Workshop on Ageing Management considerations in Codes and Standards

28-29th June 2023 – Tokyo (JAPAN)

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Opening remarks

Dr Tomoya ICHIMURA

Deputy Secretary-General for Technical Affairs
Nuclear Regulation Authority (NRA)

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Opening remarks

Dr Seiji ASADA

Mitsubishi Heavy Industry

Chair of Standard Development Organisation Convergence Board

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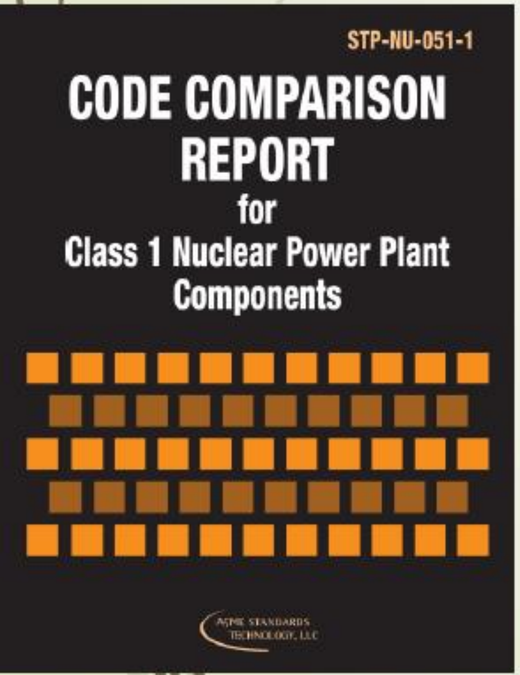
Introduction of Standard Development Organization (SDO) Convergence Board

Chair: Seiji Asada

Co-Chair: Paul Donavin

Technical Secretary: Ronan Tanguy

SDO Convergence Board



- The Multinational Design Evaluation Programme (MDEP) Code Comparison Project was initiated in late 2006 in response to a request by the MDEP Codes and Standards Working Group (CSWG) consisted of international nuclear regulatory bodies.
- The CSWG requested the various Standard Development Organizations (SDOs) to develop a comparison of their codes & standards for Nuclear Power Plant Class 1 components.
- "Code Comparison Report for Class 1 Nuclear Power Plant Components" (STP-NU-051-1) was published in 2012.
- The SDOs understood the necessity of their collaboration through this project.

SDO Convergence Board

- started from February 2013.
- Members: AFCEN (France), ASME (U.S.A), CSA (Canada), ISNI (China), JSME (Japan), KEPIC (KOREA), NIKET (Russia), NTD (Czech), WNA CORDL, R5/R6 (UK)
- Charter
  - The Standards Development Organizations (SDO) Convergence Board will facilitate the following objectives for Nuclear Power Plant Codes and Standards:
    - ✓ **Limit divergence on individual requirements**
    - ✓ **Achieve harmonization on individual requirements, where realistic and practical**
  - The SDO Convergence Board will collaborate with Working Group on Codes and Standards (WGCS in CNRA of OECD/NEA), CORDEL and other global stakeholders to identify and facilitate implementation of activities leading to nuclear code harmonization and minimization of code divergence.

3 year Work Plan

- From 2018, the SDO CB started "3 year Work Plan" to discuss potential topics.
- The 1st "3 year Work Plan" for 2018 – 2020 is;
  - ✓ **1: Pressure Testing**
  - ✓ **2: Welding residual stresses**
  - ✓ **3: Verification and validation in Codes**
  - ✓ **4: Heterogeneities in thick section forgings and flocking**
  - ✓ **5: Alternatives to radiography**
- The 2nd "3 year Work Plan" for 2021 – 2023 is;
  - ✓ **1: Advanced Manufacturing Techniques**
  - ✓ **2: UT in lieu of RT**
  - ✓ **3: Transition from construction stage to operating stage**

Communication

- **WNA** (World Nuclear Association) **CORDEL** (Cooperation in Reactor Design Evaluation and Licensing)
  - ✓ CORDEL is making great contribution to SDO CB as a technical secretary and a member.
  - ✓ CORDEL is preparing and has been published valuable reports for codes, and they are good inputs for SDO CB.
    - Certification of NDE Personnel (October 2014)
    - Comparison of pressure vessel fatigue codified design rules based on S-N approach (July 2020)
    - Non-Linear Analysis Design Rules: Part 1 Code Comparison (February 2017)
    - Non-Linear Analysis Design Rules Part 2a (July 2019)
    - Non-Linear Analysis Design Rules Part 2b: Assessment of Non-Linear Benchmark Results (September 2020)
    - Non-Linear Analysis Design Rules Part 3: Recommendations for Industrial Practices (July 2021)
    - etc.
  - ✓ WNA CORDL Non-Linear Analysis Workshop was held on August 4th 2019 in the ASME Code Week as an activity of SDO CB.
    - Typical benchmarks based on nozzles
    - Share the results, review the methodologies and agree to a consistent analysis methodology
    - Discuss common challenges, opportunities for cooperation, concerns and issues

Communication

- The MDEP CSWG consisted of international nuclear regulatory bodies moved to the OECD/NEA and Working Group on Codes and Standards (WGCS).
  - ✓ WGCS invites the SDO CB to their meeting, and we have good communication with each other.
  - ✓ The outcome of this International Workshop on Ageing Management will be shared in the SDO Convergence Board.
- The SDO Convergence Board will collaborate with global stakeholders to identify and facilitate implementation of activities leading to nuclear code harmonization and minimization of code divergence.

Thank you for your attention!

Introduction and logistics

Mr David RUDLAND

Senior Technical Advisor for Nuclear Power Plant Materials
US Nuclear Regulatory Commission

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Introduction - Objectives

The objectives of the workshop are to provide **better understanding of the global framework and to initiate discussion between stakeholders to enable a better consideration of ageing phenomena in codes and standards**.

The workshop will bring together regulators and stakeholders, including industry representatives, and other international organizations to share information and discuss the challenges related to ageing phenomena in mechanical codes and standards.

Presentations will be made available on the webpage of the event as soon as practically possible

Introduction - Programme

Day 1 – 28th June 2023	
09:00	Welcome and Registration
09:30	Opening remarks • **Dr Tomoya Ichimura (NRA Japan)** • **Dr Seiji Asada (SDO CB chair)**
Session 1 : Comparison of national regulatory requirements related to ageing phenomena RCPB	
09:50	Introduction • **Dr David Rudland** US NRC
10:00	Regulatory requirements for ageing phenomena of safety-related components in Korea • **Dr Sangmin Lee,** KINS Republic of Korea
10:20	Regulatory framework in Czech Republic • **Ms Jolana Rydlova,** SUJB Czech Republic
10:40	Coffee break
10:50	Regulatory frameworks in Japan • **Ms Haruko Sasaki,** NRA Japan
11:10	Regulatory framework in France • **Ms Rachel Vaucher,** ASN France
11:30	Regulatory framework in the US • **Dr David Rudland,** US NRC
11:50	Q/A session
12:30	Lunch break

Introduction - Programme

Session 2: Discussion on code requirements on RCPB to address/prevent ageing phenomena	
14:00	**Introduction** • **Mr Ronan Tanguy** (World Nuclear Association)
14:10	**ASME Code Requirements for Fatigue** • **Mr Paul Donavin,** Becht Engineering - ASME
14:30	**Outline of JSME rules on fitness for service for nuclear power plants** • **Dr Kiminobu Hojo,** Mitsubishi Heavy Industries, Ltd, JSME
14:50	**Code Requirements for Managing Aging of Pressure Boundary Components of CANDU Reactors in Canada** • **Mr Suqiang XU,** CNSC Canada
15:10	**Initiatives of Japan on standardizing new findings on ageing management** • **Pr Kenta Murakami**, University of Tokyo (Japan)
15:30	Coffee break
15:50	**Experience on the ageing phenomena with mechanical design** • **Mr Jongin Kim,** Doosan
16:10	**Q/A session**
16:50	**Conclusion of day 1**
17:05	End of Day 1

Introduction - Programme

Day 2 – 29th Jun 2023	
09:00	Welcome and Registration
Session 3 : Operating Experience related to ageing phenomena on RCPB	
09:30	Introduction • **Dr Yinsheng Li,** JAEA Japan
09:40	Stress corrosion cracking experience in France • **Ms Rachel Vaucher,** ASN France
10:00	Stress corrosion cracking experience in Japan • **Dr Takumi Terachi,** ATENA (KEPCO)
10:20	Coffee break
10:35	SCC Growth Rate Behavior of Stainless Steels in PWR Primary Water Environments • **Dr Do Jun Shim,** EPRI
10:55	Fatigue Phenomena Experience • **Mr Paul R. Donavin,** Becht Engineering
11:15	Integrity Assessment of Reactor Pressure Vessel Against Irradiation Embrittlement in Japan • **Dr Takatoshi Hirota,** Mitsubishi Heavy Industries, Ltd
11:35	Q/A session
12:25	Lunch break
Session 4: challenges of ageing phenomena in C&S applied to SMRs/AMRs	
13:55	Introduction • **Mr Suqiang Xu,** CNSC Canada
14:05	Degradation of Materials for High Temperature Small Modular Reactors • **Dr. Suraj Persaud,** Queen's university (Canada)
14:25	Material Aging Challenges for Small Modular Reactors, a Regulatory Perspective • **Mr Suqiang Xu,** CNSC Canada
14:45	Aging Management Considerations for Advanced SMRs and Non-LWR Reactors • **Mr Chris Wax,** EPRI
15:05	Q/A session

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Introduction - Programme

15:35	Coffee break
Closing session: Harmonization Discussion	
15:55	Panel discussion with speakers from the 4 sessions Moderator: Dr David Rudland
16:55	Conclusions • Dr Sangmin LEE, KINS
17:15	End of day 2

Introduction - logistics

- No drink/eat in the conference room;
- No eating in the building (drink only);
- A list of selected restaurants for lunch is available in the programme
- If you have any practical question, feel free to ask NRA/NEA's staff.

Introduction of session 1

Dr David RUDLAND

Senior Technical Advisor for Nuclear Power Plant Materials
US Nuclear Regulatory Commission

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Introduction of Session 1:

Comparison of National Regulatory Requirements Related to Ageing Phenomena on Reactor Coolant Pressure Boundary

David Rudland (NRC, USA)

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Objective and Scope

Scope of Session 1:

In this session each participating regulator will present their country's perspective and requirements for aging management. These discussions will include any relationship between these requirements and codes and standards, and thoughts on aging management gaps within codes and standards requirements.

Presentations in Session 1

There are 5 presentations in Session 1.

1	Sangmin Lee (KINS, Korea)	Regulatory requirements in Korea
2	Jolana Rydlova (SUJB, Czech Republic)	Regulatory requirements in Czech Republic
3	Haruko Sasaki (NRA, Japan)	Regulatory requirements in Japan
4	Rachel Vaucher (ASN, France)	Regulatory requirements in France
5	David Rudland (US NRC, USA)	Regulatory requirements in United States

We will have a Q/A session after the presentations.

Regulatory requirements for aging phenomena of safety-related components in Korea

Dr Sangmin LEE

Korean Institute for Nuclear Safety

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The Int'l WS on Aging Management Considerations in Mechanical Codes and Standards
28~29 June 2023
Hitotsubashi Hall, National Center of Science Building, Tokyo, Japan

Regulatory Requirements for Aging Phenomena of Safety-Related Components in Korea

Sangmin Lee

Korea Institute of Nuclear Safety

Contents

Overview – NPP Status in Korea

- As of June 2023

In Operation **(commercial)**

25 Units

Permanent Shutdown

2 Units

Under Construction

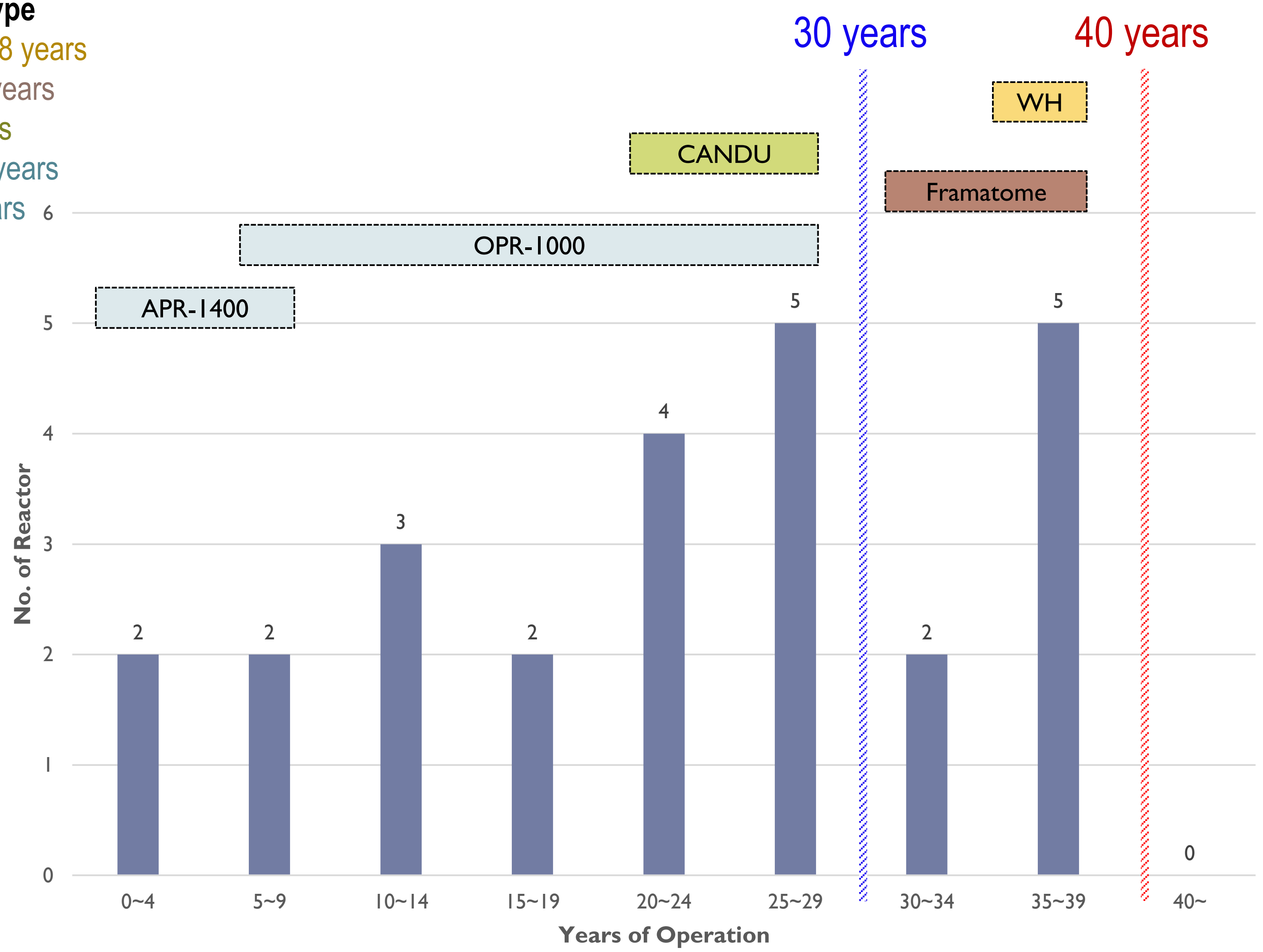
3 Units

Overview – Operating NPPs (25 Units)

