Introduction of session 2

Mr Ronan Tanguy

CORDEL Programme Lead at World Nuclear Association

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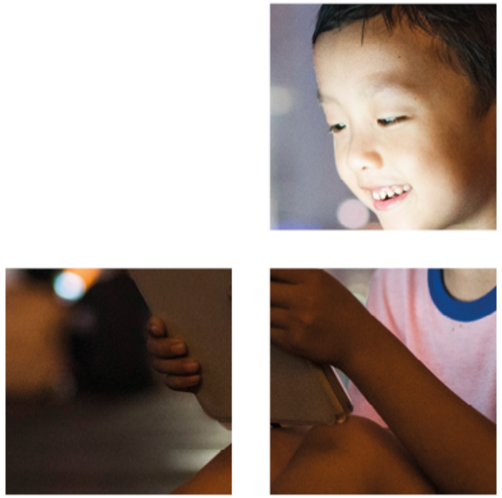

Session 2: Code requirements on RCPB to address/prevent ageing

Ronan Tanguy

Background on codes & standards

- Engineering codes & standards provide the technical requirements to which nuclear power plant SSC must conform
- IAEA SSR-2/1 lists requirements applicable to the design of nuclear power plants

Requirement 18: Engineering design rules

The engineering design rules for items important to safety at a nuclear power plant shall be specified and shall comply with the relevant national or international codes and standards and with proven engineering practices, with due account taken of their relevance to nuclear power technology.

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Background on codes & standards

- SSR-2/1 Requirement 18 allows the use of a wide variety of relevant codes and standards.
- Many countries favour their national codes and standards when available.
- The selection of codes and standards for a reactor design often varies between countries and can even vary between projects within a country if the regulatory framework allows it.
- Differences between codes and standards can require regulators, designers and manufacturers to adapt and learn a range of codes which is time consuming, expensive and limits export opportunities.

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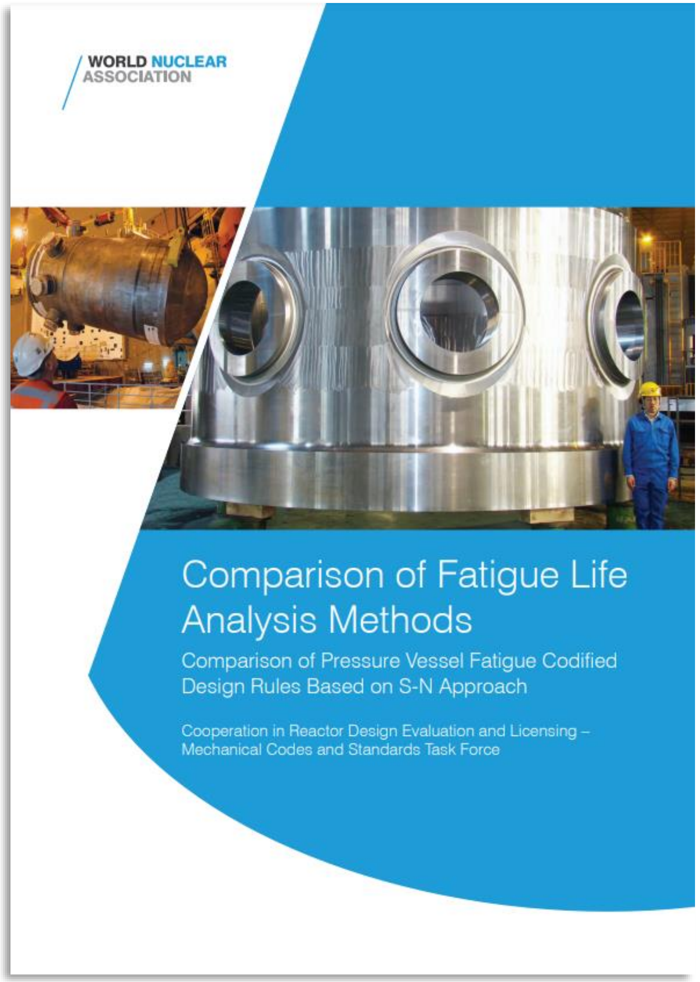
Understanding differences in code requirements

- Comparison exercises have undertaken to understand the differences and overlaps between codes and standards to assist stakeholders
- World Nuclear Association published a report examining how fatigue analysis for Class 1 vessels is approached in BPVC, RCC-M, JSME and KEPIC
- The Standard Developing Organisations Convergence Board works to minimise divergence of codes and converge where possible
- IAEA Nuclear Harmonization & Standardization Initiative is undertaking a broad code comparison exercise



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Aims of the session

- Provide an overview of ageing management requirements in nuclear mechanical C&S
- Highlight differences between C&S with regards to ageing phenomena for RCPB (materials, loading conditions, degradation mechanisms, approaches considered)
- Understand how and why different C&S were developed
- Understand the differences in approaches to RCPB ageing within the various C&S for the various stakeholder groups
- Understand how SDOs could incorporate lessons learnt and could undertake future work in a harmonised manner

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ASME Code Requirements for Fatigue

Paul R. Donavin, P.E.

Becht Engineering

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ASME Code Requirements for Fatigue

Paul R. Donavin, P.E.

International Workshop on Ageing Management Considerations In Mechanical Codes And Standards
Tokyo, Japan 28-29 June 2023

Background

- Reactor Pressure Boundary Components
  - Designed to the ASME Boiler & Pressure Vessel Code, Section III Rules for Construction of Nuclear Facility Components
- Design ASME Section III
  - Subsection NB (Class 1) (NB-3000)
  - Subsection NG (Core Support Structures)
- Fabrication ASME Section III
  - Subsection NB-4000

ASME
SETTING THE STANDARD

Basic Design Analyses

- Design Analyses include:
  - (a) internal pressure;
  - (b) impact loads, including rapidly fluctuating pressures;
  - (c) weight of the component and normal contents
  - (d) vibrations and earthquake
  - (f) reactions of supporting lugs, rings, saddles, etc
  - (g) temperature effects.

ASME
SETTING THE STANDARD

Background

- Fatigue in Section III has been consolidated into Appendix XIII Design Based On Stress Analysis

a) Stress Differences

b) Local Structural Discontinuities

c) Design Fatigue Curves

d) Effect of Elastic Modulus

e) Cumulative Damage

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SETTING THE STANDARD

Requirements

- Time Limited Aging Analyses
  - TLAAs must be
    - (a) verified to bound the renewal period;
    - (b) reanalyzed (recalculated) to determine if it will bound the renewal period; or
    - (c) the applicant must show that the aging effects encompassed by the calculation will be managed.

ASME SETTING THE STANDARD

Fatigue in License Renewal

- Original Analyses Had Conservative Assumptions
  - Severity and number of operating transients
  - Little actual power plant operating experience
- Design Approach Conservative
  - Limited modeling detail
  - Conservative Fatigue Design Curves
  - Assumed if fatigue usage factor equaled 1 the component’s fatigue life ended
- Environmental Assisted Fatigue
  - ASME Issued Various Code Cases
  - Relied on NRC Issued NUREG Series Documents

License Renewal Approach

- Reviewed actual operating experience
  - Reviewed plant operating records
  - Reviewed plant operating procedures and training
  - Interviewed Reactor Operators (Licensed Operators)
- Remodeled operating transients based on actual experience
- Assessed the actual cumulative usage based on modified cycle count
- Adjusted final usage factors by environmental fatigue factors (Fen)

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Results

- Fatigue usage was less than assumed
  - Most components had very little calculated fatigue damage
  - Certain components had to reduce the anticipated number of design cycles than originally assumed.
    - For example, the number of heatups was reduced from 200 to 142
- No replacement of RCPB components were needed

ASME Fatigue

- **Any Questions?**

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Outline of JSME rules on fitness for service for Nuclear Power Plants

Dr Kiminobu Hojo

Mistubishi Heavy Industry, Ltd.

JSME Codes and Standards Committee

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OUTLINE OF JSME RULES ON FITNESS FOR SERVICE FOR NUCLEAR POWER PLANTS

Nuclear Energy Agency Workshop
June 28, 2023, Tokyo

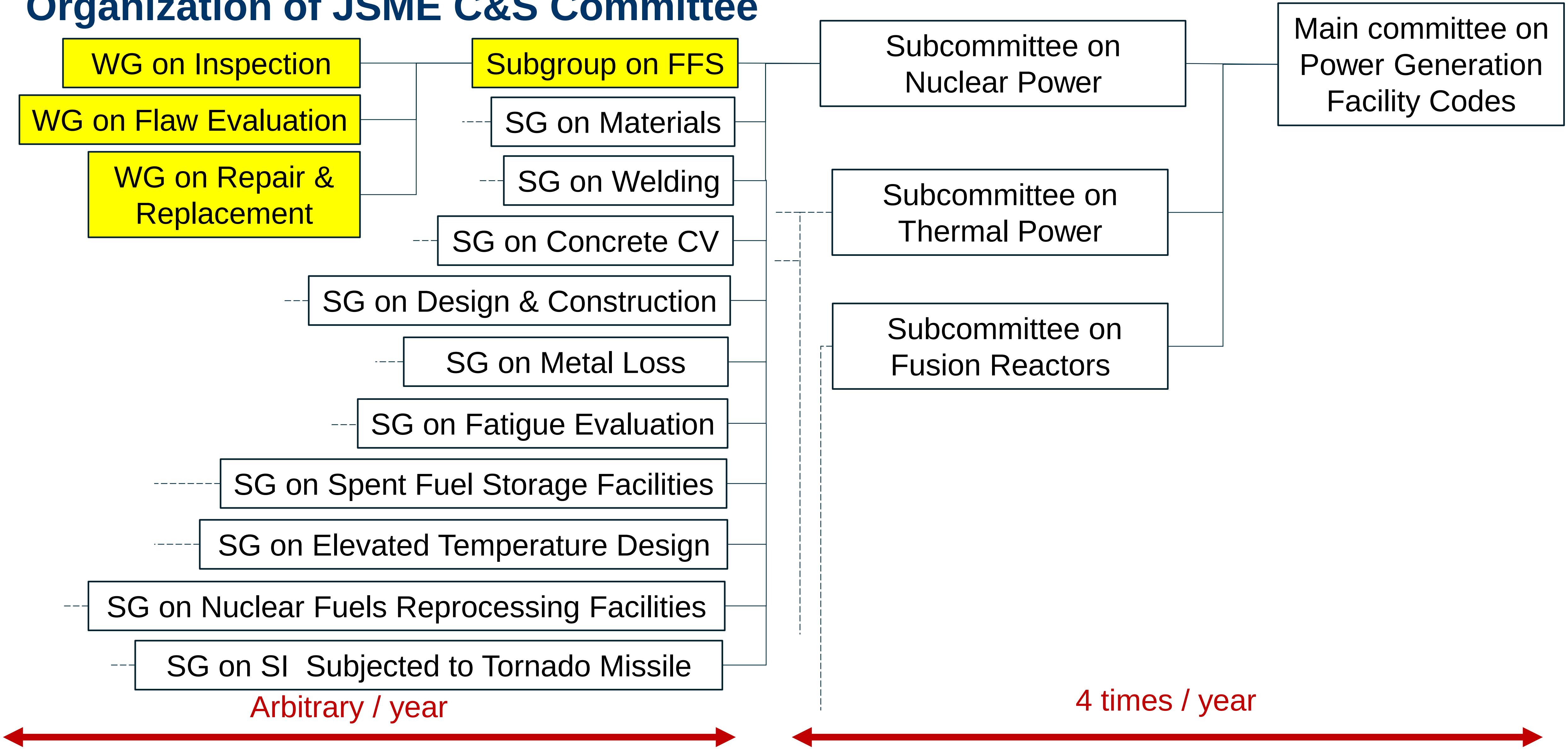
Presenter : Kiminobu Hojo
(Member of SG on FFS, Chair of WG on Flaw Evaluation, MHI)
Subgroup on Fitness for Service
Nuclear Subcommittee
JSME Codes and Standards Committee

