Enhancement of PRA at Ikata Unit 3
- Enhancement of PRA and Transition of CDF -

November 12, 2024

Shikoku Electric Power Company, Incorporated
Akihiko Kagawa

Contents

1. Introduction
2. Status of Ikata Nuclear Power Plant, and the organization of the Nuclear Power Division
3. Enhancement of PRA in the Ikata Unit 3 Project
    3.1 Initiatives taken in response to proposals from the TAC
    3.2 Initiatives taken in response to overseas specialists' reviews
4. Transition in CDF following enhancement of PRA
    4.1 Results of previously disclosed PRA regarding level one internal events
    4.2 Sensitivity analysis for PRA conducted after the establishment of AM
    4.3 Sensitivity analysis for PRA conducted as SAR
    4.4 Results of level one external event PRA and sensitivity analysis
    4.5 Insights regarding the transition of CDF, etc.
5. Future challenges
6. Summary

Reference: Examples of initiatives regarding RIDM

1. Introduction

- After conducting probabilistic risk assessment (PRA*[1]) in 1994 during the review to establish accident management, we have taken initiatives such as selecting significant accident sequence for periodic safety review and application for permission of reactor installment license, action for Ikata Unit 3 project, etc. to enhance PRA and in decision-making which utilizes risk information.
- As an activity with the goal of developing a more practical PRA (Good PRA*[2]), the Ikata Unit 3 project was initiated in January 2015 as the pilot plant for all domestic PWR operators with support from the Nuclear Risk Research Center (NRRC) and Technical Advisory Committee (TAC), etc.

Enhancement of PRA and application of RIDM in PRA at our company

▼Review establishment of accident management (1994)
▼Periodic safety review (2006-), shutdown risk management (2007-) , use in maintenance activities (2010-)
▼Application for permission of reactor installment license (selection of significant accident sequence, etc.) (July 2013)

▼ Disclosed future initiatives to autonomously improve safety for nuclear power (June 2014)

- Promote use of PRA in risk assessment (establishment of Nuclear Safety & Risk Assessment Group, expanding human resources, etc.)
- Strengthen risk management system (establishment of nuclear power safety risk management committee, etc.)

▼Initiated review of technical tasks for the Ikata Unit 3 project (PRA improvement activity) (January 2015)
▼Initiated review by overseas specialists (2017-)

▼Disclosed PRA model to NRA (October 2018-)
▼Submitted and disclosed 1st safety assessment report (SAR) (May 2019)
▼Applied to existing business process in stages (June 2019-)
▼Appropriateness of PRA model confirmed by NRA (March 2020)
▼Submitted and disclosed 2nd SAR (July 2022)
▼Submitted and disclosed 3rd SAR (December 2023)

*1: Initially referred to as probabilistic safety assessment (PSA)
*2 Defined as PRA which satisfies a level (state of practice) comparable with current international precedents in level 1 PRA and level 2 PRA

2. Status of Ikata Nuclear Power Plant, and the organization of the Nuclear Power Division

Overview of Ikata Nuclear Power Station

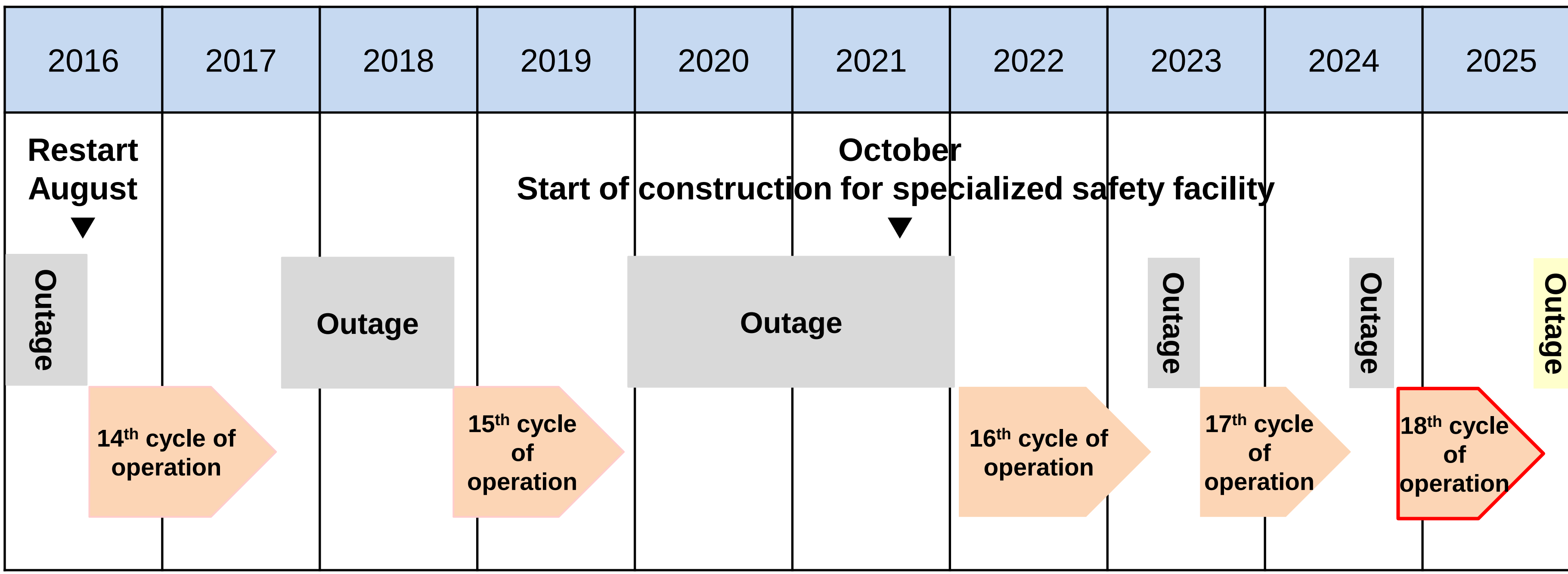
- Location: Ikata Town, Nishiuwa District, Ehime Prefecture

- Operational status and facility overview

	Unit 1 **(Decommissioning in progress)**	Unit 2 **(Decommissioning in progress)**	Unit 3 **(In operation)**
Rated power output	566MW	566MW	890MW
Reactor type	PWR	PWR	PWR
Start of commercial operation	September 30, 1977	March 19, 1982	December 15, 1994

Operational status of Ikata Nuclear Power Station

- In August 2016, Ikata Nuclear Power Station Unit 3 restarted operations after passing the screening to verify conformance with new regulatory requirements, and <u>is currently in the fifth cycle of operation after restart.</u>
- With the specialized safety facility coming online in October 2021, all tangible/intangible preparations to conform with new regulatory requirements have been completed.



Organization of the Nuclear Power Division

- The organization chart for the Nuclear Power Division is as follows. Initiatives regarding PRA and RIDM is being taken for groups and sections with red frame.