Site	Unit	Capacity(MWe)	Reactor Type	Commercial Operation
Kori	2	650	Westinghouse	July 1983
	3	950	Westinghouse	Sep. 1985
	4	950	Westinghouse	April 1986
Shin-Kori	1	1,000	OPR-1000	April 2011
	2	1,000	OPR-1000	July 2012
Saeul	1	1,400	APR-1400	Dec. 2016
	2	1400	APR-1400	Aug. 2019
Wolsong	2	700	CANDU	July 1997
	3	700	CANDU	July 1998
	4	700	CANDU	Oct. 1999
Shin-Wolsong	1	1,000	OPR-1000	July 2012
	2	1,000	OPR-1000	July 2015
Hanbit	1	950	Westinghouse	Aug. 1986
	2	950	Westinghouse	June 1987
	3	1,000	OPR-1000	Mar. 1995
	4	1,000	OPR-1000	Jan. 1996
	5	1,000	OPR-1000	May 2002
	6	1,000	OPR-1000	Dec. 2002
Hanul	1	950	Framatome	Sep. 1988
	2	950	Framatome	Sep. 1989
	3	1,000	OPR-1000	Aug 1998
	4	1,000	OPR-1000	Dec. 1999
	5	1,000	OPR-1000	July 2004
	6	1,000	OPR-1000	April 2005
Shin-Hanul	1	1,400	APR-1400	Dec. 2022

Overview - Age of Operating NPPs

- **Average age by reactor type**
  - Westinghouse (5 units): 38 years
  - Framatome (2 units): 35 years
  - CANDU (3 units): 25 years
  - OPR-1000 (12 units): 19 years
  - APR-1400 (3 units): 4 years



Overview – Different Codes

Reactor Type	Construction Code	ISI Code
Westinghouse	ASME Sec. III	ASME Sec. XI
Framatome	RCC-M	ASME Sec. XI
CANDU	CSA-N285.0	CSA-N285.4
OPR-1000, APR-1400	KEPIC MN (ASME Sec. III)	KEPIC MI (ASME Sec. XI)

- References: Each NPP's FSAR (Final Safety Analysis Report) Chap. 5.2 (Integrity of Reactor Coolant Pressure Boundary)
- ASME Sec. III, Rules for Construction of Nuclear Facility Components
- ASME Sec. XI, Rules for Inservice Inspection of Nuclear Power Plant Components
- RCC-M, Design and Construction Rules for Mechanical Components of PWR Nuclear Islands
- CSA-285.0, Requirements for Class 1, 2, and 3 Pressure-Retaining Systems and Components in CANDU Nuclear Power Plants
- CSA-N285.4, Periodic Inspection of CANDU Nuclear Power Plant Components

Regulatory Requirements - Framework

Nuclear Safety Act (NSA)

General fundamentals
◇ Article 23(Periodic Safety Review)

Enforcement Decree of the NSA

Authorization procedures
◇ Article 37~39(Details of Periodic Safety Review)

Enforcement Regulations of the NSA

Detailed authorization procedures, tables, and forms
◇ Article 20(Details of PSR), 21(Standards for PSR)

Regulations on Tech. Standards for Reactor Facilities

Basic technical standards
◇ Article 63(Testing, Monitoring, Inspection, Maintenance)

Notices of the NSSC*

Detailed technical standards
◇ 2017-29(Continued Operation), 2023-2(ISI)

Regulatory Guidance of the KINS

- Detailed review and inspection guidance

*** NSSC: Nuclear Safety and Security Commission**

Regulatory Requirements - Act

- Nuclear Safety Act

Chapter 3 – Construction and Operation of Nuclear Reactor and Related Facilities

- **Section 1 : Construction of Nuclear Reactor and Related Facilities**
  - Article 10 (Construction Permit)
  - Article 12 (Approval for Standard Design)
  - Article 16 (Inspection)
- **Section 2 : Operation of Nuclear Reactor and Related Facilities**
  - Article 20 (Operating License)
  - Article 22 (Inspection)
  - Article 23 (Periodic Safety Review)
  - Article 26 (Safety Measures for Operation)
  - Article 27 (Suspension of Use of Nuclear Reactor and Related Facilities)
  - Article 28 (Decommissioning of Nuclear Reactor and Related Facilities)
- **Section 3 : Construction and operation of Nuclear Research Reactor, etc.**

- Enforcement Decree
- Enforcement Regulation
- Regulation of the NSSC
- Notice of the NSSC

- In-Service Inspection
- Continued Operation (Aging Management Program, Time-Limited Aging Analysis)

Regulatory Requirements - Notices (1/5)

- Notices of the NSSC
  - **No. 2017-29(reactor.35), Guidelines on Application of Technical Standards for Assessment of Continued Operation of Nuclear Reactor Facilities**
    - Article 4 (Application of Technical Standards for Assessment of Continued Operation)

      1. Matters reflecting the recent operating experience and research results

         a. assessment of scoping and screening results of aging management

         b. assessment of aging management program

         c. assessment of time-limited aging analysis for continued operation

         d. matters necessary for reflecting operating experience and research results
    - Table 1 (Matters concerning Assessment of the Scoping and Screening Methodology for Aging Management)

      □ NUREG-1800 (SRP for Review of License Renewal Applications)

      □ NUREG-18011 (GALL, General Aging Lessons Learned)

Regulatory Requirements - Notices (2/5)

- Table 3 (Matters concerning the Lifetime Evaluation for Continued Operation)

Details	Reference Regulations and Technical Standards
1. Identification of Time-Limited Aging Analysis	- 10 CFR 54.21
2. Analysis of Neutron Irradiation Embrittlement in Reactor Vessel	- 10 CRF 50, Appendix G
3. Metal Fatigue Analysis	- 10 CFR 54.21
4. Environmental Qualification of Components	- 10 CFR 54.49
5. Evaluation of Tendon Pre-stress of Concrete Containment	- 10 CFR 50.21
6. Fatigue Analysis of Containment Liner Plate, Metal Containment, and Penetrations	- 10 CFR 50.21
7. Other Reactor-specific Time-Limited Aging Analyses	- 10 CFR 50.21

Regulatory Requirements - Notices (3/5)

- Table 4 (Matters Necessary for Reflecting Operating Experience and Research Results)

Details	Reference Regulations and Technical Standards
1. Fire Protection	- 10 CFR 50.48 (CANDU: CAN/CSA-N293)
2. Dynamic and Seismic Qualification of Equipment	- US NRC Regulatory Guide 1.100 (CANDU: CAN3-N289.1)
3. Pressurized Thermal Shock of Reactor Vessel	- 10 CFR 50.61
4. Anticipated Transient without Scram	- 10 CFR 50.62
5. Management Program for Active Components	- ASME OM, KEPIC MO and IEEE Std. 279, 308, and 603 (CAN3-N290-1, CAN/CSA-N290.5)
6. Evaluation of Thermal Stratification of Piping	- US NRC Bulletin 88-11
7. Safety Assessment of Combustion Gas	- IAEA Safety Guide NS-G.1.10
8. Evaluation of Capability to Cope with Station Blackout	- 10 CFR 50.63

Regulatory Requirements - Notices (4/5)

- Notices of the NSSC
  - **No. 2023-2(reactor.16), Regulation on In-Service Inspection of Nuclear Reactor Facilities**
    - Article 6 (Inspection Standards)

Code Req.

Typical ISI Program (10 year)

- ☐ ASME Code Section XI (KEPIC MI)

  Reduced Frequency

  \+ Risk-Informed Inspection for Piping (Non-Mandatory Appendix R)
- ☐ CAN/CSA-N285.4/-N285.5

- ☐ Regulatory Guide 1.14 (RCP Flywheel Integrity)
- ☐ Regulatory Guide, 1.65 (Inspections for Reactor Vessel Closure Studs)
- ☐ Regulatory Guide 1.147 (Acceptability of ASME Section XI Code Case)

Additional Req.

Regulatory Requirements - Notices (5/5)

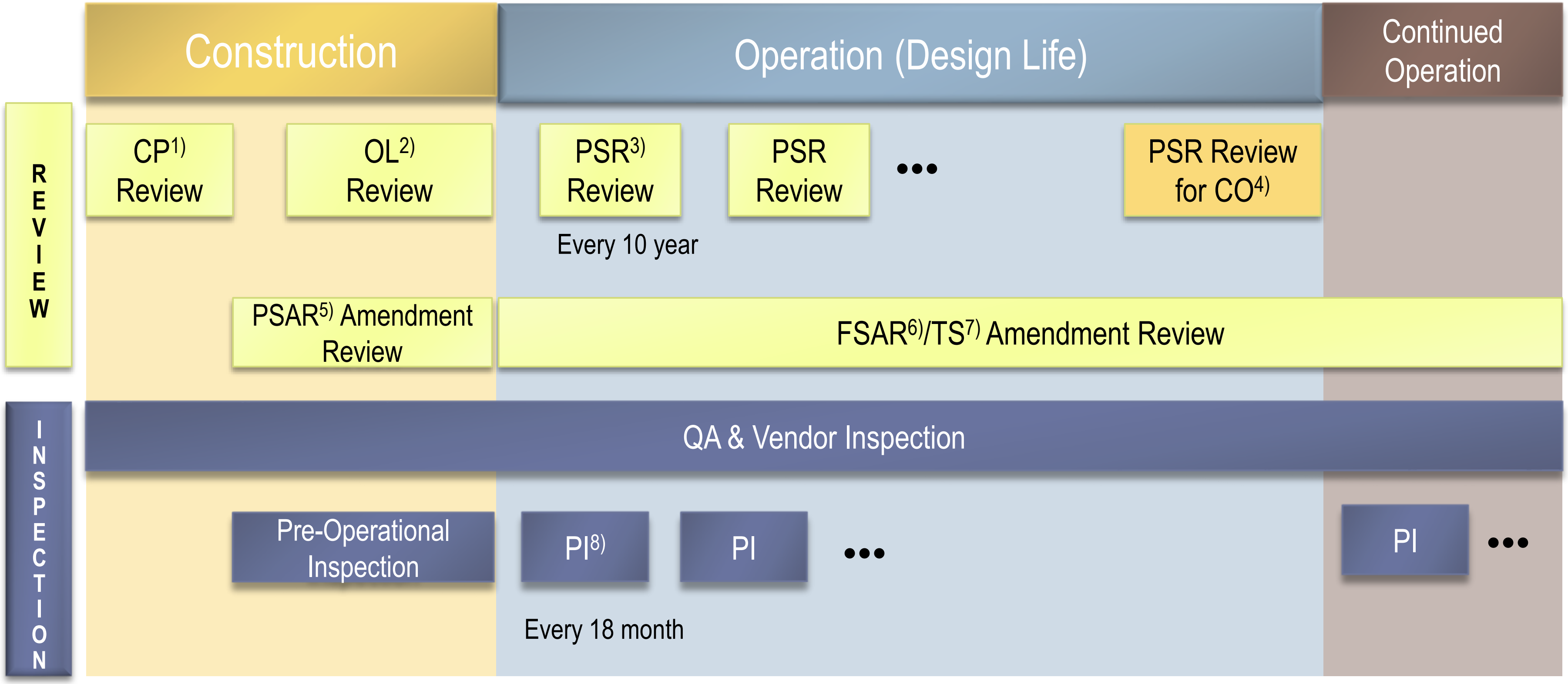
- Article 7 (Reflection of Operation Experience)

- *Examples of Augmented Inspection based on Operating Experience*
  - □ *Thermal stratification/Thermal fatigue (NRC Bulletin 88-08, 88-11)*
  - □ *Alloy 600 dissimilar metal weld joint (10 CFR 50.55a, ASME CCN-770-1)*
  - □ *Thermal sleeve (OE of Hanbit 5,6 at 2003)*
  - □ *Break exclusion area of high energy line (if adopted SRP 3.6.2)*
  - □ *Wall thinning of carbon steel pipe (Generic Letter 89-08)*
  - □ …

$$\text{Long-Term ISI Program (LTP)} = \text{Code Req.} + \underbrace{\text{Additional Req.} + \text{Augmented Req.}}_{\textbf{Reg. Req.}}$$

Regulatory Activities - Review & Inspection (1/5)

Regulatory Review & Inspection



1) Construction Permit
2) Operating License
3) Periodic Safety Review
4) Continued Operation

5) Preliminary Safety Analysis Report
6) Final Safety Analysis Report
7) Technical Specification
8) Periodic Inspection

Regulatory Activities - Review & Inspection (2/5)

- Construction Stage

  - **Regulatory Review (CP & OL)**

    - NUREG-1800(L.R.),1801(GALL)
    - ASME Code Sec. III, App. W (Environmental Effects on Components)

    - Based on safety review guidance and operating experience, confirm whether materials are appropriately selected and whether aging effects in structures, systems, and components (SSCs) are managed during the operation stage
    - Review the adequacy of the program related to aging: material selection, in-service inspection (ISI) program, water chemistry control plan, and equipment qualification considering environmental conditions, etc.

