

# ***Status of Nuclear Power Plants in Japan and Future Research Strategy for Nuclear Safety and Materials***

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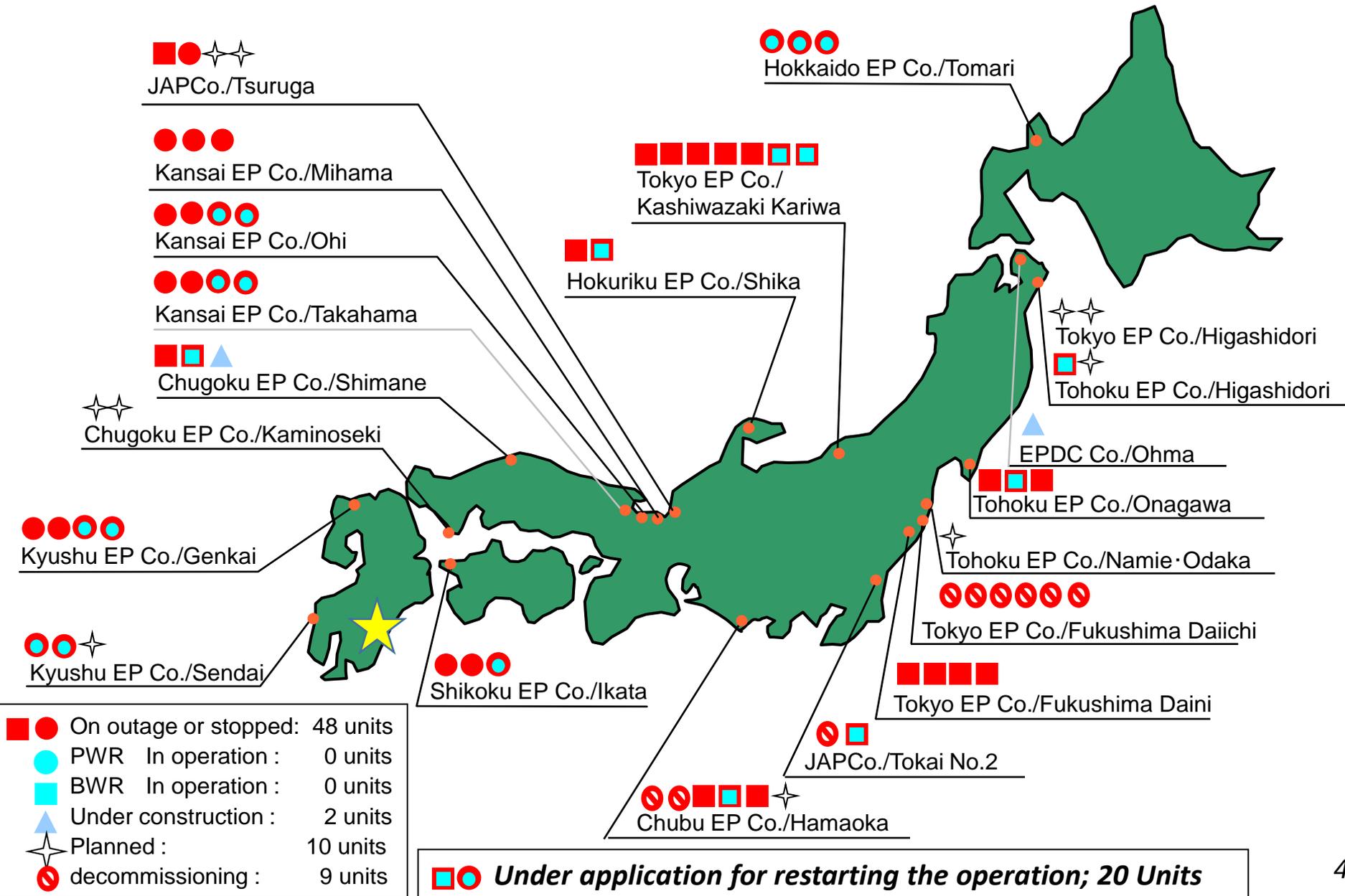
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# ***1. Current Status of NPPs in Japan***

# Nuclear Power Plants in Japan as of January, 2015



# ***The NRA has received 20 Applications for Restarting Reactors***

<b>Applicants (Electric Utility Company)</b>	<b>Nuclear Power Plant and Unit #</b>	<b>Start of Operation (Age)</b>	<b>Date of Application</b>	<b>Completion of NRA Review</b>
Hokkaido	Tomari #1, #2	1989/1991 (25/23)	July 8, 2013	
	Tomari #3	2009 (4)	July 8, 2013	
Kansai	Ohi #3,#4	1991/1993 (22/21)	July 8, 2013	
	Takahama #3,#4	1985/1985 (29/29)	July 8, 2013	
Shikoku	Ikata #3	1994 (19)	July 8, 2013	
Kyushu	<b>Sendai #1,#2</b>	1984/1985 (30/28)	July 8, 2013	<b>July 16, 2014</b>
	Genkai #3, 4	1994/1997 (20/17)	July 12, 2013	
Tokyo	Kashiwazaki-Kariwa #6,#7	1996/1997 (17/17)	Sep. 27, 2013	
Chugoku	Shimane #2	1989 (25)	Dec. 25, 2013	
Tohoku	Onagawa #2	1995 (19)	Dec. 27, 2013	
Chubu	Hamaoka #4	1993 (21)	Feb. 14, 2014	
Japan Atomic Power Company	Tokai-Daini	1978 (35)	May 20, 2014	
Tohoku	Higashidori #1	2005 (8)	June 10, 2014	
Hokuriku	Shika #2	2006 (8)	Aug. 12,2014	
total	20 Units		-	2 Units

# ***Restarting of Sendai Unit 1 & Unit 2***

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**PWRs, 890 MWe each, Start of Commercial Operation; July 4, 1984 (Unit 1)**

**Nov. 28, 1985 (Unit 2)**



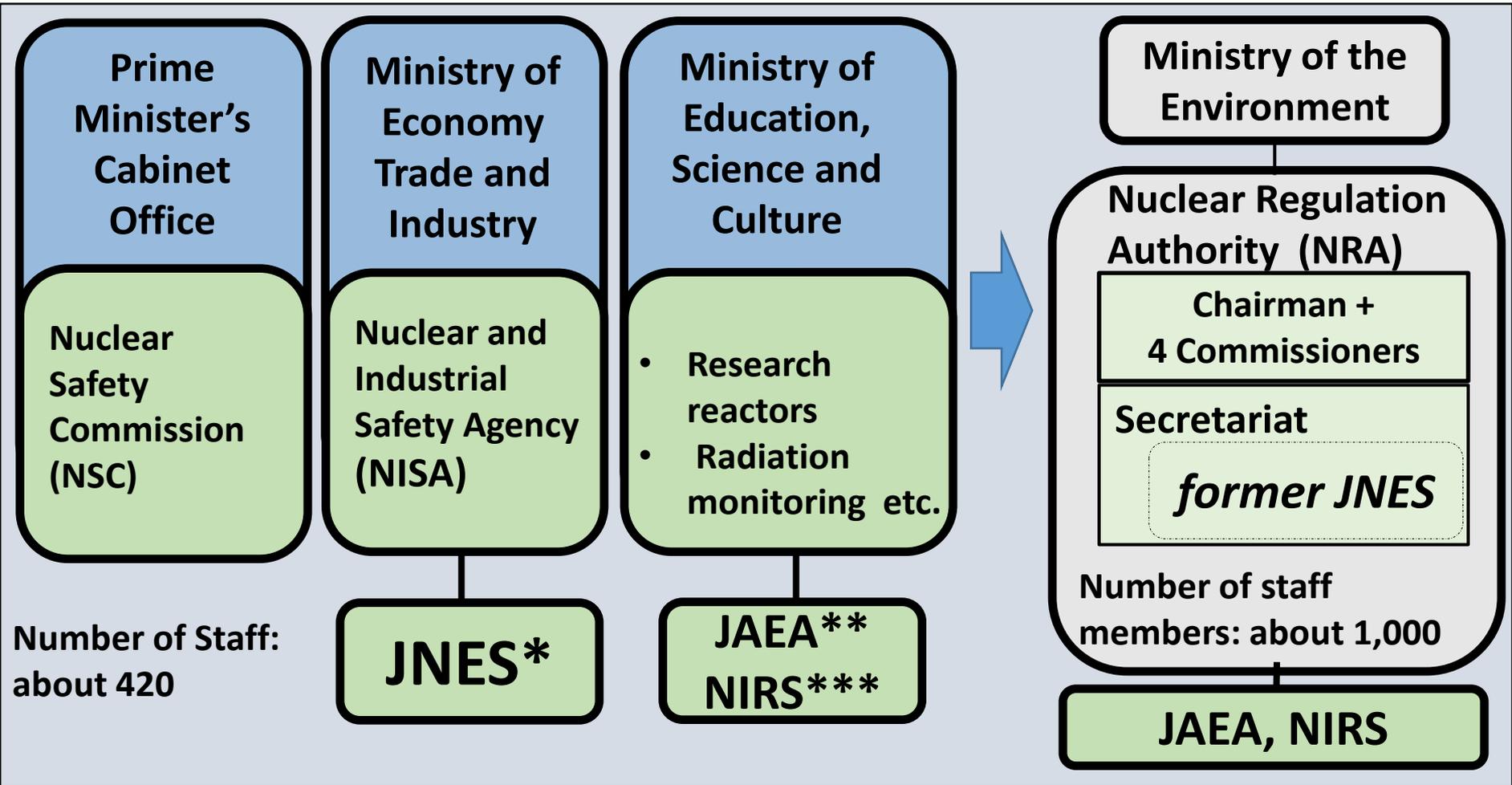
<http://www.reuters.com/article/2014/07/16/us-japan-nuclear-restart-idUSKBN0FL02R20140716>

*“An employee of Kyushu Electric Power Co walks in front of reactor buildings at the company’s Sendai NPS on April 3, 2014”*

***2. New Regulatory System in Japan,  
Regulatory Approach for Ageing  
Management and Long Term  
Operation in Japan***

# Establishment of Nuclear Regulation Authority (NRA)

NRA was established on September 19, 2012, and merged with JNES on March 1, 2014



\*JAPAN Nuclear Energy Safety Organization  
( merged to the NRA on March 1, 2014)

\*\* JAPAN Atomic Energy Agency

\*\*\* National Institute of Radiological Sciences<sup>8</sup>

# New Safety Standards in Japan

## New Safety Standards Effective from July 8, 2013

### < Previous Safety Standards >

Design Basis Standard to prevent Severe Core Damage (Only assuming single failure etc.)

Consideration for Natural Events
Consideration for Fire
Consideration for Reliability
Reliability of power source
Performance of cooling equipment
Performance of other equipment
Tolerability for Earthquake & Tsunami

Preventing Large Scale Release
Intentional Airplane Crash
Preventing CV Failure
Preventing Severe Core Damage (Assuming multiple malfunction)
Consideration for Natural Events
Consideration for Fire
Consideration for Reliability
Reliability of Power Source
Performance of Cooling Equipment
Performance of Other Equipment
Tolerability for Earthquake & Tsunami

**New**

(Countermeasures against Severe Accident)

**Enhancement**

**Enhancement**



# ***New Nuclear Regulation Systems in Japan***

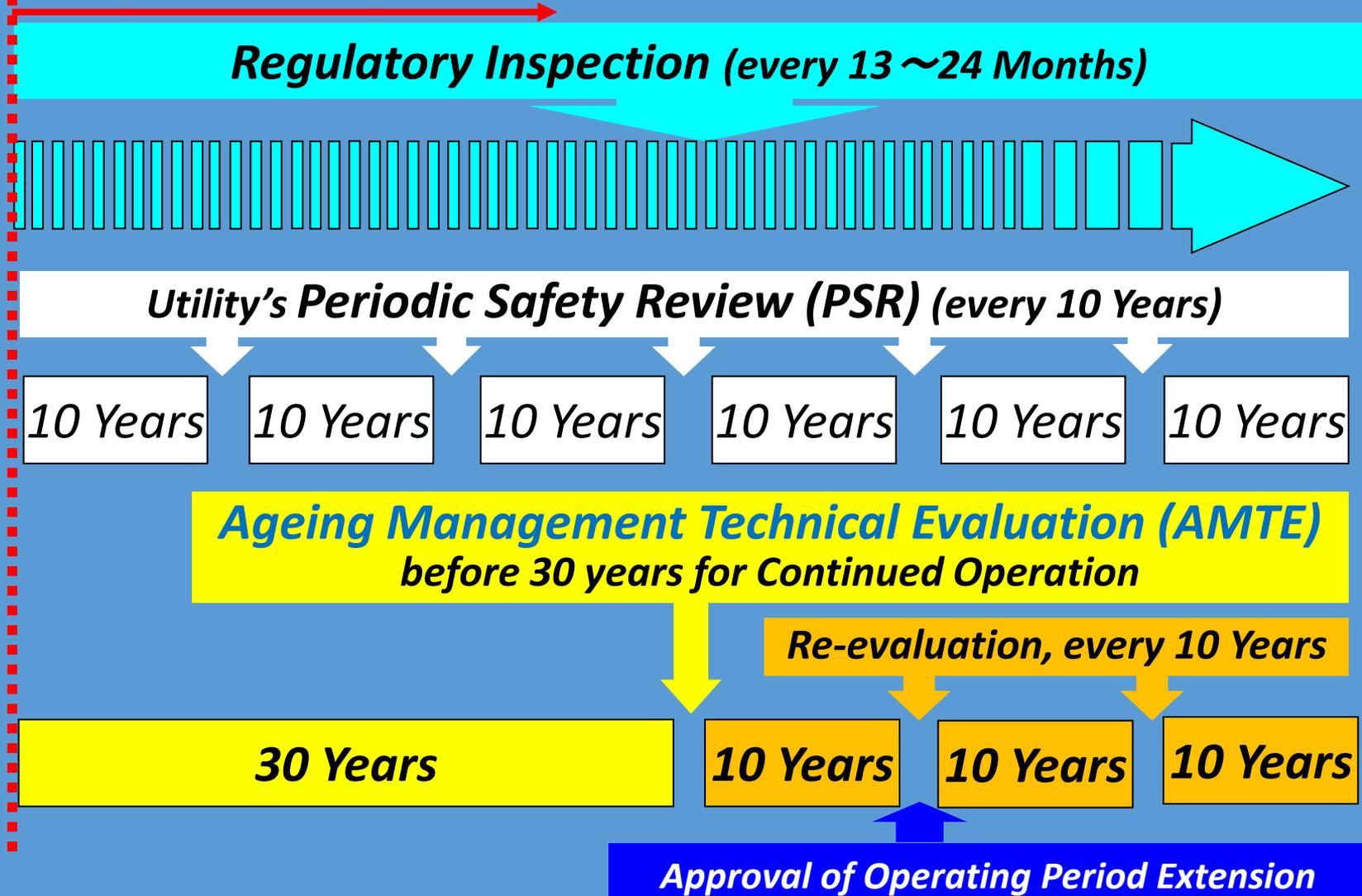
***Became Effective from July 8, 2013***

The NRA (Nuclear Regulatory Authority) implement new regulatory systems stipulated in the amended Nuclear Regulation Act ;

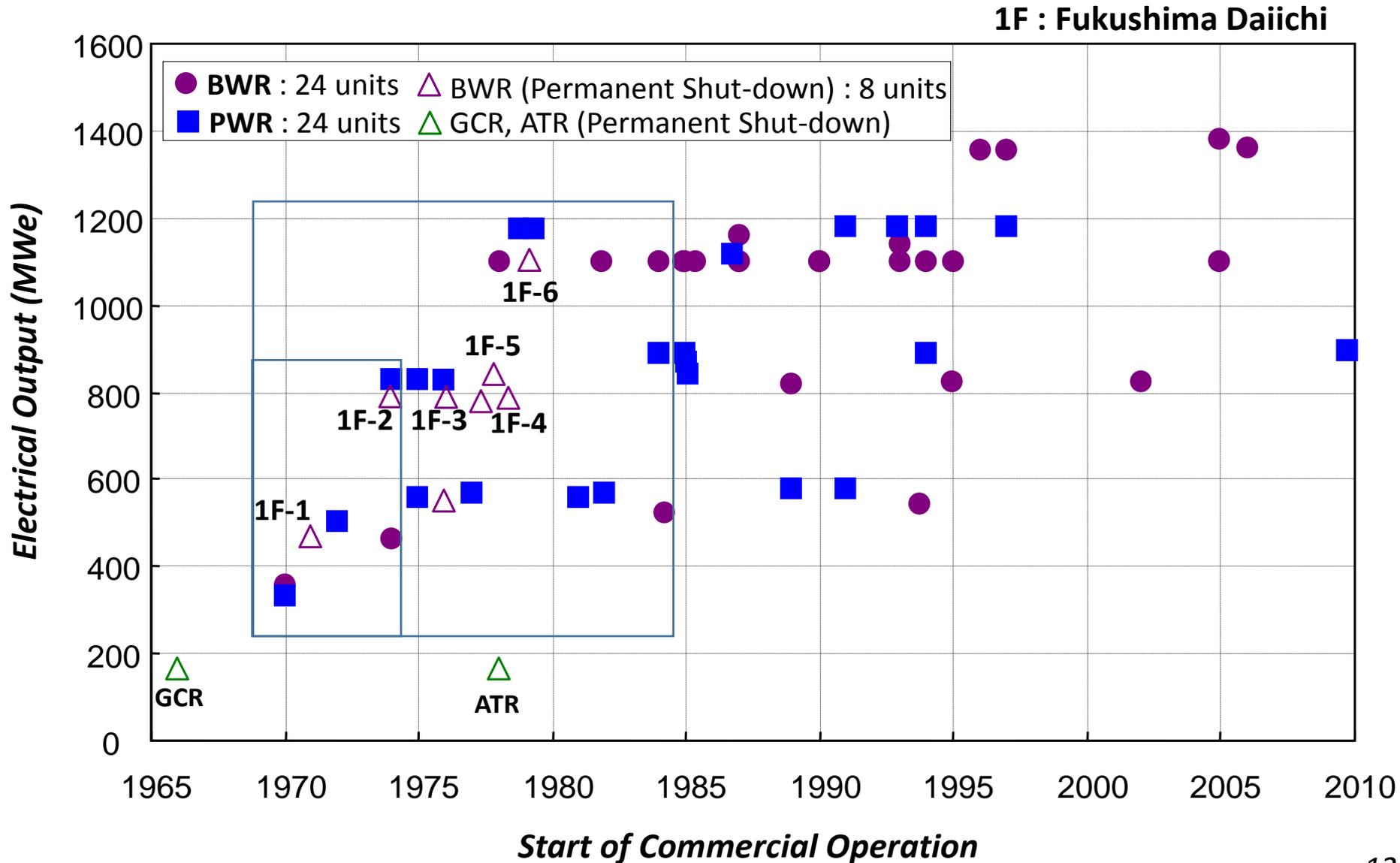
- **Regulation taking severe accidents into consideration**
  - Legally request severe accident measures to the licensees
- **Introduction of back-fitting system : Regulation applying latest scientific/technical knowledge on safety issues to existing facilities**
  - Apply new technical standards to existing licensed nuclear facilities as a legal obligation
- **An operation limit of 40 years to deal with aged reactors**
  - As an exception, **extension (<20 years) will be approved**, only when compliance with the regulatory standards by the Government Order is confirmed.
- **Special safety regulation in Fukushima Daiichi**