Nuclear Power Division

Nuclear Power Division [Matsuyama City, Ehime Prefecture]
- Strategic Administration Group
- Management Group
- Research Group
- Energy Public Relations Group
- Nuclear Research & Training Center
  - General Affairs & Public Relations Group
  - Nuclear Safety & Risk Assessment Group
  - Operator Training Group
  - Maintenance Training Group
  - Maintenance Technology Improvement Group

Head Office [Takamatsu City, Kagawa Prefecture]
- Nuclear Power Department
  - Nuclear Power Planning Group
  - Operation Management Department General Manager
    - General Affairs Group
    - Operating Management Group
    - Maintenance Management Group
    - Nuclear Safety Group
    - Aseismic Designing Group
    - Physical Protection & Construction Group
  - Nuclear Fuel Cycle Department General Manager
    - Planning & Back-End Contract Group
    - Fuel Purchasing Group
    - Fuel Cycle Engineering Group
    - Fuel Engineering Group
    - Decommissioning Group

Ikata Nuclear Power Station [Ikata Town, Nishiuwa District, Ehime Prefecture]
- Ikata Nuclear Power Station
  - Licensed Reactor Engineer
  - Decommissioning Office
    - Decommissioning Section
  - Safety Management Department
    - Safety & Engineering Section
    - Nuclear Fuel Section
    - Radiation & Chemistry Control Section
    - Physical Protection Section
  - General Affairs & Public Relations Department
    - General Affairs Section
    - Public Relations Section
  - Power Generation Department
    - Operation Section
  - Civil & Architectural Engineering Department
    - Civil & Architectural Engineering, Maintenance Section
    - Civil & Architectural Engineering, Construction Section
  - Maintenance Department
    - Maintenance Section
    - Primary Systems Mechanical Maintenance Section
    - Secondary Systems Mechanical Maintenance Section
    - Electrical Maintenance Section
    - Instrumentation and Control Maintenance Section
    - Equipment Improvement Section
  - Quality Assurance Department
    - Quality Assurance Section
    - Safety Inspection Section
    - Periodic Inspection Section
    - Personnel Training Management Section
    - Process Management Section

[Key division of roles regarding PRA and RIDM]
- Nuclear Safety Group: administration of risk management
- Nuclear Safety & Risk Assessment Group: Conduct of PRA, promotion of its use
- Process Control Section: Promotion of RIDM in the field

YONDEN

3. Enhancement of PRA in the Ikata Unit 3 Project

- After receiving proposals based on the results of the first and second TAC meeting, initiatives were taken regarding **the five technical tasks** below.
- Also, **the results of initiatives above were applied to the PRA model as necessary** and submitted to the Nuclear Regulation Authority through the safety improvement evaluation, etc. and **details were disclosed on the company website**.

Item	Overview of proposal
① **Enhancement of PRA event-tree**	General with small numbers of initiating event. Also, assumed initiating event and scenarios unique to the plant are inadequate. It is crucial to select comprehensively, including plant specific initiating events.
② **Enhancement of PRA parameter**	Operating experience unique to the power station (plant specific data) should be applied to initiating event frequency, component failure rate, component unavailability rate.
③ **Enhancement human reliability analysis**	The THERP method is outdated for evaluating human performance in complex event evolutions. The latest model used in the U.S. should be implemented.
④ **Enhancement of seismic hazard evaluation**	Should be conducted using strict procedures stipulated by U.S. specialists (process above SSHAC level 3)
⑤ **Enhancement of seismic fragility evaluation**	Evaluation methods being developed at the NRRC should be considered for application.

➢ Initiatives for the five technical tasks are as follows.

①Enhancement of PRA event-tree

- Regarding initiating events, FMEA* was used to add additional initiating events, considering plant configuration specific to Ikata Unit 3, to the 12 events in the application for permission of reactor installment license, and 44 events were selected, establishing the ET.
- In the first safety improvement evaluation, the enhanced ET was applied as a base case.

*: FMEA: Failure Mode and Effect Analysis

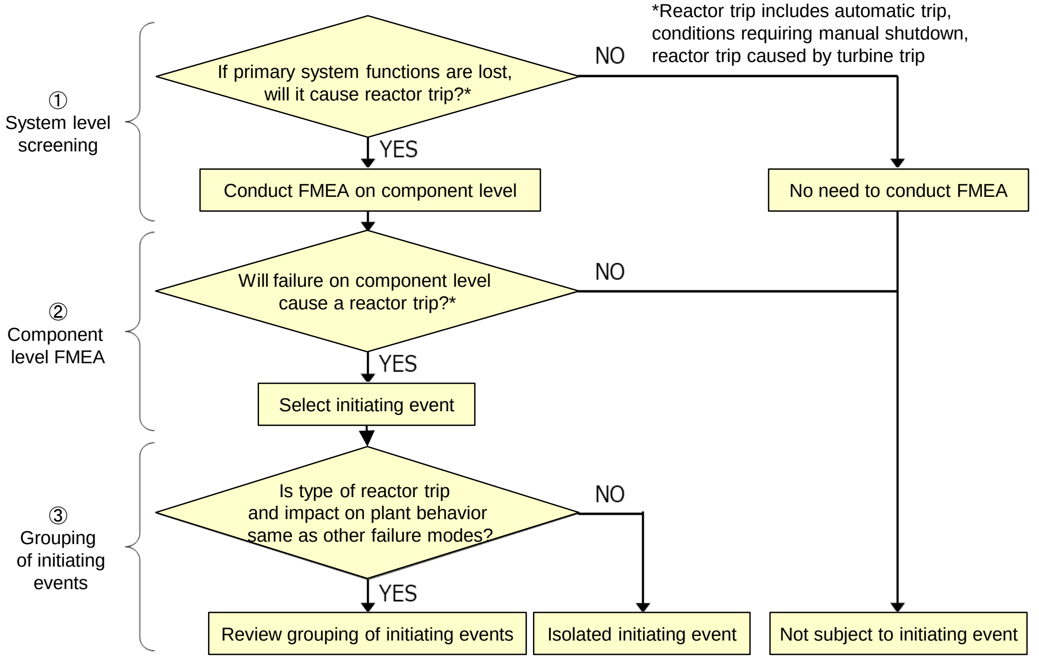


Fig. The implementation Flow of FMEA

② Enhancement of PRA parameter

- In order to utilize plant-specific data such as the number of component failures and operation time, EAM (component maintenance information database) and operation logs were surveyed, <u>data on the number of component failures and operation time between FY2004-FY2010 were collected and analyzed, and these were applied in the sensitivity analysis for the first safety improvement evaluation.</u>
- Regarding the domestic general component failure rate data* newly established by the NRRC, <u>data for failure rate at Ikata Unit 3 after FY2011 were applied, and also applied as a base case for internal event PRA for the third safety improvement evaluation.</u>

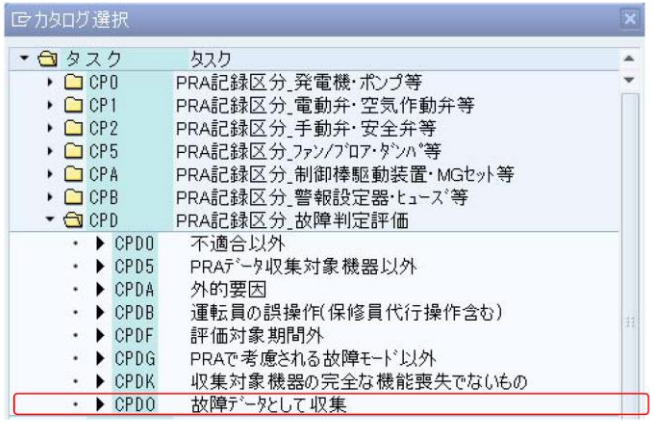


Fig. Identify maintenance requests from EAM, and collect component failure data

*: Established based on data of failure rates occurring in 27 domestic plants between FY2004 and FY2010.