    Long-Term ISI Program
    - Augmented ISI (Operating Experience)
    +
    - ASME Code Sec. XI

  - **Pre-Operational Inspection**

    - Inspect that licensee's activities, such as the pre-service inspection (PSI) program, are adequately implemented

Regulatory Activities - Review & Inspection (3/5)

- Operational Stage (1/2)
  - **Regulatory Review (PSR on Aging Management Program)**
    - In the periodic safety review (PSR) report, it should be verified that aging in the SSCs will be effectively managed for the next 10-year operation according to the plant-specific aging management program (AMP).
    - Following ten (10) elements should be considered for each aging management program:

      (1) Scope, (2) Prevention actions, (3) Monitoring/Insp. parameter, (4) Detection of aging effects, (5) Monitoring and trending, (6) Acceptance criteria. (7) Corrective actions, (8) Confirmation process, (9) Administrative control, (10) Operational experience

Regulatory Activities - Review & Inspection (4/5)

- Operational Stage (2/2)
  - **Regulatory Inspection for the Plants within Design Life**
    - To monitor aging degradation, a periodic in-service inspection (ISI) program should be performed for every plant overhaul (period: 12~18 months) according to industrial inspection codes such as ASME Code Sec. XI.
    - Additional enhanced inspection programs should be appropriately performed according to regulatory requirements to monitor aging degradation as follows;
      - ☐ Reactor surveillance program
      - ☐ Flow accelerated corrosion (FAC) program
      - ☐ Boric acid corrosion (BAC) program
      - ☐ Water chemistry control program

Regulatory Activities - Review & Inspection (5/5)

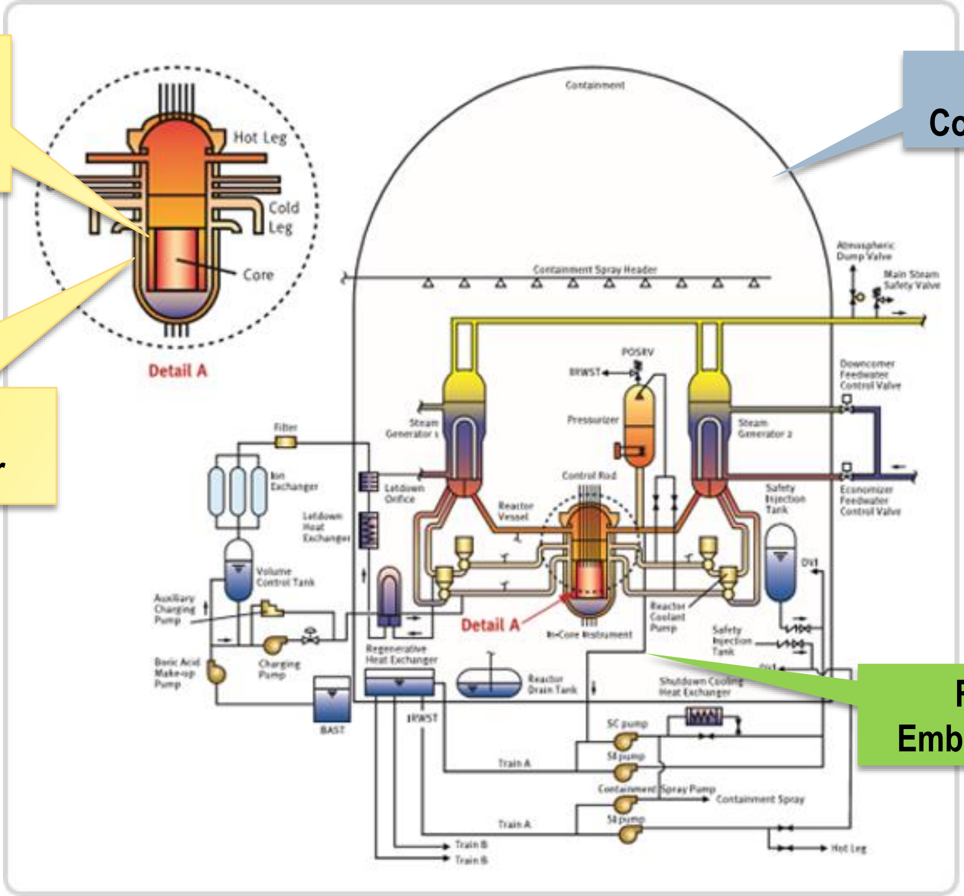
- Continued Operation Stage
  - **Regulatory Review for Aging Effect**
    - In addition to the typical PSR review, the aging effect should be re-evaluated considering the extended life of the SSCs.
    - Time-limited aging analysis (TLAA), operating experience, and new research results on aging should be considered in the aging evaluation.
    - Special concern should be given in the following TLAA items;
      - Radiation embrittlement of reactor vessel
      - Fatigue of safety class 1 component including environmental effect
      - Environmental qualification
      - Containment tendon pre-stress analysis
      - Fatigue in containment penetration
  - **Periodic Inspection**
    - Inspect that licensee's activities are adequately implemented in accordance with a new FSAR chapter.

Regulatory Activities - Aging Issues

- Major Aging Issues of PWR

Irradiation Assisted Stress Corrosion Crack (IASCC) of Reactor

Neutron Irradiation Embrittlement of Reactor



Corrosion of Containment Structure

Radiation, Temperature Embrittlement of Electric Cable

KINS 한국원자력안전기술원 KOREA INSTITUTE OF NUCLEAR SAFETY

Regulatory Activities - Plant-specific AMP

- Plant-specific AMP
  - **After 5-year feasibility study performed, the KHNP (Korea Hydro & Nuclear Power Co., Ltd.) implemented the integrated AMP of all Korean NPPs in 2016**
    - About plant-specific 30~40 AMP procedures
    - The aging data collected from the plants is intensively analyzed by experts of KHNP CRI (Central Research Institute)
  - **The regulatory body inspects the implementation status of plant-specific AM procedures as a sample during every periodic inspection**

Regulatory Activities - AM Procedures (1/3)

- 36 AM Procedures of Hanbit Units 1&2 (1,000MWe, PWR, WH Type)

AM Procedure	Aging Mechanism
1. ASME Sec. XI In-service Inspection	Fatigue, PWSCC, BAC, FAC
2. Safety class 1, 2, and 3 piping and metal containment components supports	GC(General Corrosion), Fatigue
3. Reactor vessel surveillance	Radiation Embrittlement
4. Reactor head closure studs	Fatigue, Over-Load(OL)
5. Ni-alloy nozzles and penetrations	PWSCC
6. PWR vessel internals	IGSCC
7. In-service inspection for containment steel elements	GC
8. In-service inspection for reinforced and pre-stressed concr ete containments	GC
9. Containments leak rate test	GC
10. Protective coating monitoring and maintenance program	GC

Regulatory Activities - AM Procedures (2/3)

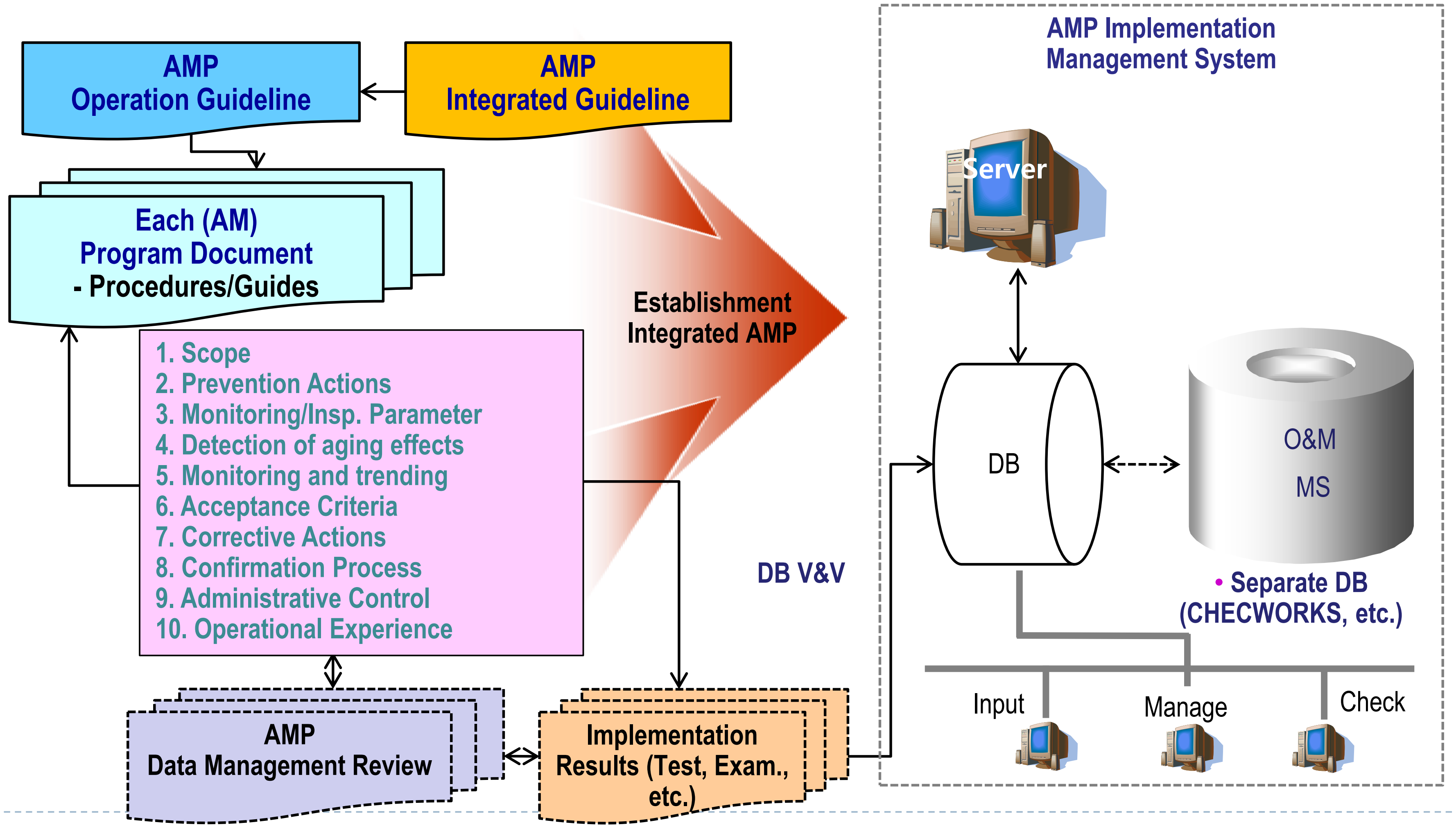
11. Flux thimble tube inspection	IGSCC
12. Monitoring of neutron-absorbing materials other than Boraflex	Radiation Embrittlement
13. Inspection of overhead heavy load and light load handling systems	GC, OL
14. Steam generator tube integrity	PWSCC, ODSCC,
15. One-time inspection of ASME Code class 1 small bore-piping	Fatigue, PWSCC, BAC, FAC
16. Ground steel tanks	GC
17. Buried and underground piping and tanks	GC
18. Fuel oil chemistry	GC
20. Flow-accelerated corrosion	FAC
21. Open-cycle cooling water system	GC, Wall Thinning Deformation, Crack FAC, Leak
22. Closed-cycle cooling water system	
23. Compressed air monitoring	
24. Fire protection system	
25. Fire water system	

Regulatory Activities - AM Procedures (3/3)

26. Insulation material for electrical cables and connections Non-EQ	Thermal Embrittlement Radiation Embrittlement GC, Wall Thinning Deformation, Crack
27. Electrical connections Non-EQ	
28. Metal material for electrical connections Non-EQ	
30. External surfaces monitoring of mechanical components	GC, BAC
31. Water chemistry	SCC, GC
32. In-service inspection of water-control structures associated with nuclear power plants	Corrosion
33. Structures monitoring	GC
34. Bolting integrity	Corrosion
35. Boric acid corrosion	BAC
36. Fatigue monitoring	Fatigue

Regulatory Activities - AMP Management System

- The Integrated AMP Management System of the KHNP



Concluding Remarks

- The regulatory requirements and activities on aging management in Korea were explained at each stage of construction, operation, and continued operation
  - The regulatory body adopts some industrial codes and standards as a requirement to ensure or confirm the integrity of structures, systems, and components at a nuclear power plant
- The plant-specific aging management program of the KHNP was briefly introduced
  - Plant-specific aging management program suitable for each reactor type has been implemented based on their feasibility study
  - The aging data from the plants is collected and analyzed for preparing the technical basis (or applications) of in-service inspection, periodic safety review, and continued operation

Thank you for your attention!

sangmin.lee@kins.re.kr

Requirements of the Czech Legislations on Ageing of RCPB, and Supervision of their Fulfillment

Ms Jolana Rýdlová

State Office for Nuclear Safety (SÚJB), Czech Republic

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International Workshop on Ageing Management Considerations in Mechanical Codes and Standards

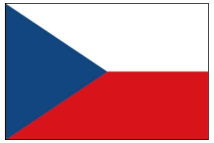
Requirements of the Czech Legislations on Ageing of RCPB, and Supervision of their Fulfillment

Jolana Rýdlová
State Office for Nuclear Safety (SÚJB), Czech Republic

June 28-29, 2023 Tokyo, Japan

Czech Republic

Area: 78 866 km2
Population: 10,8 mil.
Capital: Prague (1,2 mil.)



