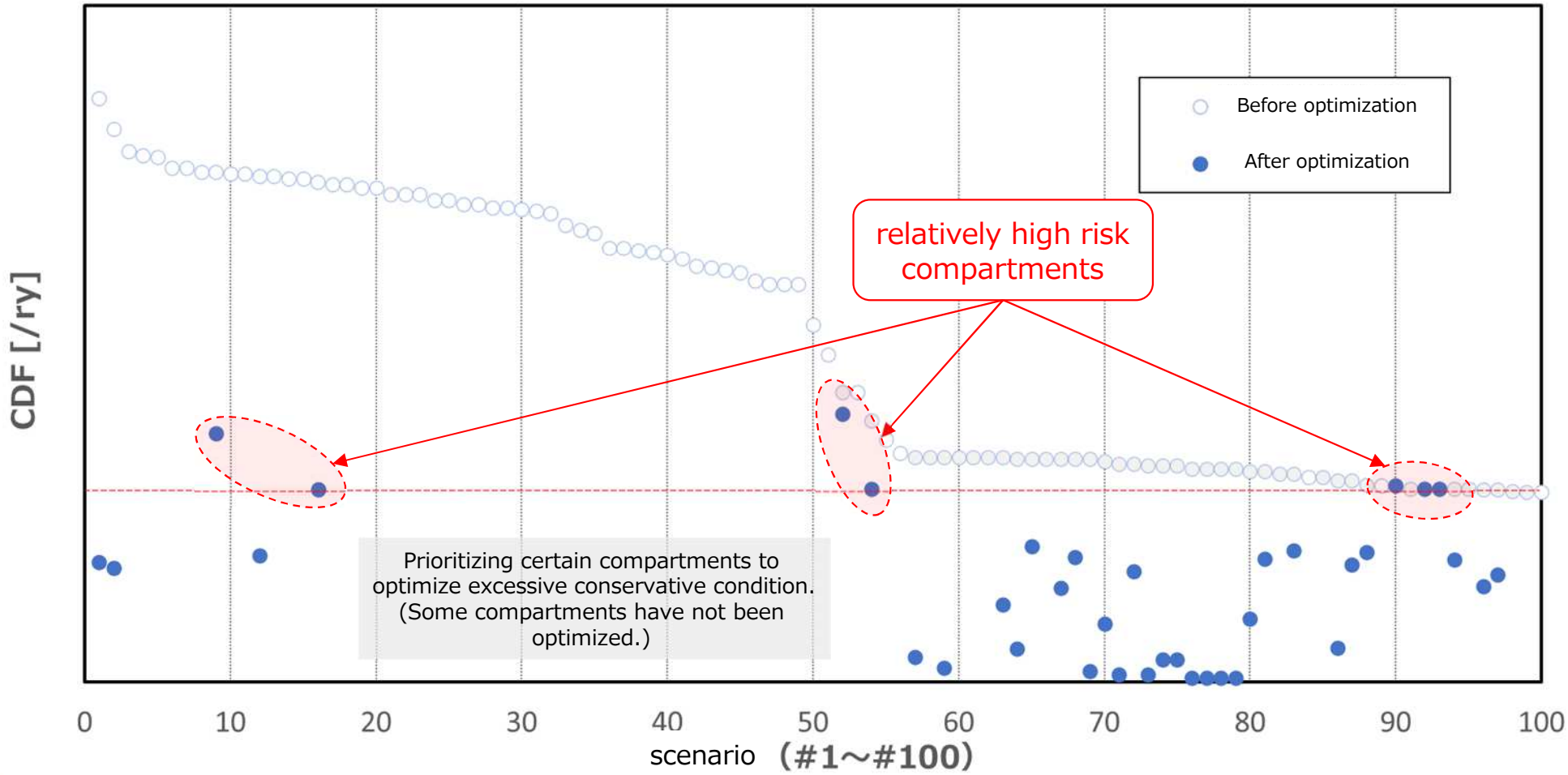


(Initiative②) Improvement of reliability of PRA (Fire PRA in the US)

- ○ Some US plants have significantly reduced the CDF of internal fire PRA in recent years.
- ○ From the perspective of continuing to use and evolve PRA, we are currently studying effective risk reduction measures, etc., utilizing the fire PRA currently under development and learning from US examples.