# Regulatory Systems for Inspection, Periodic Safety Review and Ageing Management Technical Evaluation

*Start of Commercial Operation*



# Nuclear Power Plants in Japan



# ***Ageing Management Technical Evaluation before 30 years of Operation***

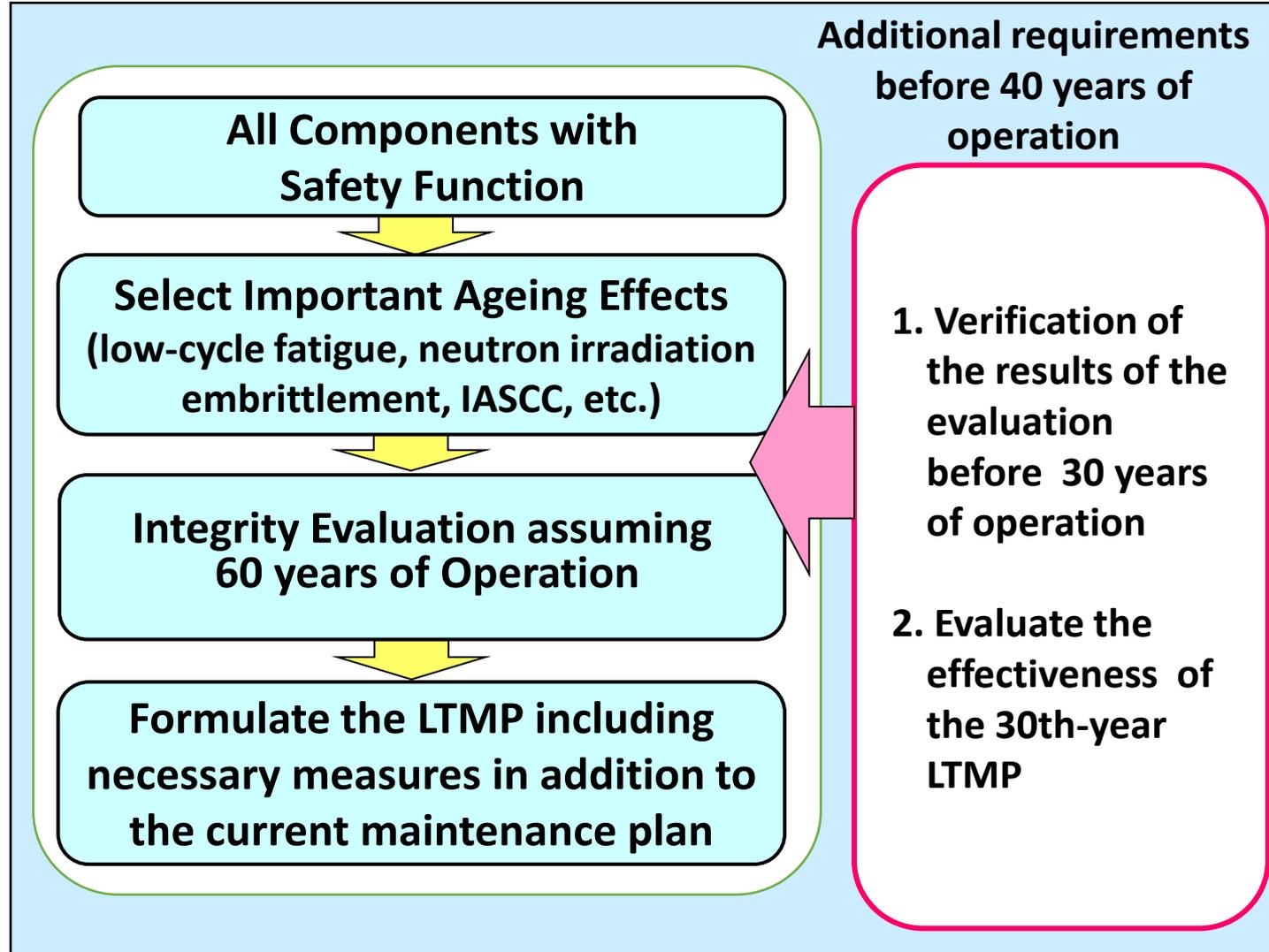
- 1. Review Operational Experience and Possible Ageing Degradation in All the Safety-related SSCs***
- 2. Reflect Latest Knowledge and Engineering Database***
- 3. Evaluate Integrity of SSCs considering Ageing Degradation Assuming the Service for 60 years***
- 4. Evaluate Seismic Safety Analysis considering Ageing Degradation Assuming the Service for 60 years***
- 5. Evaluate Validity of Current Maintenance Program***
- 6. Establish Long-Term Maintenance Program in the next 10 years, including R&D Plans***

# Ageing Management Technical Evaluation

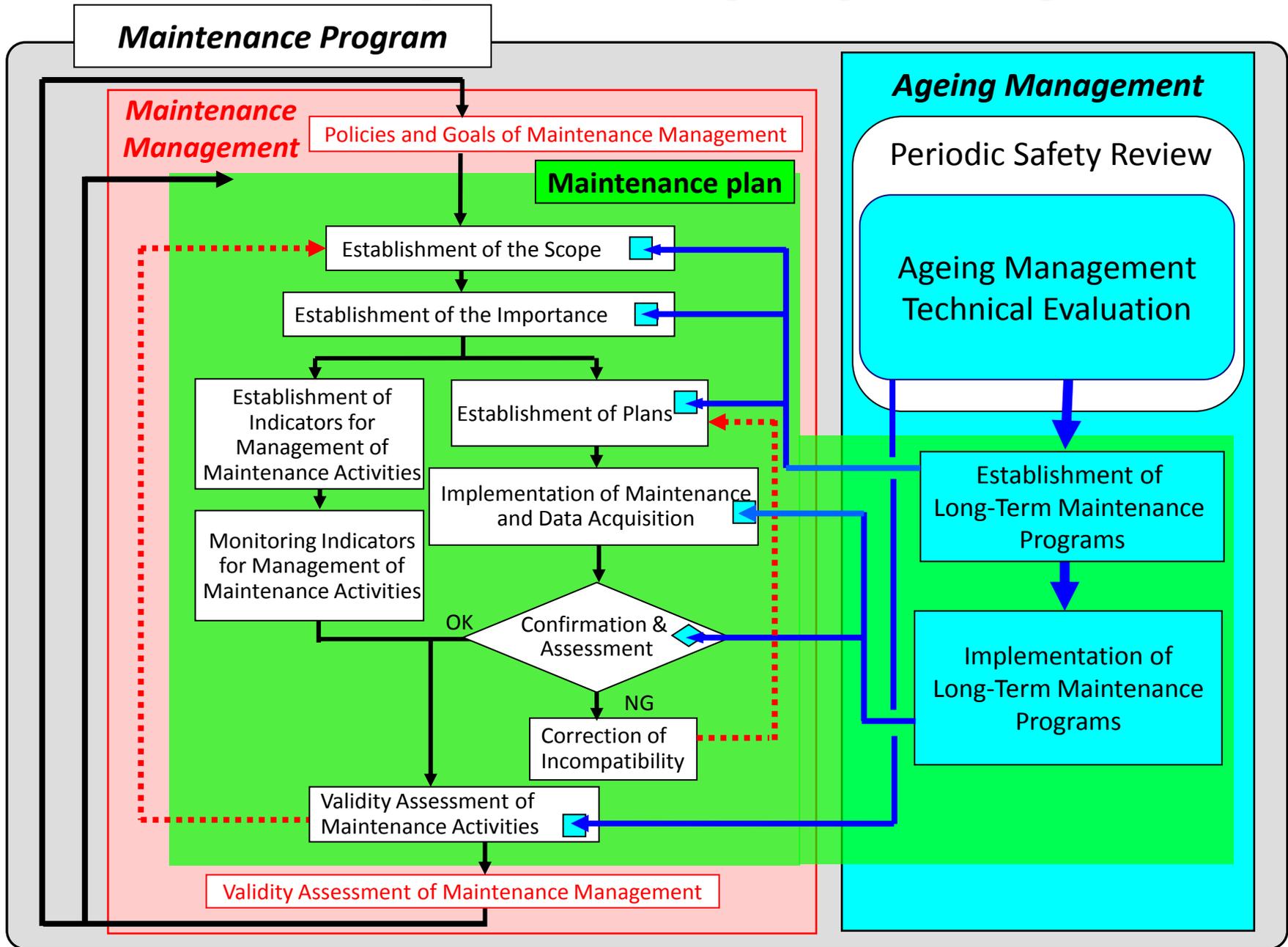
*Before 30 years of operation and following every 10 years*

## Regulatory Review

- Results of the evaluation
- Adequacy of the Long-Term Maintenance Program (LTMP)



# Maintenance Program and Ageing Management



# ***New Nuclear Regulation Systems in Japan***

***Effective from July 8, 2013***

**The NRA (Nuclear Regulatory Authority) implement new regulatory systems stipulated in the amended Nuclear Regulation Act ;**

- **Regulation taking severe accidents into consideration**
  - Legally request severe accident measures to the licensees
- **Introduction of back-fitting system : Regulation applying latest scientific/technical knowledge on safety issues to existing facilities**
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# Approval of Operational Period Extension and Ageing Management Technical Evaluation

## Extension of Operational Period beyond 40 years

### <Requirements>

- (1) Implementation of **special safety inspection** of the NPPs
- (2) Technical evaluation on the degradation for the extension period
- (3) Formulation of maintenance management policy for the extension period

Regulatory Review

40<sup>th</sup> year

Approval

Extension period

## Ageing Management Evaluation before 30 years and beyond

### <Requirements>

- (a) Technical evaluation on ageing degradation
- (b) Formulation of the long-term maintenance management policy

Regulatory Review

Approval

### <Requirements>

The appropriate maintenance is implemented according to the standard technical specifications.

# ***Special Inspections to Extend Operational Period***

## **Regulatory Requirements of Special Additional Inspections for Long Term Operation beyond 40 years up to 60 years**

<b>Components</b>	<b>Current Inspection</b>	<b>Additional Inspection</b>
<b>Reactor Pressure Vessel</b>	<b>Ultrasonic Tests of Welded Zone</b>	<b>100% UT Examination of Base Metal in addition to Welded Zone</b>
<b>Primary Containment Vessel (Steel)</b>	<b>Leak Rate Tests</b>	<b>Visual Test (Appearance of coating film)</b>
<b>Civil Concrete Structure</b>	<b>Visual Tests &amp; NDT</b>	<b>Core Sampling (Strength, Neutralization, Salt intrusion, etc.)</b>

- **Special Inspections are requirements to extend the operational period beyond 40 years by up to 20 years ( 60 years maximum ) in addition to;**
  - **Ageing Management Technical Evaluation (AMTE) for every 10 years**
  - **Maintenance Program**

# ***Ageing Management Technical Evaluation after the Fukushima Accident***

Unit	Ageing Management Technical Evaluation	Date (30 or 40 years)	Submitted Date of the Report	Date of Approval by the NRA
Genkai-2	30 years	Mar. 30, 2011	Mar. 26, 2010	Nov. 25, 2010
Ikata-2	30 years (cold shutdown)	Mar. 19, 2012	Mar. 11, 2011	Mar. 15, 2012
Fukushima Daini-1	30 years (cold shutdown)	Apr. 20, 2012	Jan. 13, 2012	Apr. 19, 2012
Mihama-2	40 years (cold shutdown)	July 25, 2012	July 22, 2011	July 26, 2012
(Establishment of NRA on Sep. 19, 2012)				
Fukushima Daini-2	30 years (cold shutdown)	Feb. 2, 2014	July 31, 2013	Jan. 22, 2014
Onagawa-1	30 years (cold shutdown)	June 1, 2014	Nov. 6, 2013	May 21, 2014
<b>Sendai-1</b>	<b>30 years</b>	July 4, 2014	Dec. 18, 2013	○
Shimane-1	40 years (cold shutdown)	Mar. 29, 2014	Sep. 27, 2013	Feb. 26, 2014
Takahama-1	40 years (cold shutdown)	Nov. 14, 2014	Nov. 12, 2013	Nov. 12, 2014
<b>Takahama-3</b>	<b>30 years</b>	Jan. 17, 2015	Jan. 15, 2014	○
<b>Takahama-4</b>	<b>30 years</b>	June 5, 2015	June 3, 2014	○
Fukushima Daini-3	30 years (cold shutdown)	June 21, 2015	June 20, 2014	
Kashiwazaki-1	<b>30 years</b>	Sep. 18, 2015	Sep. 18, 2014	
Genkai-1	40 years (cold shutdown)	Oct. 15, 2015	Oct. 15, 2014	
Takahama-2	40 years (cold shutdown)	Nov. 14, 2015	Nov. 14, 2014	
<b>Sendai-2</b>	<b>30 years</b>	Nov. 28, 2015	Nov. 28, 2014	○

○ : Application to the NRA for restarting the reactors 19

# ***The NRA has received 20 Applications for Restarting Reactors***

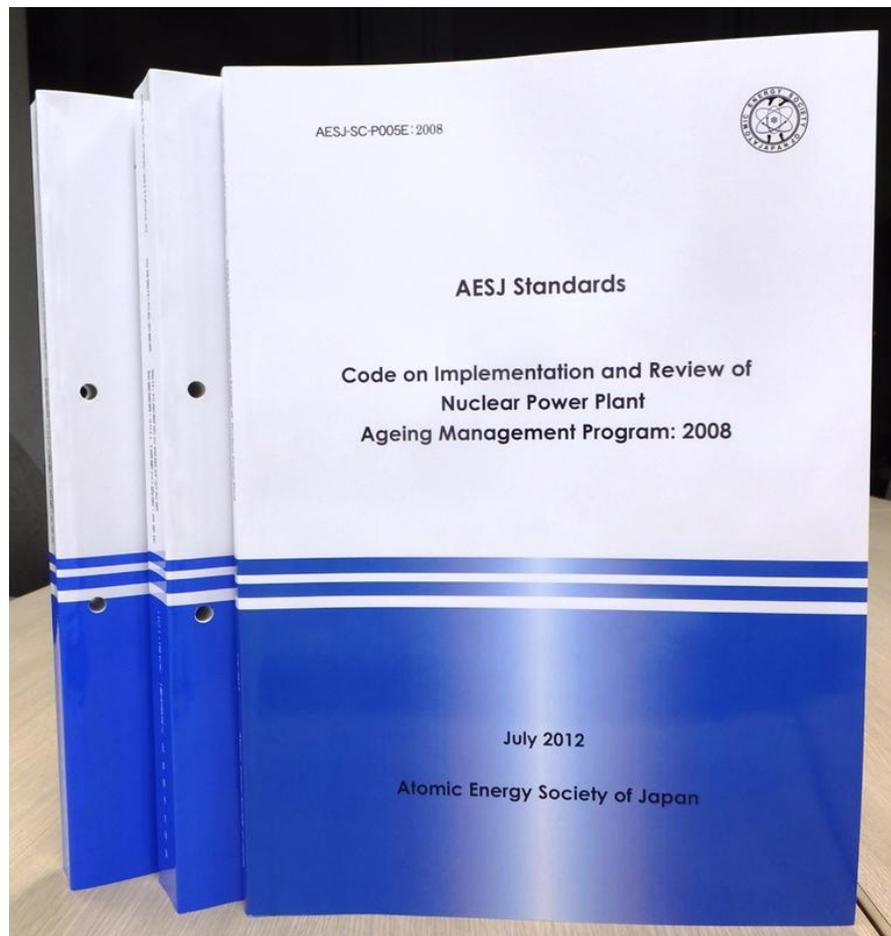
<b>Applicants (Electric Utility Company)</b>	<b>Nuclear Power Plant and Unit #</b>	<b>Start of Operation (Age)</b>	<b>Date of Application</b>	<b>Completion of NRA Review</b>
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Shikoku	Ikata #3	1994 (19)	July 8, 2013	
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	Genkai #3, 4	1994/1997 (20/17)	July 12, 2013	
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Tohoku	Onagawa #2	1995 (19)	Dec. 27, 2013	
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Hokuriku	Shika #2	2006 (8)	Aug. 12, 2014	
total	20 Units		-	2 Units

### ***3. Reflection of the IAEA IGALL Knowledge-base and AESJ Code on Implementation and Review of Nuclear Power Plant Ageing Management Program***

# ***Code on Implementation and Review of Nuclear Power Plant Ageing Management Program***

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***Atomic Energy Society of Japan (AESJ) has established the “Code on Implementation and Review of Nuclear Power Plant Ageing Management Programs” (AESJ-SC-P005: 2008) in 2008 and continuously updated in 2010, 2011, 2012 and 2013.***