CONTENTS

1. INTRODUCTION
2. STRUCTURE OF JSME RULES ON FITNESS FOR SERVICE FOR NPP
3. EXAMPLES OF CHAPTER ‘FLAW EVALUATION’ IN JSME RULES ON FFS
4. HARMONIZATION WITH ASME SEC. XI AND OTHER CODES
5. CONCLUSIONS

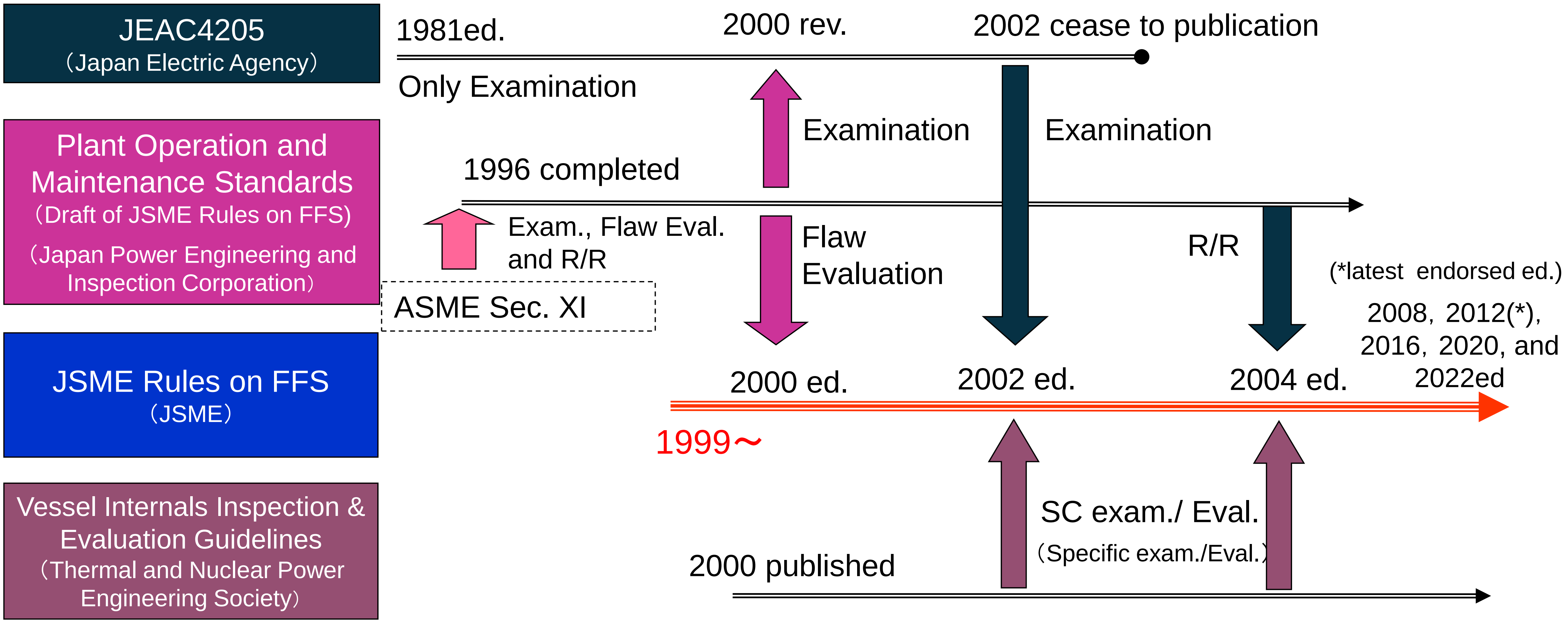
1. INTRODUCTION

Organization of JSME C&S Committee



1. INTRODUCTION

History of JSME Rules on Fitness For Service



2. STRUCTURE OF JSME RULES ON FITNESS FOR SERVICE

<table>
<tr><th colspan="2">Chapter Examination</th><th colspan="2">Chapter Flaw Evaluation</th><th colspan="2">Chapter Repair and Replacement</th></tr>
<tr><td colspan="2">IA: General Requirements for Examination</td><td colspan="2">EA: General Requirements for Evaluation</td><td colspan="2">RA: General Requirements for Repair/Replacement</td></tr>
<tr><td rowspan="6">I*: Standard Examination</td><td>IB: Class 1 component</td><td rowspan="6">E*: Flaw Evaluation of Class Component</td><td>EB: Class 1 Component</td><td rowspan="8">RB: Repair Technique and Procedure</td><td rowspan="4">RB-1000: General requirement for repair technique</td></tr>
<tr><td>IC: Class 2 component</td><td>EC: Class 2 component</td></tr>
<tr><td>ID: Class 3 Component</td><td>ED: Class 3 Component</td></tr>
<tr><td>IE: Class MC Component</td><td>EE: Class MC Component</td></tr>
<tr><td>IF: Support structure</td><td>EF: Support structure</td><td rowspan="4">RB-2000: Procedure</td></tr>
<tr><td>IG: Core Structure</td><td>EG: Core Structure</td></tr>
<tr><td rowspan="2">IJ*: Specific Examination</td><td>IJB: Class 1 component</td><td rowspan="2">EJG: Flaw Evaluation of Core Structure by Specific Examination</td><td>EJG-1000: Basic requirement</td></tr>
<tr><td>IJG: Core Structure</td><td>EJG-1300: Flaw Evaluation</td></tr>
</table>

Nonmandatory Appendices for Flaw Evaluation (2020ed)

E*: Flaw Evaluation of Class Component

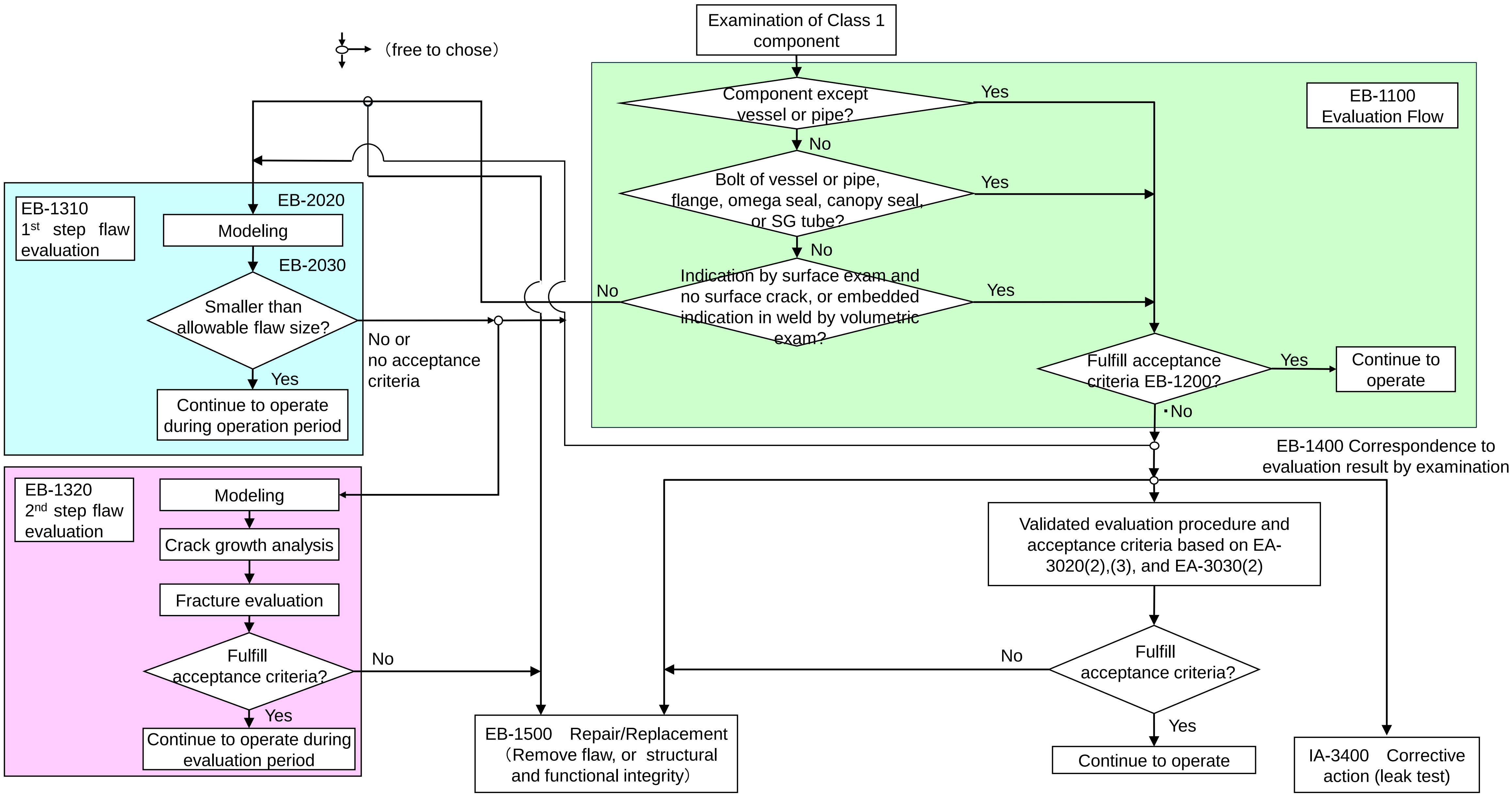
- E-1 Flaw modeling
- E-2 Crack Growth Rate
- E-3 Flaw Shape Evaluation Method
- E-4 Proximity Rule
- E-5 Stress Intensity Factors
- E-6 K_{Ia} and K_{Ic} Curves
- E-7 Loads for Flaw Evaluation
- E-8 Limit Load Evaluation Procedure
- E-9 Elastic Plastic Fracture Mechanics Procedure
- E-10 Two Parameters Method
- E-11 Selection of Fracture Evaluation
- E-12 Fracture Toughness for Flaw Evaluation of Ferritic Steel Piping
- E-13 Weld Joint Modeling for Core Structure
- E-14 Selection of Fracture Evaluation of Core Structure
- E-15 Linear Elastic Fracture Mechanics Procedure of Core Structure
- E-16 Twice Slope Method
- E-17 Function Evaluation of Core Structure
- E-18 Material Properties for Fracture Evaluation of Cast Stainless Steel Piping