③Enhancement of human reliability analysis

- The HRA Calculator, a human reliability analysis tool widely used in the U.S., was implemented.
- Also, input parameters for the HRA Calculator was prepared through analysis of procedures and interview with operators while referring to NRRC's Guideline Regarding Human Reliability Analysis.
- In the first safety improvement evaluation, PRA sensitivity analysis was conducted for internal event level one during power output which applied HRA Calculator based human reliability evaluation results; also, results were applied to the internal event PRA as a base case in the 3rd safety assessment report (SAR).
- In the future, internal events during earthquakes, tsunamis, and shutdowns will be applied in PRA in the 4th SAR.

Fig. Interview with operators

Table. Example of interview sheet

個別操作（事象発生前）に対する質問				
起因事象発生前人的過誤事象（手動弁の操作）の記載内容を確認した上で、A）モデル化情報に記載の状況との想定で回答願います。				
操作失敗に係る質問				
No	質問項目	回答欄		備考
現場での操作性				
事故前 1-1	習熟度や体力に係らず、操作に工具（ウィルキー、ユニハンドラー等）が必要である。	☐ YES	☐ NO	
事故前 1-2	【↑の質問YESを選択した時のみ】 操作を実施するのに工具は十分な数が備わっている。	☐ YES	☐ NO	
事故前 1-3	操作に部品（着脱式のハンドルや、ガスケット等）が必要である。	☐ YES	☐ NO	
事故前 1-4	【↑の質問YESを選択した時のみ】 操作を実施するのに部品は十分な数が備わっている。	☐ YES	☐ NO	
体制				
事故前 2-1	手順書にチェック欄があり、チェックが実施されるか。 （注）仮に手順書にチェック欄が無かったとしてもチェックが実施されるのであればYESをチェック。 逆に、チェック欄が有ったとしてもチェックがされないのであればNOをチェック。	☐ YES	☐ NO	
事故前 2-2	運転員、作業員は操作についてトレーニングを受けている。 （注）当該操作に限ったトレーニングでなく、類似のトレーニングでもよい	☐ YES	☐ NO	
事故前 2-3	操作が適切に行われたことを確認する指示が手順書に記載されている。 （注）一般注意事項のような記載でも良い。	☐ YES	☐ NO	
事故前 2-4	操作が適切に実施されていることを、操作担当者以外の人物が直接確認を行う。 （注）直接確認を行うのであれば、確認のタイミングに指定は無い。直接確認を行わない（無線・電話により遠隔地から口頭で確認する等）場合はNOを回答する。	☐ YES	☐ NO	

④Enhancement of seismic hazard evaluation

- In March 2016, for the purpose of enhancing the probabilistic seismic hazard analysis (PSHA), a project*1 was initiated for Ikata Unit 3 to apply the U.S. SSHAC*2 guideline level 3, which stipulates PSHA evaluation procedures at nuclear facilities.
- While the guideline is widely implemented overseas as part of regulatory requirements for nuclear power facilities, <u>subject implementation was the first in Japan.</u>

*1: Ikata SSHAC project
*2: Senior Seismic Hazard Analysis Committee

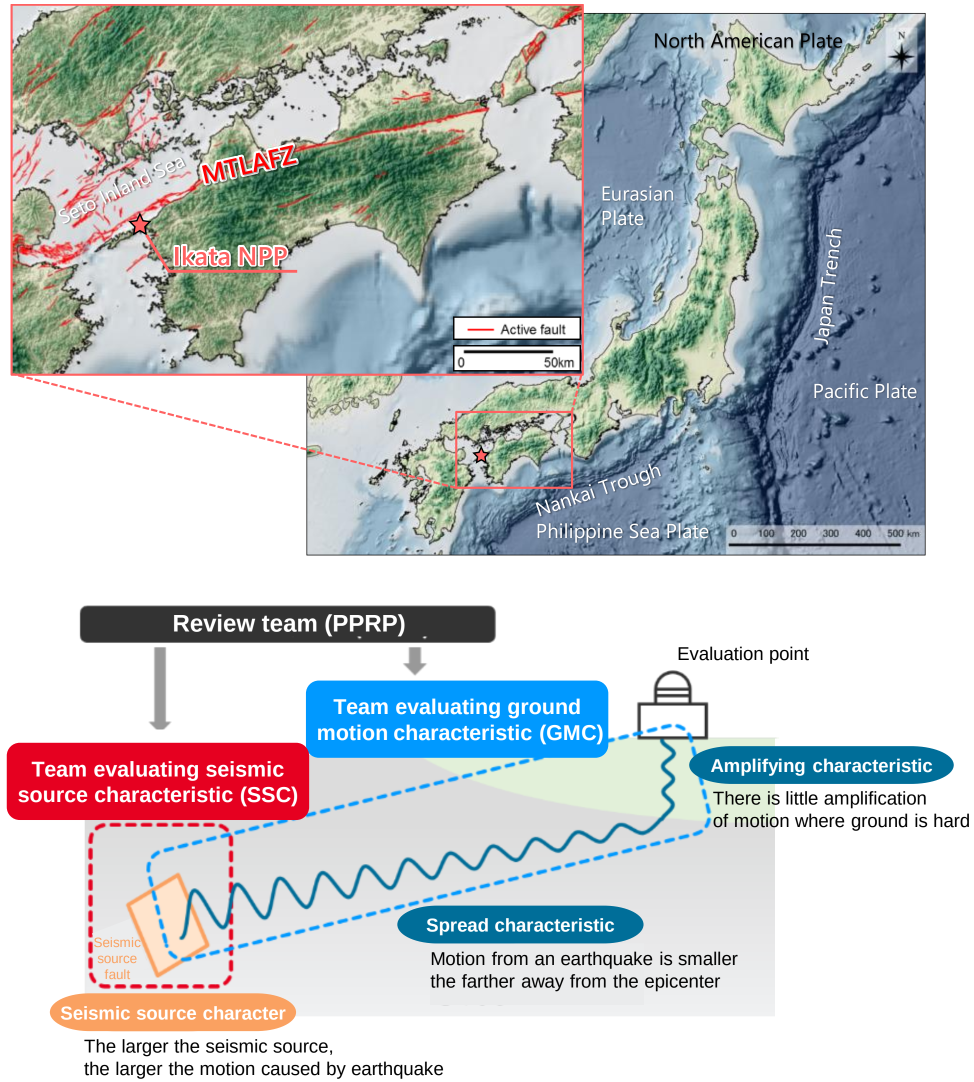


Fig. Fault belt zone around Ikata Nuclear Power Plant and overview of SSHAC project