Dukovany NPP

- WWER 440/213, 4 units
- PWR, 6 loops
- In operation since 1985 -1987

Now: LTO

Temelín NPP

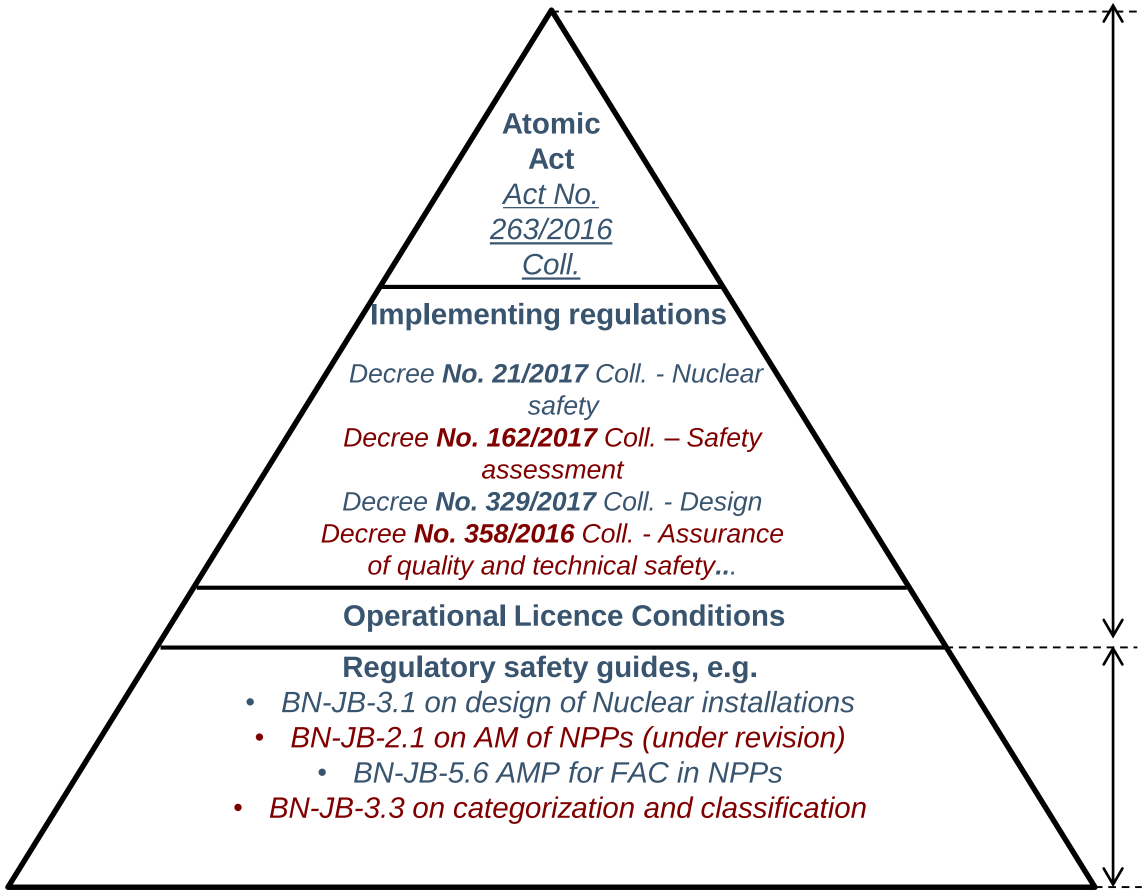
- WWER 1000/V320, 2 units
- PWR, 4 main circulation loops
- In operation since 2000 – 2002

Now: Unlimited licence, req: approval of documents

Outline

- Czech legislation (ageing – RCPB)
- Licence
- How do we supervise...

Czech Regulatory Framework (nuclear safety)



Legally binding (requirements)

Decrees are issued by SÚJB

(In English: https://www.sujb.cz/en/legal-framework/nuclear-law)

Non-legally binding (recommendations)

(In Czech: https://www.sujb.cz/dokumenty-a-publikace/navody-a-doporuceni)

Act No. 263/2016 Coll. (Atomic Act) and
Decrees No. 21/2017 Coll. and No. 162/2017 Coll.

§ 49 General obligations of licence holders - AMP is required – as one of the fundamental documents for the activity to be licensed

Details: Decree No. 21/2017 Coll. (e.g. on AM process and AM programme, see §11 and 12)

§ 48 Requirements on safety assessment

Details: Decree No. 162/2017 Coll. (e.g. on PSR, on safety analyses, on PSA, on special safety review for design modification incl. in case of operation beyond designed lifetime)

Act No. 263/2016 Coll. (Atomic Act) and Decree No. 329/2017 Coll.

§ 44 Categorization of safety functions: 1, 2, 3; safety classes : 1, 2, 3 - to ensure quality

§ 46 Requirements for nuclear installation design and the design process of nuclear installation

Details: ***Decree No. 329/2017 Coll., on design***

Reactor coolant pressure boundary (RCPB)

- 2nd physical safety barrier in DiD (1st fuel cladding, 3rd containment)
- SSCs of Safety Class 1 and 2 (SSC in RCPB is of SC 2 if failure of such SSC does not need the safety system intervention)

(RCPB – in Czech legislative as “primary circuit”)

Requirements on primary circuit are in §36 - §38, e.g.:

determination of requirements so as to ensure

1. their resistance to initiation of material flaws, including rupture,
2. low velocity of the material flaws propagation,
3. resistance to brittle fracture of the material and
4. that the pressure vessel rupture is a practically eliminated matter,

etc.

Act No. 263/2016 Coll. (Atomic Act) and Decree No. 358/2016 Coll.

§ 56 Requirements on quality assurance for selected equipment

§ 57 Requirements on technical safety of selected equipment

§ 58 and 59 Assessment and verification of conformity of selected equipment with technical requirements

Details: ***Decree No. 358/2016 Coll., on requirements for assurance of quality and technical safety and assessment and verification of conformity of selected equipment***

Note: Selected equipment = SSC of safety class 1, 2 or 3

Technical requirements – in Annex 1, contains specific requirements for selected equipment (i.e. SSCs of SC 1, 2 or 3), e.g. on integrity, functionality / loadings, materials...

> The **calculation method** must apply a **conservative approach in accordance with requirements in technical standards** for strength calculations, ageing monitoring and assessment etc.

Following the design of basic pressure equipment dimensions, a **check calculation** must be performed that must demonstrate e.g. strength for static, cyclic and seismic loading, resistance to brittle fracture.

Licence

Atomic Act

§9 - activities requiring the license (siting, construction etc.)

§13 - licensing and registration prerequisites

§16 - licence application

§19 - procedure for issuing licence

Before 2017: limited licence

Since 2017: unlimited licence

- SSCs in Czech NPPs were designed for 30 years of operation, exception: RPV - 40 years

-> LTO in CR means operation beyond 30 years

Note: Czech legislative does not use term “long term operation”; license can be approved for the “operation”

Development of regulatory requirements for AM in **safety guide BN-JB-2.1** - inputs from

- USA – 10 CFR 50.54, NUREG 1800, 1801 and 1833, NEI 95-10...
- IAEA EBP SALTO, IGALL, SSG-48 (AM and Development of a Programme for LTO in NPPs)...
- Catalogue of DM and ageing effects (NRI Rez)
- etc.

SÚJB duties

SÚJB shall conduct **inspections** to verify compliance with the Atomic Act and Decrees

SÚJB

- issues licences
- approves packaging assemblies for the carriage, storage or disposal of radioactive materials...
- approves documentation for licensed activities
- issues binding opinions relating to nuclear installations under the building act
- etc.

- How do we supervise...

(currently operating NPPs)

... ageing of SSCs within RCPB

Regularly:

- Annually: Operational SAR – summary update describing the current plant condition with highlighted changes since the last year
- 5 years interval: Safety demonstration that installation and personnel is prepared for further operation of the nuclear installation (results of AM and SCCs condition monitoring process based on AM review etc.)
- 10 years interval: PSR - e.g. Areas 2 (actual state of SSCs) and 4 (ageing)

- SÚJB internal guideline for review
- SÚJB Chairperson Directive for the review process

...resistance to initiation of material flaws/ fatigue

- Czech NPPs:
  - **AMP low cycle fatigue**

- Regulatory review:
  - within annual review of Operational SAR
  - within Safety demonstration for further operation
  - in case of a modification at NPP
  - specific inspection – process in general and selected cases

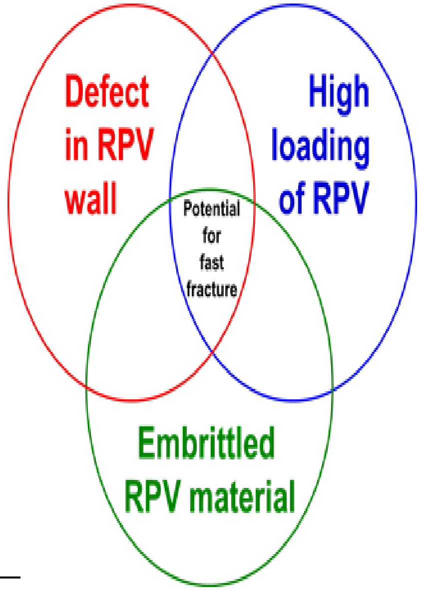
- **Technical standard**: NTD A.S.I., Section IV
  (based on Russian approach in PNAE G-7-002-86)

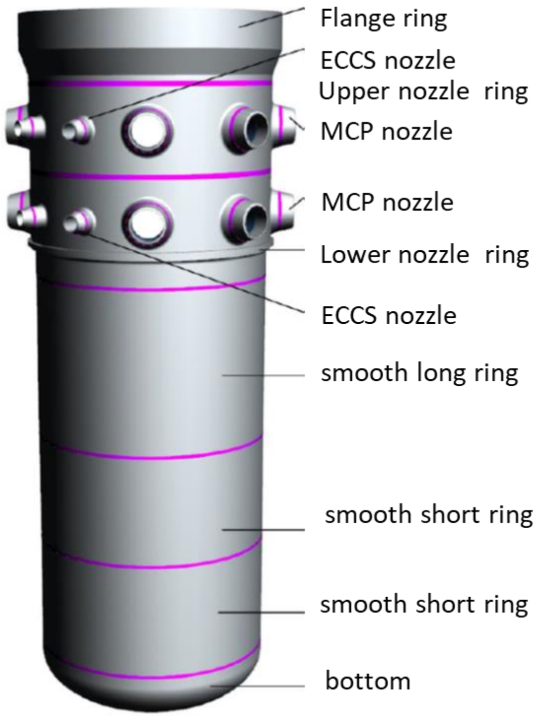
Questions we can ask:

- *All relevant DM/ ageing effects (AE) accounted for?*
- *Influence of DM/ AE on loading, material is significant?*
- *All design loads?*
- *Material characteristics used – source? Ageing - assessed?*
- *Environmental effects on fatigue – assumed?*
- *Critical locations were appropriately defined and assessed?*
- *Correctly used methodology?*

... RPV resistance to brittle fracture

- **Pressurized thermal shocks**
- Regulatory review:
  - annually within Operational SAR
  - in case of a modification at NPP with possible impact to the current PTS analyses – e.g. modification of PRZ safety valve (to improved safety)
  - change in computational SW– e.g. FLUENT for TH analyses, coupling of codes





Technical standard: NTD A.S.I., Section IV / based on VERLIFE - specifically for WWER

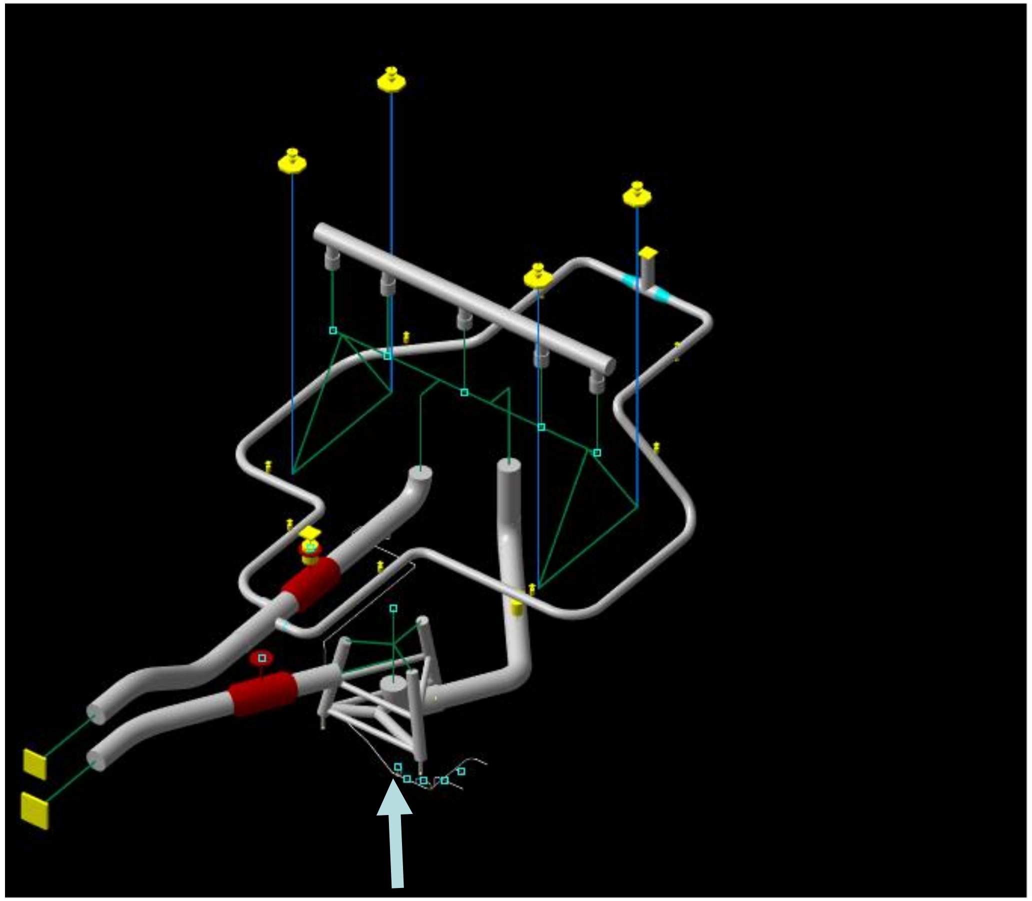
...a safety related event

Requirements on event reporting – in the Decree No. 21/2017 Coll.