Nonmandatory Appendices for Flaw Evaluation (2020ed)

EJG: Flaw Evaluation of Core Structure by Specific Examination

EJG-B-1	Flaw Evaluation of Core Shroud Support
EJG-B-2	Flaw Evaluation of Core Shroud
EJG-B-3	Flaw Evaluation of Upper Grid Plate
EJG-B-4	Flaw Evaluation of Jet Pump Pipe
EJG-B-5	Flaw Evaluation of Core Spray Pipe
EJG-P-1	Damage Prediction of Baffle Former Bolt
EJG-P-2	Flaw Evaluation of Barrel Former Bolt
EJG-P-3	Flaw Evaluation of Core Barrel
EJG-P-4	Wear Evaluation of Guide Tube of Control Rode

Flow of Flaw Evaluation of Class 1 Component



3. EXAMPLES OF CHAPTER 'FLAW EVALUATION' IN JSME RULES ON FFS

Appendix E-2
Crack growth rate (2020ed)

Crack growth mode	Component and material	Environment and flaw shape
Fatigue	Ferritic steel for vessels and piping	Surface flaw and subsurface flaw in air
		Surface flaw in LWR env. (BWR, PWR)
	Austenitic steel for piping and core structures	Surface flaw and subsurface flaw in air
		Surface flaw in BWR env.
		Surface flaw in PWR primary water env.
	Nickel alloy and weld metal for piping and core structures	Surface flaw and subsurface flaw in air
		Surface flaw in BWR env.
		Surface flaw in PWR primary water env.
SCC	Austenitic steel for piping (sensitized SS304)	Normal BWR water env. (-70≦ECP≦+50mV$_{SHE}$ and conductivity<20μS/m）
	Austenitic steel for piping and core structures (sensitized SS304, low carbon stainless steel)	Normal BWR normal water chemistry (NWC) (ECP≧150mV$_{SHE}$ and Conductivity< 20μS/m)
		BWR hydrogen water chemistry (HWC) (ECP≦-100mV$_{SHE}$ and conductivity<20μS/m)
	Nickel alloy and weld metal for piping and core structures	Normal BWR env. (ECP>100mV$_{SHE}$)
		Hydrogen injected BWR env.
		PWR primary water env.
	Austenitic steel for core structures (sensitized SS304, low carbon stainless steel)	BWR env. (neutron irradiation >5×10^{24}n/m^2 (E＞1MeV))

FCGR of ferritic steel in LWR env. (BWR, PWR)

$R \leqq 0.25$

$\Delta K \leqq 19.48$ $\quad da/dN = 1.48\times10^{-14}\Delta K^{5.95}$

$\Delta K > 19.48$ $\quad da/dN = 2.13\times10^{-9}\Delta K^{1.95}$

$0.25 < R < 0.65$

$$\Delta K_a = 19.49\left(\frac{3.75R+0.06}{26.9R-5.725}\right)^{0.25}$$

$\Delta K \leqq \Delta K_a$ $\quad da/dN = 1.48\times10^{-14}(26.9R-5.725)\Delta K^{5.95}$

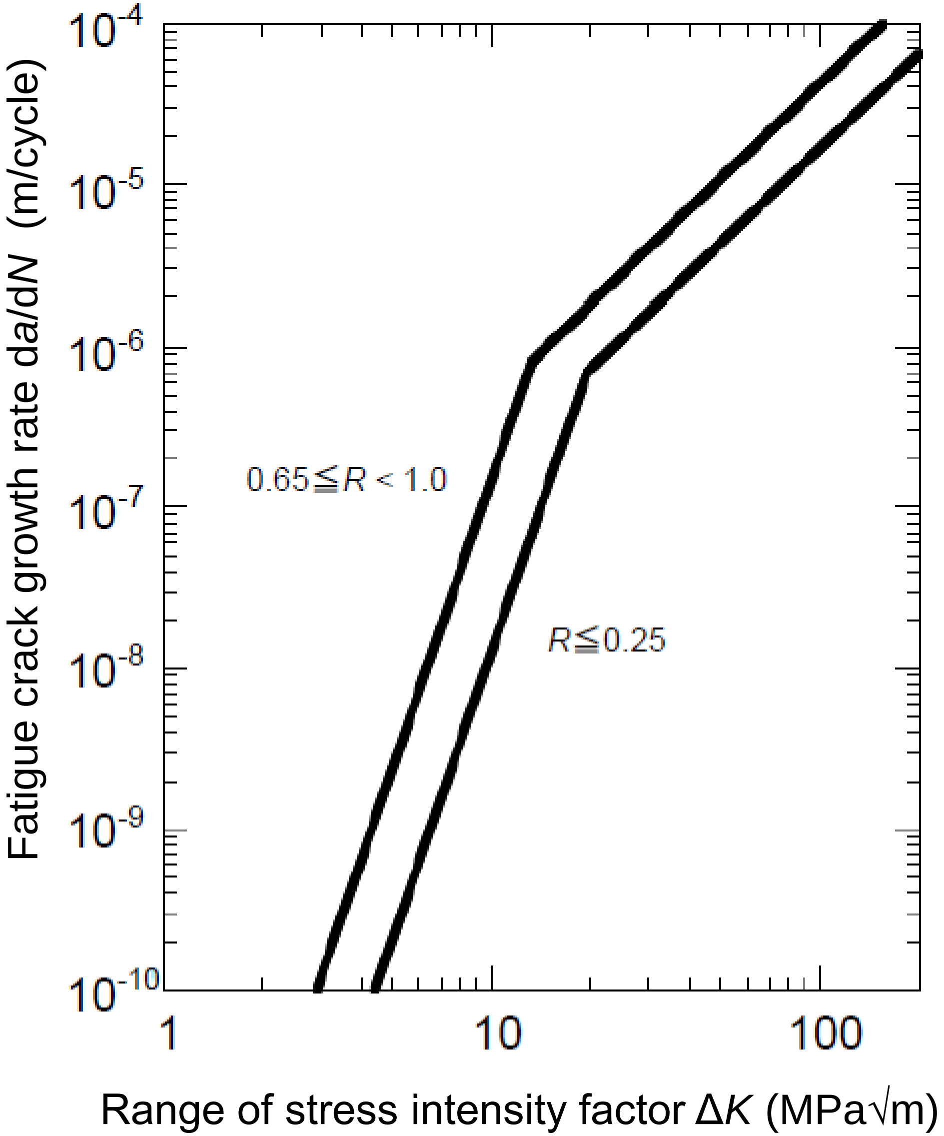
$\Delta K > \Delta K_a$ $\quad da/dN = 2.13\times10^{-9}\,(3.75R+0.06)\Delta K^{1.95}$

$0.65 \leqq R < 1.0$

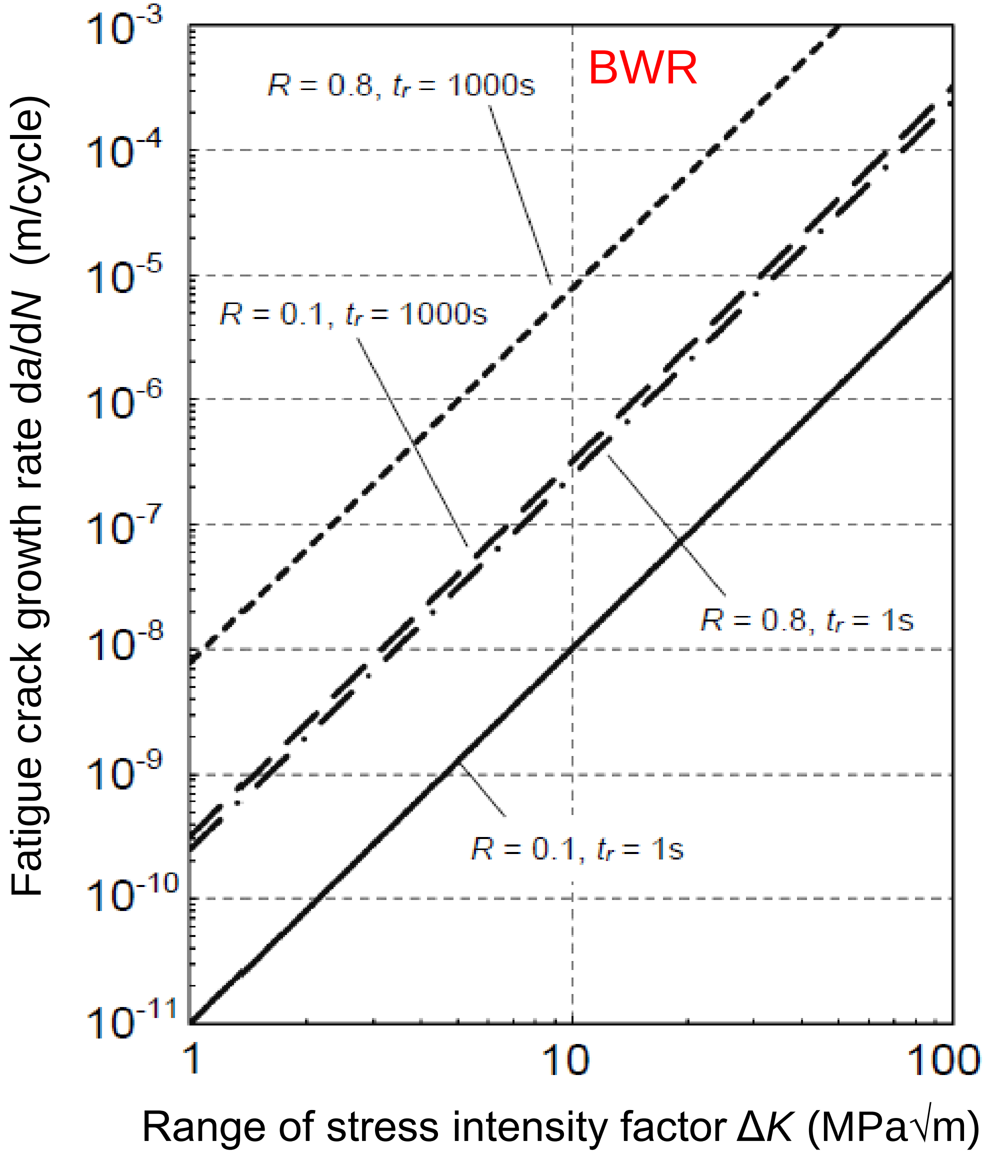
$\Delta K \leqq 13.23$ $\quad da/dN = 1.74\times10^{-13}\Delta K^{5.95}$