- ○ The fire PRA for the pilot plant has revealed that there are compartments that pose a relatively high risk.
- ○ From the perspective of continuing to use and evolve the PRA, the plan is to utilize the fire PRA currently under development and, based on the knowledge gained so far, to improve safety by setting prioritized inspection areas and taking effective risk reduction measures.



(Initiative②) Improvement of reliability of PRA (2/2)

- ○ Regarding the PRA model of PWR plants in Japan, US experts were invited to conduct level 1 (internal event) peer review at Ikata Unit 3 of Shikoku Electric Power as a representative plant, in a situation where peer review standards for the PRA model have not been established.
- ○ Comments from the peer review were deployed to plants of Kansai Electric Power to efficiently improve reliability.
- ○ We plan to conduct a peer review of our plant as well next year. Taking into account the similarities in models between PWR plants, we aim to conduct an effective review that focuses on the essential differences from the representative plant.

(November 2023)
Peer review requirements were confirmed through US benchmarks*

*US reviewer requirements stipulated in NEI17-07 and implementation of peer review when new methods are incorporated into the PRA model in the US (=not to be implemented periodically) were confirmed.

(October 2024)
Implementation standards for peer reviews in Japan are being prepared at NRRC.
After the implementation standards are established, peer review of plants of Kansai Electric Power is to be conducted (scheduled for FY2025 H2)

★**Peer review of plants of Kansai Electric Power** (scheduled for FY2025 H2)

★**Establishment of domestic peer review implementation standards**

★**Present**

(2017-2018 & 2022)
Peer review of the PRA model of level 1 internal events was conducted at Ikata Unit 3 of Shikoku Electric Power as a representative PWR plant in Japan

Peer review implementation standards have not been established in Japan, and US experts were invited to conduct the peer review with an all-Japan framework.
Comments are incorporated into plants of Kansai Electric Power.

(Record of restart)
2023.9 Takahama Unit 2
2023.7 Takahama Unit 1
2021.7 Mihama Unit 3
2018.5 Ohi Unit 4
2018.3 Ohi Unit 3
2016.2 Takahama Unit 4
2016.1 Takahama Unit 3

PRA model is developed in parallel with response to restart (internal event, earthquake, tsunami)

(Initiative③) Establishment of mechanism to promote RIDM (1/3)

○ From the perspective of encouraging RIDM, we are considering to introduce an assessment check sheet so that each individual can naturally consider RIDM in their daily safety activities.

○ The assessment check sheet will cover not only PRA but also deterministic viewpoints, and will be a truly user-friendly system for power plant personnel.

Flow of RIDM (NRRC "On-line Maintenance Guideline")

Selection of maintenance work, etc.

Organization of information such as subject equipment, work period, etc.

Initial screening with PRA

CDF<10^{-4} & CFF<10^{-5}

No → Cannot be performed

Yes

Assessment of the 7 key elements of RIDM

1. Standards and good practices
2. Operational experience
3. Deterministic considerations
4. Probabilistic considerations (PRA)
5. Personnel and organizational considerations
6. Considerations of interface with nuclear security
7. Other consideration (assumption of defects, development of response policies, etc.)

RIDM through risk review meeting, etc.