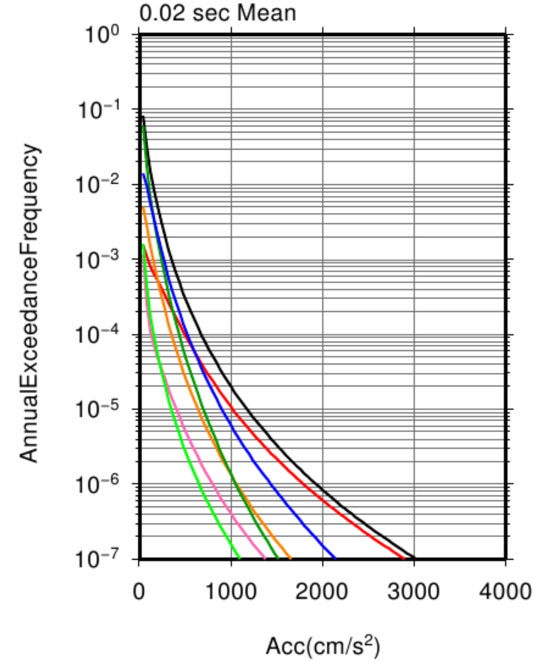
④**Enhancement of earthquake hazard evaluation** (continued)

- The Ikata SSHAC project was completed in October 2020 after approx. 4.5 years of discussion. In November of the same year, the Ikata SSHAC Project Final Report was disclosed on the company website*.

- A sensitivity analysis which applied the results from the 2nd SAR was conducted. We planned to apply this as the base case for the seismic PRA in the 4th safety improvement evaluation.

⑤**Enhancement of seismic fragility evaluation**

- Reviews are currently in progress through nation-wide research to solve issues, and results shall be implemented if necessary.



[Horizontal direction]

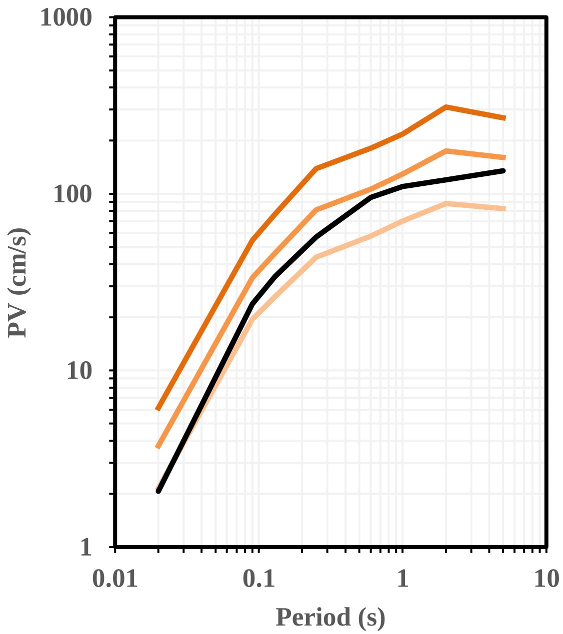


Fig. Comparison between hazard curve by seismic source and design basis seismic motion Ss-1

*: https://www.yonden.co.jp/energy/atom/safety/sshac_project/index.html

- The Ikata Unit 3 project initially focused on response to TAC proposals, but in the 2015 4th meeting, the original role of TAC (technical advisory to R&D of NRRC) was confirmed.
- As activities which replace TAC proposals, **reviews conducted by overseas specialists** have been conducted from 2017 with support from NRRC as listed in the table below.
- In reviews conducted by overseas specialists, **conformance to ASME/ANS PRA standards (Category II) was confirmed.**

No.	Details	Period, duration
No. 1	Seismic level 1 and 1.5	February 2017
No. 2	Internal event level 1.5 during at-power operation	August 2017
No. 3	Internal event level 1 during at-power operation (first time)	February 2018
No. 4	Internal event level 1 during at-power operation (second time)	August 2018
No. 5	Internal event level 1 during shutdown	October-November 2019
No. 6	Follow-up for past reviews*	November-December 2020
No. 7	Confirm course of action for past comments*	December 2021
No. 8	Confirm status of response to past comment * (confirm to conclude action for findings)	December 2022
No. 9	Confirm status of response to past comment * (confirm to conclude action for findings)	December 2023

*Review and comment regarding 3rd and 4th online internal event level 1 PRA

3.2 Initiatives taken in response to reviews conducted by overseas specialists

➢ At the time of concluding the 9th overseas specialist review, the status of conformance regarding support requirements (SR) in accordance with ASME/ANS PRA standards and number of F&O* issued are as follows. * : Fact & Observation

Technical elements	SR	Status of conformance to SR				Status of response to F&O			
		○	△	×	Other*	○	△	×	Total
Initiating event (IE)	33	20	2	6	5(2)	11	8	3	22
Accident sequence (AS)	21	14	1	4	2(1)	5	7	12	24
Success criteria (SC)	16	7	1	4	4(3)	2	5	2	9
System analysis (SY)	41	31	2	5	3(2)	7	5	9	21
Human reliability analysis (HR)	38	26	0	7	5(5)	3	4	6	13
Data analysis (DA)	33	23	2	5	3(1)	3	1	7	11
Quantitative analysis (QU)	33	23	0	9	1(1)	10	9	7	26
Total	215	144	8	40	23(15)	41	39	46	126

【Status of conformance to SR】
○: Conforms to category above performance category II
△: Performance category I
× : Does not conform to any category
∗: Not subject to review, numbers in () are subjects that have not yet been reviewed

[Response status to F&O]
○: closed(response complete)
△: partially closed(partially complete)
× : open(incomplete)