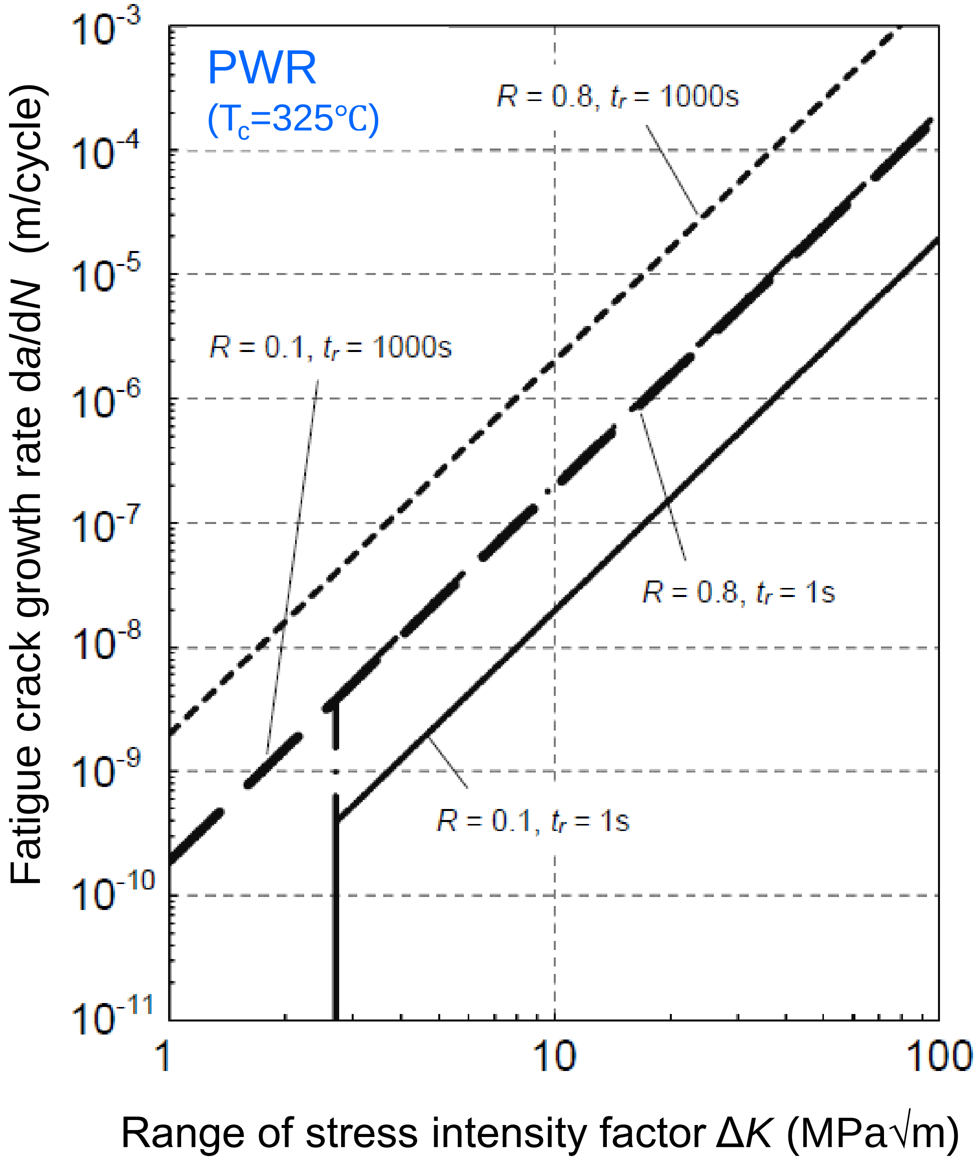
$\Delta K > 13.23$ $\quad da/dN = 5.33\times10^{-9}\Delta K^{1.95}$

Unit : da/dN (m/cycle), ΔK (MPa$\sqrt{\text{m}}$)