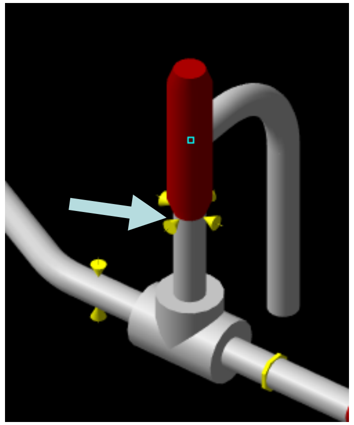
All safety related events are analysed and reported by the licensee

Licensee is obliged to determine the direct and root causes of the event

SÚJB: meetings with licensee, specific inspections, performing calculations



Leaking crack in weld



Issues and Challenges for the Regulator

- Requirements on experts are quite challenging:
  - Broad scope of degradation mechanisms/ ageing effects to be assessed
  - Knowledge of standards requirements
  - Knowledge of relevant NPP (equipment, processes..)
  - Knowledge of all areas of review (mechanical, chemistry…)
  - Knowledge of strength / fatigue / … calculations
- Increased requirements on safety
- Development in NDT area
- New technologies
- New materials

Conclusion

- Legislation (safety of nuclear utilities) in the Czech Republic

... new since 2017, needs to be assessed for covering all new items (materials, technologies..)

- Supervision

... is demanding

... not always easy

at the same time - not boring!

Thank you for your attention

Coffee break

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Regulatory requirements for aging phenomena and related Codes and Standards in Japan

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Nuclear Regulatory Authority, Japan

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Session 1: Comparison of national regulatory requirements related to ageing phenomena on reactor coolant pressure boundary (RCPB)

Regulatory requirements for aging phenomena and related Codes and Standards in Japan

June 2023

Haruko Sasaki

1. Regulatory Requirements (1/2)

<u>Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors</u>

Article 43-3-14

A licensee of power reactor operations must maintain the power reactor facilities to comply with the technical criteria specified by the rules of the NRA; provided, however, that this does not apply to power reactors for which the approval set forth in Article 43-3-34, paragraph (2) has been obtained, unless otherwise specified by the rules of the NRA.

1. Regulatory Requirements (1/2)

Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors

Article 43-3-22

(1) *A licensee of power reactor operations, pursuant to the provisions of the rules of the NRA, must take necessary measures for operational safety concerning the following particulars (including the particulars concerning measures to be taken in the event of a severe accident):*

(i)maintenance of the power reactor facilities;

(ii)operation of the power reactors;

(iii)transport, storage, or disposal of nuclear fuel material or material contaminated by nuclear fuel material

1. Regulatory Requirements (2/2)

Ordinance Related to Technical Standards for Commercial Power Reactors and Their Auxiliary Facilities

Article 14(safety equipment)

Safety equipment shall be installed so that it can perform its functions under all estimated environmental conditions in the event of a design basis accident and until the period to a design basis accident.

Article 18 (Prevention of destruction due to cracks, etc. during use)

Class 1 components, class 1 support structures, class 2 components, class 2 support structures, class 3 components, class 4 tubes, reactor containment vessels, reactor containment support structures and core supports under operation shall be free of cracks or other defects that could cause of failure.

2. Main ageing phenomena on RCPB

- *The NRA requires licensees to conduct aging management technical evaluation once per 10 years after 30 years during operation.*
- *Important reference is the following;*

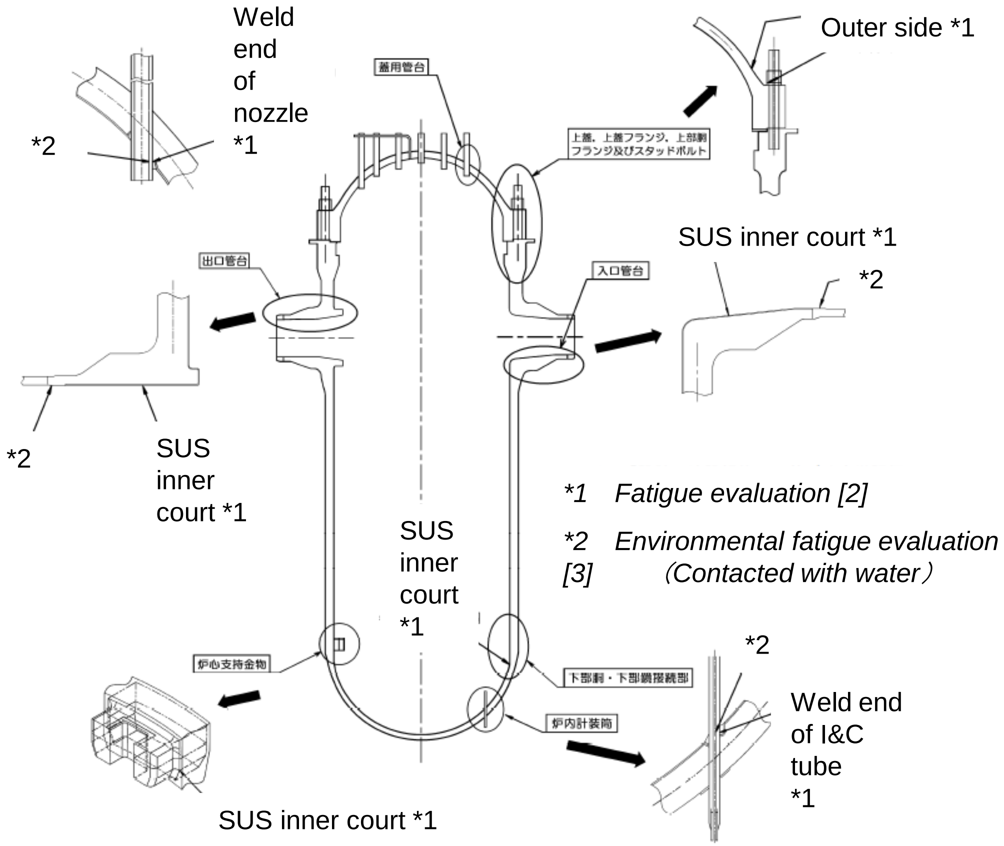
  [1] AESJ Code on implementation and review of nuclear power plant aging management program (see Session 2)
- *Aging degradation prediction shall be conducted in the evaluation.*
- *Main ageing phenomena on reactor coolant pressure boundary (RCPB) is the followings;*
  - ✓ *Low cycle fatigue*
  - ✓ *Irradiation embrittlement*
  - ✓ *Thermal aging of duplex stainless steel*
  - ✓ *Irradiation Assisted Stress Corrosion Cracking*
- *Main ageing phenomena without RCPB is the followings;*
  - ✓ *Reduced insulation resistance of electrical and I&C components*
  - ✓ *Degradation of concrete structure*

2.1 Low cycle fatigue

In the case of RPV, the following Codes and Standards are typically referred;

[2] JSME Design and Construction Code (endorsed)

[3] JSME Codes for Nuclear Power Generation Facilities -Environmental Fatigue Evaluation Method for Nuclear Power Plants-



- ✓ Licensees conduct **a fatigue evaluation in an atmospheric environment based on the JSME design and construction code** and confirmed that the fatigue accumulation factor (Uf) was below the allowable value (Uf ≤ 1).
- ✓ In addition, as a result of conducting **a fatigue evaluation in an environment contacted with the liquid phase based on the JSME environmental fatigue evaluation method**, it is confirmed that the fatigue cumulative coefficient was below the allowable value (Uen ≤ 1).

Presenter made addition to “Takahama Unit 1, 2 Degradation evaluation (low cycle fatigue), Supplementary explanatory materials 16th June KEPCO”

2.2 Irradiation embrittlement

The following Codes and Standards are referred;

[4] JEAC4206 Method of verification test of the fracture toughness for nuclear power plant component (endorsed)(see session 3)

[5] JEAC4201 Method of surveillance tests for structural materials of nuclear reactors (endorsed) (see session 3)

- ✓ ***K_{Ic} > K_I*** *as a result of the PTS evaluation at beltline region of the reactor vessel .*
- ✓ ***USE shall be 68J*** *or more.*

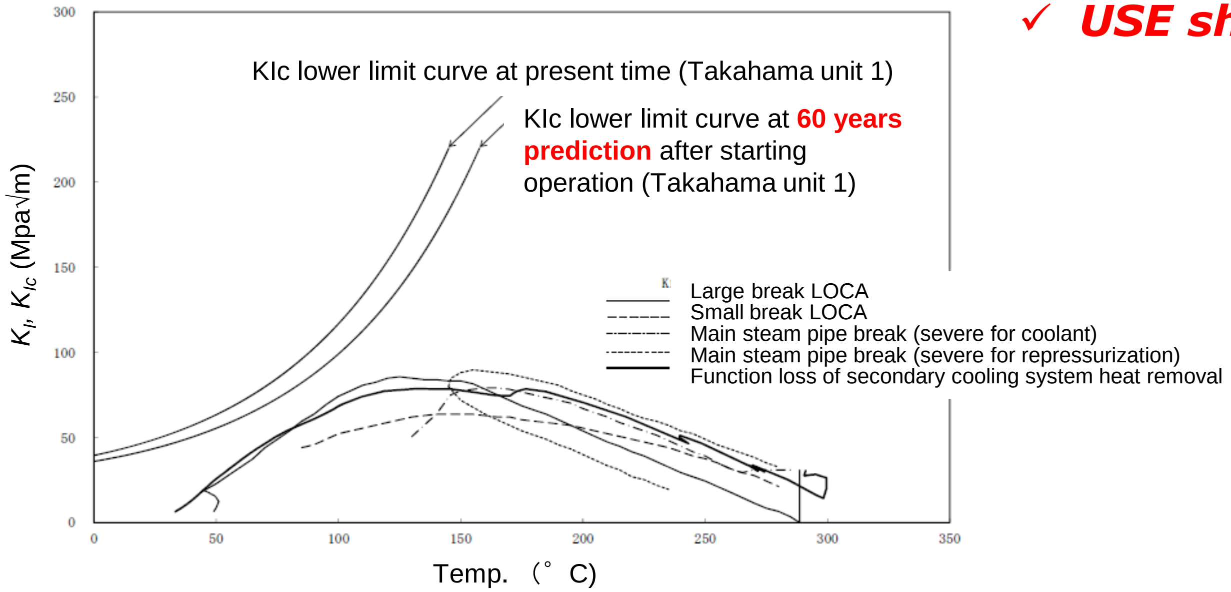
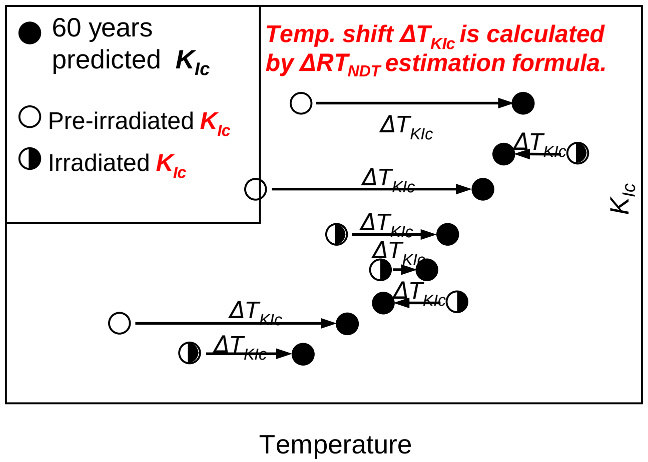


Fig. PTS evaluation result of Takahama unit1 (evaluation using 10mm depth postulated defect)



Presenter made addition to "Takahama Unit 1, 2 Degradation evaluation (Irradiation embrittlement), Supplementary explanatory materials 16th June KEPCO"