Determine whether to perform RIDM

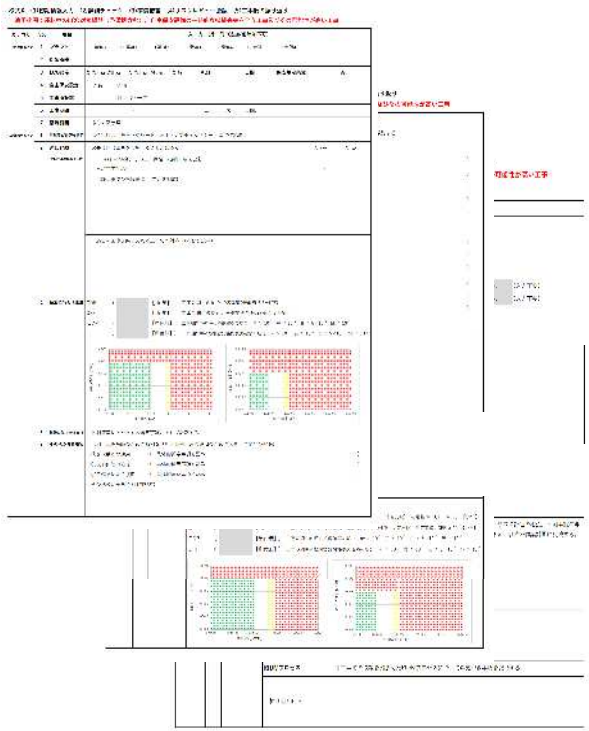
No → Cannot be performed

Yes

Execution and management

To execution and management phase

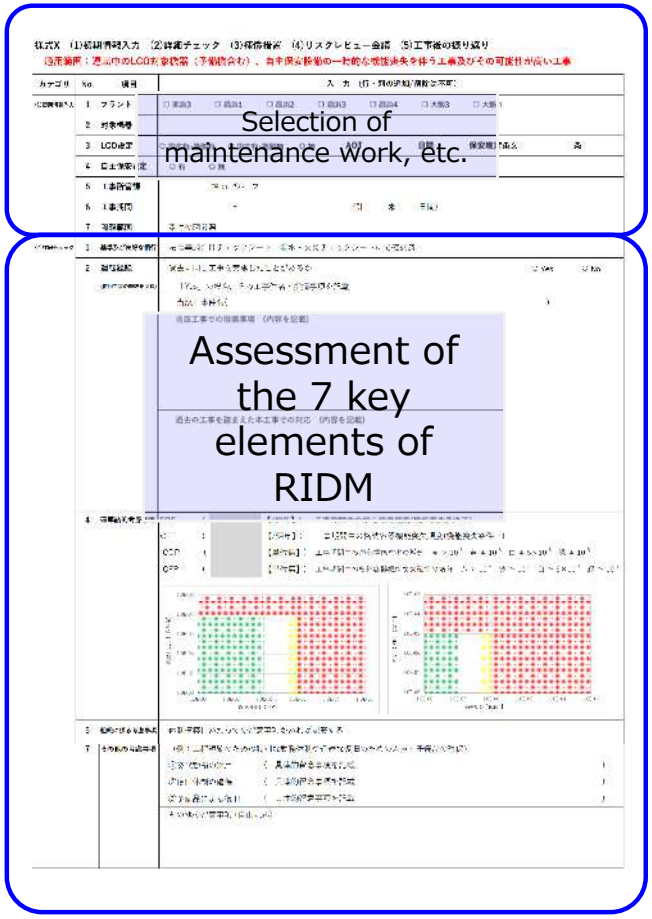
Assessment check sheet (under review)