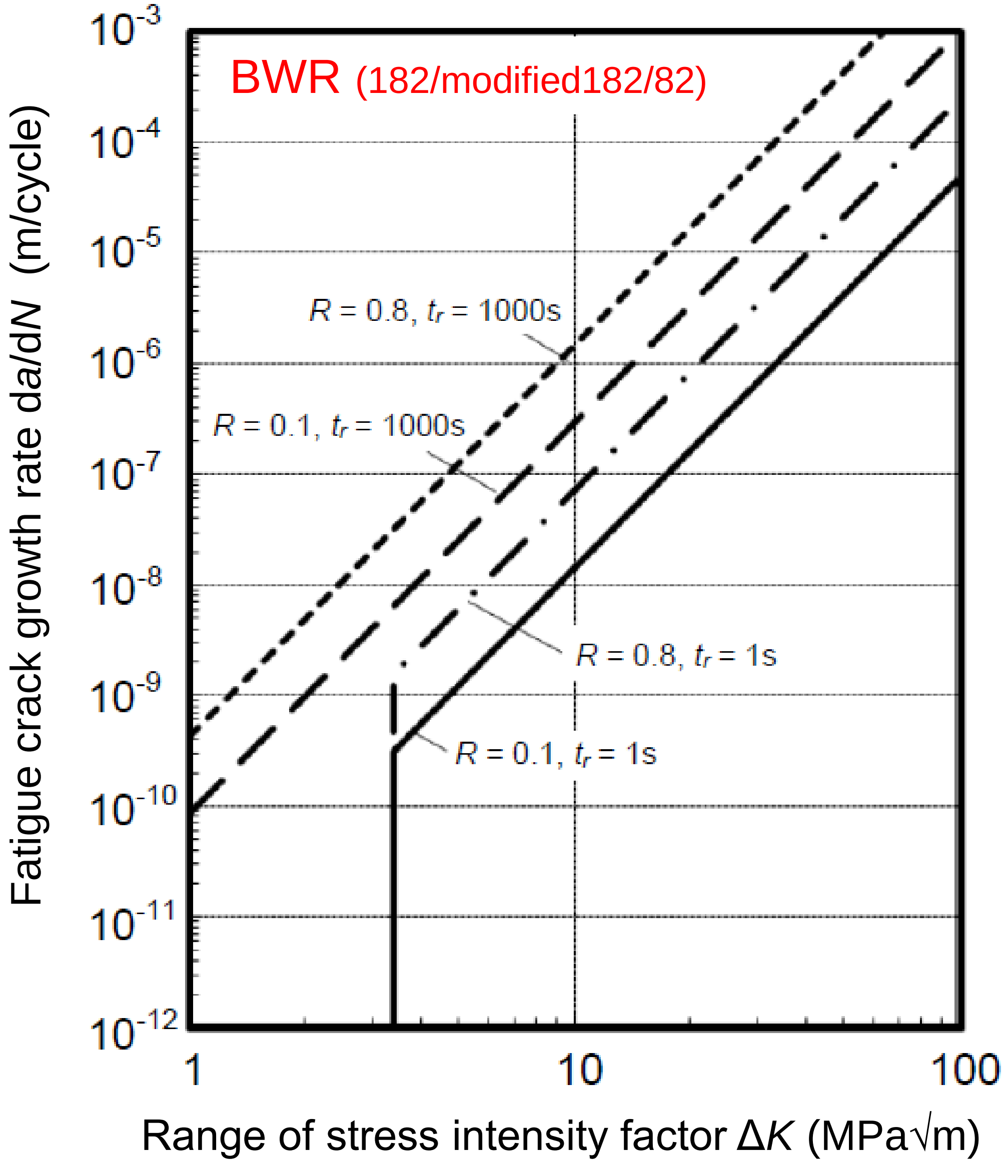


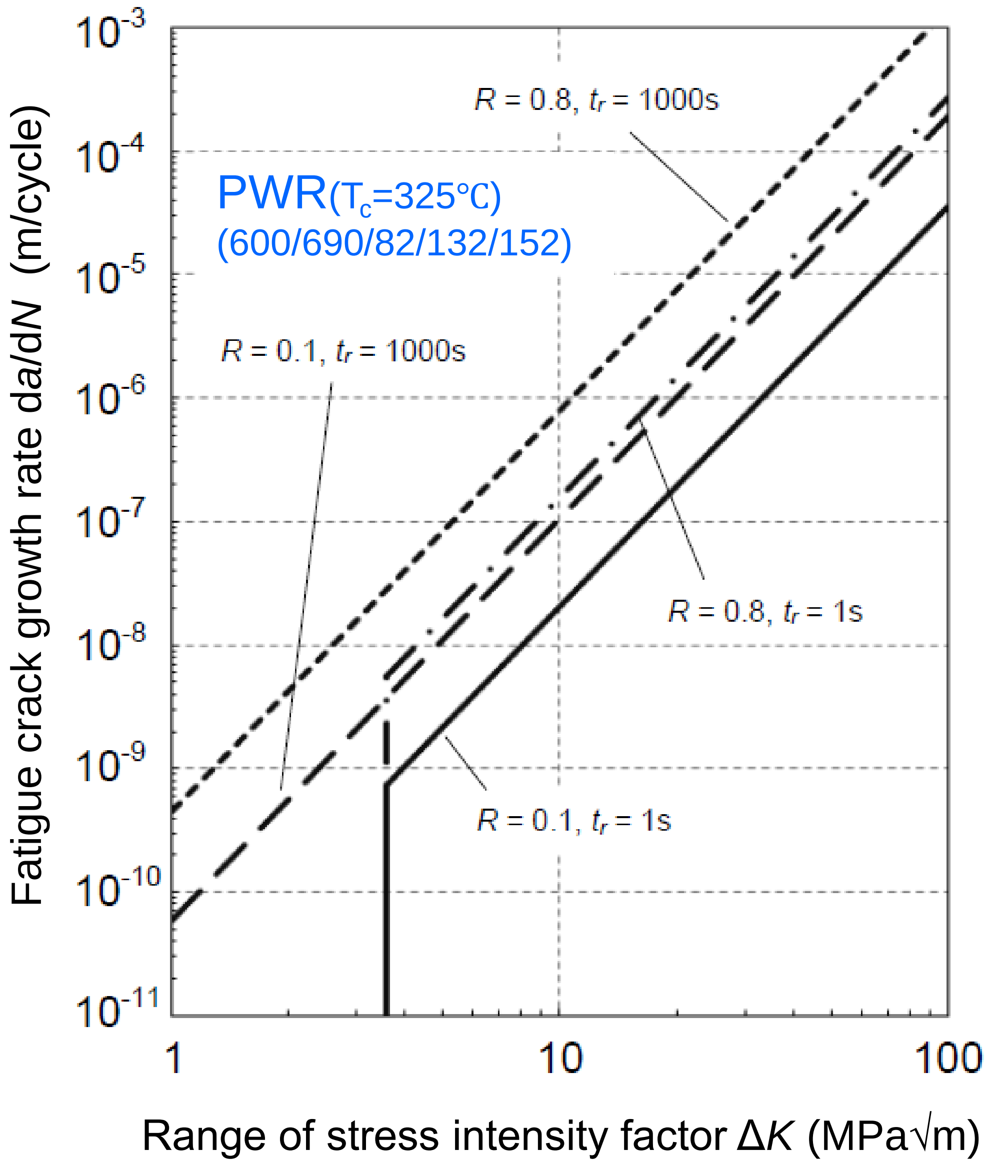
FCGR of austenitic stainless steel in LWR env.



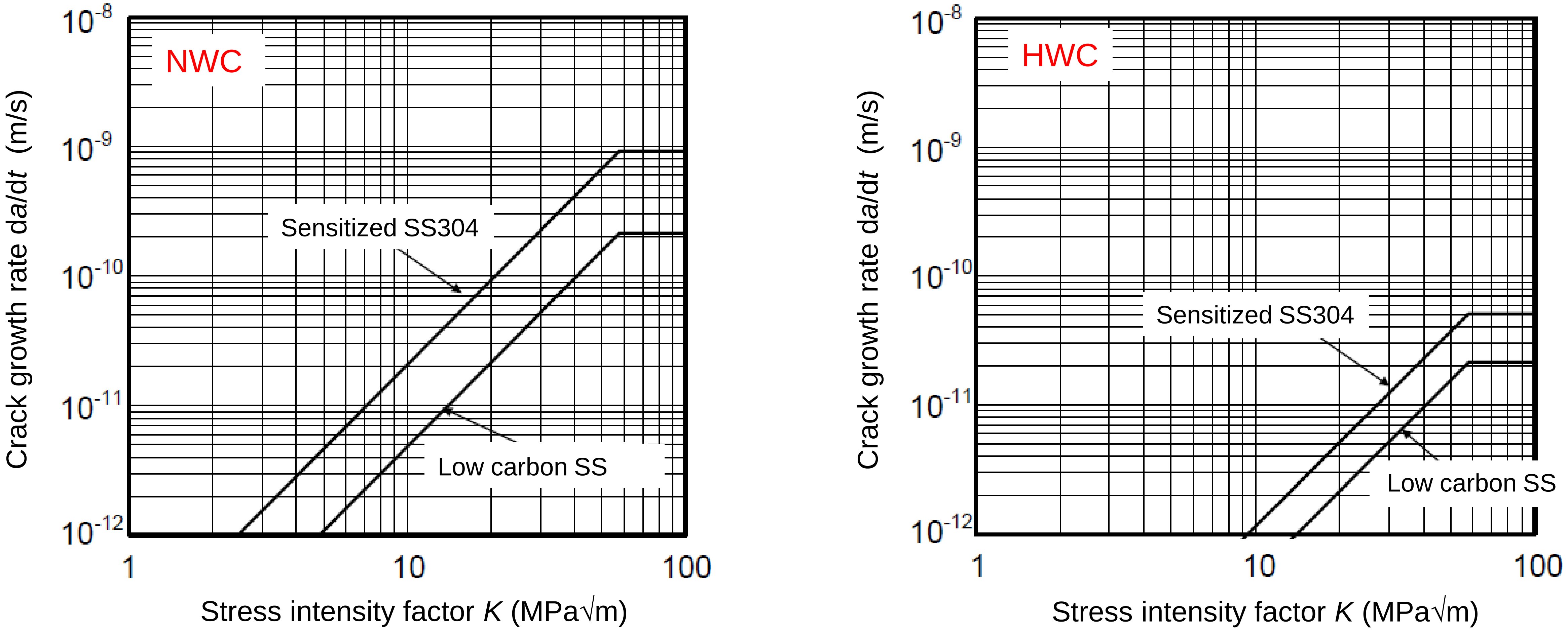


FCGR of nickel alloy and weld metal in LWR env.