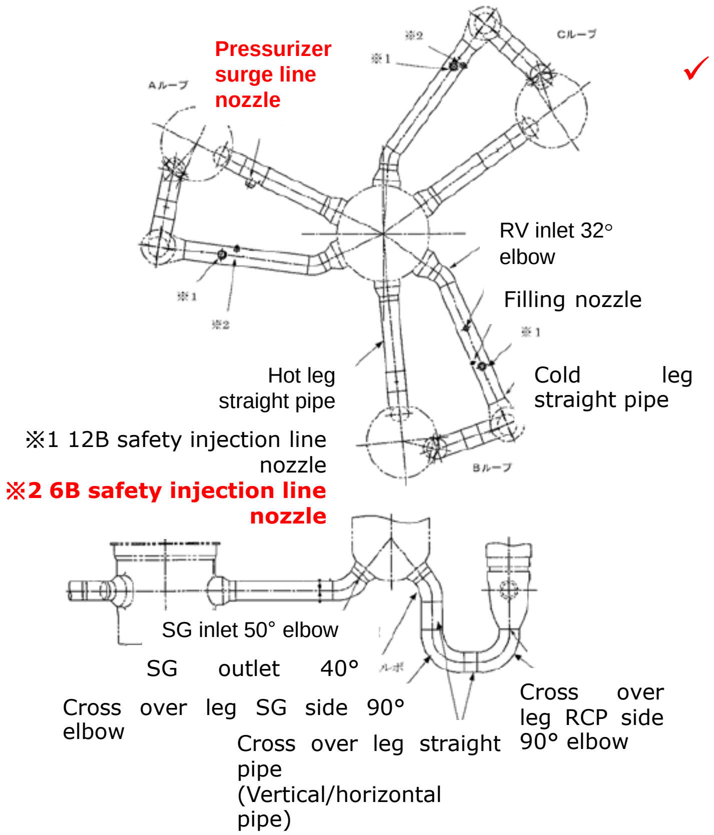
2.3 Thermal aging of duplex stainless steel

The following Codes and Standards are typically referred;

[6] JEAG4613; Technical guidelines for protection design against postulated piping failures in nuclear plants

- ✓ *Embrittlement prediction formula H3T was established in the research study “Development of technology to extend plant life; Thermal aging of duplex stainless steel test (PWR), March 1994”*

- ✓ **The crack growth (CG) resistance shall be exceeded the CG force** at the evaluation at target part as a result of the ductile CG evaluation.
- ✓ **The micro-change rate of CG resistance shall be exceeded the micro-change rate of CG force** in the state where CG resistance and CG force are equal at the evaluation target site as a result of the crack instability evaluation.



Portion	Ferrite [%]	Temp. [C°]	Stress [MPa]	Selection
Hot leg straight pipe	Approx. 13.9	322.8	Approx. 179	
SG inlet 50° elbow	Approx. 12.8	322.8	Approx. 133	
SG outlet 40° elbow	Approx. 10.3	288.6	Approx. 162	
Cross over leg straight pipe (Vertical pipe)	Approx. 14.1	288.6	Approx. 127	
Cross over leg SG side 90° elbow	Approx. 12.7	288.6	Approx. 116	
Cross over leg straight pipe (Horizontal pipe)	Approx. 14.1	288.6	Approx. 116	
Cross over leg RCP side 90° elbow	Approx. 14.8	288.6	Approx. 101	
Cold leg straight pipe	Approx. 14.8	288.6	Approx. 108	
RV inlet 32° elbow	Approx. 15.3	288.6	Approx. 115	
Pressurizer surge line nozzle	Approx. 13.7	322.8	Approx. 215	✓
12B safety injection line nozzle	Approx. 13.7	288.6	Approx. 171	
Filling nozzle	Approx. 11.6	288.6	Approx. 152	
6B safety injection line nozzle	Approx. 15.5	322.8	Approx. 208	✓

Presenter made addition to “Takahama Unit 1, 2 Degradation
June KEPCO”

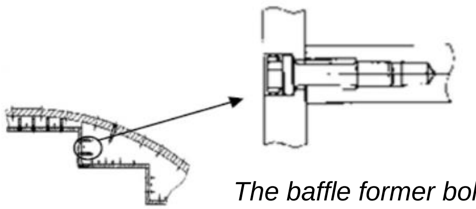
2.4 Irradiation Assisted Stress Corrosion Cracking

The following Codes and Standards are typically referred;

[7] JSME Rules on fitness-for-service for nuclear power plants (endorsed) (see session 2)

Related guideline and research study ;

- ✓ *JANSI Guidelines for Inspection and Evaluation of Reactor Vessel Internals [Baffle former bolts]*
- ✓ *The Japan Power Engineering and Inspection Corporation (JAPEIC) report "Development of technology to extend plant life"*
- ✓ *JNES report "Report on irradiation assisted stress corrosion cracking (IASCC) evaluation techniques*



The baffle former bolt
1088 pcs

- ✓ ***The baffle former bolt** was selected as the most severe evaluation portion.*
  - *the highest neutron irradiation dose and temperature*
  - *the highest stress level*
  - *Lessons learnt from overseas*

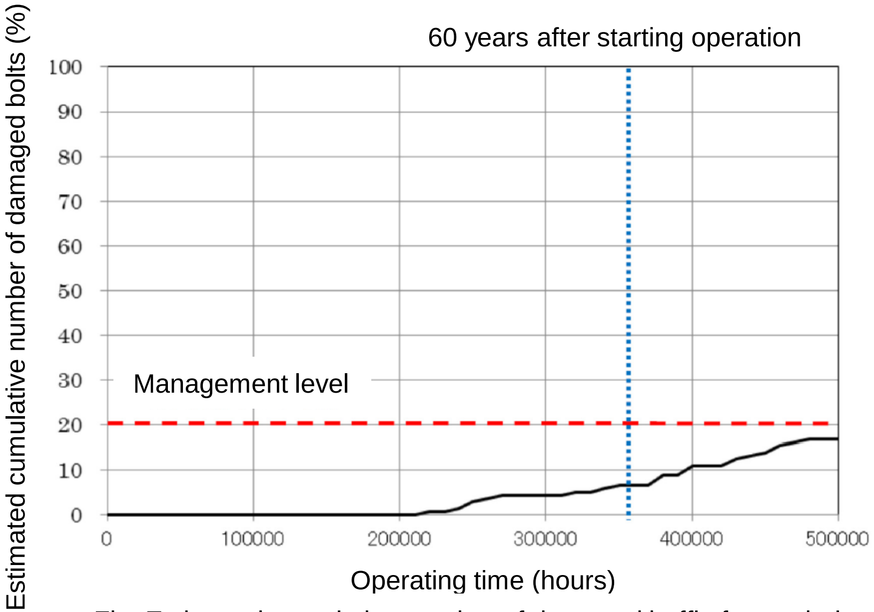


Fig. Estimated cumulative number of damaged baffle former bolts

Presenter made addition to "Takahama Unit 1, 2 Degradation evaluation (IASCC), Supplementary explanatory materials 16th June KEPCO"

3. Role of Codes and Standards in Japan

- *In recent years, unexpected aging phenomena have been occurred in unexpected portions.*
- *Only the organization concerned can investigate the cause of the phenomena. And it is important not only to investigate the cause and to conduct countermeasures, but also to disclose the information.*
- *Japanese domestic Codes and Standards provide appropriate measures to prevent similar phenomena and the NRA has been incorporated the Codes and Standards in the regulation.*
- *Activities to collect information on aging phenomena all over the world and incorporate the obtained knowledge into the Codes and Standards are very effective for both the industrial side and regulatory side.*
- *I hope that the SDO will update the Codes and Standards timely and expand to play an active role in the future .*

4. Summary

- *The NRA has incorporated many Codes and Standards such as JSME Code, JEAC, AESJ into regulation.*
- *Important aging phenomena, such as Low-cycle fatigue, irradiation embrittlement, thermal aging, and IASCC has been evaluated based on the Codes and Standards.*
- *I hope that the information related to aging phenomena will be actively exchanged all over the world and SDO will update the Codes and Standards timely and expand to play an active role in the future .*

Regulatory requirements for aging phenomena and related Codes and Standards in France

Dr Rachel VAUCHER

Autorité de Sûreté Nucléaire, France

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28 June 2023

REGULATORY REQUIREMENTS IN FRANCE

INTERNATIONAL WORKSHOP ON AGEING MANAGEMENT CONSIDERATIONS IN MECHANICAL CODES AND STANDARDS - TOKYO, JUNE 2023

Rachel VAUCHER

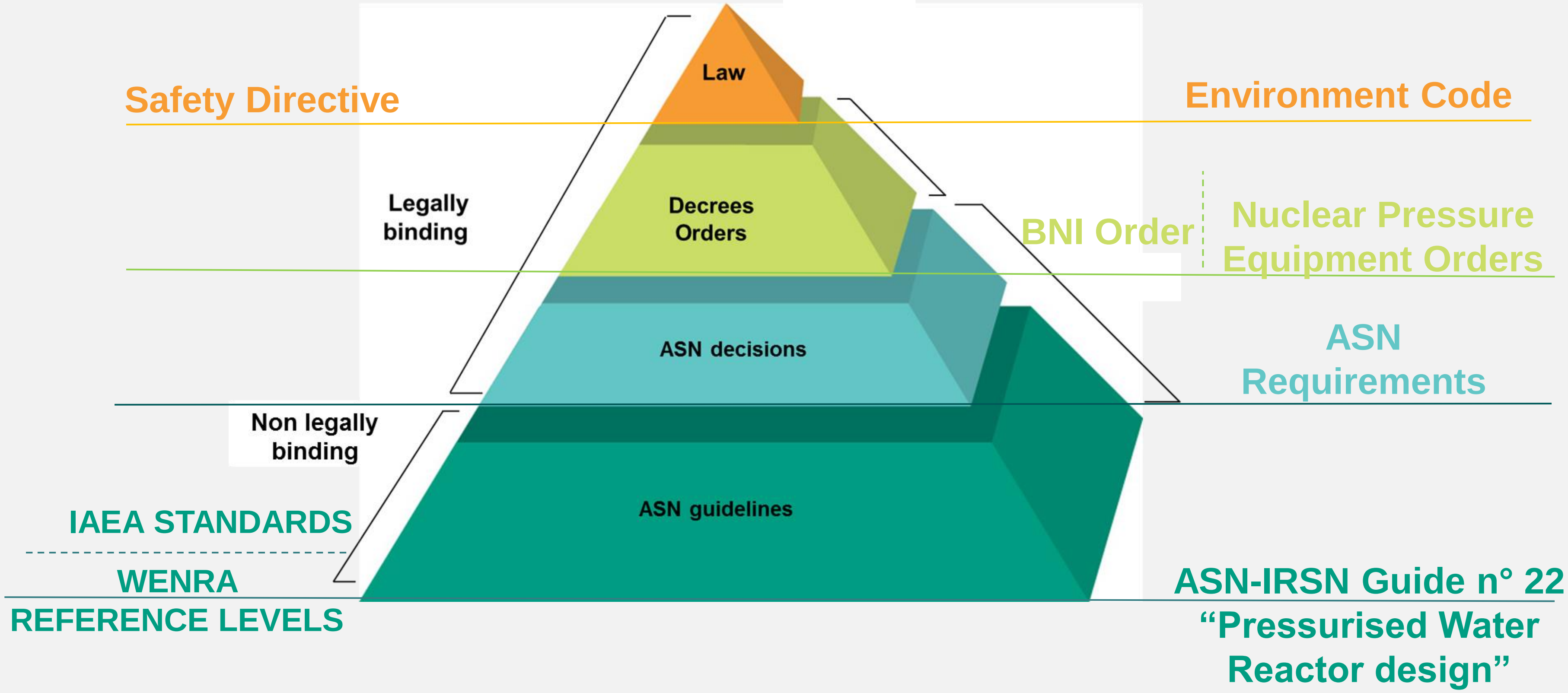
ASN – Nuclear pressure equipment department