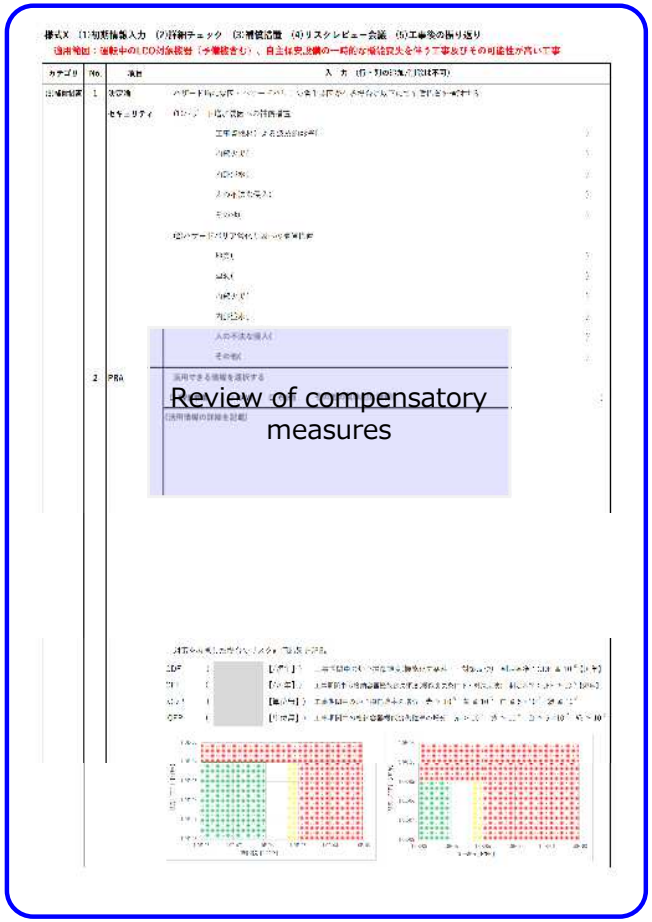


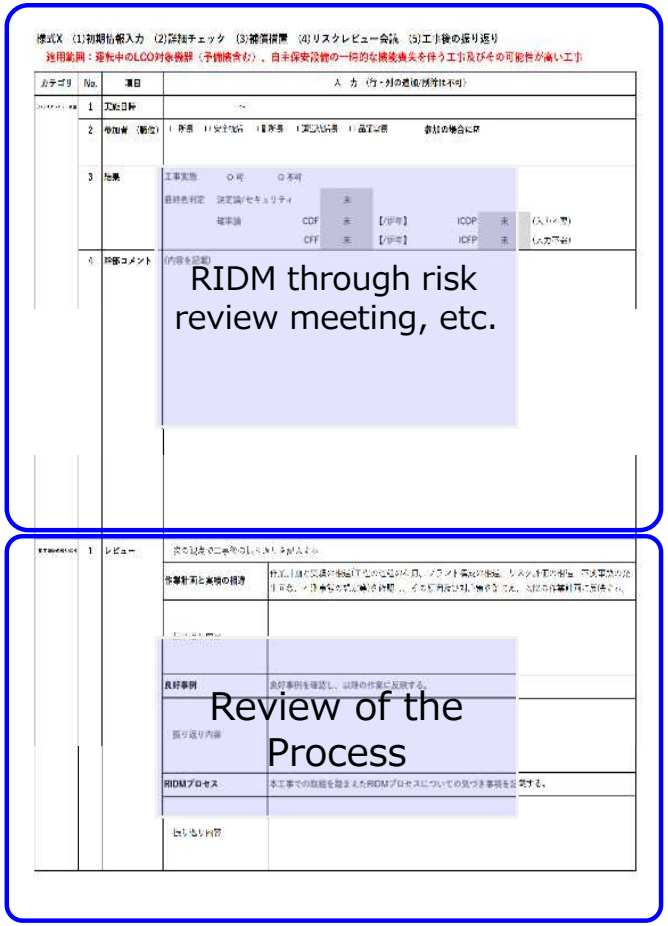
See next page for detailed composition

(Initiative③) Establishment of mechanism to promote RIDM (1/3)

○ The composition of the assessment check sheet is being reviewed so that necessary information is input along the flow of RIDM in order to be consistent with the approach of NRRC's On-line Maintenance Guideline.