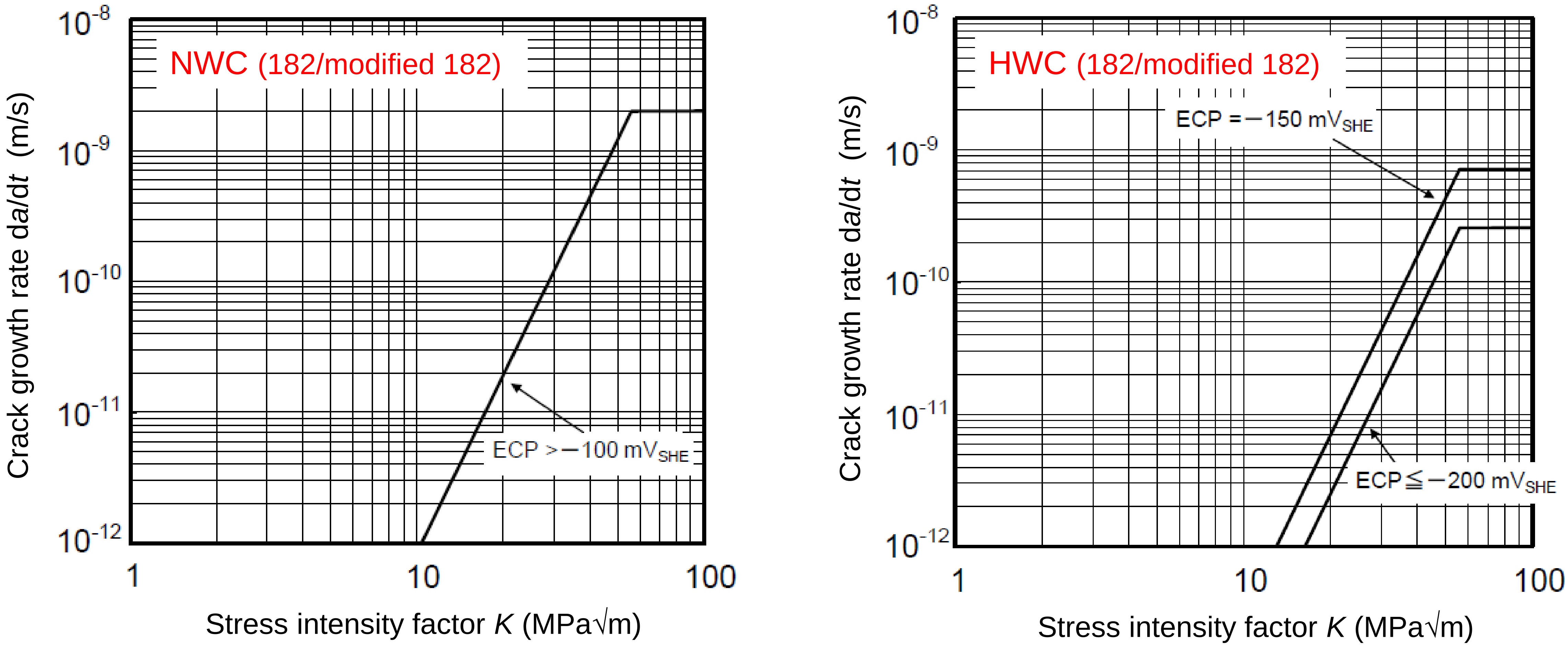




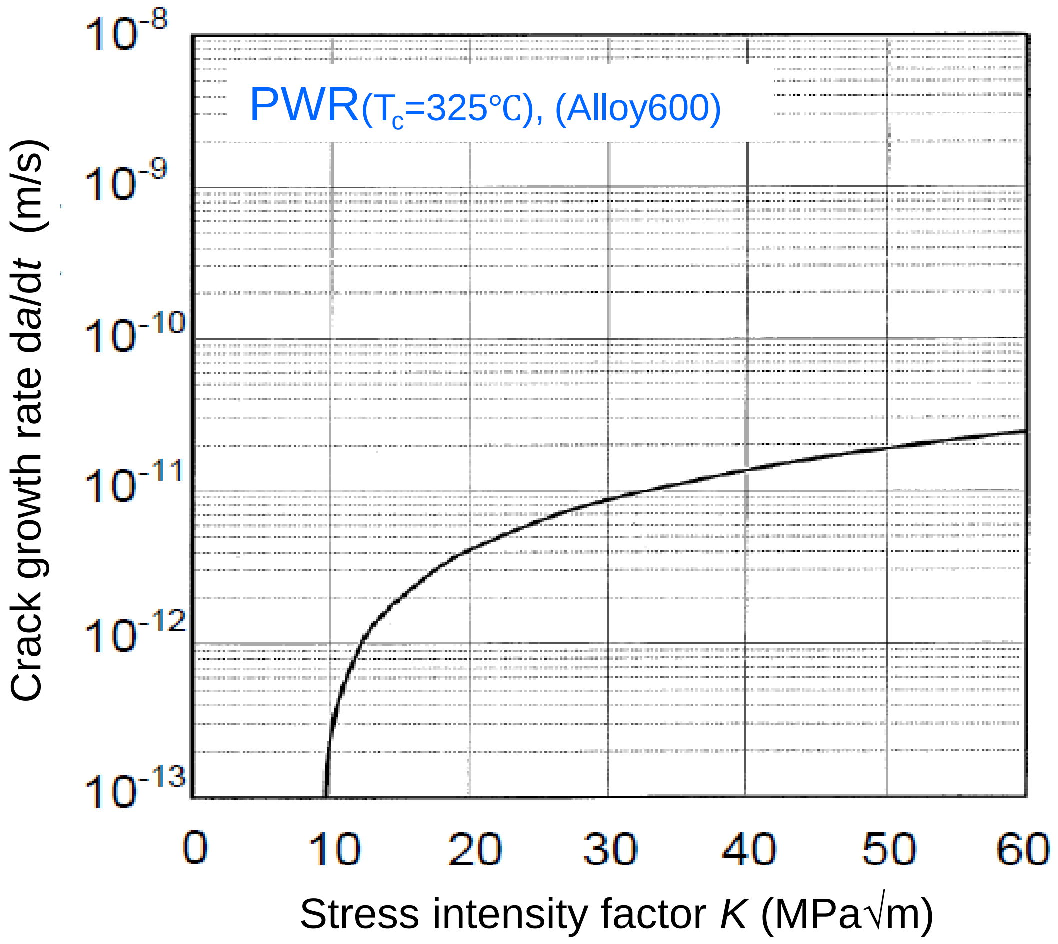
SCCCGR of austenitic steel(BWR, sensitized SS304, low carbon SS)

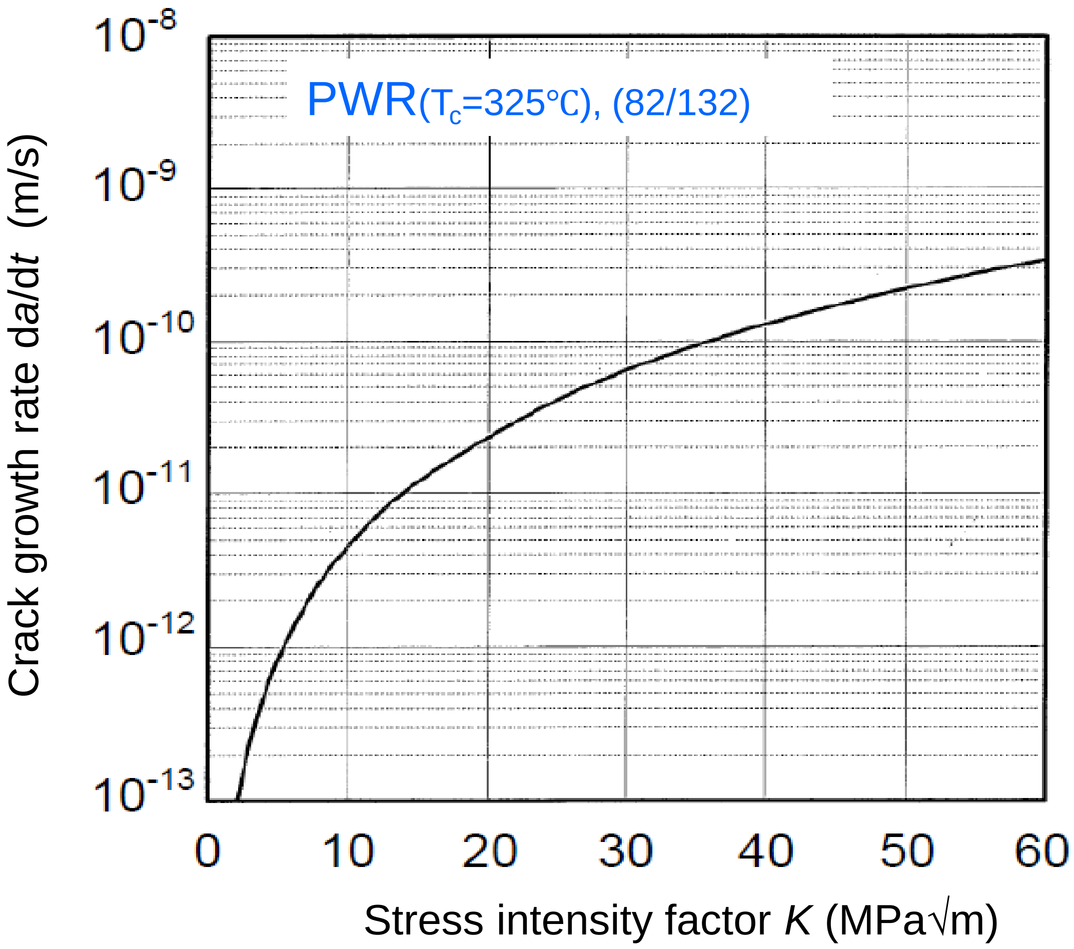


SCCCGR of nickel alloy weld metal in BWR env.



SCCCGR of nickel alloy (base metal and weld metal) in PWR





Appendix E-18 Material properties for facture evaluation of CASS pipes

Material

JIS SCS13A and SCS14A, comparable to A351 Gr. CF-8 and CF-8M) ≦ferrite content 23.5%

Incorporated aging effect on material properties considering aging condition (since 2018ad.)

Degradation model

Tensile properties ← TSS model

Fracture toughness ← H3T model

Kawaguchi et al. , PVP2005-71528

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Tensile properties – Determination of σ_y and σ_f

Ferrite content F

$$F = -4.834366 - 56.80472X + 134.641X^2 - 99.90123X^3 + 30.02316X^4$$

$$X = Cr_{eq} / Ni_{eq} \qquad Cr_{eq} = Cr + 1.5Si + 1.4Mo + Nb - 4.99$$

$$Ni_{eq} = Ni + 30C + 0.5Mn + 26(N - 0.02) + 2.77$$

Initial σ_y and σ_f (σ_{y0} and σ_{f0})

$$\sigma_{y0} = 6.653 + 5.385F + 10.007Mo + 1353.385C$$

$$\sigma_{f0} = 105.472 + 6.96F + 16.062Mo + 1535.398C$$

Thermal aging parameter P

$$P = \log\left(t_{aged}\right) + 0.4343\frac{Q}{R}\left(\frac{1}{673.2} - \frac{1}{T_{aged}}\right)$$

T_{aged} : aging temp. (K)

t_{aged} : aging time (h)

Q : activation energy (kJ/mol)

R : gas constant (kJ/molK)

σ_y and σ_f after aging

$$\frac{\sigma_y}{\sigma_{y0}} = \frac{1+1.071}{2} - \frac{1-1.071}{2}\tanh\left(\frac{P-1.617}{0.916}\right) \qquad \frac{\sigma_f}{\sigma_{f0}} = \frac{1+1.161}{2} - \frac{1-1.161}{2}\tanh\left(\frac{P-2.996}{0.929}\right)$$

$$\sigma_y = \sigma_{y0}\left(\sigma_y/\sigma_{y0}\right) \qquad \sigma_f = \sigma_{f0}\left(\sigma_f/\sigma_{f0}\right)$$

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Tensile properties – Determination Stress-strain curve after aging

σ_f after aging → Selection of two categories whose σ_f s covers σ_f of aged material →

Calculation of σ at each stain point using the same interpolation factor as σ_f for the two categories

Table Appendix E-18-1 Database of S-S curve

(ε (%), σ (MPa))

Category	σ_f		Point 1	Point 2	Point 3	Point 4	Point 5	Point 6	Point 7
1	282	ε	0.2	0.5	1	3	6	12	20
		σ	134	166	185	227	274	366	473
2	318	ε	0.2	0.5	1	2	4	8	15
		σ	149	192	222	254	297	371	484
3	377	ε	0.2	0.3	0.5	1	2	4	10
		σ	178	204	235	269	308	360	487
4	423	ε	0.2	0.3	0.4	0.5	1	2	5
		σ	182	211	235	255	314	365	462

Fracture toughness – Determination of J_{Ic}

Ferrite content F

$$F = -4.834366 - 56.80472X + 134.641X^2 - 99.90123X^3 + 30.02316X^4$$

$$X = Cr_{eq}/Ni_{eq}$$

$$Cr_{eq} = Cr + 1.5Si + 1.4Mo + Nb - 4.99$$

$$Ni_{eq} = Ni + 30C + 0.5Mn + 26(N - 0.02) + 2.77$$

Calculation of coefficients at 325℃

$$\log(A_J) = 3.2961 - 0.0530\,F - 0.5036$$

$$\log(B_{J325}) = 5.7869 + 0.9256\,Mn - 0.3028$$

$$\log(t_{F325}) = 4.3047 - 19.1095\,N - 0.5464$$

$$\log(D_{J325}) = 1.5354 + 0.2062\,Ni + 0.2834$$

$$B_J = B_{J325}\exp\left[\frac{Q}{R}\left(\frac{1}{T_{aged}} - \frac{1}{598.2}\right)\right]$$

$$t_F = t_{F325}\exp\left[\frac{Q_F}{R}\left(\frac{1}{T_{aged}} - \frac{1}{598.2}\right)\right]$$

$$D_J = D_{J325}\exp\left[\frac{Q}{R}\left(\frac{1}{T_{aged}} - \frac{1}{598.2}\right)\right]$$

Q =100 kJ/mol

Q_F =100 kJ/mol

R =0.0083145kJ/molK

J_{Ic} after aging

$$J_{Ic} = A_J + \frac{B_J}{t_{aged} + D_J - t_F}$$

(kJ/m^2)