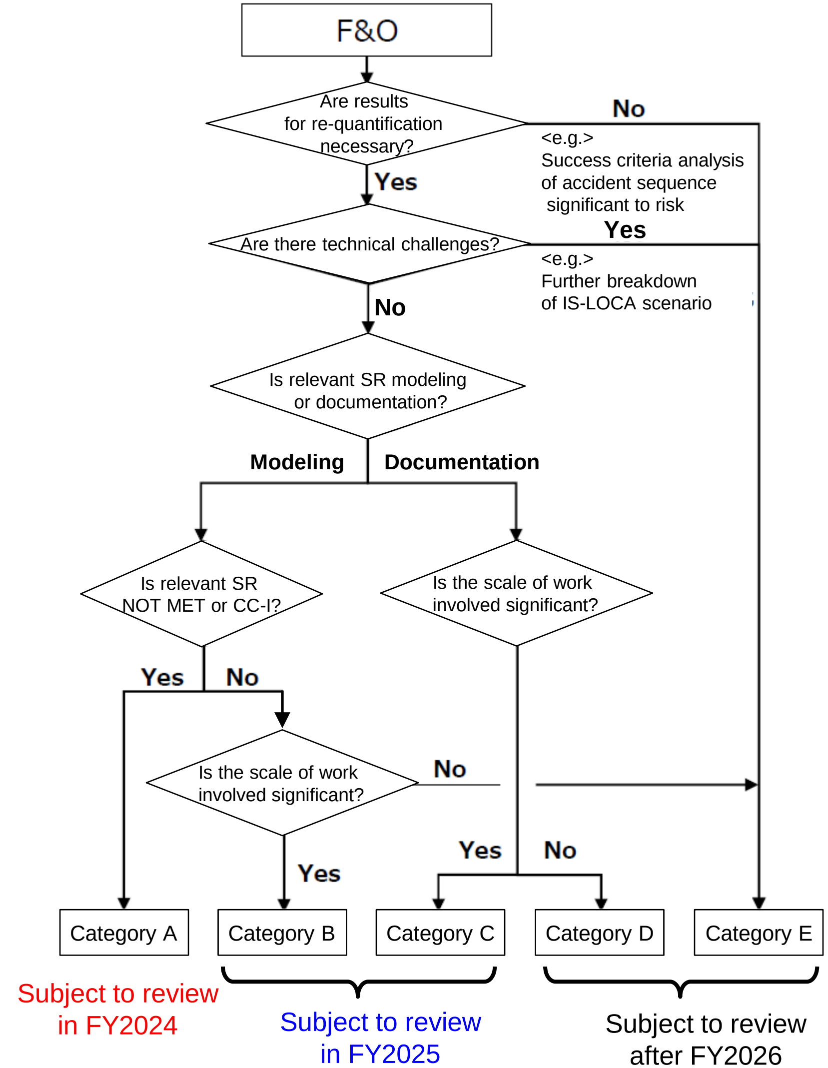
✓ Of the 192 cases excluding “others”, number of items conforming to SR are:
- 144 cases above MET and capability category II (75%)
- 8 cases were capability category I (4%)
- 40 cases were not met (21%)

✓ Regarding F&O, of the 126 cases,
- closed: 41 cases (33%)
- partially closed: 39 cases (31%)
- open: 46 cases (37%)

YONDEN

➢ **Of the unresolved 85 F&O cases**, streamline response by prioritizing F&O cases to be subjected to review.

- F&O cases requiring review that consider re-quantification and involve technical challenges shall be responded to after FY2026 (category E).
- Findings regarding modelling impacts the result of quantification, becoming a higher priority than documentation related F&O; therefore, subject cases shall be responded to FY2024 or FY2025 (category A, B)
- If relevant SR satisfies CC-II or above, the response shall be taken in FY2025 or FY2026, depending on the scale of work. (category B, E)
- F&O regarding documentation shall be responded to in FY2025 or FY2026 depending on the scale of work. (category C, D)



➢ Example of response to comments from review ① Modelling systems operating alternately

✓ Before enhancement of PRA

- Train involved in initiating events and systems running normally are asymmetrical.
  - ・Initiating events such as LOCA and SGTR always occur in a specific loop
  - ・Fix operating train of system under normal operation (periodic switching of trains not modelled)

[Comment from overseas specialist]

- Asymmetry in the model impacts risk significance of each component, and results in evaluation of risk significance being unrealistic. (AS, SY)

✓ After enhancement of PRA

- Train involved in initiating events and systems running normally are symmetrical.
  - ・Initiating events such as LOCA and SGTR occur in both loops
  - ・Models reflect actual operating status of normally operating systems.

- **Example of response to comments from review ② Improving frequency of secondary system rupture event occurrence**

✓ **Before enhancement of PRA**

- Bayesian updating was applied to U.S. NRC's data on initiating event occurrence frequency to calculate the frequency of secondary system rupture event*

*Main feedwater line rupture, main steam tube rupture (upstream of main steam isolation valve), main steam tube rupture (downstream of main steam isolation valve)

[Comment from overseas specialist]

- The database for U.S. NRC initiating event occurrence frequency database has not been confirmed to be conforming with ASME PRA standards
- EPRI report should be used as it assigns overflow frequency for each unit length in accordance with the scale of overflow of each system

✓ **After enhancement of PRA**

- Calculated secondary system rupture initiating event occurrence frequency using the EPRI report (in progress)

➢ Example of response to comments from review ③ Implementation of success criteria which uses realistic conditions

✓ Before enhancement of PRA

- Accident sequence and success criteria (number of component, margin time for operator manipulation) set based on licensing analysis conditions which reflect conservative bias

[Comment from overseas specialist]

- There are success criteria which do not stipulate the minimum number of component (one pump, one train, one valve)
- Success criteria for risk significant accident sequence should be set based on most probable conditions

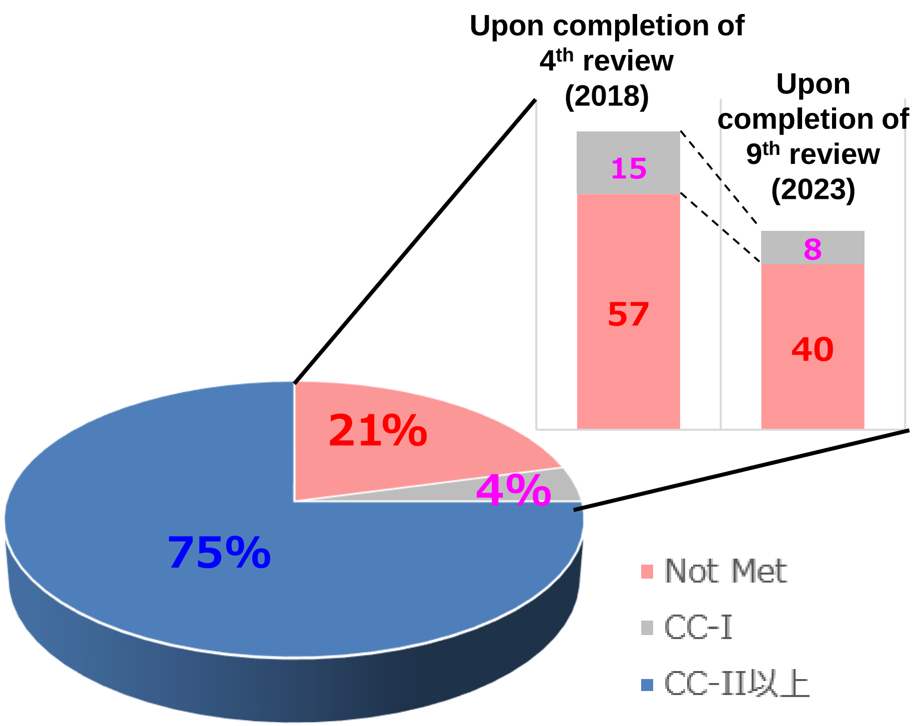
✓ After enhancement of PRA

- Risk significant accident sequence identified, and success criteria set based on analysis implementing most probable conditions (in progress)