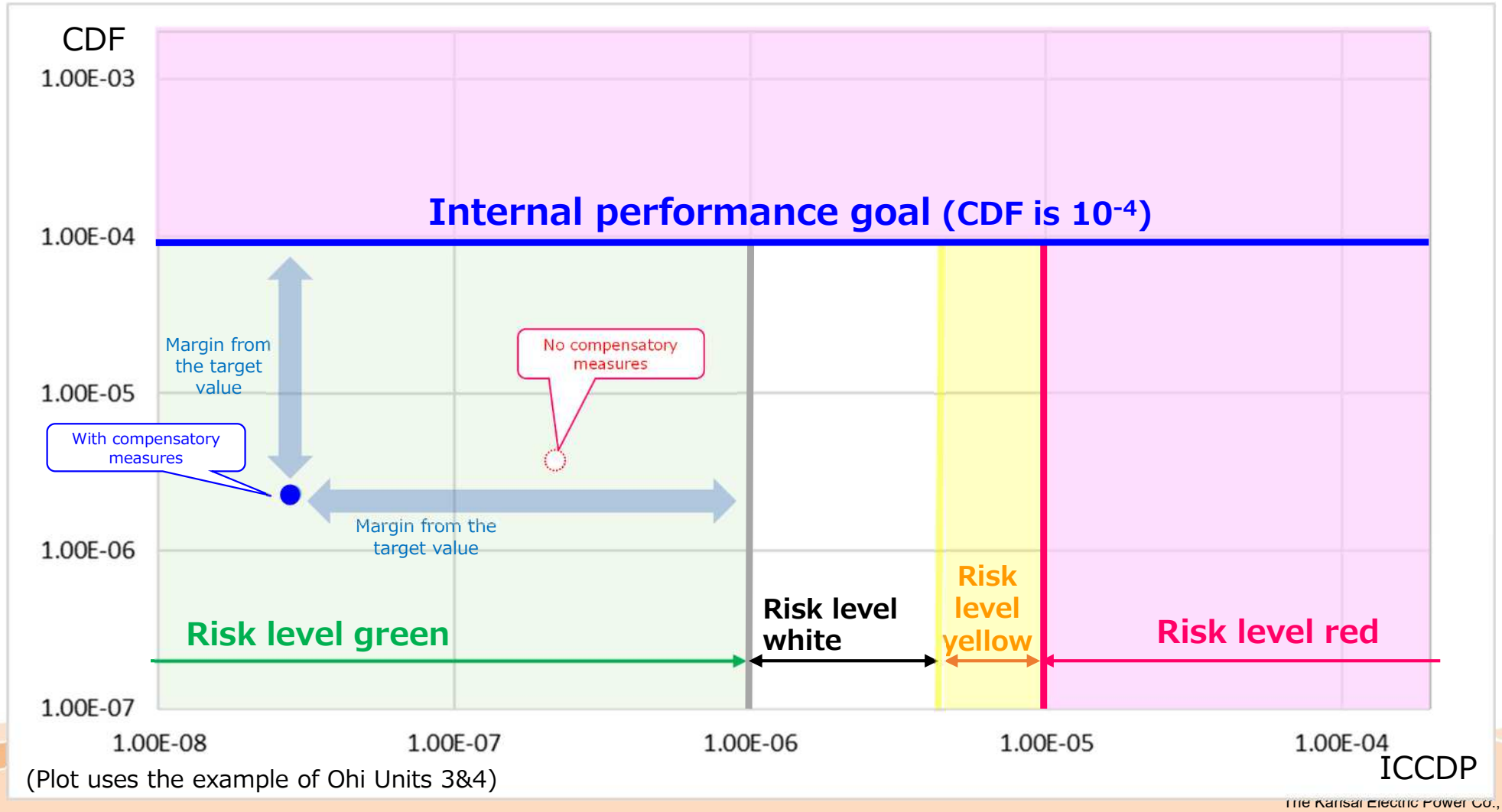
(Initiative③) Establishment of mechanism to promote RIDM (1/3)

- ○ The assessment check sheet follows the NRRC's OLM guidelines and sets quantitative judgment criteria and risk levels according to the results of internal events. For external events, risk levels are set according to the factors that increase hazards and the deterioration of hazard barriers.
- ○ The risk levels are coded in four levels, from green to red, so that even those who are not familiar with PRA can correctly recognize them.
- ○ A graded approach is also aimed at compensation measures, taking into account the risk levels.

Risk level	Definition	Internal event*		External event
		CDF	ICCDP	Hazard increasing factors and hazard barrier deterioration
Red	Level not to conduct work	$\geqq 10^{-4}$ (Performance goals)	$>10^{-5}$	There are hazard increasing factors and hazard barrier deterioration that affect standby SSC with safety function, and effective risk management measures cannot be set
Yellow	Level to conduct work upon taking risk management measures to compensate for functions	–	$\leqq 10^{-5}$	There are hazard increasing factors and hazard barrier deterioration that affect standby SSC with safety function, but the same level of risk as it was assumed in design can be maintained by implementing risk management measures
White	Level to conduct work upon taking risk management measures	–	$\leqq 5\times 10^{-6}$	–
Green	Level to perform risk management in accordance with normal work management	–	$\leqq 10^{-6}$	There are no hazard increasing factors and hazard barrier deterioration that could affect standby SSC with safety function

*CFF and ICCFP are managed one order of magnitude lower

- ○ For the case of “Ohi Units 3&4 Isolation work for spare seawater pumps”, CDF and ICCDP were mapped and margin from the target value was visualized, which reconfirmed that the compensatory measures implemented at the time were appropriate.
- ○ This case utilized FV importance based on internal event PRA, but it is intended to actively utilize PRA also for external events to the extent possible, even if the PRA is still in the process of development, to identify important scenarios as supplementary information for qualitative assessment.



(Plot uses the example of Ohi Units 3&4)

Conclusion

○ While recognizing the imperfections of PRA, it is important to continue using PRA while evolving it and to use it to improve safety and efficiency through more sophisticated Risk-Informed decision-making.