***English version is available***

# Cross Table for Ageing Mechanisms and (Location and Material in) Components

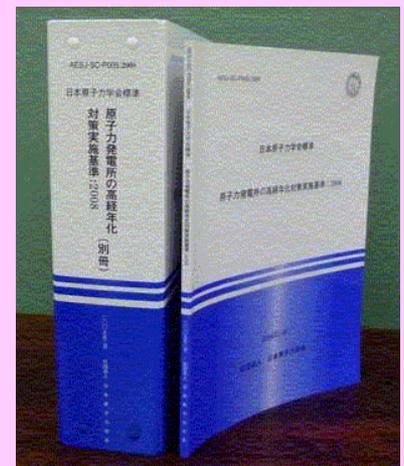
## Summary Sheets of Ageing Mechanisms (Attachment A of the AESJ Code)

- **Based on the experience in Ageing Management Technical Assessment in 14 Plants (7 PWRs and 7 BWRs).**
- **Cross tables of ageing mechanisms for 300 components in each PWR and BWR have been successfully summarized and published in the AESJ Code as a mandatory requirement.**

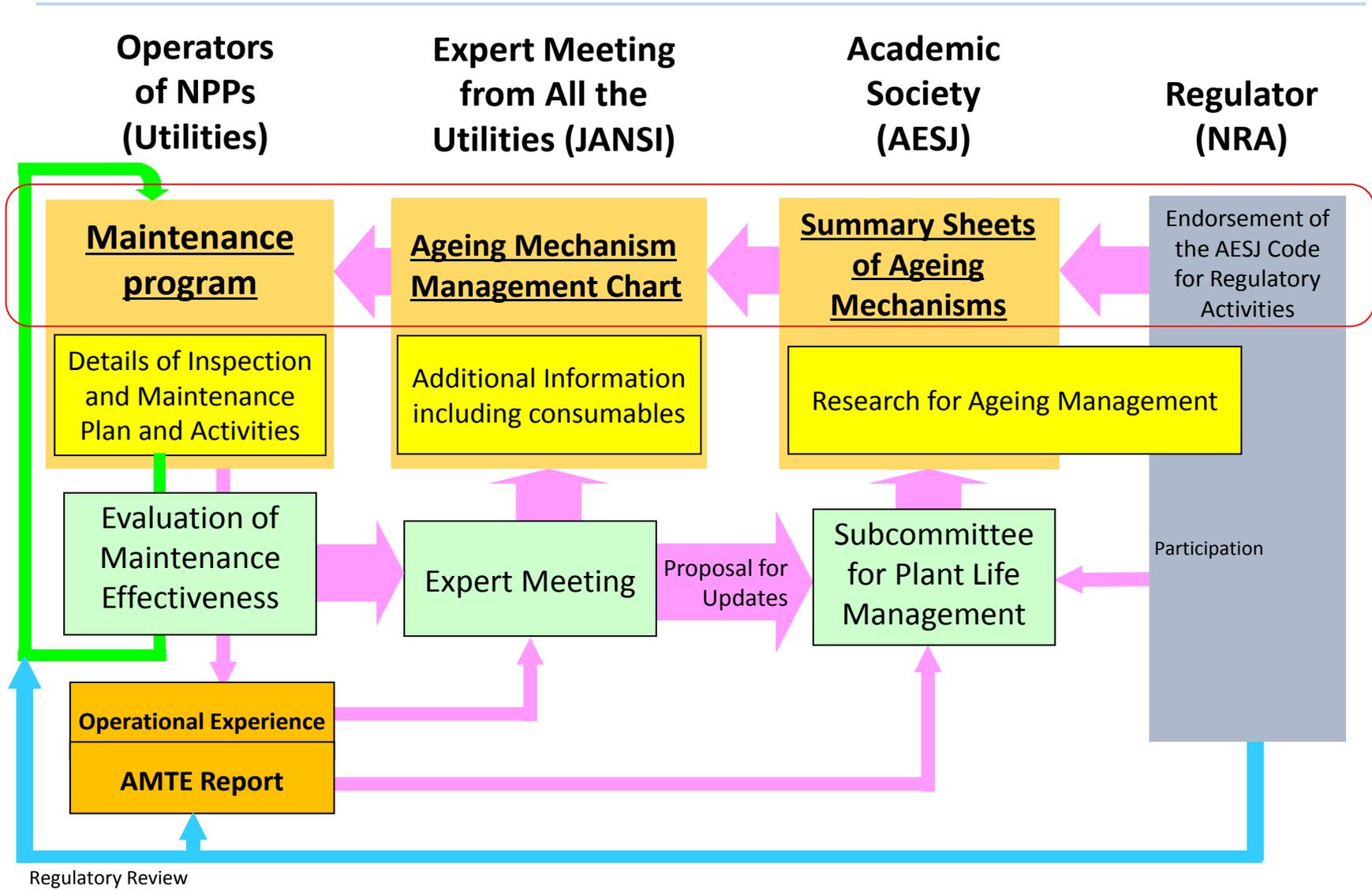
### Summary Sheet of Ageing Mechanisms for Turbo Pumps

P01-01 Pump (vertical axial flow turbo pump/seawater/SS)

N o.	Issues required to achieve intended functions	Part	Material	Ageing phenomena	Remarks
1	Assurance of pump capacity (head)	Main shaft	SS	Wear	
2		Main shaft	SS	Corrosion (pitting, etc.)	
•		• • •	• • •	• • •	• • •
13	Maintenance boundary of	Discharge elbow	Cast iron	Corrosion (pitting, etc.)	
14		Discharge pipe	Cast iron	Corrosion (pitting, etc.)	
•		• • •	• • •	• • •	• • •
23	Support of component	Support plate	SS	(N/A)	
24			SS	(N/A)	
•		• • •	• • •	• • •	• • •



# Sharing and Updating Knowledge for Ageing Management with Summary Sheets of Ageing Mechanisms



Regulatory Review

# Example (1)

## PWR Reactor Vessel

No.	Function	Location	Material	Ageing phenomenon
1	<b>Boundary for Pressure and Coolant</b>	Upper head, upper shell, intermediate shell, lower shell, Bottom head	Low alloy steel (overlaid with stainless steel)	Fatigue cracking
2				Neutron irradiation embrittlement (intermediate and lower shells)
3				Under clad cracking (UCC)
4		Top head flange, upper shell flange	Low alloy steel (overlaid with stainless steel)	Pitting
5				Fatigue cracking
6				Under clad cracking (UCC)
7		Coolant inlet nozzle	Low alloy steel (overlaid with stainless steel) [safe end is stainless steel; weld metal is Inconel 600 alloy]	Fatigue cracking
8				Stress corrosion cracking (weld metal)
9				Fatigue cracking
10		Coolant outlet nozzle	Low alloy steel (overlaid with stainless steel) [safe end is stainless steel; weld metal is Inconel 600 alloy]	Stress corrosion cracking (weld metal)
11				Fatigue cracking
12				Stress corrosion cracking (weld metal)
13		Top head nozzle	Inconel 600 alloy	Fatigue cracking
14				Stress corrosion cracking
15			Inconel 690 alloy	Fatigue cracking
16		Stress corrosion cracking (including weld metal)		
17		Air vent nozzle	Inconel 600 alloy	Fatigue cracking
18				Stress corrosion cracking
19			Inconel 690 alloy	Fatigue cracking
20		Stress corrosion cracking (including weld metal)		
21		Bottom mounted instrumentation nozzle	Inconel 600 alloy [weld metal is Inconel 600 alloy]	Fatigue cracking
22				Stress corrosion cracking (including weld metal)
23		Radial support	Inconel 600 alloy	Fatigue cracking
24				Stress corrosion cracking (including weld metal)
25		Stud bolt	Low alloy steel	Corrosion
26				Fatigue cracking
27		Vessel support metal	Low alloy steel	Fatigue cracking
28		O-ring	—	(Consumables and periodic replacement parts)

# Example(2)

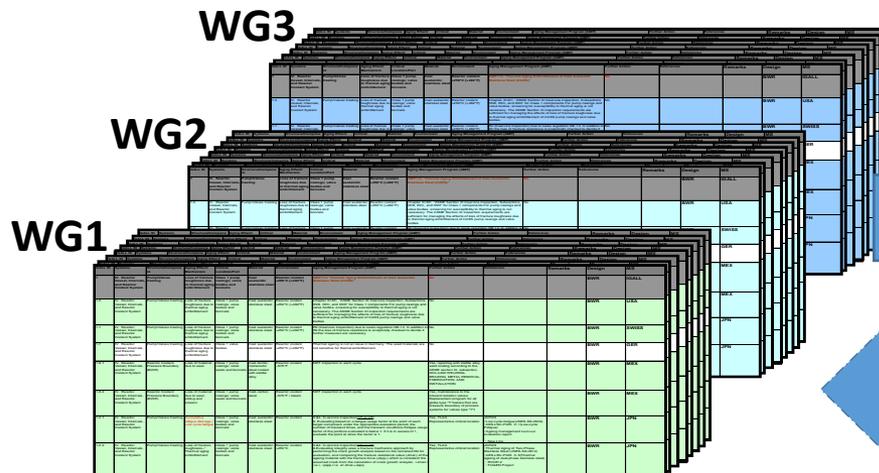
## PWR Pressurizer

No.	Function	Location	Material	Ageing phenomenon	
1	<b>Boundary for Pressure and Coolant</b>	Upper head, upper shell, intermediate shell, lower shell, Bottom head	Low alloy steel (overlaid with stainless steel)	Under clad cracking (UCC)	
2		Manhole seat	Low alloy steel (overlaid with stainless steel)	Pitting	
3		Manhole cover	Low alloy steel	Pitting	
4		Manhole bolt	Low alloy steel	Corrosion	
5		Instrumentation nozzle	Stainless steel	Stress corrosion cracking	
6		Heater sleeve	Stainless steel	Stress corrosion cracking	
7		Spray line nozzle	Low alloy steel (overlaid with stainless steel) [safe end is stainless steel]	Low alloy steel [safe end is stainless steel]	Fatigue cracking
8				Stress corrosion cracking	
9				Fatigue cracking	
10				Stress corrosion cracking (weld metal)	
11				Fatigue cracking	
12				Stress corrosion cracking	
13				Fatigue cracking	
14				Stress corrosion cracking (weld metal)	
15		Surge nozzle	Low alloy steel (overlaid with stainless steel) [safe end is stainless steel; weld metal is Inconel 600 alloy]	Stress corrosion cracking	
16				Fatigue cracking	
17				Stress corrosion cracking (weld metal)	
18				Stress corrosion cracking	
	Safety valve nozzle	Low alloy steel [safe end is stainless steel]	Stress corrosion cracking		

# Knowledge Transfer between IGALL and the AESJ Code

- The combinations of intended and required safety functions, portion of SSCs and the ageing mechanism/effect are summarized in the summary sheet in the Attachment A of the AESJ code, based on the experience from the 17 AMTEs in Japan.
- Knowledge-base from IAEA IGALL is currently under review by AESJ members be reflected in the next major revision, including the lessons learned from the Fukushima accident.

## IGALL (Spread sheets)



## Attachment A of the AESJ code

## Summary Sheets of Ageing Mechanisms

No.	機能達成に必要な項目	部位	材料	経年化対象	経年化技術評価値不降の条件		経年安全上の影響	
					静的機能	動的機能	前歴上の影響	
1	ポンプの容量・構造確保	主軸	ステンレス鋼	疲労割れ (高サイクル疲労割れ)	②		☆	▼
3	羽根車	羽根車	ステンレス鋼/鋳鉄、銅合金鋳物	腐食 (キャビテーション)	②			▼
4	羽根車リソング	羽根車リソング	—	(消耗品・定期取替品)	—			▼
5	圧熱性能の確保	圧熱管	インコネル690合金	疲労割れ (フレッディング疲労割れ) 管板/ボリス部応力腐食割れ	②	★		▼
7				スケール付着				▼
9		駆止め金具	ステンレス鋼	疲労		★		▼
10		冷却材出入口管台セーフエンド (ステンレス鋼)	ステンレス鋼 (インコネル肉盛)	応力腐食割れ		★		◎
12		1次側マンホール	低合金鋼 (インサートプレートはステンレス鋼)	(想定されず)	—	★		◎
13		パウダ	低合金鋼	疲労割れ		★		◎
14		管板	低合金鋼 (インコネル肉盛)	内盛部応力腐食割れ		★		◎
15		リの維持		内盛下部のき裂	②			◎
16		ガスケット	—	(消耗品・定期取替品)	—	★		◎
17		仕切板	インコネル600合金	応力腐食割れ		★		◎
18			インコネル690合金	応力腐食割れ	①	★		◎

1,000 pages for  
PWR and BWR

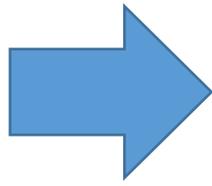
# Knowledge Transfer from IGALL to the AESJ code

**In the AESJ code, specific methods for implementing a technical evaluation of the ageing phenomena are defined in Attachment C.**

**Knowledge from these evaluation methods in AMPs and TLAAs of the IGALL is reviewed and taken into Attachment C if necessary.**

## IGALL (AMP, TLA)

IGALL AMPs	Lead Author
AMP101 "Fatigue Monitoring"	Brazil (P. Filho)
AMP102 "In-service Inspection/Periodic Inspection"	Japan (M. Kanno)
AMP103 "Water Chemistry"	Switzerland (J. Held)
AMP104 "Reactor Head Closure Stud Bolting"	Germany (Fr. Michel)
AMP105 "BWR Vessel ID Attachment Welds"	Japan (M. Kanno)
AMP106 "BWR Feedwater Nozzle"	Japan (M. Kanno)
AMP107 "BWR Stress Corrosion Cracking"	Germany (Fr. Michel)
AMP108 "BWR Penetrations"	Japan (M. Kanno)
AMP109 "BWR Vessel Internals"	Mexico (Griz Cruz)
AMP110 "Boric Acid Corrosion"	USA (A. Hiser)
AMP111 "Cracking of Nickel-Alloy Components and Loss of Material Due to Boric Acid-Induced Corrosion in RCPB Components (PWRs Only)"	Belgium
AMP112 "Thermal Aging Embrittlement of Cast Austenitic Stainless Steel (CASS)"	France (F. Termon-Morin)
AMP113 "PWR Vessel Internals"	USA (A. Hiser)
AMP114 "Flow-Accelerated Corrosion"	Czech (R. Krivanek)
AMP115 "Bolting Integrity"	Japan (M. Kanno)
AMP116 "Steam Generators"	China (G. Chun)
AMP117 "Closed Treated Water Systems"	USA
AMP118 "Reactor Vessel Surveillance"	Belgium (M. De Smet)
AMP119 "One-Time Inspection"	Switzerland (J. Held)
AMP120 "Selective Leaching"	USA (A. Hiser)
AMP121 "One-Time Inspection of ASME Code Class 1 Small-bore Piping"	Switzerland
AMP122 "Flux Thimble Tube Inspection"	USA (A. Hiser)
AMP123 "BWR Control Rod Drive Return Line Nozzle"	Mexico
AMP124 "Open-Cycle Cooling Water System"	Switzerland
AMP125 "Buried and Underground Piping and Tanks"	USA/ Japan (M. Kanno)
AMP126 "Boraflex Monitoring"	
AMP127 "Inspection of O-Rings (Related to Refueling) Handling Components"	
AMP128 "Compressed Air Systems"	
AMP129 "BWR Reactor Wall"	
AMP130 "Fire Protection"	
AMP131 "Fire Water System"	
AMP132 "Aboveground Metal Piping"	
AMP133 "Fuel Oil Chemistry"	
AMP134 "External Surfaces"	
AMP135 "Inspection of Interconnecting Components"	
AMP136 "Lubricating Oil Analysis"	
AMP137 "Monitoring of Neutron Flux"	
AMP138 "Use of Consumables"	
AMP139 "Pressure Tube"	
AMP140 "Feeder Pipes"	
AMP141 "CANDU/PHWR RCPB"	
AMP142 "CANDU/PHWR RCPB"	
AMP143 "CANDU/PHWR Valves"	
AMP144 "CANDU/PHWR Pumps"	
AMP145 "CANDU/PHWR Heat Exchangers"	
AMP146 "CANDU/PHWR ISUPT"	



## Attachment C of the AESJ code

- Low-cycle fatigue
- Neutron irradiation embrittlement
- Irradiation assisted stress corrosion cracking
- High-cycle thermal fatigue
- Thermal ageing of duplex stainless steel
- Fretting fatigue
- Insulation degradation of electric and instrumentation equipment
- Reduced strength and shielding performance of concrete

# ***Domestic Steering Team Members for IGALL***

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# Reflection of IGALL Knowledge in the NRA

- The Regulatory Standards and Research Department of Secretariat of Nuclear Regulatory Authority (NRA) has participated in the IAEA IGALL (International Generic Ageing Lessons Learned program) project to investigate technical Knowledge on ageing management of SSCs used in nuclear power plants.
- NRA will issue a Technical Report, which summarizes the results of surveys on the issues of electrical and I&C equipment discussed in the IAEA IGALL, identifying the importance of technical basis for environmental qualification (EQ) program for electrical and I&C equipment.



原子力発電所の電気・計装設備の環境認定  
に関する調査報告

Report on Environmental Qualification of Electrical and I&C  
Equipment for Nuclear Power Plants

技術基盤グループ

原子力規制委員会  
Nuclear Regulation Authority

平成26年XX月  
Month 2014

NTEC-2014-1002

Report on Environmental Qualification of Electrical and I&C Equipment  
for Nuclear Power Plants

Regulatory Standards and Research Department,  
Secretariat of Nuclear Regulation Authority (S/NRA/R)

This report summarizes the results of surveys on the issues of electrical and I&C equipment discussed in the IAEA International Generic Ageing Lessons Learned program. The Regulatory Standards and Research Department of the Secretariat of Nuclear Regulation Authority has participated in to investigate technical information on ageing management of SSCs used in nuclear power plants.

It also provides the results of a study to introduce a review of environmental qualification (EQ) program for electrical and I&C equipment, which has been identified as an item of importance in the survey above mentioned, into the Ageing Management Technical Evaluation (AMTE) review, based on additional surveys on relevant regulatory standard in United States Nuclear Regulatory Commission, codes and standards, research, and technical document of international organization, etc.