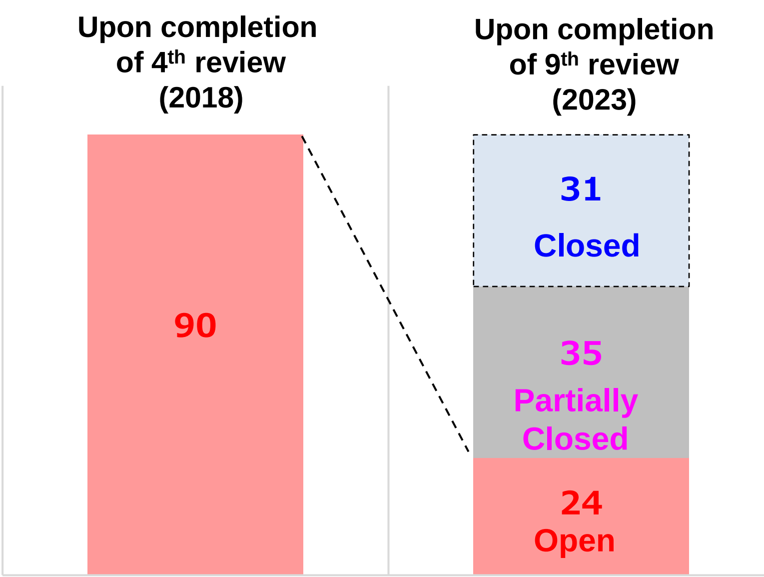
  E.g.) Number of low-pressure injection pumps during mid/small rupture LOCA + high-pressure injection failure: 1/2 units (before enhancement: 2/2)
  Margin time for manipulation of secondary system forced cooling during SBO/LUHS + RCP seal LOCA: 70min. (before enhancement: 30min.)

➢ Transition of conformance status regarding ASME/AMS PRA standard SR as review progresses

- ✓ SR conformance status: **Capability category (CC) II or above at 75%**
  By solving F&O, ”Not Met” and “CC-I” now conform with CC-II
- ✓ Of F&O, response status to findings*:
  Closed: 31 cases, Partially Closed: 35 cases, Open: 24 cases



SR conformance status
(excluding items not subject to review, items not yet reviewed)



Status of F&O (findings*)

*F&O includes recommendations and good practices, but only the number of findings are indicated here
Total number of recommendations and good practices are:
closed：41 cases (33%), partially closed: 39 cases (31%), open: 46 cases (37%)

4. Transition in CDF following enhancement of PRA

4.1 Results of previously disclosed PRA regarding level one internal events

- Regarding Ikata Unit 3, after establishment of accident management (AM), PRA has been conducted and results disclosed for periodic safety review (PSR), application for change of installation permit regarding inspection to verify conformance to new regulatory requirements and in submission for safety assessment report (SAR).

- CDF is as indicated in the table below, and regarding ② disclosed in 2004 and ⑦ disclosed in 2023, **CDF has increased by approx. 20 times despite additional measures being considered**, and the impact of difference in analysis conditions of each case shall be reviewed in the next page.

Items		PSA report after establishing AM (March 2004)		③ PSR report (September 2006)	④Application for permission of reactor installment license (July 2015)	③'PSR report (September 2016)	1st SAR (May 2019)		⑦ 3rd SAR (December 2023)
		①Prior to establishment of additional AM	②After establishing additional AM				⑤ No SA measures	⑥ SA measures taken	
CDF[/reactor/year]		2.9×10^{-7}	**1.5×10^{-7}**	1.4×10^{-7}	2.2×10^{-4}	1.4×10^{-7}	1.8×10^{-3}	1.8×10^{-6}	**2.8×10^{-6}**
Analysis condition	Initiating event occurrence frequency	-FY2002 data	←	←	-FY2010 data	-FY2002 data	-FY2015 data	←	-FY2021 data
	Component failure rate	U.S. data	←	←	Domestic data (21 years)	U.S. data	Domestic data (29 years)	←	New domestic data + Individual data
	CCF	NUREG-1150	←	←	CCF 2010	NUREG-1150	CCF 2012	←	CCF 2015
	Mitigating measures	No AM*	AM taken	←	No AM No SA measures	AM taken	No AM No SA measure	SA measures taken (includes AM)	SA measures taken + Consideration for specialized safety facility
	Human error dependency	None	←	←	Between trains: considered Between sequences: none	None	Between trains: complete dependence Between sequences: considered	←	←
	Comment from TAC	–	–	–	–	–	Applied	←	←

*: Expected for feed and bleed, secondary system forced cooling established before 1992

4.2 Sensitivity analysis for PRA conducted after establishment of AM

- Regarding PRA conducted after establishing AM, sensitivity analysis was conducted to confirm impact of each analysis condition.
- Based on comparison between ② and ③, update of data regarding initiating event occurrence frequency data had little impact. Also, comparison between ② and ④ revealed that the impact of human error dependency was significant, increasing CDF by approx. 2.7 times. Furthermore, comparison between ①, ⑤ and ②, ⑥ revealed that impact of component failure rate data was medium, with ⑤ seeing a reduction of approx. 59% and ⑥ seeing a reduction of 36%.

Item		PSA report after establishing AM (March 2004)		Sensitivity analysis to confirm impact of analysis conditions			
		①Prior to establishment of additional AM	②After establishing additional AM	③ Impact of initiating event occurrence frequency	④ Impact of human error dependency	⑤ Impact of component failure rate (no AM)	⑥ Impact of component failure rate (AM taken)
CDF[reactor/year]		2.9×10^{-7}	1.5×10^{-7}	1.5×10^{-7}	4.0×10^{-7}	1.2×10^{-7}	9.6×10^{-8}
Analysis condition	Initiating event occurrence frequency	-FY2002 data	←	-FY2015 data	←	-FY2002 data	-FY2015 data
	Component failure rate	U.S. data	←	←	←	Domestic data (21 years)	Domestic data (29 years)
	CCF	NUREG-1150	←	←	←	←	←
	Mitigating measures	No AM[*1]	AM taken	←	←	No AM[*]	AM taken
	Human error dependency	None	←	←	Between trains: complete dependence	None	←
	Comment from TAC	–	–	–	–	–	–

*1: Expected for feed and bleed, secondary system forced cooling established before 1992
*2: Colored areas in the table indicate impact of analysis conditions. Green = no impact, Blue = reduction, Red = increase

YONDEN

4.3 Sensitivity analysis for PRA conducted as safety improvement evaluation

- There is little impact of updating initiating event occurrence frequency and failure rate data based on ⑤, ⑥, ⑦ and ⑧.
- Based on ②, impact of enhancing PRA ET was significant, with CDF increasing by approx. 86 times.
- Based on ④, the impact of implementing NRRC's human reliability analysis method was significant, increasing CDF by approx. 2.3 times.
- Based on ⑨ and ⑩), the impact of enhancing model of success criteria analysis, which is a condition to achieve highest probability, and severe accident response facilities was medium, with ⑨ seeing a reduction of approx. 20% and ⑩ seeing a reduction of approx. 60% (CFF).