○ Toward the realization of truly effective use of risk information, operators intend to not only improve the reliability of PRA models and sophisticate evaluation methods through peer reviews, but also to carefully build up a track record of using risk information.

○ In Japan, the new nuclear regulatory inspection system was introduced in 2020. We would like to have our initiatives confirmed under this inspection system, and to continue close communication with regulatory authorities to improve common issues.

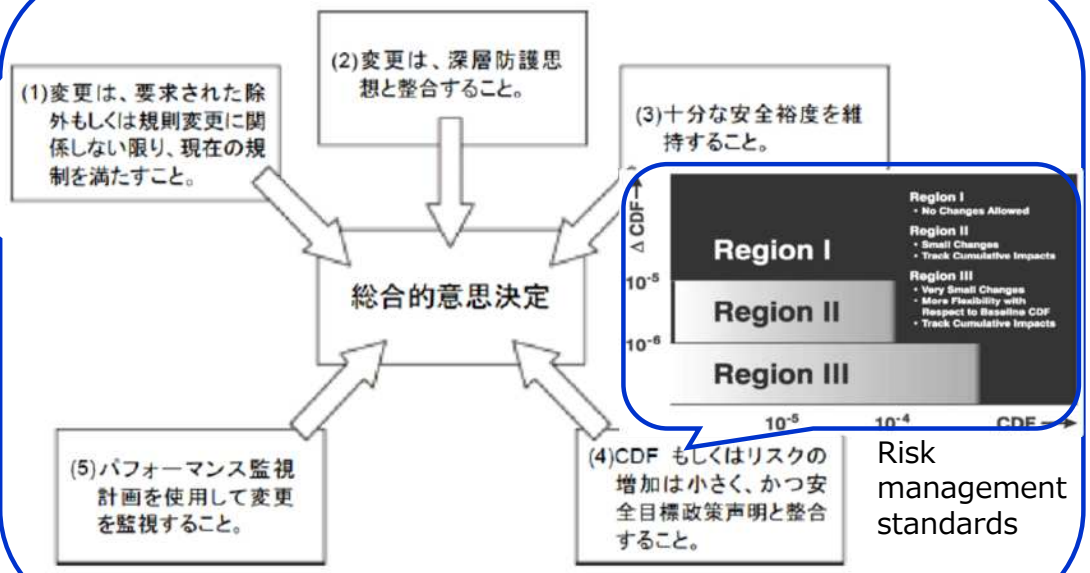
Reference

Utilization of risks in the US

・The regulatory authority stipulates major principles on change of license using risk information.
・Comprehensive assessment* is performed from 5 perspectives.

* Decisions cannot be made only with risk information.
The operator cannot freely make changes at their discretion just because CDF (absolute value) satisfies the performance goal.
(Note) There were plants where performance goals were exceeded.

・The regulatory authority determines whether backfit is required using ΔCDF (relative value)①. Additionally, the regulatory authority approves the application of the operator using ΔCDF (relative value)②.



Risk management standards

PRA Policy Statement in August 1995

① ②

①Example where the regulatory authority determines whether backfit is required using ΔCDF (relative value)

The regulatory authority performs comprehensive assessment from 5 perspectives including ΔCDF (relative value) and cost.

1. Example where backfit is not required
Backfit~~it~~ is not required for filter vent of Mark I/II BWR, when uncertainty of cost-benefit assessment is large and safety can be enhanced with other measures.

2. Example where backfit is required
Backfit is justified for ATWS, SBO, etc. based on cost-benefit assessment.

② Example where ΔCDF (relative value) is within a certain range in the application of the operator, and the regulatory authority has approved it

1. On-line maintenance
Maintenance inspection of multiple systems is conducted during plant operation, in consideration of ΔCDF. (in risk management standards, Region III: safeguards not required, Region II: safeguards required, Region I: on-line maintenance not allowed)
⇒Outage is shortened, and workers and cost are reduced.

2. Change of importance category utilizing risk information
Since it is important to safety, target equipment subject to special handling requirements (test, inspection, quality assurance, etc.) is changed based on risk information.
⇒Maintenance cost for equipment of low risk importance of those conventionally categorized as safety system equipment is reduced.