CONTENTS

1

REGULATORY FRAMEWORK & CONTEXT

REGULATORY FRAMEWORK



✚ ASN requests by specific letters

REGULATORY FRAMEWORK

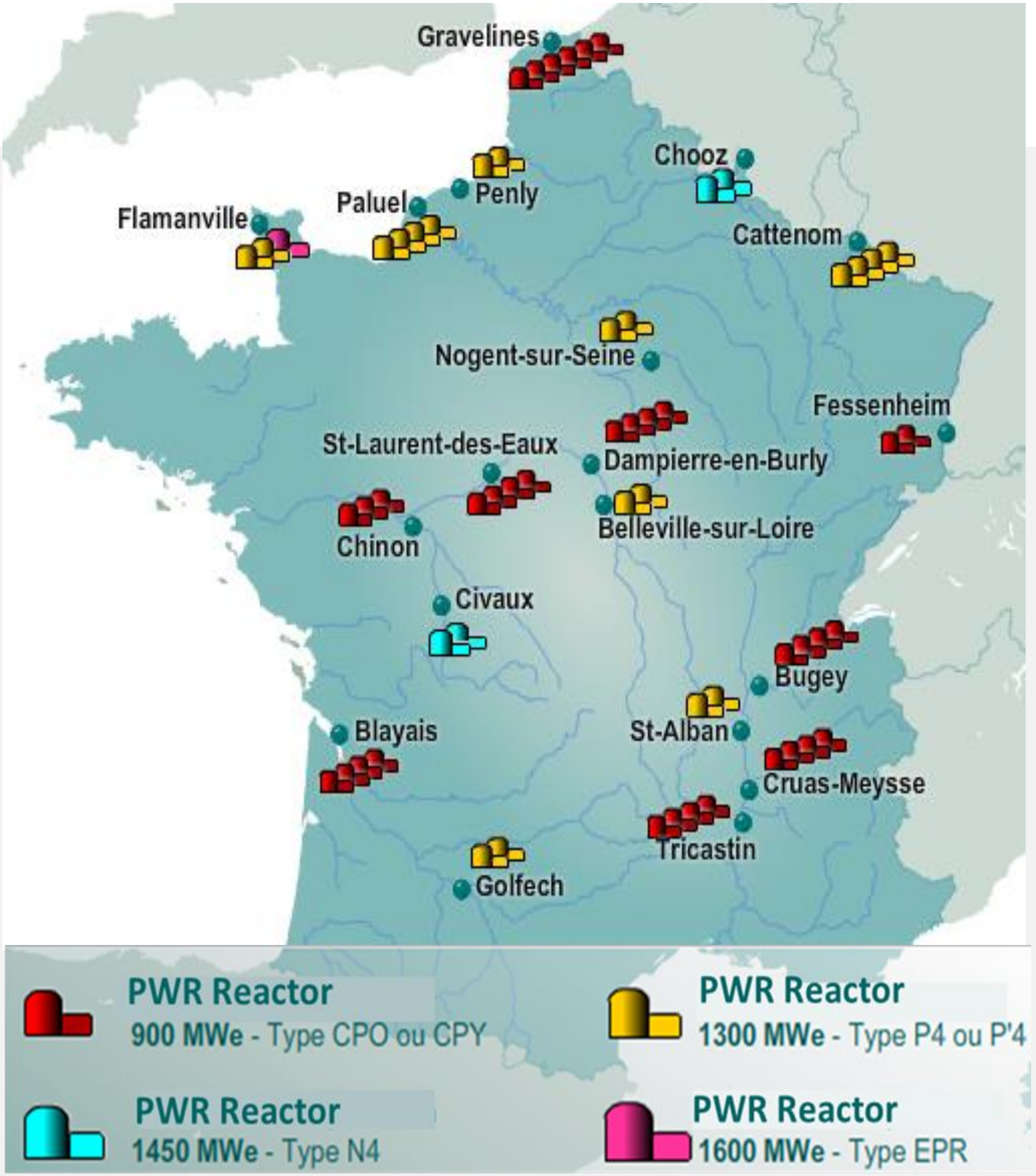
- **Regulatory framework in France:**
  - The operator is responsible for the safety of its installation
  - Continuous supervision performed by the regulator (ASN)

If serious and immediate hazard: ASN can stop nuclear installation operation at any time

  - **No legal time limit for service operation of a nuclear installation**
  - **Every 10 years: a mandatory periodic safety review** (PSR) **to determine the conditions of operation for the 10 following years**
  - **Three main goals:**
    - ✓ To operate a conformity check with current safety requirements
    - ✓ To check the ageing management programmes
    - ✓ To improve the safety of the installation with regards to the new safety objectives (currently Gen. 3+ – EPR type – objectives)

FRENCH NPP FLEET

- Standardized fleet:
  - 56 PWRs (-2, +1)
  - 18 sites
  - 1 vendor
  - 1 licensee
- 1979 - 1990: 3/4 of the fleet built
- Oldest reactor: 45 year old
- NPPs average age (/1st criticality)
  - 900 MWe: 32 reactors ⇨ 40 years
  - 1300 MWe: 20 reactors ⇨ 35 years
  - 1450 MWe: 4 reactors ⇨ 25 years



AGEING MANAGEMENT PROCESS

EDF's approach for ageing management = formal organisation, at the request of ASN, from the third ten-yearly inspections of the 900 MWe reactors from 2009 for:

- ✓ **identifying the different modes of degradation;**
- ✓ **define the associated parades;**
- ✓ **incorporate feedback.**

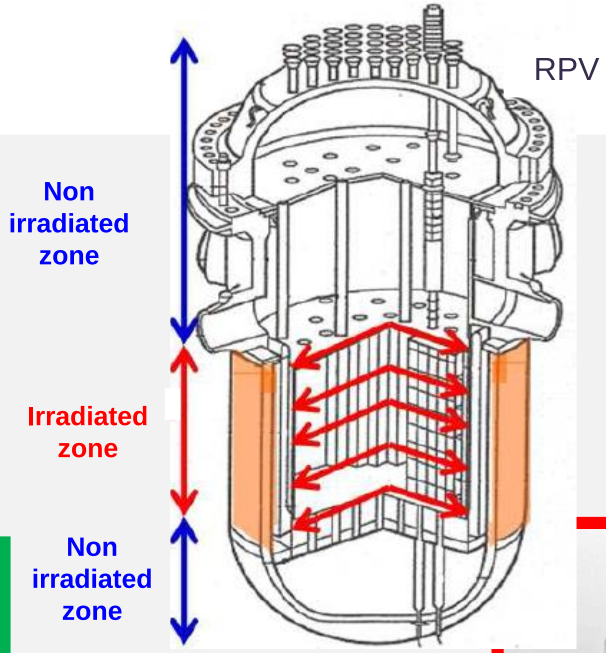
➡ **Generic** phase

➡ Application **reactor by reactor** for their **PSR3**, then **PSR4**, then…

➡ **Replaceable** and **irreplaceable** components

AGEING MANAGEMENT PROCESS

Replaceable



Irreplaceable

REGULATORY FRAMEWORK FOR THE NPE/RCPB IN SUMMARY

- Article R. 557-14-2 of the environment code
- Article 2.5.1 of the order of 7 February 2012 setting the general rules relative to basic nuclear installations
- Appendices 1 & 5 of the order of 30 December 2015 relative to nuclear pressure equipment (+ order of 12/12/2005)
- Articles 7, 8, 11, 12, 14, 15 of the order of 10 November 1999 relative to the main primary system and the main secondary systems of nuclear pressurized water reactors
- Guide ASN n°22 on Pressurised Water Reactor design

2

PROVISIONS OF THE ORDER OF 30 DECEMBER 2015 RELATIVE TO NUCLEAR PRESSURE EQUIPMENT

PROVISIONS OF THE ORDER OF 30 DECEMBER 2015 RELATIVE TO NUCLEAR PRESSURE EQUIPMENT

Article R. 557-14-2 of the Environment Code requests that *“The equipment is maintained in good condition and checked as often as necessary”.*

The order of 30 December 2015 concerning nuclear pressure equipment (NPE) details this objective by first of all making provision for ageing management measures as of the design of the equipment.

➡ **As early as the design stage, the NPE manufacturer must take account of alteration of the materials over time and of the ageing phenomenon, in particular irradiation-induced ageing, especially in Appendix 1 of the order of 30 December 2015 relative to nuclear pressure equipment**

Example:

2. Design (appendix 1)

The equipment is designed so as to minimise the risk of loss of integrity in the light of the foreseeable alterations to the materials…

This order then asks the NPE manufacturer to **meet a level of manufacturing quality consistent with the importance of the NPE built**, more specifically with regard to the production of the materials used in the various components and the methods of assembling these components. The purpose of these manufacturing provisions is **to minimise the risk of flaws forming in service**, as a result of mediocre manufacturing conditions.

It also specifies that the NPE **manufacturer must notify the licensee of all information allowing operation of the equipment in the conditions specified for its design**.

Example:

3.7. Service instructions

Pressure equipment is accompanied by an instruction manual. The instruction manual gives the particular design characteristics that are decisive factors in the service lifetime of the equipment. These characteristics comprise at least:

- *for creep, the theoretical number of operating hours at specified temperatures;*

…

This information must appear in an **instruction manual specifying the phenomena taken into consideration at the design stage and which must not be exceeded during operation, along with the operating and ageing conditions**.

Finally, this order defines the in-service monitoring conditions for the NPE with a relatively high risk, on the basis of the quantity and nature of the fluid contained (harmfulness, activity) and its pressure.

➡ **For equipment belonging to the main primary system and main secondary systems of nuclear power reactors, the order of 30 December 2015 relative to nuclear pressure equipment makes reference to the order of 10 November 1999 concerning the main primary system and main secondary systems of pressurised water reactors.**

For the NPE (excluding main primary system and main secondary systems) with a relatively high level of risk, the in-service monitoring procedures are determined by appendices 5 and 6 of the order of 30 December 2015 relative to nuclear pressure equipment. These procedures are based on three principles:

1. For each equipment item, the licensee must draft a programme of maintenance and monitoring operations, the aim of which is to manage the possible deterioration of the equipment considered in order to prevent it from failing; the licensee is required to keep this programme up to date (see *2.4 of Appendix 5);*

2. the equipment with the highest risk in terms of fluid contained and fluid pressure must systematically be periodically inspected by the licensee (every 40 months) with periodic requalification by an independent organisation (every ten years):

   a. the periodic inspections consist of an examination of the outer and inner walls of the pressurised compartments and verification of the operation of valves;

./.

b. the periodic requalifications consist of the same checks as the periodic inspections, plus containment pressure tests of the pressurised compartments.

3. equipment repairs and modifications are carried out in accordance with the rules in force for the design and manufacture of new equipment.

The regulations authorise certain adaptations of the in-service monitoring rules. It is sometimes difficult to carry out internal inspections and pressure tests on equipment not originally designed to meet relatively recent regulatory requirements (January 2006). In this case, the licensee takes compensatory measures and carries out more frequent and/or more precisely targeted monitoring.

3

PROVISIONS OF THE ORDER OF 10 NOVEMBER 1999 RELATIVE TO THE MAIN PRIMARY SYSTEM AND THE MAIN SECONDARY SYSTEMS OF NUCLEAR PRESSURIZED WATER REACTORS

THE PROVISIONS OF THE ORDER OF 10 NOVEMBER 1999 RELATIVE TO THE MAIN PRIMARY SYSTEM AND THE MAIN SECONDARY SYSTEMS OF NUCLEAR PRESSURIZED WATER REACTORS

The order of 10 November 1999 relative to the monitoring of operation of the main primary system and the main secondary systems of pressurized water nuclear reactors requires monitoring of the NPE making up these systems. The systems are called “equipment” in the order.

This monitoring comprises:

- **periodic equipment monitoring programmes to verify the absence of defects or, if manufacturing defects are indeed found, to check that they do not develop;**
- **a programme to monitor the degradation modes of the properties of the materials;**
- **a precise documentary system precisely identifying the actions to which the equipment has been subjected and indicating all the observations liable to affect its maintained integrity.**

In-service monitoring of the main primary system and the main secondary systems is regulated by articles 14 and 15.

The **in-service monitoring provisions are stipulated in article 14**.

Operations regarding periodic requalification are regulated by article 15. They more specifically include a complete inspection and a pressure test. The maximum interval of ten years between two periodic inspections is consistent with the time interval between two periodic safety reviews.

As of the third ten-yearly outage inspection, the regulations require partial requalification comprising an in-depth inspection of the equipment five years after the ten-yearly outage inspection, this being applicable to all subsequent ten-yearly outage inspections.

Articles 11 and 12 contain provisions for monitoring the chemical conditions which could have an impact on the equipment and provisions for monitoring degradation phenomena of the properties of the materials.

Finally, **article 7 provides for a documentary record of the findings made on the pressure equipment liable to affect their integrity and the actions to which they were subjected**.

Pursuant to **article 8, the NDT processes must be qualified prior to implementation on the site by an independent, recognised organisation**.

As a type B internal inspection organisation as defined by standard NF EN ISO / CEI 17020, EDF NDT Qualification Commission has been accredited by COFRAC (French Accreditation Committee) since 2002.

The role of this Commission is to provide completely independent confirmation of the compliance of the NDT processes to be qualified with the Licensee’s functional requirements baseline.
This compliance review results in the issue of a certificate of compliance in the form of an attestation of qualification.

4
WHAT ABOUT CODES AND STANDARDS?

ASN GUIDE N° 22 ON THE DESIGN OF PRESSURISED WATER REACTORS

Guide n° 22 “Design of pressurised water reactors” published on 18 July 2017, was produced jointly by ASN and IRSN: it presents the ASN and IRSN recommendations for the design of pressurised water reactors and is intended for future PWR licensees in France, who are responsible for managing the risks and drawbacks of their facility in accordance with article L.593-6 of the Environment Code.

Its article 1.3 states that “As the recommendations of this guide apply primarily to the design of new PWRs, they may also be used, for reference, to seek improvements to existing reactors, for example on the occasion of their periodic safety reviews, in accordance with article L. 593-18 of the Environment Code and articles 8a and 8c introduced by the European Directive of 8 July 2014.”

Part IV.2.5 specifically deals with taking account of industrial practices, maintenance, in-service monitoring and the constraints relative to their ageing in the design of EIPs. **Article 4.2.5.3 states that “measures must be taken at the design stage to facilitate monitoring of the planned ageing mechanisms and to detect any deterioration or unexpected behaviour that could arise during operation of the BNI**.”

Even if it expects the industry to follow the regulatory requirements, ASN does not asses the codes although they are supposed to take into account such requirements.

For EPR, its conformity is assessed according to regulations, not to codes.

However, interactions exist at different levels between regulation and codes-standards:

- Manufacturing for construction and repair (including replacement) ➔ RCC-M
  - ✓ During the design evaluation, demonstration of the mechanical resistance shall be provided (fatigue, fracture,..)
  - ✓ Materials properties shall take into account ageing (irradiation, corrosion…)
- NDT qualification ➔ RSE-M
- Flaw treatment ➔ RSE-M
- Risk of obsolescence ➔ Quality of manufacturing and supply chain

Thank you for your attention!

Any question?