Item	1st SAR (May 2019)				2nd SAR (July 2022)		3rd SAR (December 2023)			
	①No SA measures	② Impact of TAC comment	③SA measures taken	④Impact of human reliability analysis tool	⑤Impact of initiating event occurrence frequency	⑥ Impact of component failure rate	⑦ Impact of initiating event occurrence frequency	⑧ Impact of individual plant failure rate	⑨ Impact of other model enhancements	⑩Impact of severe accident response facilities
CDF[reactor/year]	1.8×10^{-3}	2.1×10^{-5}	1.8×10^{-6}	4.2×10^{-6}	4.1×10^{-6}	3.8×10^{-6}	3.7×10^{-6}	3.5×10^{-6}	2.8×10^{-6}	2.8×10^{-6}
CFF[reactor/year]	–	–	5.7×10^{-7}	–	9.3×10^{-7}	1.1×10^{-6}	1.1×10^{-6}	9.2×10^{-7}	6.7×10^{-7}	2.8×10^{-7}
Analysis conditions: Initiating event occurrence frequency (period)	-2015 fiscal year data	←	←	←	-2017 fiscal year data	←	-2021 fiscal year data	←	←	←
Analysis conditions: Component failure rat	Domestic data (29 years)	←	←	←	Domestic data (29 years) +Individual data	New domestic data	←	New domestic data＋Individual data	←	←
Analysis conditions: TAC comment	Applied	None	Applied	←	←	←	←	←	←	←
Analysis conditions: PRA enhancement: Human reliability analysis	THERP	←	←	HRA Calculator	←	←	←	←	←	←
Analysis conditions: PRA enhancement: Initiating event occurrence frequency (method)	Not considering maximum estimated capacity factor	←	←	←	Consider mean capacity factor	←	Plant specific loss of offsite power	←	←	←
Analysis conditions: PRA enhancement: Alternating operation, other *1	None	←	←	←	←	←	←	←	Yes	←
Analysis conditions: Mitigating measures: Emergency GTG	None	←	←	←	←	Yes	←	←	←	←
Analysis conditions: Mitigating measures: Specialized safety facility No. 3 battery	None	←	←	←	←	←	None	←	←	Yes

*1: In addition to modeling of systems operating alternately, includes application of the most probable condition success criteria analysis, application of other latest knowledge, and updates to design information.

*2: Colored areas in the table indicate impact of analysis conditions. Green = no impact, Blue = reduction, Red = increase

4.4 Results of level one external event PRA and sensitivity analysis

- For earthquakes, comparison between ①, ② and ③ confirmed that CDF increased due to impact of seismic hazard and TAC comments, and comparisons between ① and ④ confirmed that CDF decreased significantly due to impact of fragility.
- For tsunamis, comparison between ① and ② confirmed that impact of hazards and flood routes were significant.
- Earthquake and tsunami **to be reevaluated in the 4th SAR.**

[Earthquake]

Item			①Application for permission of reactor installment license (July 2015)	②Sensitivity analysis (Seismic hazard impact)	③Sensitivity analysis (Impact of TAC comment)	④Sensitivity analysis (Fragility impact)	⑤1st SAR [No SA measures] (May 2019)
Earthquake	CDF[reactor/year]		3.2×10^{-5}	4.0×10^{-5}	4.6×10^{-5}	8.7×10^{-6}	8.5×10^{-6}
	Difference in conditions	Earthquake hazard	Hazard at the time of application in July 2015	Hazard at the time of approval in Jully 2017	←	←	←
		Sequence added in response to TAC comment	Not considered	←	Considered	←	←
		Fragility	Not refined	←	←	refined	←

[Tsunami]

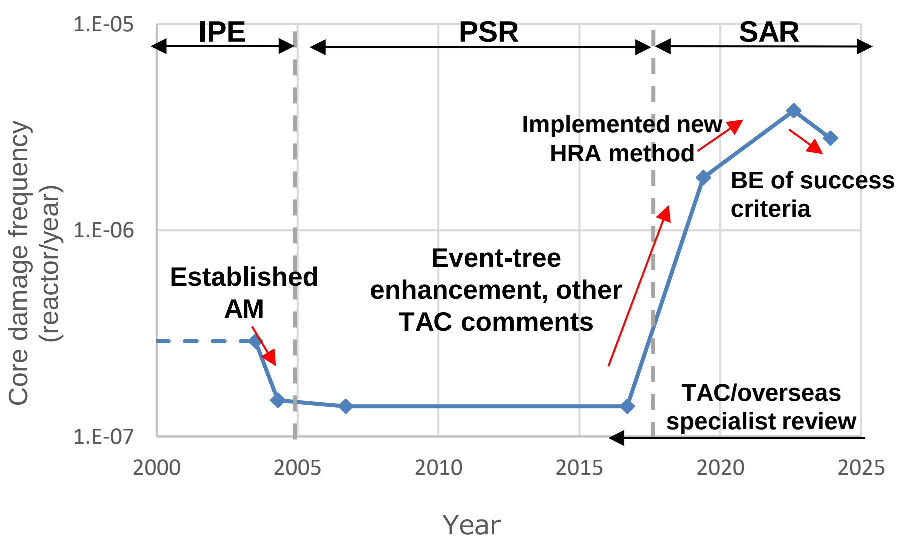
Item			①Application for permission of reactor installment license (July 2015)	②Sensitivity analysis (Impact of tsunami hazard, flood route)			⑤1st SAR [No SA measures] (May 2019)
Tsunami	CDF[reactor/year]		1.3×10^{-5}	1.9×10^{-5}			1.9×10^{-5}
	Difference in conditions	Tsunami hazard	Hazard at the time of application in July 2015	Hazard at the time of approval in Jully 2017			←
		Flood route (elevation of opening)	Not considered (3.8m)	Considered (5.9m)			←

*: Colored areas in the table indicate impact of analysis conditions, Blue = reduction, Red = increase

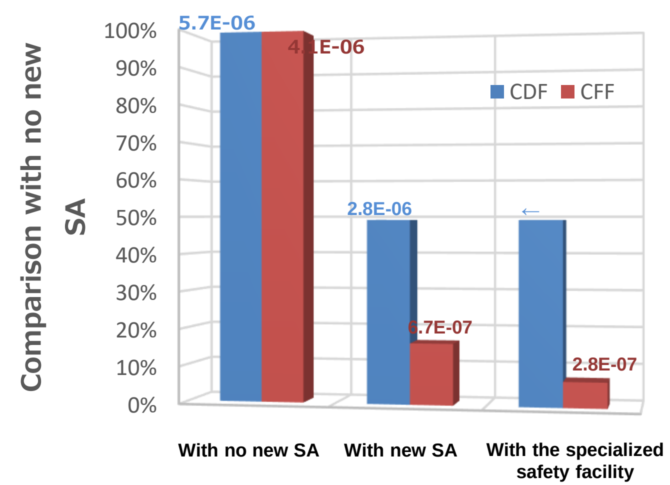
4.5 Insights regarding transition of CDF, etc.