The appendix of this report "Guide for Environmental Qualification of electrical and I&C equipment for Nuclear Power Plants" is a document which can be utilized as a guidance by the Secretariat of Nuclear Regulation Authority staff reviewers in performing a review on conformity to relevant regulatory standards, when development and implementation of the EQ program is incorporated to the regulatory requirement of the Nuclear Regulation Authority. It can also be used as acceptance criteria in reviewing adequacy of applicant's ageing management of electrical and I&C equipment subject to the EQ requirement implemented as a part of the EQ program, by being referred in the guidance of the AMTE review.

DRAFT

## ***4. Collaborative Work for Research Strategy Map for Nuclear Safety Research***

## ***Phase 2 Project***

### **JAMPSS**

***Japan Ageing Management Program on Systems Safety  
(October 2011-March 2016)***

***funded by Nuclear Regulatory Authority (NRA)***

- ***Management for Systems Safety (Defense-in-Depth)***
- ***Knowledge-base for Plant Life Management of LWRs***
- ***Research for Ageing Degradation***
  - ***Radiation Embrittlement***
  - ***Irradiation Assisted Stress Corrosion Cracking***
  - ***Cable Degradation***
  - ***Concrete Structure Degradation***
  - ***Other Issues***

# Four Major Categories in the Strategy Roadmap for Ageing Management and Safe Long Term Operation

## (4) International Collaboration

## (3) Establishment of Technical Information Basis

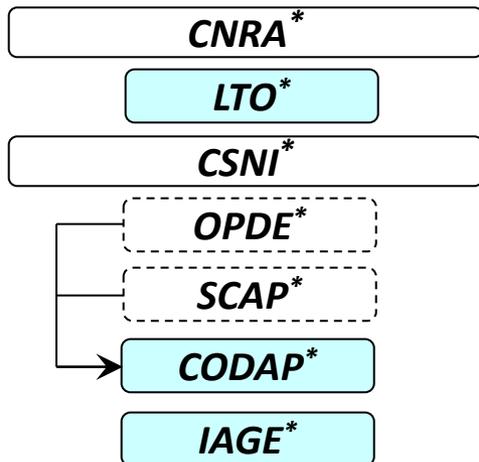
## (2) Establishment of Codes and Standards

## (1) Safety Research on Ageing Degradation

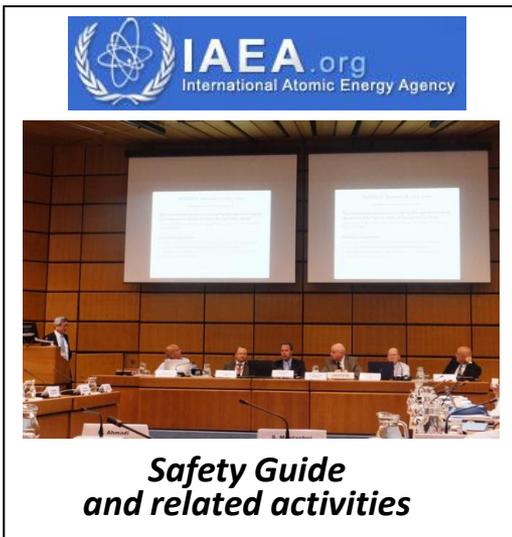
<b>Ageing Phenomena</b>	<b>Current Status</b>	<b>Mid Term Issues</b>	<b>Long Term Issues, Next Generation Reactors</b>
<b>Radiation Embrittlement</b>	Countermeasures are being taken by rule of thumb based on plant data.	Predication methods and monitoring technology will be upgraded.	Advanced designs will be developed based on past operating experience.
<b>Stress Corrosion Cracking</b>	Countermeasures suited to materials in use are being taken. Database is being constructed.	Use of SCC-resistant materials will be ensured. Database will be constructed.	Simulation methods will be established. ISI technology will be upgraded.
<b>Fatigue</b>	Countermeasures suited to materials in use or environment are being taken, and database is being constructed.	Countermeasures suited to materials in use or environment will be taken, and database will be constructed.	Countermeasures suited to materials in use or environment will be taken, and database will be constructed.
<b>Wall Thinning</b>	Countermeasures suited to materials in use are being taken, and database is being constructed.	Mechanism-based predication methods will be established. Risk-based maintenance methods will be established.	Monitoring technology will be upgraded.
<b>Cable Insulation</b>	Countermeasures suited to materials in use are being taken. Database is being constructed.	Deterioration diagnosis technology will be upgraded.	Monitoring technology will be upgraded.
<b>Strength of Concrete</b>	Intensive study is being made on scarcely known fields.	The reliability of integrity evaluation methods will be improved or enhanced. CCV integrity evaluation methods will be established.	Past records of performance will be reflected in the maintenance technology and durability designs for new plants. Recycling methods will be established for replaced structures and materials.

# International Collaboration

## OECD/NEA



## IAEA



## ISaG (International Symposium)

*International Symposium on the Ageing Management and Maintenance of Nuclear Power Plants -*



## Other International Collaboration

**PARENT\***

**RCOP-1 & 2\***

**IGRDM\***

**Bi-lateral collaboration**



- \*1 Committee on Nuclear Regulatory Activities (CNRA)
- \*2 Long Term Operation (LTO Regulatory Guidance Green Booklet)
- \*3 Committee on the Safety of Nuclear Installations (CSNI)
- \*4 OECD Piping Damage Data Exchange (OPDE)
- \*5 SCC and Cable Ageing Project (SCAP)
- \*6 Component Operation Experience, Damage, Aging Program (CODAP)
- \*7 Integrity of Components and Structures (IAGE)
- \*8 Program to Assess the Reliability of Emerging Nondestructive Techniques (PARENT)
- \*9 Feedback of Operational Experiences at NPPs in North east Asia Regional Cooperative Projects (RCOP-1) Non-Destructive Inspection, Evaluation and Repair Techniques (RCOP-2)
- \*10 International Group on Radiation Damage Mechanisms in Pressure Vessel Steels (IGRDM)

## ***5. New Coordinated Research Project in IAEA***

# ***IAEA New Coordinated Research Project***

---

***“The Evaluation of Properties of Structure and Component Materials Utilizing Actual Aged Materials from Decommissioned Reactors”***



***The first Meeting of the CRP on June 11 – 13, 2014 in Vienna***

# ***New IAEA Coordinated Research Project***

---

## ***Title of the CRP***

***Evaluation of structure's and component's material properties utilizing actual aged materials removed from decommissioned reactors for safe LTO***

## ***Project Preparation***

- ***Initial proposal to IAEA prepared by Japan in 2012***
- ***Approved in IAEA in November 2012 for 2014 -2015 biannual plan***
- ***Coordination Meetings***
  - ✓ ***February 2013 – Vienna***
  - ✓ ***July 2013 – Tokyo***
- ***Draft of “Proposal for a New Coordinated Research Project” prepared***

## ***Official Meetings***

- ***First Meeting of the CRP in Vienna in June, 2014***  
***(participants from 12 countries and 2 international institutes)***
- ***Second Meeting in Spain (Madrid and Zorita) in January 13-15, 2015***

# ***Purpose of the New IAEA CRP***

---

***To establish international collaboration aimed to collecting, measuring, recording and analysis of properties of sample materials removed from SSCs of decommissioned NPPs or replaced components, which are subject to physical ageing,***

- ***addressing synergetic effects of combination of different degradation mechanisms in real operational conditions,***
- ***providing basis for comparison with results of laboratory tests and calculations; and***
- ***providing possibility for removing of unnecessary conservatism.***

# ***New IAEA Coordinated Research Project***

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***The specific objective of the CRP will address degradation mechanisms of mechanical, electrical and I&C components and also structures,***

## ***Phase 1 (2014-2017)***

- 1. RPV***
- 2. Core internals***
- 3. Concrete structures, and other non-metallic materials***

## ***Additional Topics in Phase 2 beyond 2017 (can be initiated in 2015 or later)***

- 1. Low-cycle fatigue including environmentally assisted fatigue***
- 2. Degradation of cable insulation and electrical and I&C penetrations***

# **IAEA New CRP: Sub-group 1 on RPV**

---

## ***Subgroup for Coordination of Harvesting and Joint Analysis of RPV Materials***

### ***Scope:***

***To share opportunities for harvesting of RPV materials, discuss and coordinate exchange of specimens for joint or shared analysis and coordinate with other international technical groups focused on aging of RPV materials (such as **IGRDM**, **ASTM**, and **ASME**). Joint technical research would be better accomplished by these international groups that already meet on an annual basis.***

***Possible candidate reactors for the CRP in 4 years:***

***Hamaoka and Zion***

# IAEA New CRP Sub-group 2 on Core-internals

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## *Subgroup for Coordination of Harvesting and Joint Analysis of In-core Materials*

1. *Develop the **Library of the Harvested Materials** is the first priority*
2. *Materials of highest priority and interest will depend on reactor technology but in general **austenitic SS** recognizing that there will be microstructure differences*
  - a. *PWRS/ VVER – SS 304 and SS 316, A 321*
  - b. *BWRs – SS 304, SS 304L, SS 316L, SS347*
  - c. *CANDU – 304L, 403m and pressure tube material Zr-2.5 Nb*
3. *Short-term opportunities based on known harvested materials:*
  - a. *Thimble tubes in Sweden – high dpa SS 316*
  - b. *Zorita baffle plates – 10, 30 and 50 dpa SS 304*
  - c. *UJV – has VVER materials, A 321 from German plant*
  - d. *SCK.CEN – has flux thimble tube materials of 40 to 80 dpa SS316 (Requires approval from Utilities)*
  - e. *Romania, Canada – SS 304L and 403 m, pressure tube material Zr-2.5Nb and Incoloy 800*
  - f. *DOE has baffle former bolts material*

# **IAEA New CRP Sub-group 3 on Concrete and Others**

---

## **Subgroup for Coordination of Concrete Ageing Research**

**Topics for concrete:** Irradiation effects, NDE methods

**Schedule:** Nothing is likely within the first 2 years.

Possible results may be available in the next four years with materials from **Hamaoka, Zion and Zorita**.

**Priority of actions:**

- 1) Pursue Zorita concrete collaborative research;**
- 2) BWR Hamaoka-1 reactor in Japan.**

**Concern:** Testing will not be completed in two years.

Results are restricted, available to participants only.

For Japan reactors some of data will be available in 2 – 3 years.

For Zorita some test coupons will be available in 1Q 2015.

This sub-group would coordinate and interface with other groups such as CSNI and ICIC in OECD/NEA which have similar activities.

# Summary for the IAEA New CRP

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## **Summary and Conclusion in the First Meeting**

- *The common item for the CRP is to have database or library for operational and decommissioned reactors with their plans for harvesting and testing materials with cost estimates.*
- *The **library** should be discussed in the Meetings of the CRP and updated for the future CRP topics.*
- *The participants are welcomed to provide IAEA with their plans for harvesting and testing materials. The data will be uploaded to the CRP web folder and will serve as the basis for discussion.*
- *Communication with international groups such as **OECD/NEA CSNI IAGE, IGRDM and other community** is necessary to define the future CRP topics.*

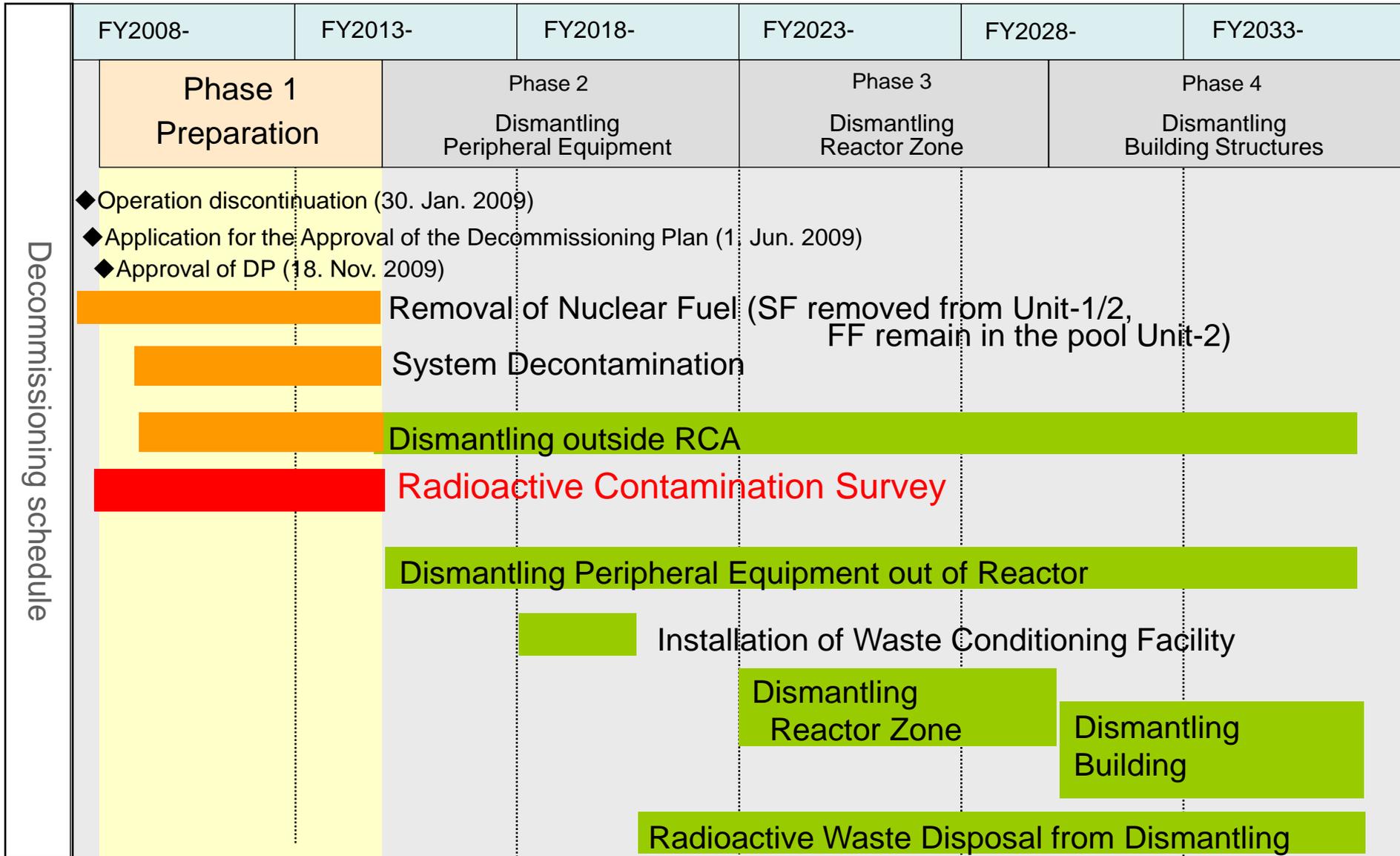
## **Second Meeting**

**January 13-15, 2015** at IETcc (Instituto Eduardo Ttroja de Ciencias de la Construcción) in Madrid & Jose Cabrera NPP in Zorita, **Spain**

# Hamaoka Nuclear Power Station (Chubu)

	Unit 1	Unit 2	Unit 3	Unit 4	Unit 5
Reactor type	BWR-4		BWR-5		ABWR
Thermal output (MWt)	1593	2436	3293	3293	3926
Containment type	Mark-1		Advanced Mark-1		RCCV
Power output (MWe)	(540)	(840)	1100	1137	1380
Total power output (MWe)			3617		
Start of construction			March 1971	March 1974	November 1982
Start of commercial operation	March 1976	November 1978	August 1987	September 1999	January 2005
Present status	Under decommissioning (Nov. 2009~)		Operation Outage	Licensing Safety Review	Operation Outage
	Countermeasures against new safety regulations are being implemented				

# Decommissioning Schedule in Hamaoka #1 and #2



# ***Radioactive Characterization in RPV***

## **Target Radionuclides : C-14, Co-60, Ni-63, I-129, Cs-137**

C-14: Most important nuclide in safety assessment for disposal of **long half life nuclide**

Co-60: Easy to analyze representative corrosion product

Ni-63: For **verification of the activation calculation** (representative corrosion product)

I-129: For verification of the activation calculation (representative fission product)

Cs-137: Easy to analyze **representative fission product**

Parent elements are also subject to analyze for activation calculation.

Radiochemical laboratory will be ready in Hamaoka NPS.



Cumulative neutron exposure can be evaluated at the target points, as a results of the above investigation.