Fracture toughness – Determination of J-R curve after aging

Calculation of coefficients at 325℃

$$\log\left(A_J{}'\right) = 3.6699 - 0.0490\,F - 0.2980$$

$$\log\left(B_{J325}{}'\right) = -1.7907 + 0.4130\,Cr - 0.3566$$

$$\log\left(t_{F325}{}'\right) = 7.6362 - 0.3670\,Ni - 16.108\,N - 0.1784$$

$$\log\left(D_{J325}{}'\right) = -2.9645 + 0.3438\,Cr - 0.1648\,Mo + 0.1404$$

$$B_J{}' = B_{J325}{}' \exp\left[\frac{Q}{R}\left(\frac{1}{T_{aged}} - \frac{1}{598.2}\right)\right]$$

$$t_F{}' = t_{F325}{}' \exp\left[\frac{Q_F}{R}\left(\frac{1}{T_{aged}} - \frac{1}{598.2}\right)\right]$$

$$D_J{}' = D_{J325}{}' \exp\left[\frac{Q}{R}\left(\frac{1}{T_{aged}} - \frac{1}{598.2}\right)\right]$$

J at Δ*a*=6mm after aging, J_6

$$J_6 = A_J{}' + \frac{B_J{}'}{t_{aged} + D_J{}' - t_F{}'} \qquad \text{(kJ/m}^2\text{)}$$

Δa_t corresponding to J_{Ic}

$$\Delta a_t = \frac{J_{Ic}}{2\sigma_f} + 0.2 \qquad \text{(kJ/m}^2\text{, mm)}$$

Calculation of C_1 and C_2

$$C_2 = \frac{\log\left(J_{Ic}/J_6\right)}{\log\left(\Delta a_t/6\right)} \qquad C_1 = \frac{J_{Ic}}{\left(\Delta a_t\right)^{C_2}}$$

J-R curve, J_R after aging

$$J_R = C_1\left(\Delta a\right)^{C_2}$$

4. HARMONIZATION WITH ASME SEC. XI AND OTHER CODES

- Based on Sec. XI for flaw evaluation of JSME Rules on FFS

  - similar evaluation flow and basic concept
- In 15 years, many improvements and new addition on JSME have been made and influenced to activities of Sec. XI

For example

(1) Revision of old *K*-solution

Replace Zahoor's *K*-solution for fan-shaped flaw in App. E-5 and screening criteria (SC) calculation in App. E-11 of JSME, to *K*-solutions for circumferential semi-elliptical flaw from CEA-R-5900 (2000) (2010 add.～2022 ed.)

⬇

Proposed Code change of App. C-4000 (SC calculation) of Sec. XI to abolish Zahoor's *K*-solution for fan-shaped flaw, and introduce closed-form *K*-solutions for circumferential semi-elliptical flaw from API 579-1 which is nearly the same as CEA-R-5900 (2000)

⬇

Published 2019ed. of Sec. XI

4. HARMONIZATION WITH ASME SEC. XI AND OTHER CODES

(2) Flaw evaluation of J-groove weld joint of BMI nozzle

JSME published Code Case ‘Flaw evaluation of J-groove weld joint of BMI nozzle’ aiming at application evaluation of CAP repair for BMI penetration nozzle of PWR

- Very new Code Case because of FEA guideline to calculate *J* or *K* for complex structure

Published in 2015 from JSME

Proposed similar Code Case to ASME Sec. XI

Application: 3/4-inch to 6.0-inch NPS and include the following J-groove attachment welds:
(1) Reactor vessel upper and bottom head penetrations
(2) Steam generator and pressurizer head and shell penetrations
(3) Primary main loop piping instrumentation nozzle penetrations

Repair options
(1) Cut and Cap Repair
(2) Half-Nozzle Repair
(3) Embedded Flaw Repair

Published new CC N-897 in 2021 from ASME

4. HARMONIZATION WITH ASME SEC. XI AND OTHER CODES

(3) CGR in PWR/BWR env.

JSME Rules have substantial CGRs in the Code, especially in LWR env.

FCGR of SS in PWR env. in App. E-2 of JSME ➡ CC N-809

FCGR of SS in BWR env. in App. E-2 of JSME ➡ Draft CC N-895 under discussion

SCGR of SS in PWR env. :under preparing technical basis by EPRI and ATENA

Harmonization between JSME and ASME is now accelerating!

5. CONCLUSIONS

- Introducing history and structure of JSME Rules on FFS
- JSME Rules have substantial CGRs in LWR env. and introduced thermal aging degradation model for CASS which is one of important aging management issues
- Harmonization between JSME and ASME is accelerating in a decade
- Need to collaborate JSME with ASME and other SDO for aging management

THANK YOU FOR YOUR ATTENTION!

Code Requirements for managing aging of pressure boundary components of CANDU REACTORS in Canada

Mr Suqiang XU

Technical Specialist

Canadian Nuclear Safety Commission

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Canadian Nuclear Safety Commission | Commission canadienne de sûreté nucléaire

Canada

CODE REQUIREMENTS FOR MANAGING AGING OF PRESSURE BOUNDARY COMPONENTS OF CANDU REACTORS IN CANADA

Suqiang Xu, Technical Specialist

Canadian Nuclear Safety Commission

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

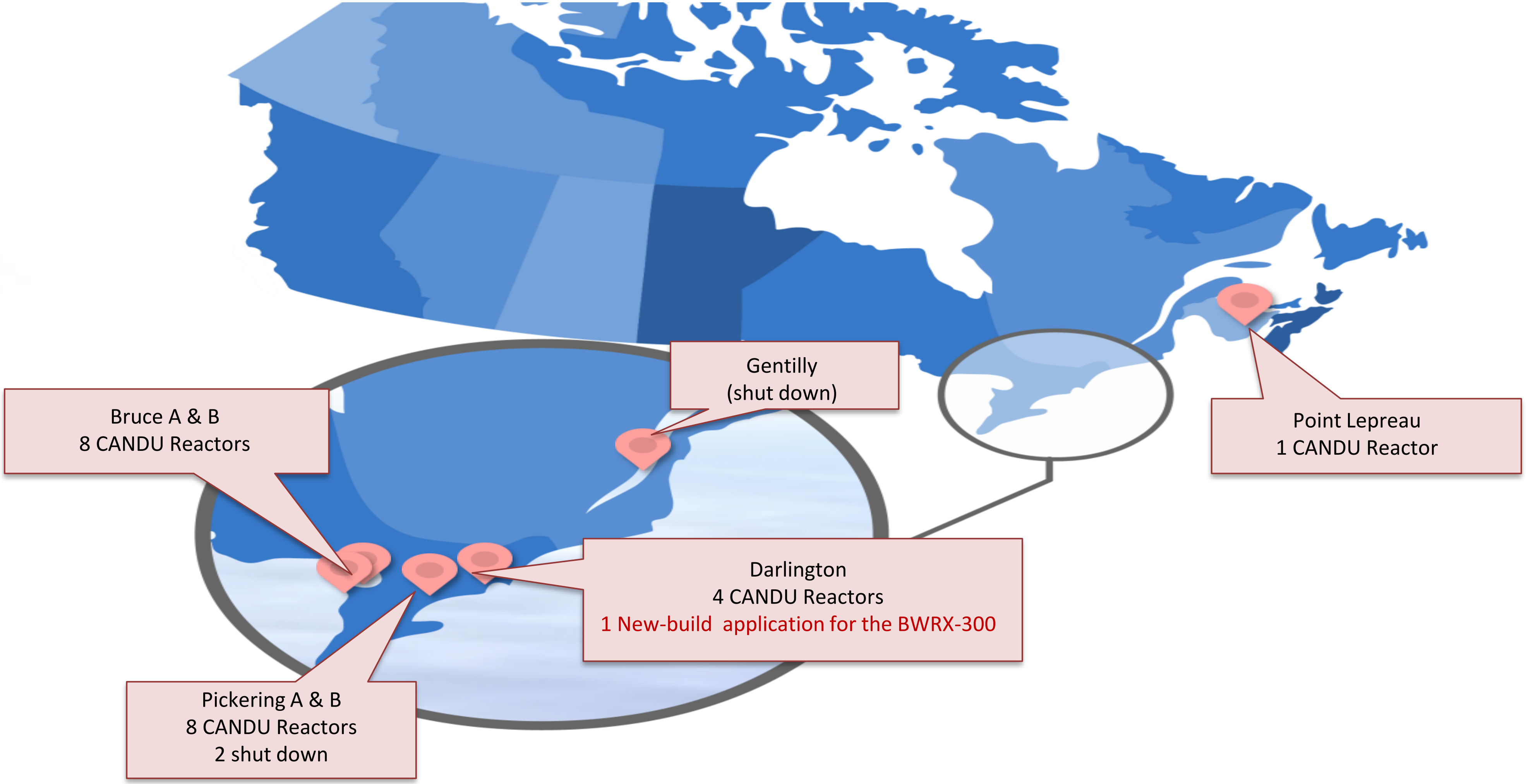
CNSC CCSN

Canadian Nuclear Safety Commission

Regulates the use of nuclear energy and materials to protect the health, safety and security of Canadians and the environment; implements Canada's international commitments on the peaceful use of nuclear energy; and disseminates objective scientific, technical and regulatory information to the public.

Regulating Since 1946!