- Regarding internal event PRA during at-power operation, when comparing the post AM model, conducted when PRA was referred to as PSA, with the model incorporating knowledge from TAC and overseas specialists, **enhancement of PRA event-tree and human reliability analysis significantly improved CDF, etc.**
- ―On the other hand, when comparing CDF and CFF while focusing only on the differences between facilities, **the addition of SA facilities, and specialized safety facilities as a response to new regulatory requirement conformance inspection, CDF decreased by approx. 1/2 and CFF by approx. 1/10.**
- As indicated above, **while CDF increased**, **safety has increased** due to the results of risk assessment being more realistic with detailed models and application of new methods and allowing additional measures to be considered.

Ikata Unit 3, CDF transition (reported value)



Comparison when using the same data and evaluation methods



4.5 Insights regarding transition of CDF, etc.

- When considering comparisons with the U.S., regarding **CDF of Surry NPP**, **a reduction of a single digit** was observed with 4.0×10^{-5}reactor/year in 1990, **decreasing to**
- **2.5×10^{-6}/reactor/year in 2017**. This indicates that trends are generally consistent with the average of the U.S. industry.
- While comparisons are not completely accurate as maintenance/management of failure data are handled differently between the U.S. and Japan, Ikata Unit 3's CDF was **2.8×10^{-6}/reactor/year and was generally at the same level as the U.S. Surry NPP.**

New/old CDF at U.S. Surry NPP

Referenced from NUREG-1150

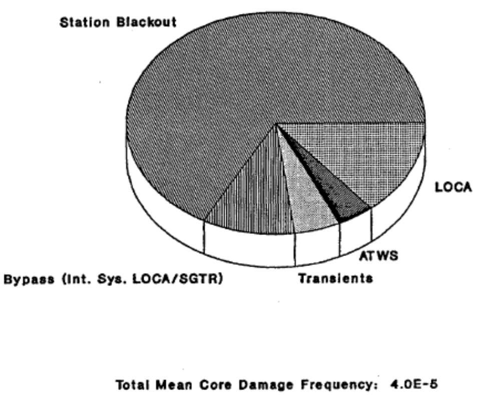


Figure 3.3 Contributors to mean core damage frequency from internal events at Surry

Referenced from PSA-2017

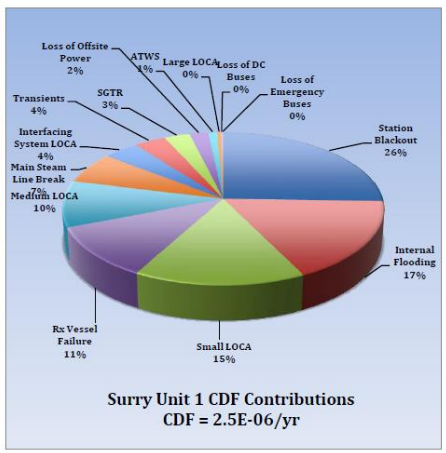
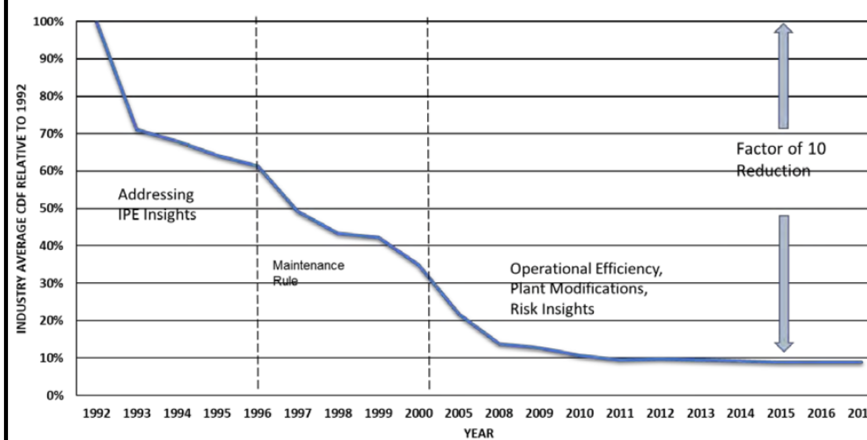


Figure 4: Surry Risk Profile after FLEX implementation

Reduction of a single digit was observed with 4.0×10^{-5}reactor/year in 1990 decreasing to 2.5×10^{-6}/reactor/year in 2017

U.S industry mean CDF: Referenced from NEI 20-04



Source: Multiple Sources including IPE submittals and ROP data for Mitigating System Performance Index

Figure 12 - Industry Average CDF Trend

Reduction of a single digit from 1990's to 2010's

5. Future challenges

- In this way, we have been working on various improvements not only for internal event PRA during at-power operation but also for seismic PRA. We have developed a PRA model that is now comparable to the international examples. To ensure its quality, initiatives shall continue to be taken to incorporate reviews by overseas specialists.

- On the other hand, RIDM application to regulatory activities is limited to the significance determination process (SDP) in the nuclear regulatory inspection.

- PRA is by no means a miracle tool, but it is an important tool for the conduct of risk management. It is important not to forget the concept of being risk informed.

- In the future, PRA application in various business processes, such as online maintenance and including regulatory activities, is expected to spread the concept of risk management.

6. Summary

- Regarding internal event PRA during at-power operation, adequate quality is to be secured by solving F&O from reviews conducted by overseas specialists, initiatives are to be taken to develop appropriate models to ensure evaluation results become more realistic, and application of RIDM in business processes are to be expanded.

- For seismic/tsunami PRA, enhanced models will be implemented in the base of internal event PRA during at-power operation. For seismic PRA, the seismic hazard established as a result of the Ikata SSHAC project is to be considered to conduct a more reliable external event PRA, which is to be applied to various business processes.

- Initiatives are to be continued for the enhancement of the PRA model, and by applying the enhanced PRA model to RIDM of various business processes, performance at Ikata NPP is to be improved, and results shall be disclosed to the society through the submission of SAR, etc.

Thank you