# Sampling Point in and around the RPV (Unit 1)

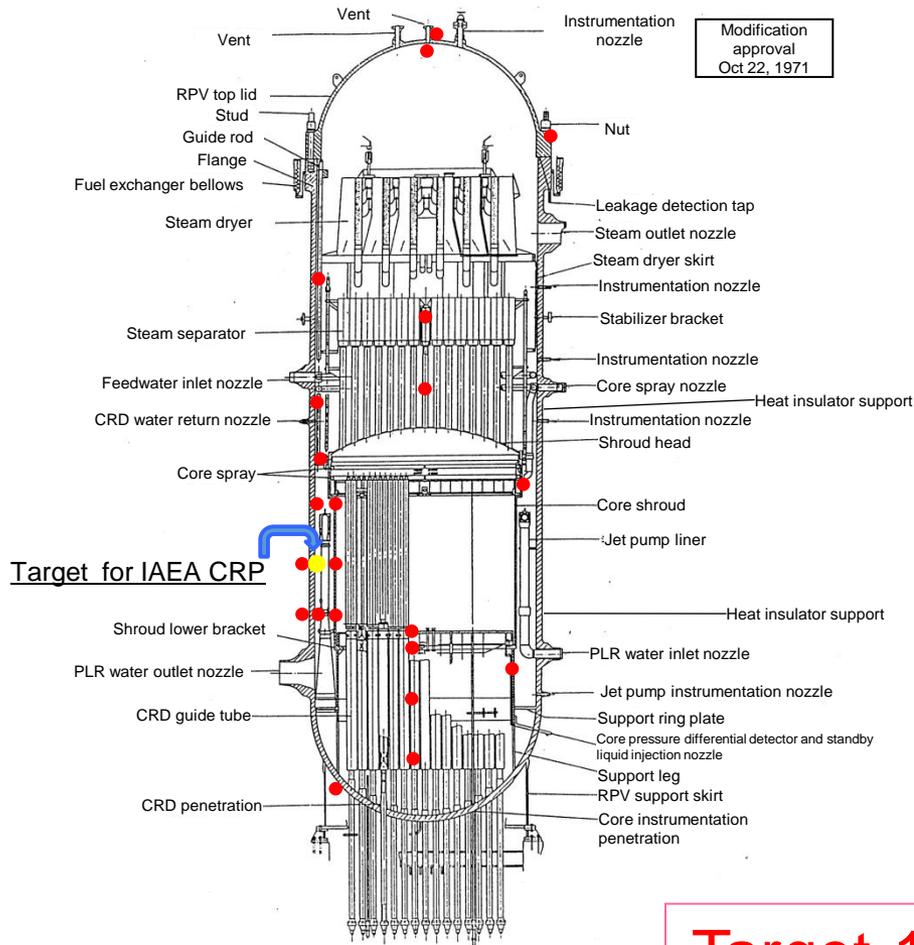
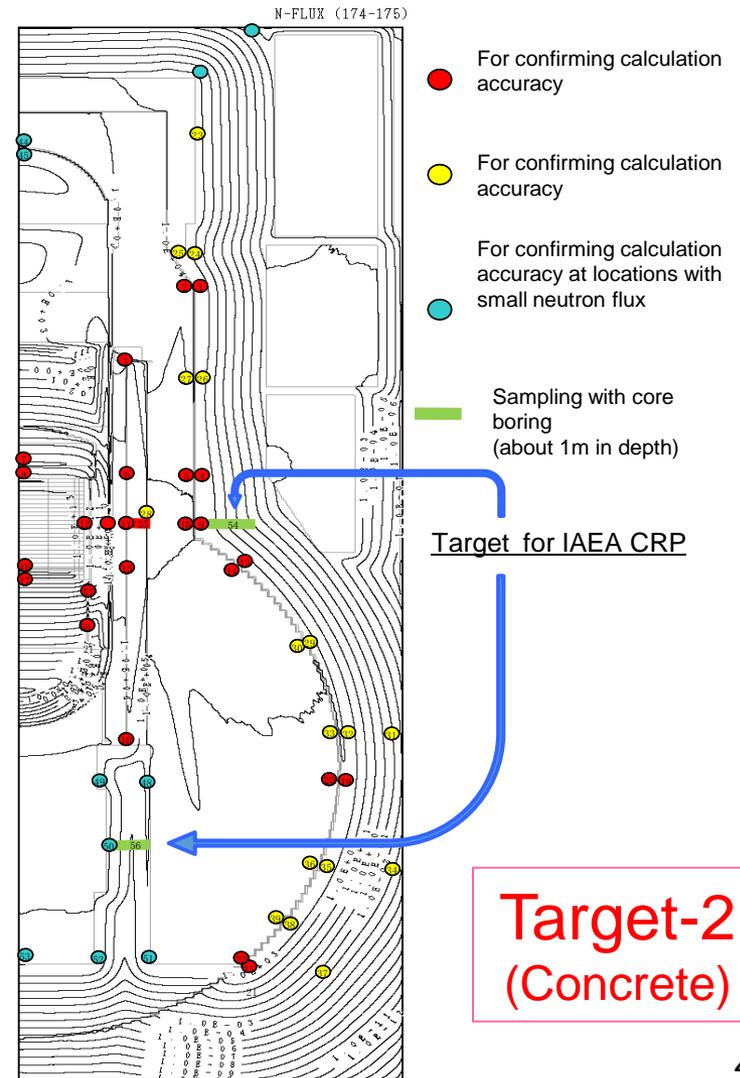


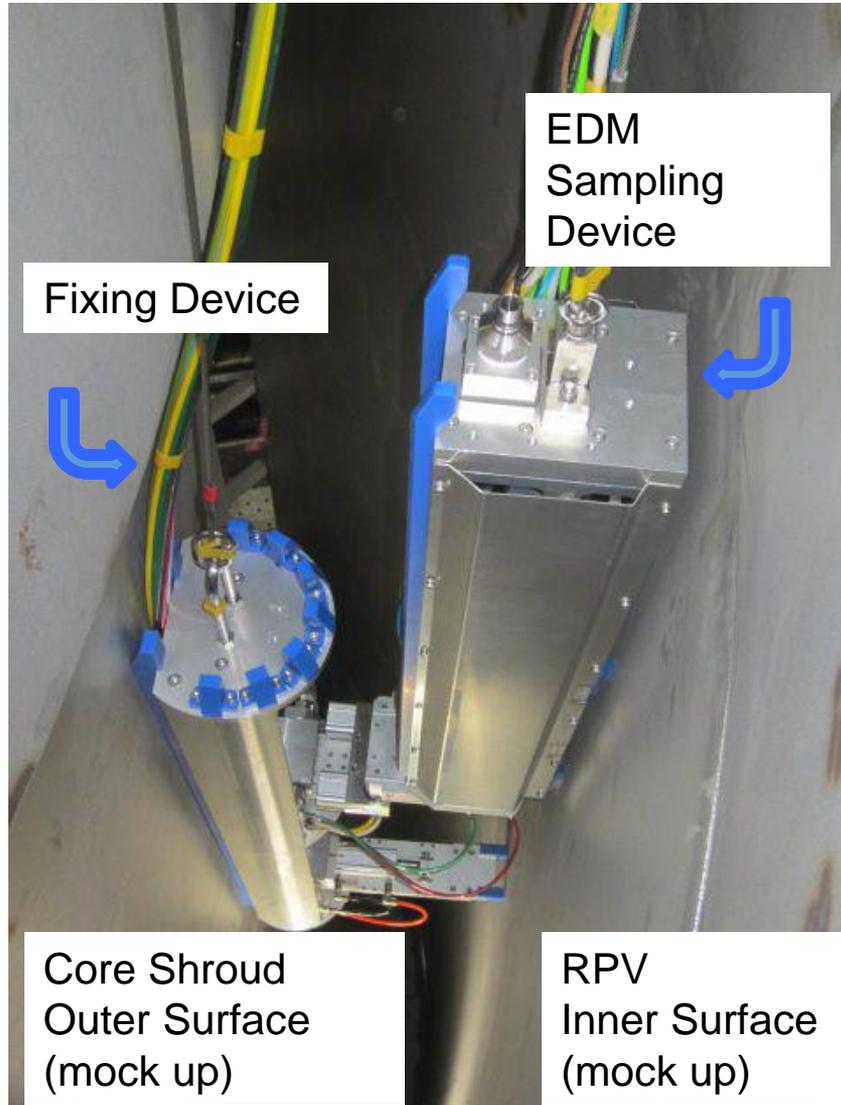
Figure 31-1 Reactor internal structure schematic diagram

**Target-1  
(RPV)**

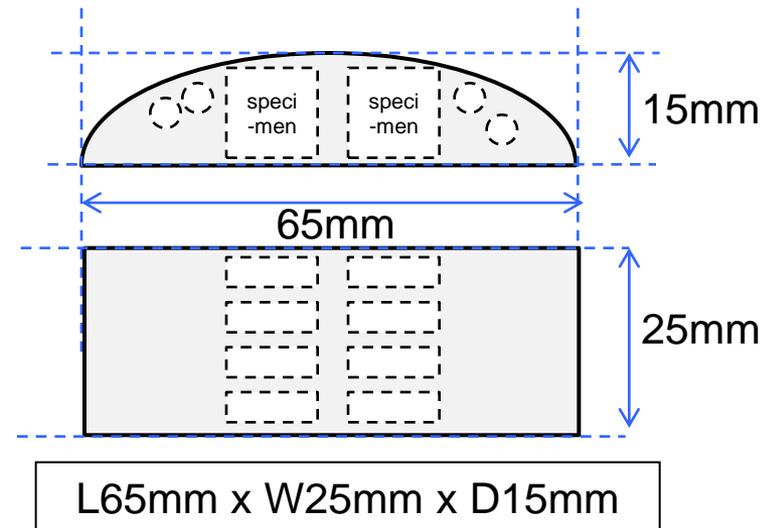


**Target-2  
(Concrete)**

# Sampling Device for RPV (Unit 1)



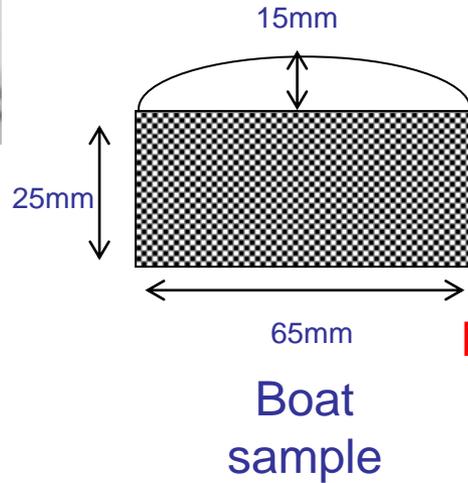
RPV (Sample)



# ***Specification of the RPV in Hamaoka Unit-1***

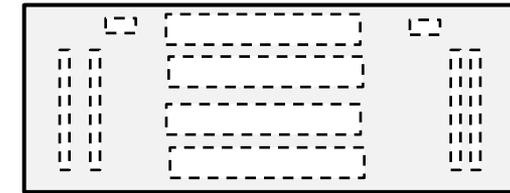
	<b>Specification</b>
<b>Reactor type</b>	<b>BWR-5</b>
<b>Thickness</b>	<b>120mm</b>
<b>Material of belt line</b>	<b>SA533 Gr.B C<math>\phi</math>.1</b>
<b>Manufacturing method</b>	<b>Welding of rolled plates</b>
<b>Material of inner clad</b>	<b>309 stainless steel (weld metal)</b>
<b>Major impurities</b>	<b>0.09wt%Cu , 0.56wt%Ni</b>
<b>Operated term</b>	<b>Mar. 1976 ~ Jan. 2009</b>
<b>Operated time</b>	<b>144,570 hours (approx. 16.5 years)</b>
<b>Neutron fluence (max.)</b>	<b><math>1.5 \times 10^{18}</math> n/cm<sup>2</sup> (E&gt;1MeV)</b>

# Specimen for Task-1 for the Validation of Prediction Formula and Previous Embrittlement Management

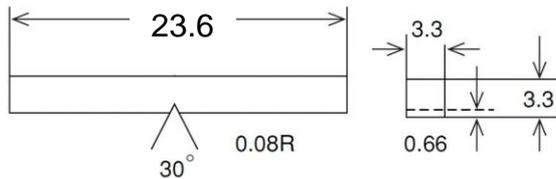


Machining to samples

Top view



Front view



1/3-Charpy specimen  
(3.3 × 3.3 × 23.6mmL)

Charpy Test



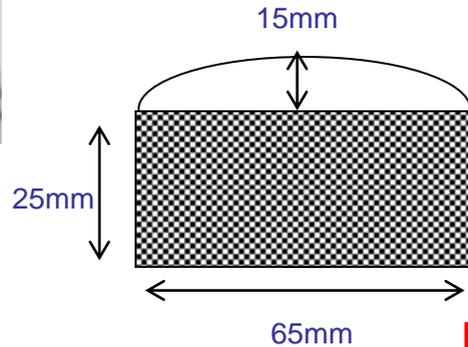
APT needle specimen  
(1mm)



TEM and PAS disk  
(3mmΦ)

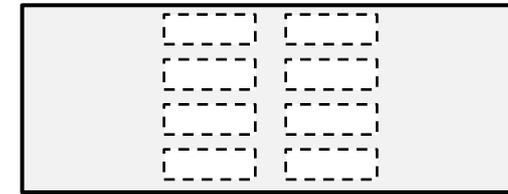
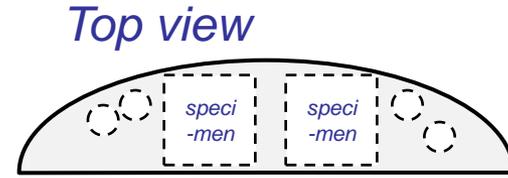
Microstructural Observation

# Specimen for Task-2 to Confirm the Safety Margin

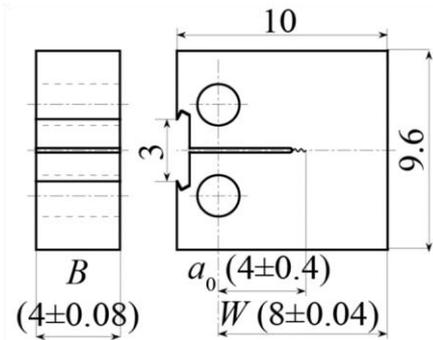


Boat  
sample

Machining to  
samples



Front  
view



CT specimen (10 × 10 × 4mmt)  
Toughness tests



Tensile specimen  
(50 × 12 × 1mmt)

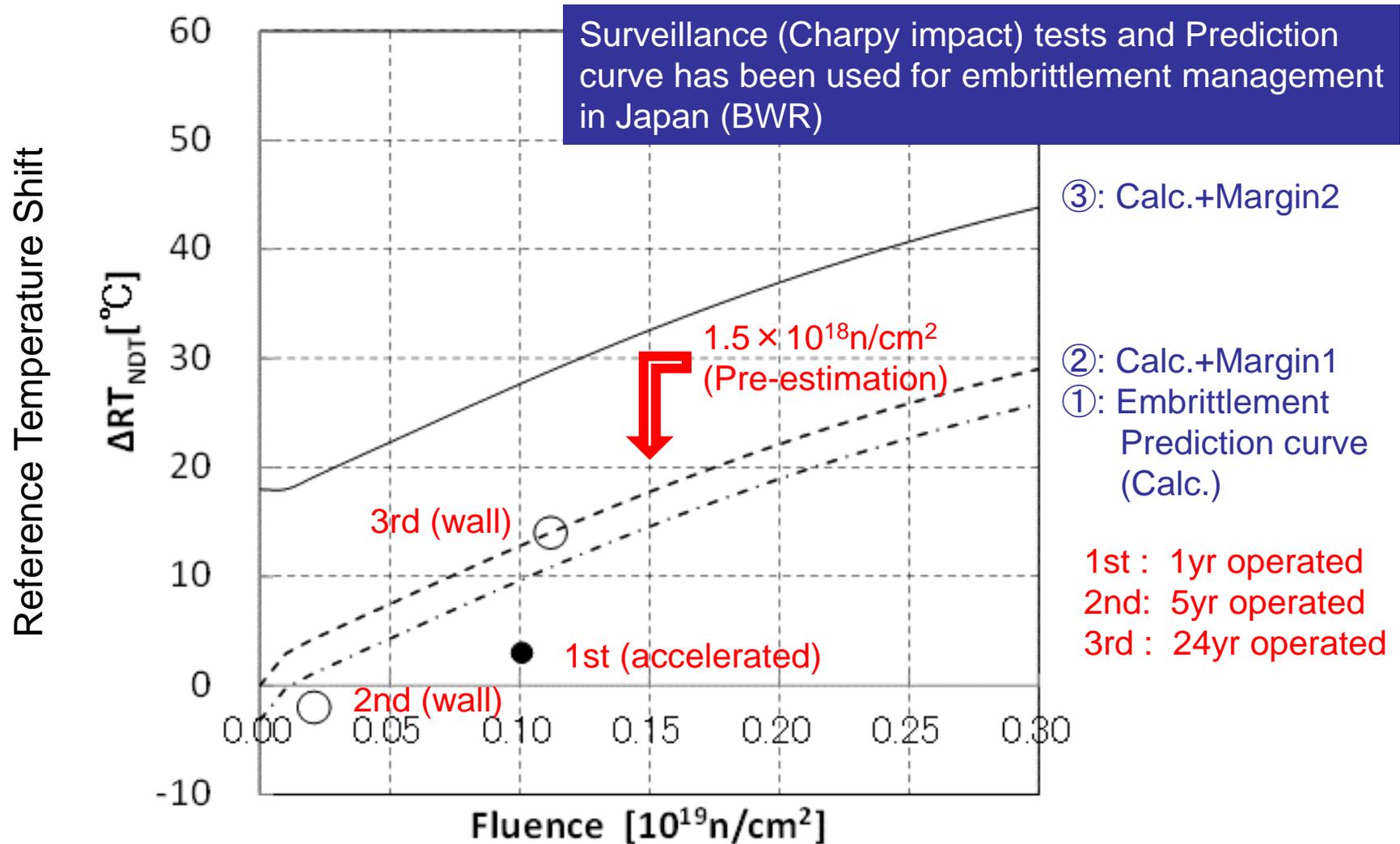
Tensile tests

# ***The Objective of Hamaoka Unit-1 Project on RPV by Chubu Electric Power Co.***

**Examination of the samples from the RPV materials of Hamaoka Unit-1 of Chubu Electric Power Co. Inc,**

- 1) To validate past irradiation embrittlement management based on surveillance specimens by confirming that  $\Delta RT_{NDT}$  of the RPV in Hamaoka unit-1**
- 2) To Assess and verify the correctness and representativeness of prediction curve formula by conducting microscopic observation**
- 3) To confirm safety margin in present management for RPV irradiation embrittlement by fracture toughness measurement**

# Managing Radiation Embrittlement in Hamaoka Unit-1



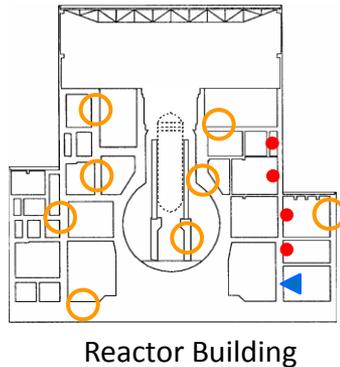
# Schedule for RPV Examination in Hamaoka

		2014	2015	2016	2017	2018
IAEA Meetings				△	△	
Task #1-1	Mechanical properties measurement <1/3 Charpy impact test>	Sampling	Unirr.	Irr.		
Task #1-2	Microstructure observation (APT etc.) followed by comparison with prediction formula		Unirr.	Irr.		
Task #2	Mechanical properties measurement <Fracture Toughness test>			Unirr.	Irr.	