AUTORITÉ
DE SÛRETÉ
NUCLÉAIRE

Codes and Standards and Ageing Management in the United States

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International Workshop on Ageing Management Considerations in Mechanical Codes and Standards
28-29 June 2023 – Tokyo, Japan

Outline

- Overview of ageing management and license renewal in the US
- Mechanical codes and standards use in the US
- Relationship between ageing management and codes and standards

Status of U.S. Reactors

- 93 operating power reactors[1]
    - 9 with 40-year licenses
    - 78 with 60-year licenses
    - 6 with 80-year licenses[2]

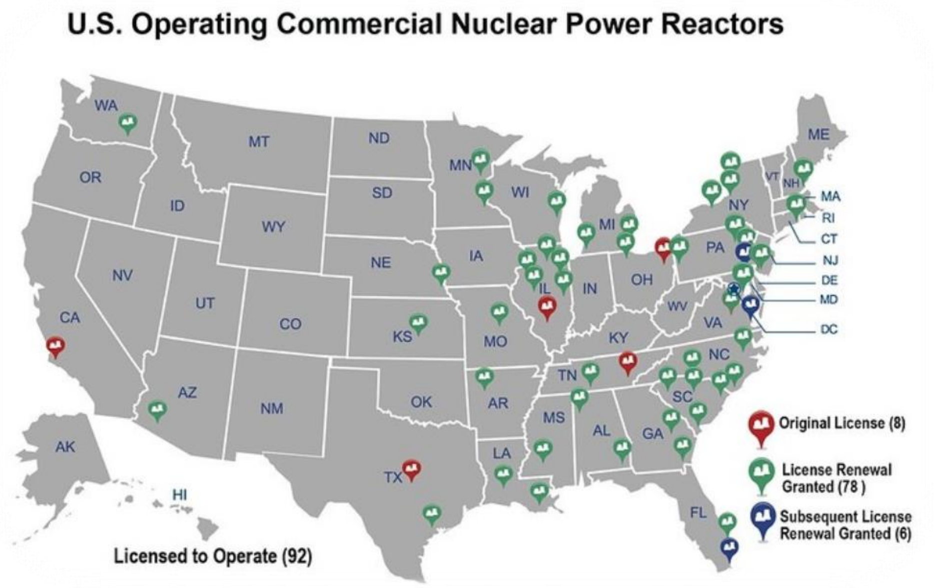
updated May 2023

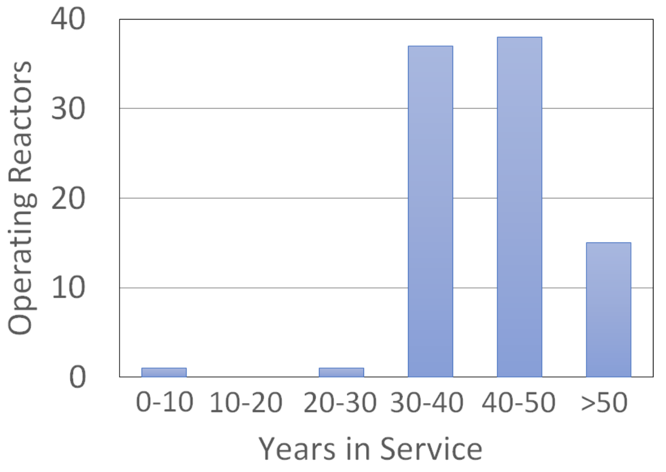
[1] Does not include one additional reactor that will soon start commercial operation (Vogtle 4)

[2] Some 80-year licenses have been reset to back to 60 years until environmental impacts are reevaluated

- 53 of the operating reactors have exceeded 40 years of service (oldest at 53 years)

- 9 additional reactors exceeded 40 years, but have been shut down





Ageing Management

- Ageing management is for proactive assurance against degradation, or proactive management of known degradation throughout the plant life cycle
- Original license term - First 40 years
  - Regulations, guidance, generic communication and codes and standards
- License renewal and subsequent license renewal – 40-60 years and 60-80 years
  - Regulations, ageing management programmes and codes and standards

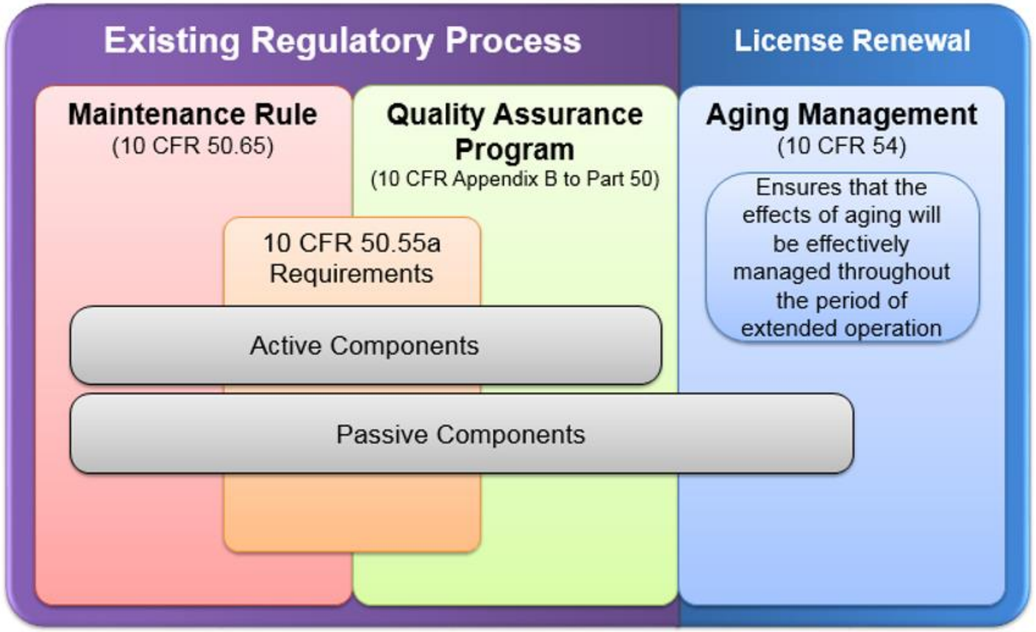
License Renewal

10 CFR Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants”

- A renewal application may not be submitted earlier than 20 years before the expiration of the current license
  - Allows for the accumulation of sufficient operating experience to inform a licensing decision
- Licenses may be renewed for up to 20 additional years
  - 20-year limit provides opportunity to reassess the understanding of age-related degradation effects and adequacy of plant programmes
- Renewed licenses may be subsequently renewed

License Renewal vs. Existing Processes

- Focus is on managing the effects of aging of long-lived, passive structures and components important to safety
- Remaining structures and components are adequately addressed by existing plant programmes to meet requirements for:
  - Maintenance
  - Quality assurance
  - Codes and standards activities (e.g., ASME)



active vs. passive: performance with or without moving parts or a change in configuration/properties

Safety Review

SSCs in Scope

- Safety-related (SR)
- Non-safety whose failure may affect SR functions
- Relied on for selected regulated events (e.g., fire protection, station blackout)

10 CFR 54.4

passive &

long-lived

54.21(a)(1)

Ageing Management Review

For each SSC:

- Identify material and service environment
- Determine aging effects requiring management

54.21(a)(3)

Ageing Management Programmes

Demonstrate aging effects will be managed

Credit existing programmes enhanced, as necessary (e.g., ASME Section XI, Water Chemistry)

Develop new programmes (typical examples: Selective Leaching, Thermal Embrittlement of CASS)

Safety Review

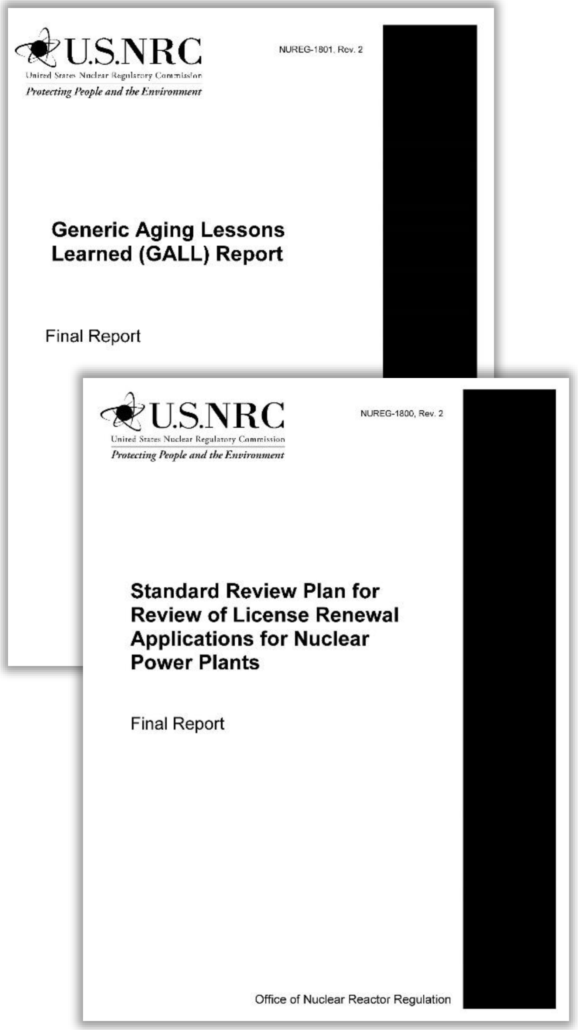
Time-Limited Ageing Analyses (TLAAs)

- Analyses in the original licensing basis that involve a time-dependent aging assumption
  - Examples: metal fatigue (including calcs for ASME Section III), reactor vessel embrittlement

- Renewal requirement – perform <u>one</u> of the following:
  1) Demonstrate the existing analysis remains valid for the extended period
  2) Project the analysis for the extended period
  3) Manage the aging effect with a programme

License Renewal Guidance Documents

- Generic Aging Lessons Learned (GALL) Report
  - NUREG-1800, Rev. 2 (License Renewal) and NUREG-2191 (Subsequent LR)
  - Provides ageing assessments, including identification of materials, environments, and ageing effects that require management
  - Identifies acceptable **ageing management programmes (AMPs)**
- Standard Review Plan (SRP)
  - NUREG-1801, Rev. 2 (License Renewal) and NUREG-2192 (Subsequent LR)
  - Provides guidance for NRC staff review of renewal applications
- NUREG-2191 and 2192 first issued in 2017; Currently undergoing revision (considering new operating experience, updates of codes and standards, lessons learned) – New revision coming this summer



NRC's Policies on Consensus Codes and Standards

- Consensus codes and standards (C&S) have been integral to the regulatory process since 1971.
- C&S promote safe operation of nuclear facilities and improve the effectiveness and efficiency of regulatory oversight
- Federal law requires agencies to use technical standards developed by voluntary consensus standards bodies and participate in their development, where possible

National Technology Transfer and Advancement Act of 1995

PUBLIC LAW 104-113 NATIONAL TECHNOLOGY TRANSFER AND ADVANCEMENT ACT OF 1995

Including Amendment by Public LAW 107-107, section 1115 on Dec 28 2001.
UTILIZATION OF CONSENSUS TECHNICAL STANDARDS BY FEDERAL AGENCIES

OFFICE OF MANAGEMENT AND BUDGET

Revision of OMB Circular No. A–119, "Federal Participation in the Development and Use of Voluntary Consensus Standards and in Conformity Assessment Activities"

AGENCY: Office of Management and Budget, Executive Office of the President.

ACTION: Notice of availability.

SUMMARY: The Office of Management

NRC Formal Endorsement Process

- Rules and regulations: Title 10 to the Code of Federal Regulations (10 CFR) – 10 CFR 50.55a for codes and standards
- Regulatory Guides (RGs)
- Standard Review Plans (SRPs)
- Generic communications (e.g., Regulatory Issue Summary)

NRC Endorsement Process: Rulemaking

- Most formal of NRC endorsement processes
- Standards ***incorporated by reference*** in rulemaking process become regulatory requirements
- Includes public comment and rigorous reviews
- Takes several years to complete a typical rulemaking

For current operating fleet, this process is used to mandate the use of ASME BPV Section III and XI and OM codes

Codes and Standards in Ageing Management

- For sake of discussion, going to only focus on Inservice inspection (ISI), but also pertains to design, construction, inservice testing, etc.

- C&S ISI requirements are referenced throughout GALL-SLR (NUREG-2191). Some examples include:
  - XI.M1 - ASME SECTION XI INSERVICE INSPECTION, SUBSECTIONS IWB, IWC, AND IWD
  - XI.M35 ASME CODE CLASS 1 SMALL-BORE PIPING
  - XI.S1 ASME SECTION XI, SUBSECTION IWE
  - XI.S2 ASME SECTION XI, SUBSECTION IWL
  - XI.S3 ASME SECTION XI, SUBSECTION IWF

- GALL often augments C&S with preventive actions and/or more stringent or extensive inspections

- Staff review future editions of the ASME Code for their adequacy for subsequent license renewal to ensure the conclusions of GALL/GALL-SLR remain valid

C&S Enhancement

- Ageing management is for proactive assurance against degradation, or proactive management of known degradation
- ISI C&S are typically reactive to operating experience,
  - How and where to inspect, and how to disposition indications
  - New code rules are formed based on OE
- Ageing management relies on C&S rules as one way to manage degradation

- Areas of possible C&S enhancement to support ageing management:
  - Components susceptible to embrittlement – reactor vessel embrittlement prediction & surveillance (neutron), monitoring for cracking of cast austenitic stainless steels (thermal)
  - Vessel & internals – additional, more stringent inspections in areas of higher safety significance or leading indicators of degradation (BWR ID attachment welds, PWR internals)
  - Bolting – preventive measures: reactor head closure studs; IWE containment bolting

Summary

- Ageing management is a critical tool for the prevention and mitigation of passive component degradation in all areas of a power plant life cycle
- In the US, regulations and guidance, supplemented by C&S, provide ageing management requirements for the US Fleet
- Enhancing C&S to provide additional ageing management requirements may promote international regulatory harmonization

Questions ? Answers !

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Lunch Break

Workshop will resume at 14:00PM

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