Unirr.: Test of un-irradiated specimens, Irr.: Test of irradiated specimens

# Materials Database for Concrete Structure

## (A) Data Collection from Core Sampling and NDT



### (A-1) Core Sampling

○ Candidate Sampling Locations

- Reactor Building
  - RPV Pedestal / Primary Shielding Wall
  - Spent Fuel Storage Pool / Walls etc.

### (A-2) Nondestructive Testing

- Impact elastic wave method

▶ Impactor  
• Detector

Example of Materials Database

Properties	RPV Pedestal			Primary Shield Wall			Seismic Wall		
	a1	a2	a3	b1	b2	b3	c1	c2	c3
Strength/Stiffness									
Carbonation Depth									
Solt Content									
Density									
Free Water Content									
Chemical Bound Water Content									
Pore Size Distribution									
X-Ray Dffraction									
Wave Propagation Rate									
...									

Sampling Locations  
a1~a3  
b1~b3  
c1~c3

# Comprehensive Soundness Evaluation Method on Compressive Strength of Concrete

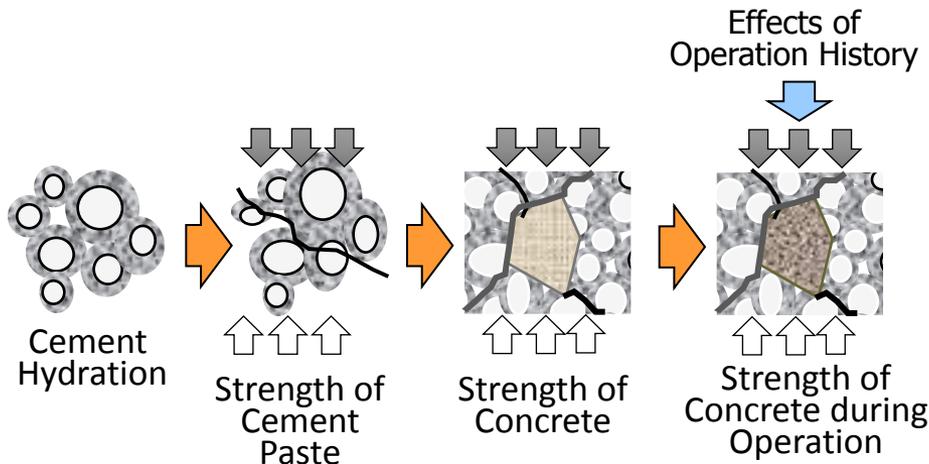
## (A) Data Collection from Core Sampling and NDT

- (A-1) Core Sampling
- (A-2) Nondestructive Testing

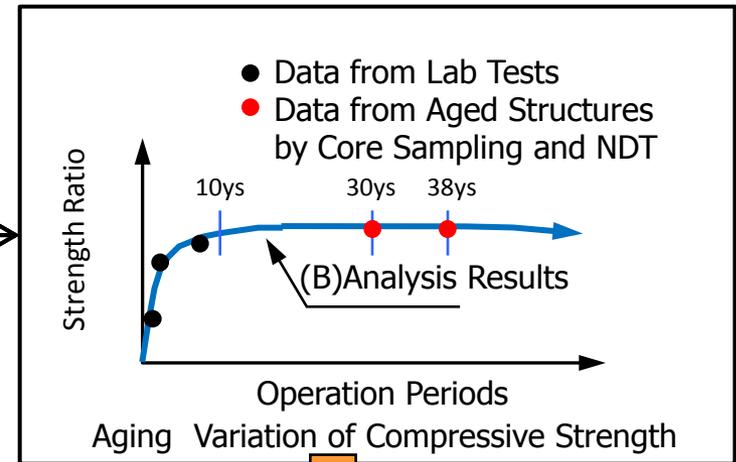
## (B) Numerical Analysis Approach

### Analysis Conditions of Reactor Building

- Material Properties (cement, aggregates etc. )
- Mix Proportion, Dimensions of Members
- Operation History (temp., humidity, flux, dose rates



## (C) Validation of Analysis Approach



## (D) Comprehensive Soundness Evaluation

- Core Sampling
- Nondestructive Testing
- Numerical Analysis Approach
- Assessment of Concrete Strength  
Strength Evaluated  $\geq F_c$

# Schedule for Concrete Examination

	2014	2015	2016	2017	2018
IAEA Meetings		↑ △	↑ ↑ △	↑ ↑ △	
(A-1)Performing <b>Core Sampling</b> and <b>Property Tests</b> for Database	Planning	Core Property Data	Core Property Data	Core Property Data	
(A-2)Performing <b>Nondestructive Test</b> and Obtaining <b>Property Data</b> for NDT Validation		Planning	Property Data by NDT	Property Data by NDT	
(B,C)Performing <b>Numerical Analysis Approach</b> for Validation and Upgrading		Planning		Outline of Approach	
(D)Proposal of <b>Comprehensive Soundness Evaluation Methods</b>		Planning		Outline of Method	

## ***6. The Fukushima Daiichi Accident and the Effect of Ageing on the Accident***

***Seismic Effects on the Integrity of Components considering  
Ageing Degradation in Fukushima Dai-ichi***

# Water Level in Unit 1

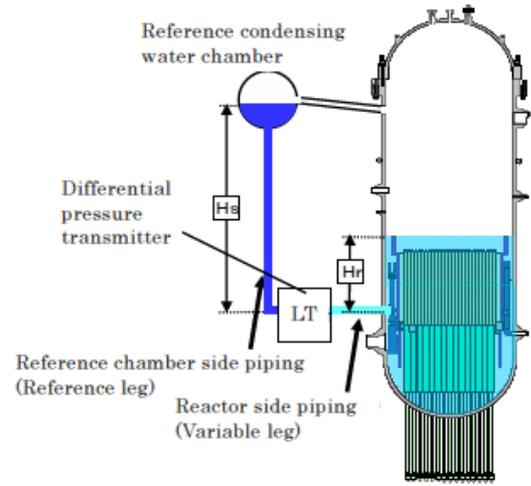
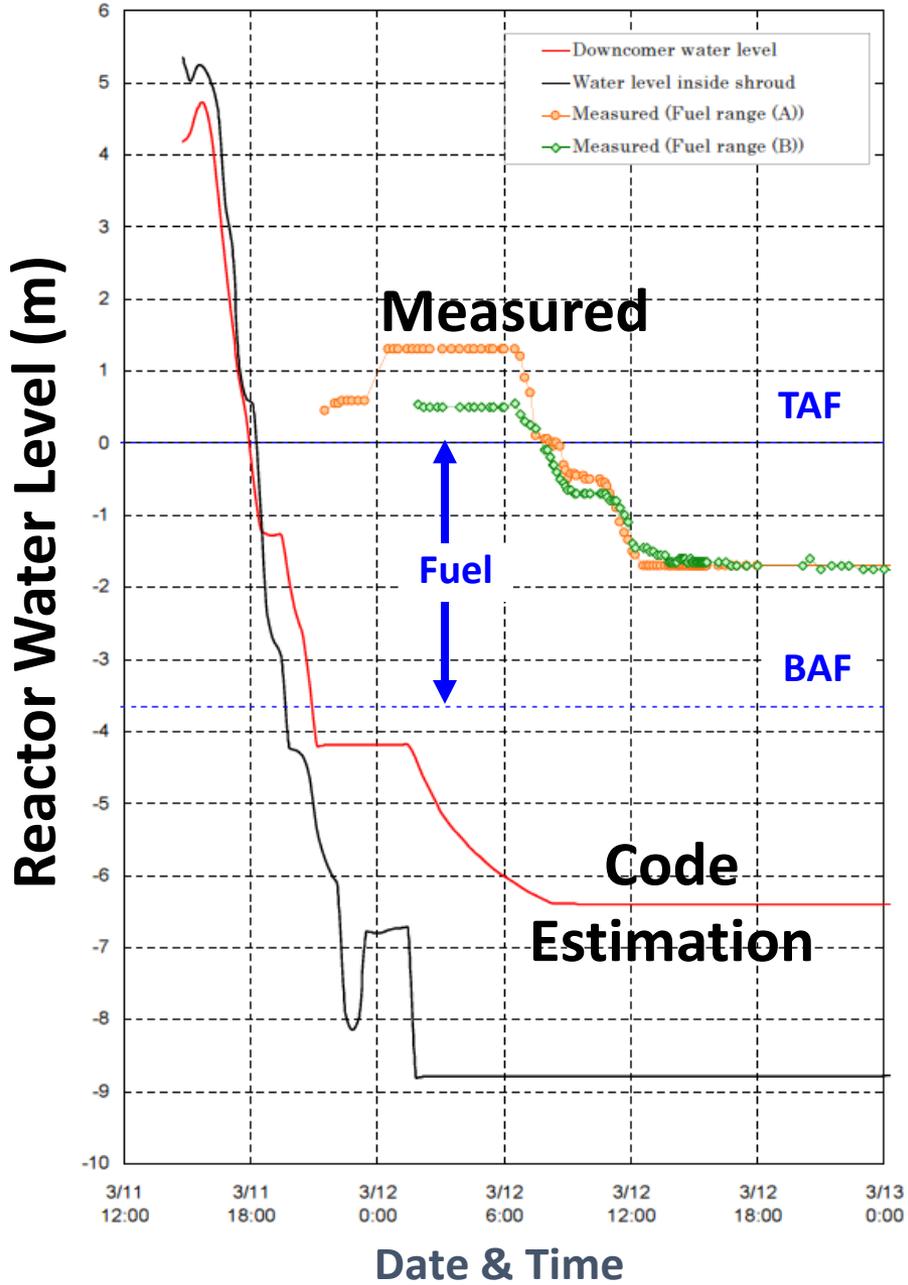
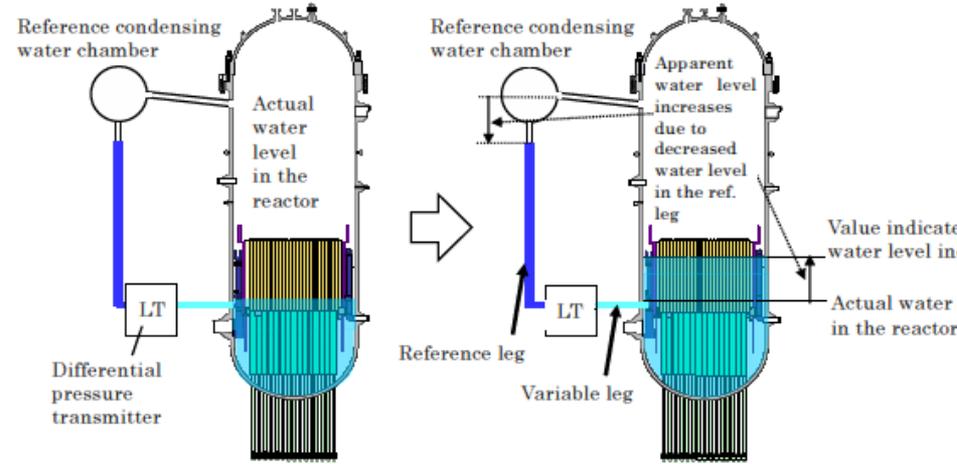
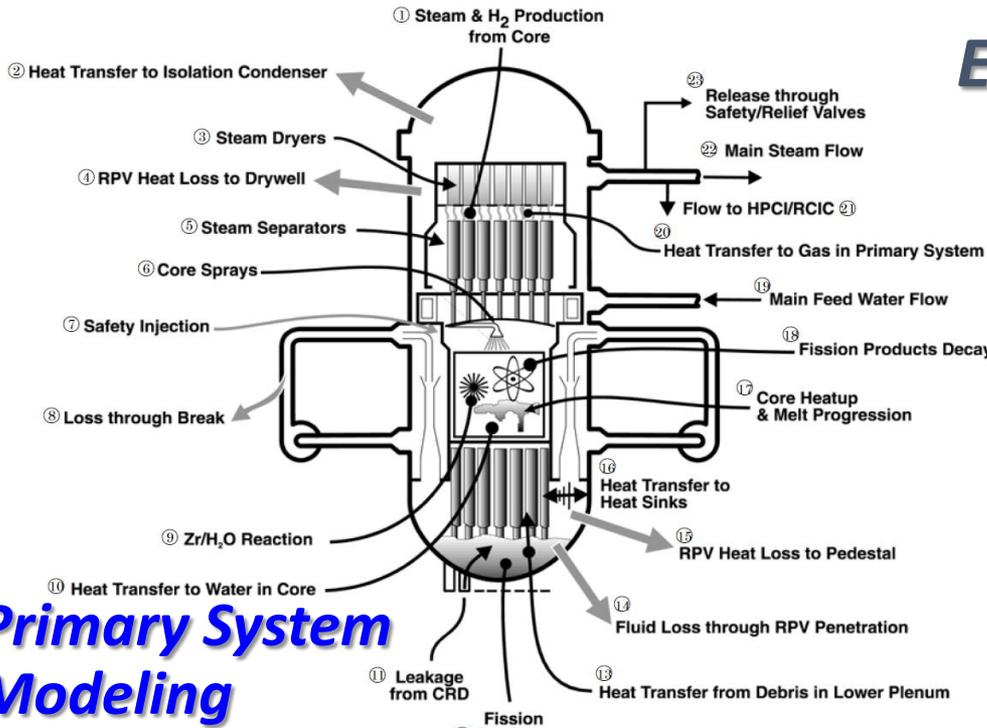


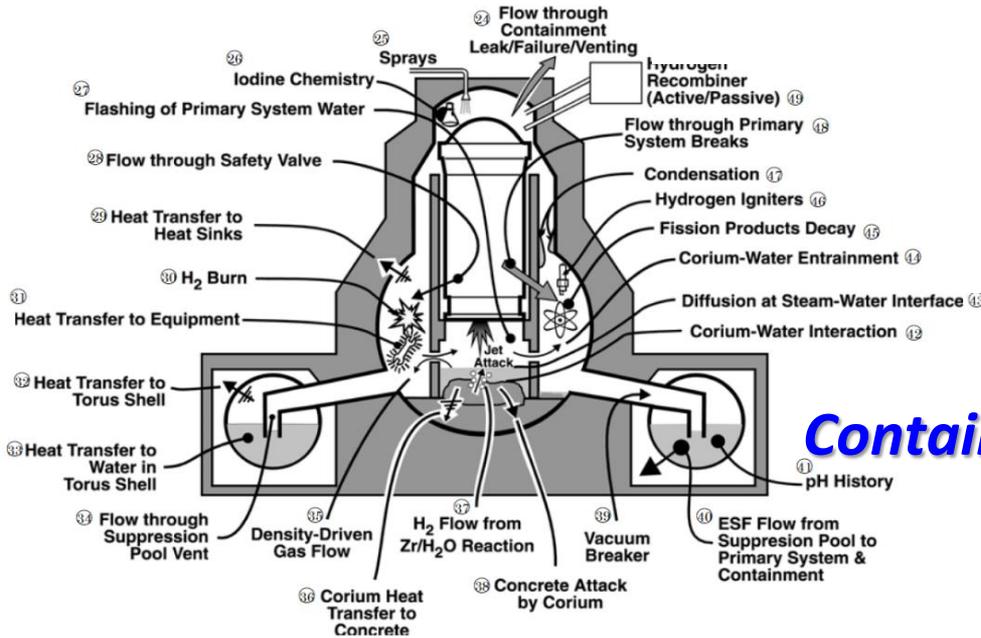
Figure 1 Fuel range water level indicator



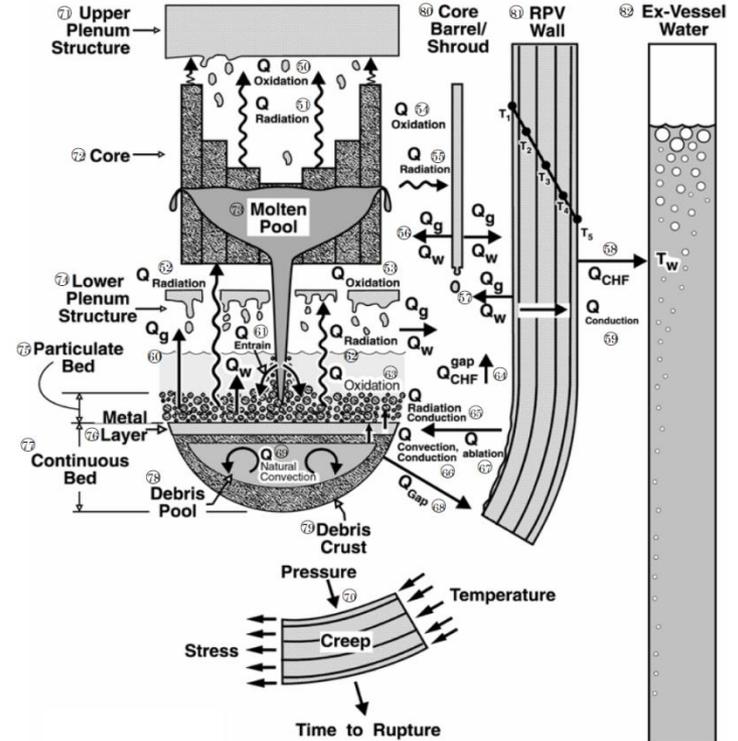
# Elementary Models in the Accident Analysis Code



## Primary System Modeling



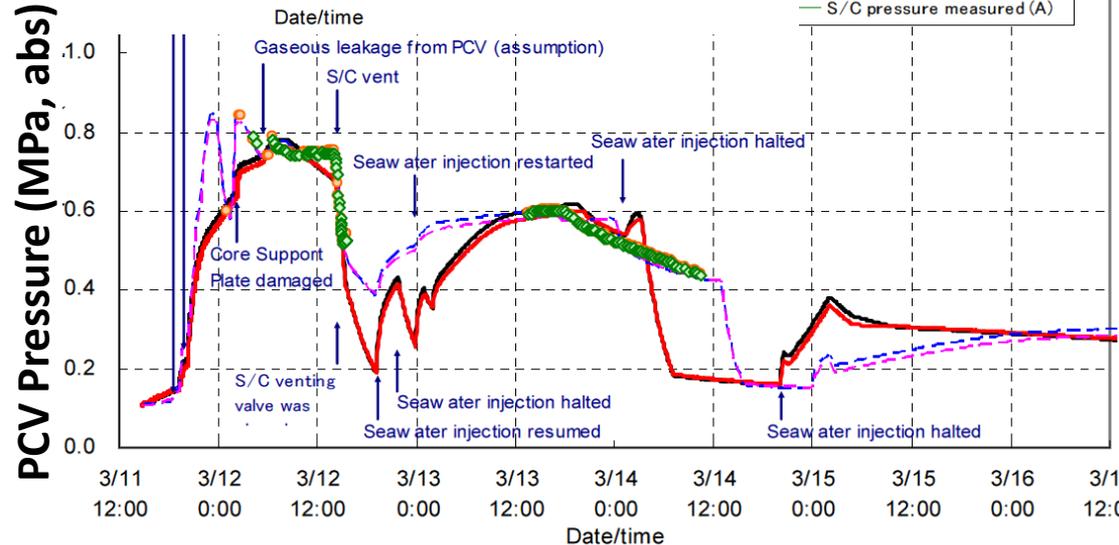
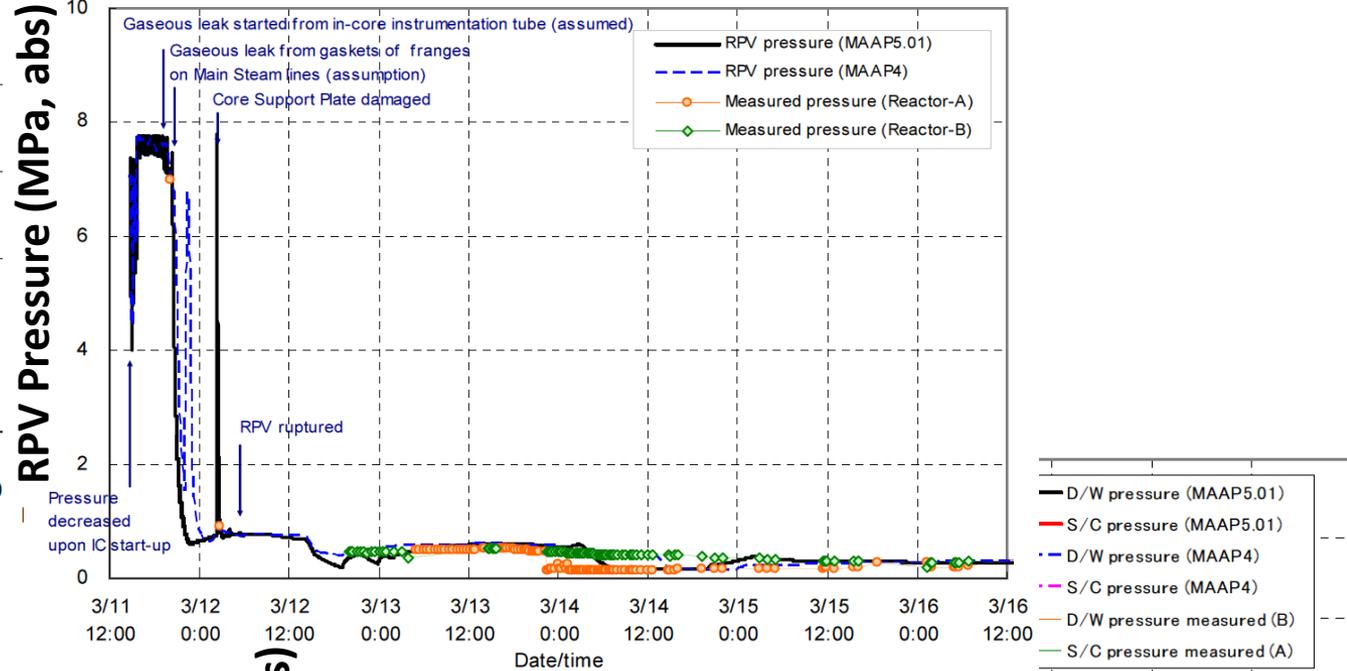
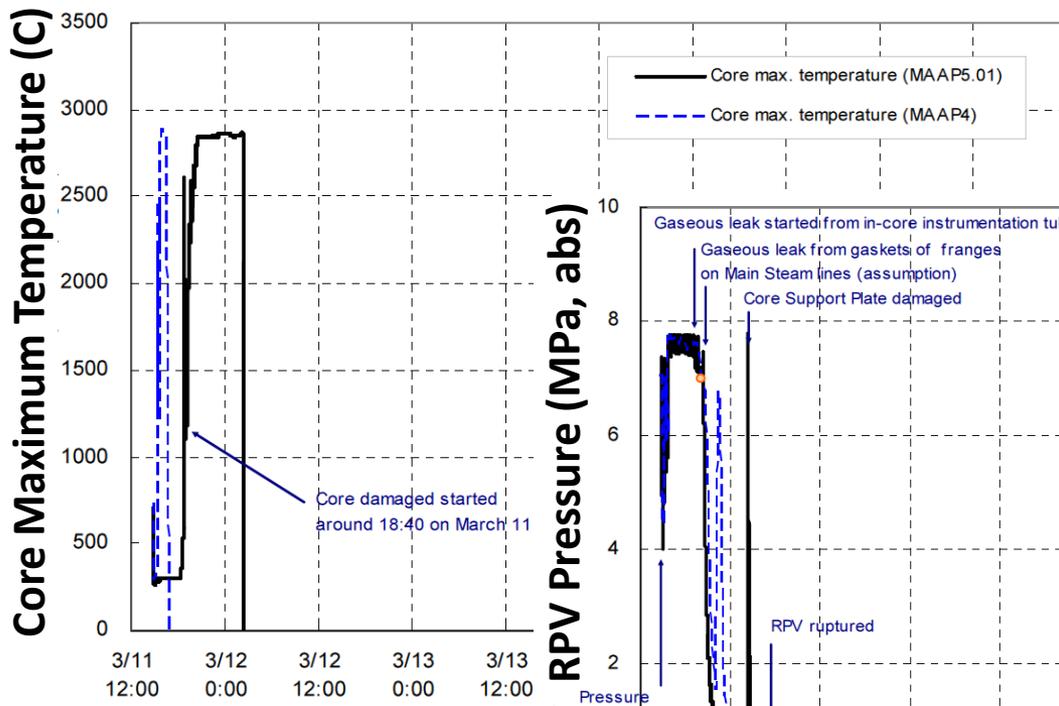
## Containment Modeling



## Lower Plenum Modeling

Example of the MAAP code

# Estimated Parameters in Unit 1

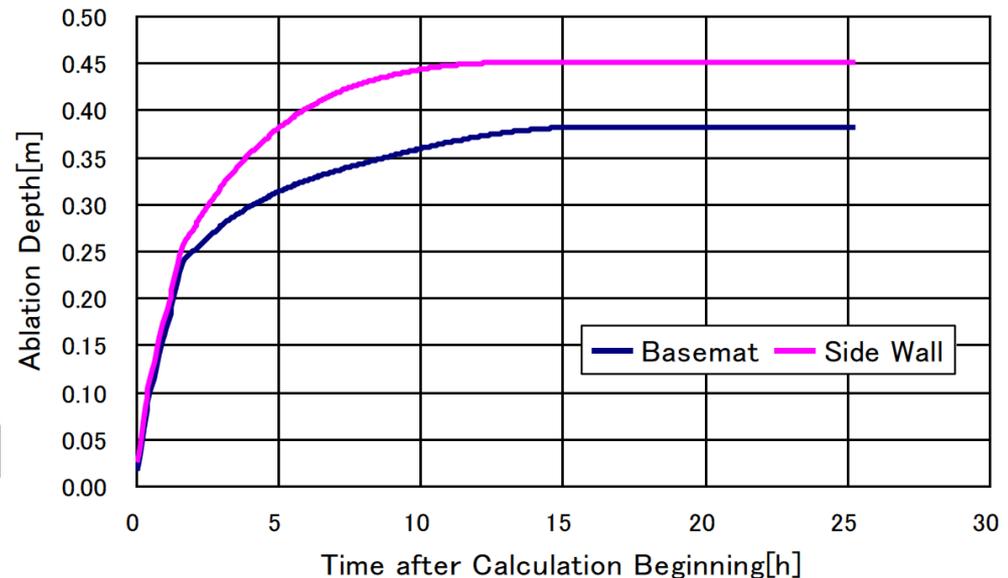
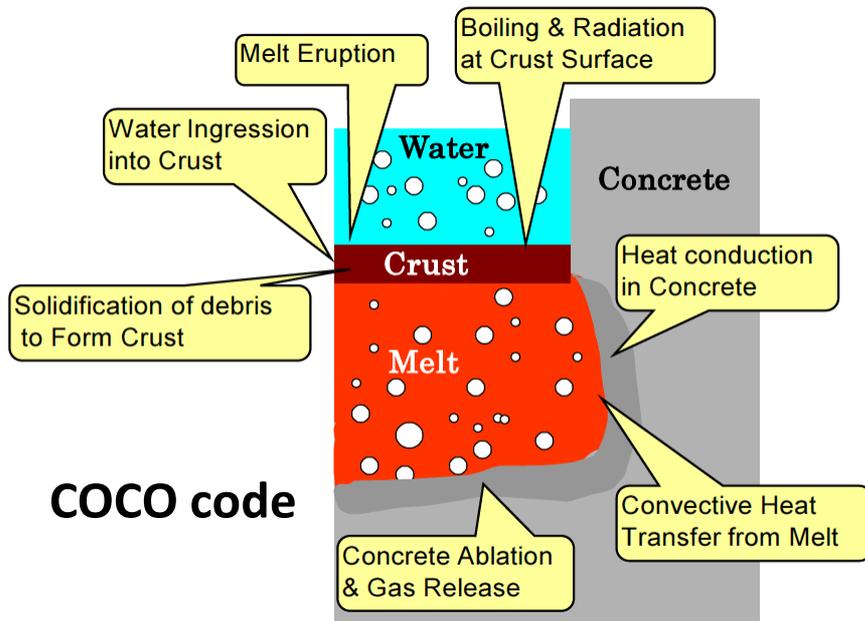
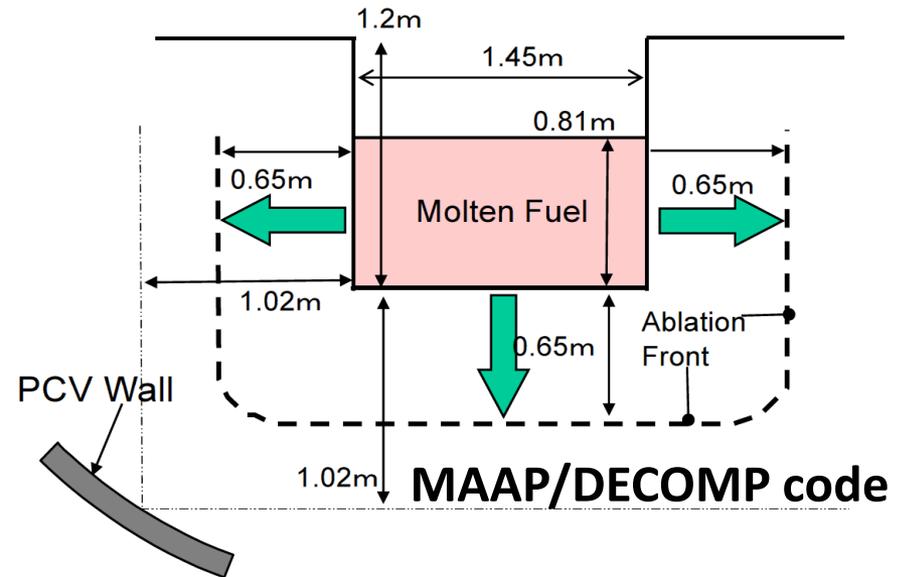


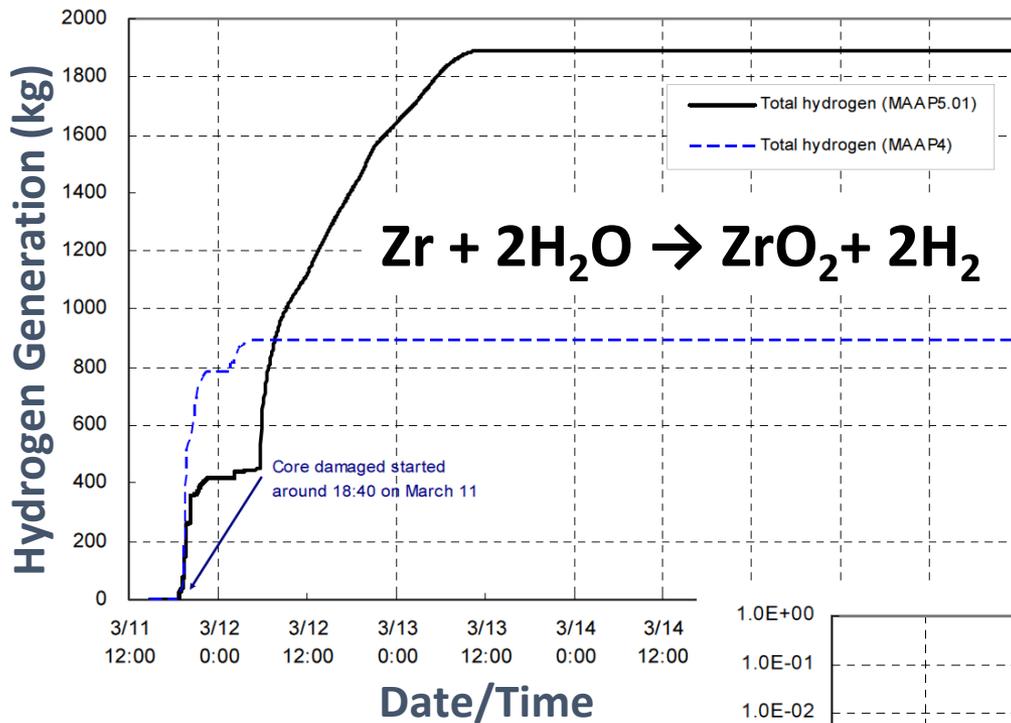
Some measure values used for model justification are also shown.

# MCCI: Molten Core Concrete Interaction in Unit 1

## Estimated Ablation Depth

- ✓ MAAP/DECOMP code  
0.65 m
- ✓ COCO code  
0.45 m : Basement  
0.38 m : Side Wall



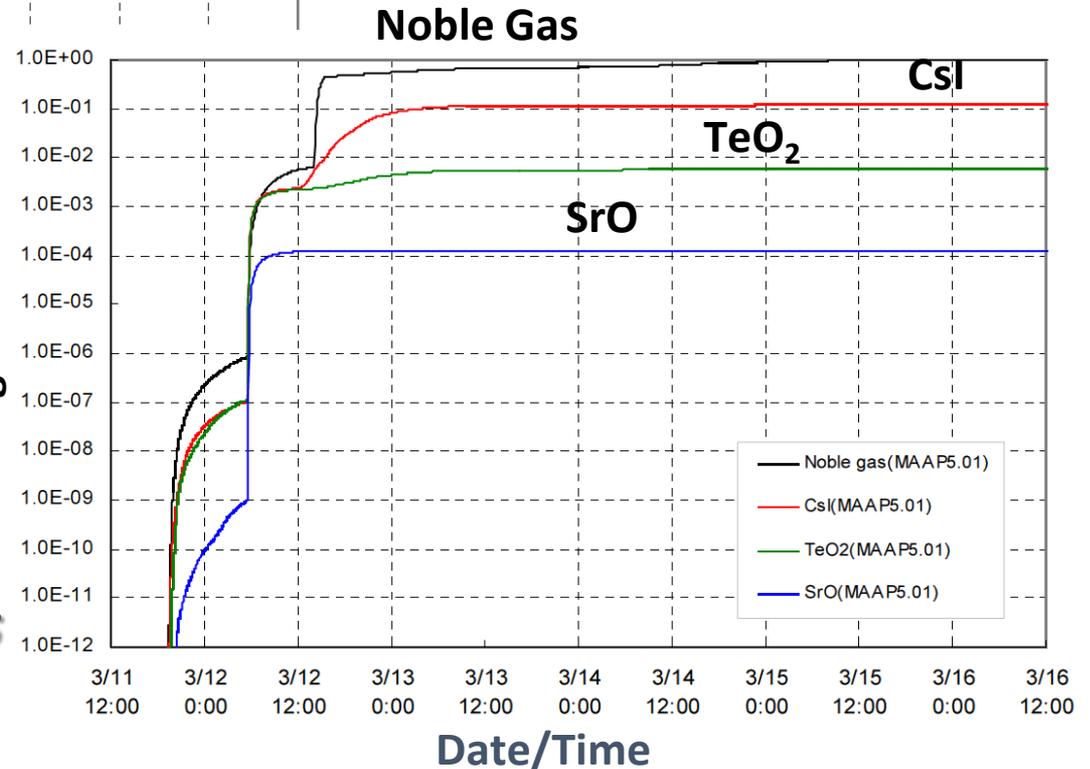


## Generation of Hydrogen in Unit 1

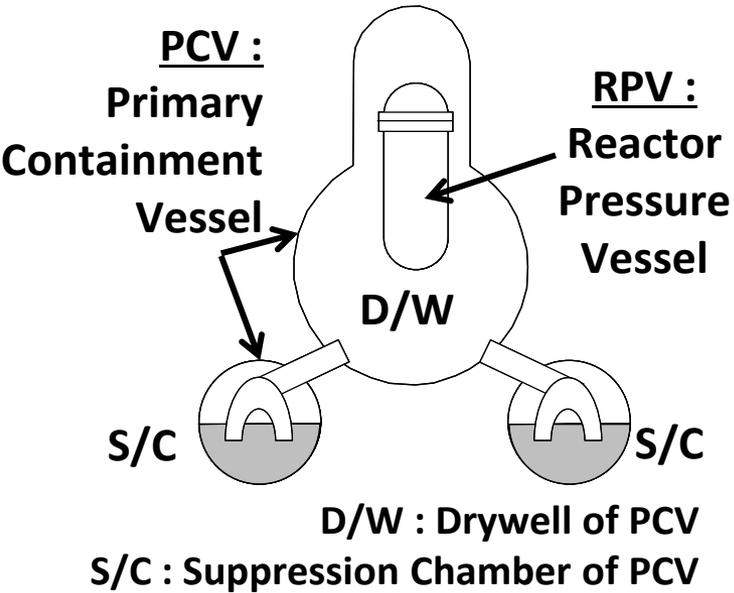
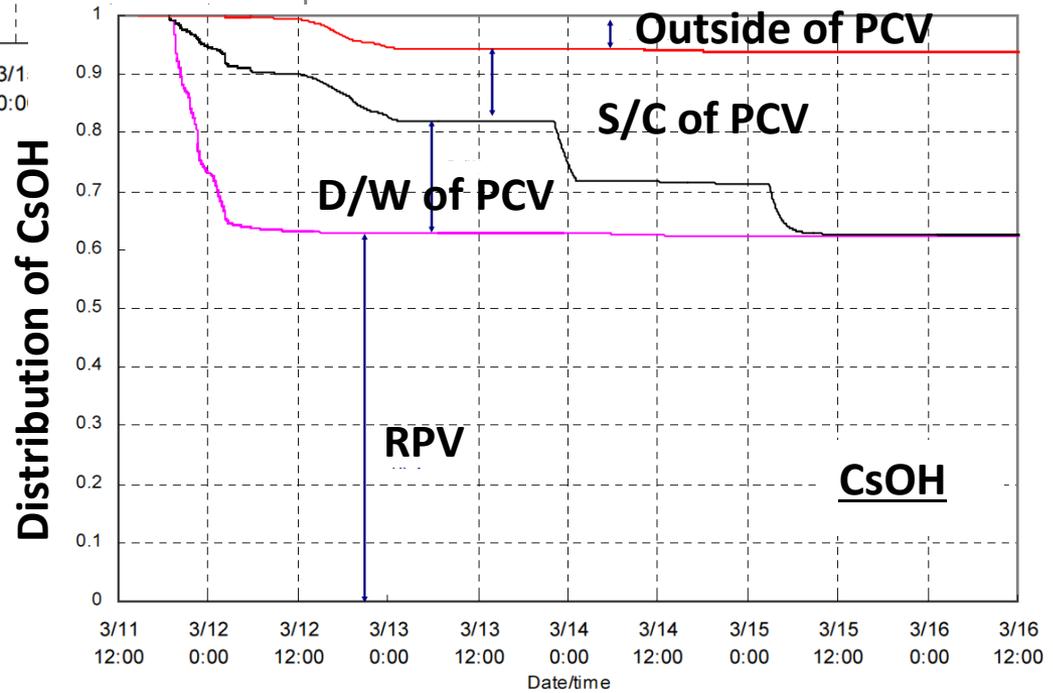
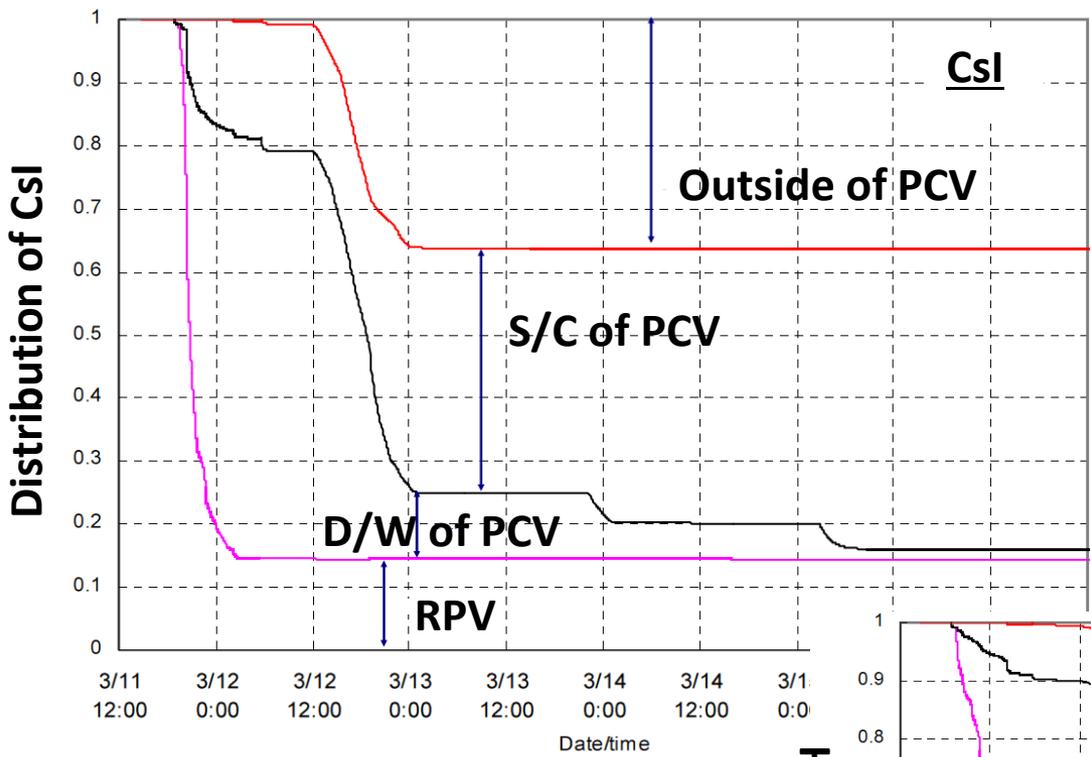
Zr-H<sub>2</sub>O reaction at high temperature is main source of H<sub>2</sub>.

## Release Rates of Fission Products in Various Chemical Forms in Unit 1

**FP Release Ratio in log scale**



# Distribution of CsI and CsOH in Various Location in Unit 1 (MAAP code)

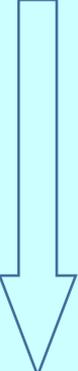


# ***Fundamental Safety Functions for Light Water Reactors***

**Shutting Down by interrupting fission chain reaction through insertion of control rods**

**Cooling Down the fuels through continuous water injection and circulation**

**Confining radioactive materials inside the boundary (RPV and PCV)**

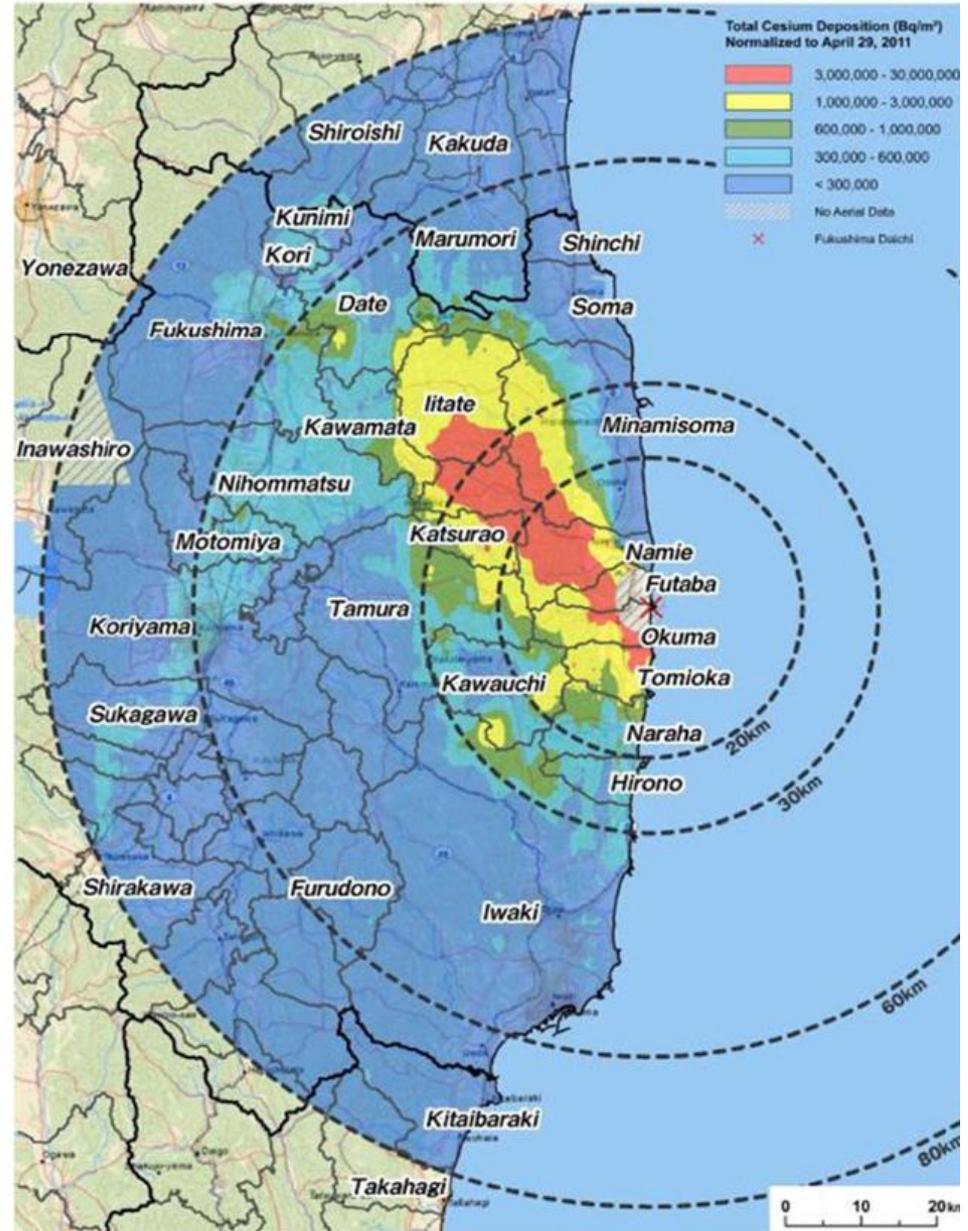
- 
- 1. High-pressure Coolant Injection**
  - 2. Depressurization**
  - 3. Low-pressure Coolant Injection**
  - 4. Heat Removal to Ultimate Heat Sink (Ocean)**
- 

# Cesium Deposition

April 29, 2011

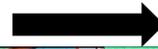
## Aerial Measuring Results

Joint US / Japan Survey Data

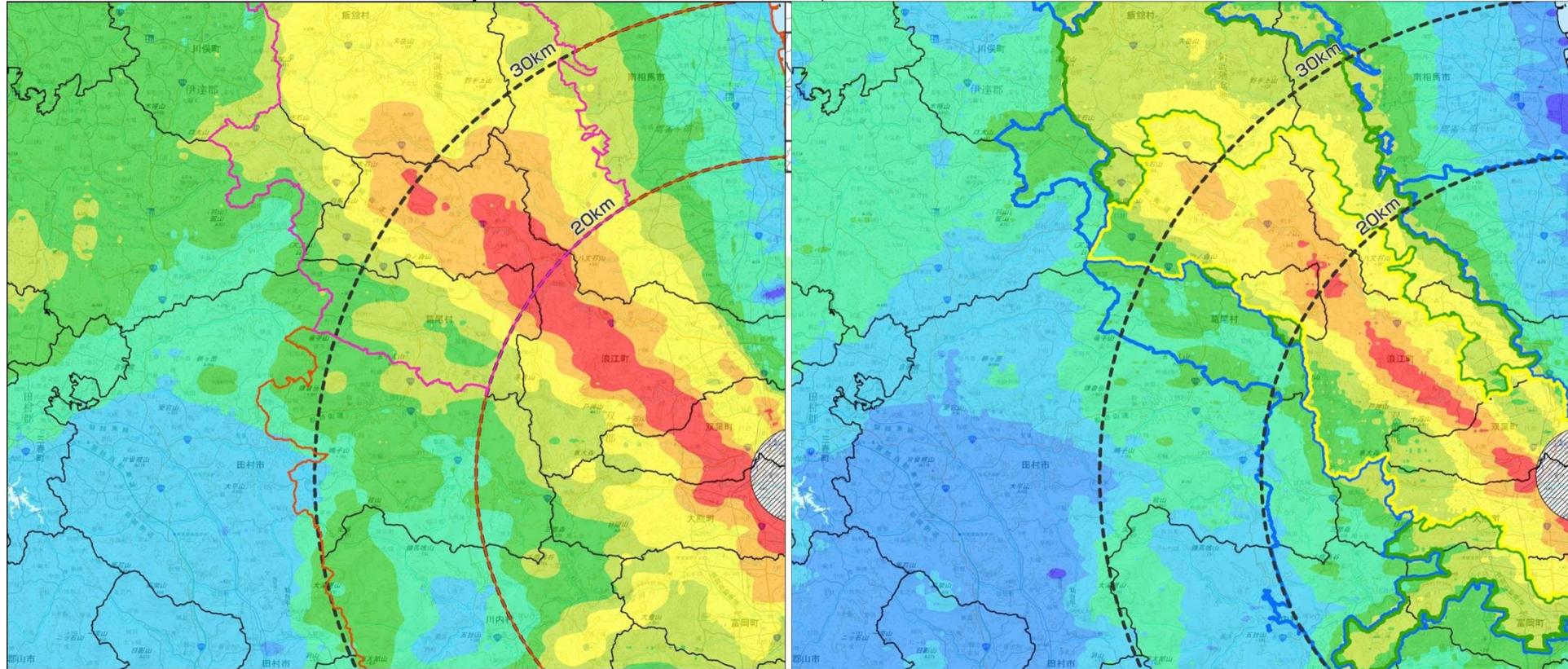


# Monitoring of Radiation Dose Change in 2 Years

November 5, 2011

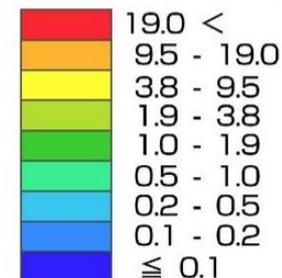


November 19, 2013



Decrease of Dose Rate through

- (1) Radioactive decay
- (2) Reduction by natural dispersion
- (3) Decontamination action



$\mu$  Sv/hr

***English version of the AESJ Final Report will be available very soon.***

Editor :  
Atomic Energy Society of Japan



# The Fukushima Daiichi Nuclear Accident

Final Report of AESJ Investigation Committee

This report was completed by the AESJ, Japanese academia in nuclear science/engineering, with concerted effort by specialists.

It covers all aspects of technical issues of the Accident.

New  
Book

# Maximum Acceleration Observed in Fukushima Daiichi

Location of Seismometer (bottom floors of the reactor buildings)		<u>Records</u>			Max. Response Acceleration to the <u>Design Basis</u> Ground Motion, S <sub>s</sub> (Gal)		
		Max. Acceleration (Gal)			NS	EW	UD
		NS	EW	UD			
Fukushima Daiichi	Unit 1	460※ <sup>1</sup>	447※ <sup>1</sup>	258※ <sup>1</sup>	487	489	412
	Unit 2	348※ <sup>1</sup>	550※ <sup>1</sup>	302※ <sup>1</sup>	441	438	420
	Unit 3	322※ <sup>1</sup>	507※ <sup>1</sup>	231※ <sup>1</sup>	449	441	429
	Unit 4	281※ <sup>1</sup>	319※ <sup>1</sup>	200※ <sup>1</sup>	447	445	422
	Unit 5	311※ <sup>1</sup>	548※ <sup>1</sup>	256※ <sup>1</sup>	452	452	427
	Unit 6	298※ <sup>1</sup>	444※ <sup>1</sup>	244	445	448	415

※<sup>1</sup>: Each recording was interrupted at around 130-150(s) from recording start time

※<sup>2</sup>: 1 Gal=0.01m/s<sup>2</sup> , 981 Gal=1G

Source: Added to “The impact of the 2011 off the Pacific coast of Tohoku Earthquake to Nuclear Reactor Facilities at Fukushima Dai-ichi Nuclear Power Station” (Sep. 9, 2011, revised Sept. 28, 2011, TEPCO)

# ***Effects of Ageing in the Fukushima Accident ?***

- **As the results of the evaluation based on the knowledge obtained so far, it is quite unlikely that there was an effect of ageing degradation on loss of functions in SSCs important to safety due to the ground motion by the earthquake.**
- **It is also unlikely that ageing degradation phenomena have caused the occurrence and enlargement of the Fukushima Daiichi accident.**
- **However, as it is difficult to confirm the status of equipment at this moment, additional investigation will be needed, when new knowledge is obtained in the future.**

**Report of the Expert Meeting on Ageing Management in NISA  
(February 7, 2012)**

# ***Evaluation of Additional Effect of the Earthquake on Low Cycle Fatigue of Main Steam Line Piping of Unit 1***

<b>Cumulative Fatigue Coefficient for 60 years</b>	<b>Fatigue Coefficient considering Earthquake</b>		
	<b>Due to Seismic Load</b>		<b>Total</b>
<b>0.064</b>	<b><math>S_2</math></b>	<b>0.252</b>	<b>0.316</b>
	<b><math>S_s</math></b>	<b>0.269</b>	<b>0.333</b>
	<b>Earthquake on March 11, 2011</b>	<b>0.264</b>	<b>0.328</b>

Equivalent number of cycle due to seismic load is conservatively assumed to be 100.  
 $S_2$ ,  $S_s$  : Design basis seismic ground motion

# ***Evaluation of Effect of the Earthquake on March 11, 2011 on Structural Integrity of Important Pumps considering General Corrosion of Anchor Bolts***

	Pumps Evaluated	Shear Stress [MPa]		Allowable Stress [MPa]
		Without Corrosion	With Corrosion for 60 years	
Unit 1	Reactor Shut Down Cooling System Cooling Pumps	8	9	127
Unit 2	Residual Heat Removal System Pumps	34	36	202
Unit 3	Residual Heat Removal System Pumps	23	24	202

- The corrosion of the anchor bolts was evaluated by multiplication of cross-section decrease rate (6.0%/y) in consideration of the corrosion amount for 60 years of 0.3 mm.
- The shear stress with consideration of the amount of corrosions for 60 years was confirmed that there was sufficient margin to the allowable stress.

*Thank you very much for your attention*