Canada's Nuclear Power Plants



Presentation Outline

- Regulatory Requirements, Codes and Standards
- Aging Considerations During Design
- Aging Management Process
- Periodic inspection of CANDU nuclear power plant components
- Example
- Summary

Regulatory Requirements, Codes and Standards

CNSC Regulatory Documents

- REGDOC 2.5.2, Design of Reactor Facilities: Nuclear Power Plants
- REGDOC 2.6.3, Aging Management

CSA(Canadian Standards Association)

- N285.0 General requirements for pressure-retaining systems and components in CANDU nuclear power plants
- N285.4 Periodic Inspection of CANDU NPP components
- N285.7 Periodic Inspection of CANDU NPP balance of plant system components
- N285.8 Technical requirements for in-service inspection evaluation of zirconium alloy pressure tubes in CANDU reactors
- N290.20 Aging Management Requirements for Nuclear Power Plants

Aging Considerations During Design as per REGDOC 2.5.2

Section 5.17, Aging and wear

The design shall take due account of the effects of aging and wear on SSCs. For SSCs important to safety, this shall include:

1. an assessment of design margins, taking into account all known aging and wear mechanisms and potential degradation in operational states, including the effects of testing and maintenance processes
2. provisions for monitoring, testing, sampling, and inspecting SSCs so as to assess aging mechanisms, verify predictions, and identify unanticipated behaviours or degradation that may occur during operation, as a result of aging and wear

Aging Considerations During Design as per CSA N285.0

Clause 7.1.7 Periodic Inspection

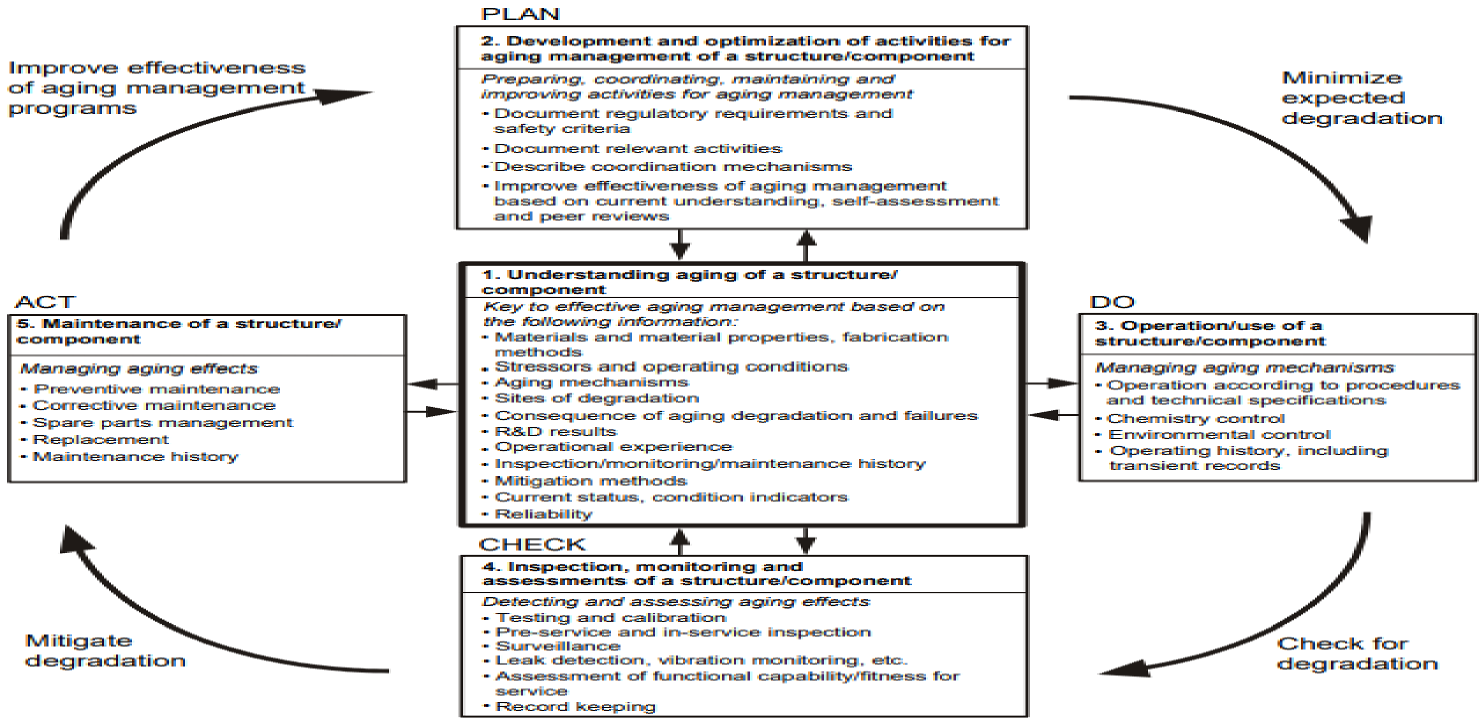
Items shall be inspected periodically in accordance with CSA N285.4, CSA N285.5 and CSA N285.7 and shall be designed to allow access as required to such inspections.

Clause 13.2 Inaugural or Baseline Inspections

When the pressure boundary of a component that is subject to periodic inspection is repaired, replaced or otherwise changed, it shall receive, before put into service, an inaugural or baseline inspection as required by CSA N285.4, CSA N285.5 and CSA N285.7

REGDOC-2.6.3, Aging Management

- Outlines the CNSC requirements on aging management through all life stages, from Design to decommissioning
- Requires a systematic and Integrated approach



REGDOC 2.6.3, Elements of AM Program

1. Organizational arrangements
2. Data collection and record keeping
3. Screening and selection process
4. Evaluations for aging management
5. Condition Assessments
6. SSC Specific AMPs
7. Management of Obsolescence
8. Interfaces with Other Supporting Plant Programs
9. Implementation of AMPs
10. Review and improvement of AMPs

CNSC REGDOC 2.6.3, Screening and Selection Process

- From a regulatory perspective, the screening and selection process for SSCs should follow a safety-based approach
- Nuclear power plants in Canada have adopted INFO AP 913, the Equipment Reliability Process Description. The AP 913 component classification is based on both safety and power production criteria. A typical four tier classification is approximately summarized as follows:
  1. High critical components (Criticality Code 1 – CC1)
  2. Low Critical Components (Criticality Code 2 – CC2)
  3. Noncritical Components (Criticality Code 3 – CC3)
  4. Run to Maintenance Components (Criticality Code 4 – CC4)

REGDOC 2.6.3, SSC-specific Aging Management Plans

1. Scope of the AM program based on understanding of aging
2. Preventive actions
3. Detection of aging effects
4. Monitoring and trending
5. Mitigating aging effects
6. Acceptance criteria
7. Corrective actions
8. Operating experience feedback & R&D results
9. Quality management

CSA N285.4, Periodic Inspection of CANDU Nuclear Power Plant Components

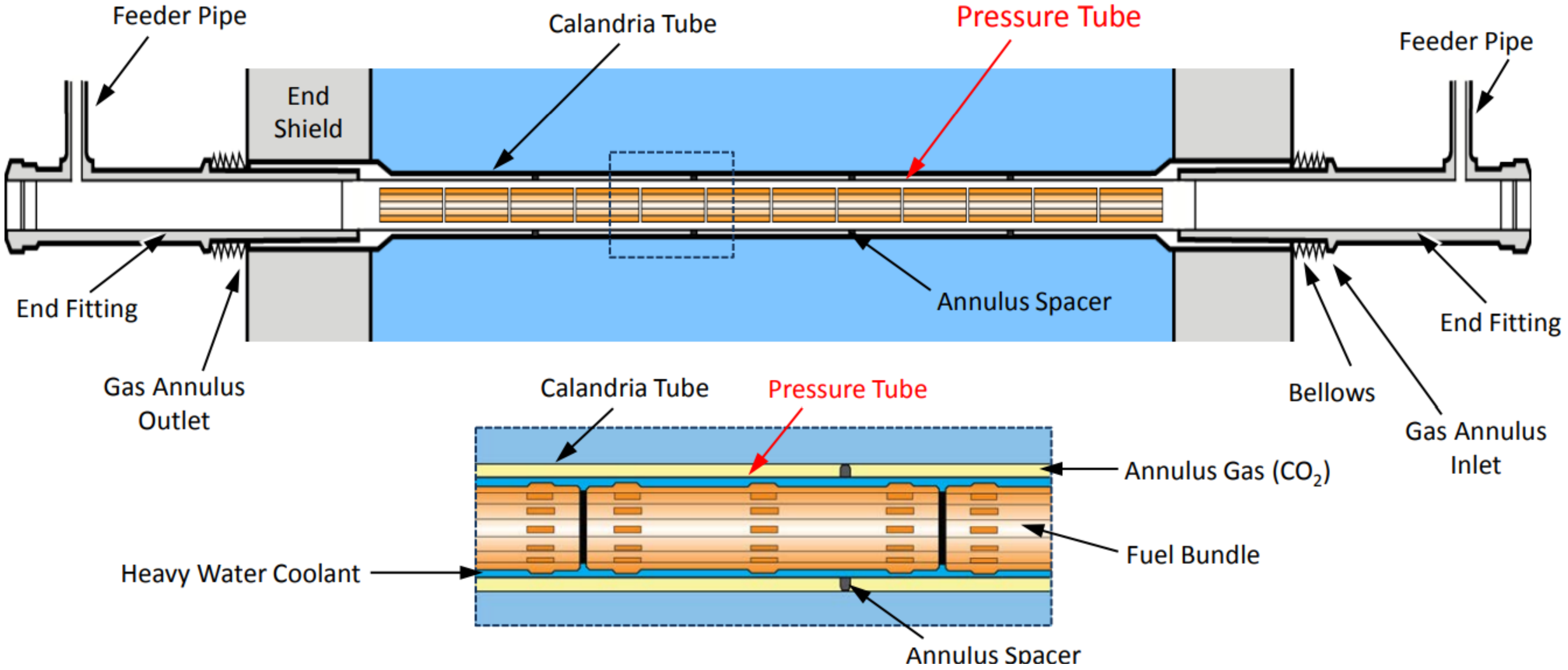
- Systems subject to inspection shall include the following systems or portions thereof:
  - systems, and systems connected thereto, containing the fluid that, under normal conditions, directly transports heat from nuclear fuel, and other systems whose failure can result in a significant release of radioactive substances
  - systems essential for the safe shutdown of the reactor and/or the safe cooling of the nuclear fuel in the event of a process system failure
  - systems and equipment whose failure or dislodgement could jeopardize the integrity of systems in Item a) or b), or both. This includes large pieces of rotating machinery (e.g., flywheels)
- Three major pressure boundary components
  - Fuel Channels, Steam Generators and Feeders

Degradation of Three Major Components

Component	Degradations
Fuel Channel	Creep, Delayed Hydride Cracking, embrittlement, wear, Pressure Tube and Calandria Tube contact, Elongation, Reduction in wall thickness, Increase in diameter, Sag, flaws, Corrosion, Degradation of annulus spacers
Steam Generator	Fretting, Denting, pitting, stress corrosion cracking, intergranular attack, flow accelerated corrosion, fouling, fatigue
Feeders	flow accelerated corrosion, intergranular stress corrosion cracking,

Aging Management Example-Flaws in Pressure Tubes

CANDU Fuel Channel (FC)



The example is from CNSC CMD 18-M4

Aging Management Example-Flaws in Pressure Tubes

Progression of flaw degradation:

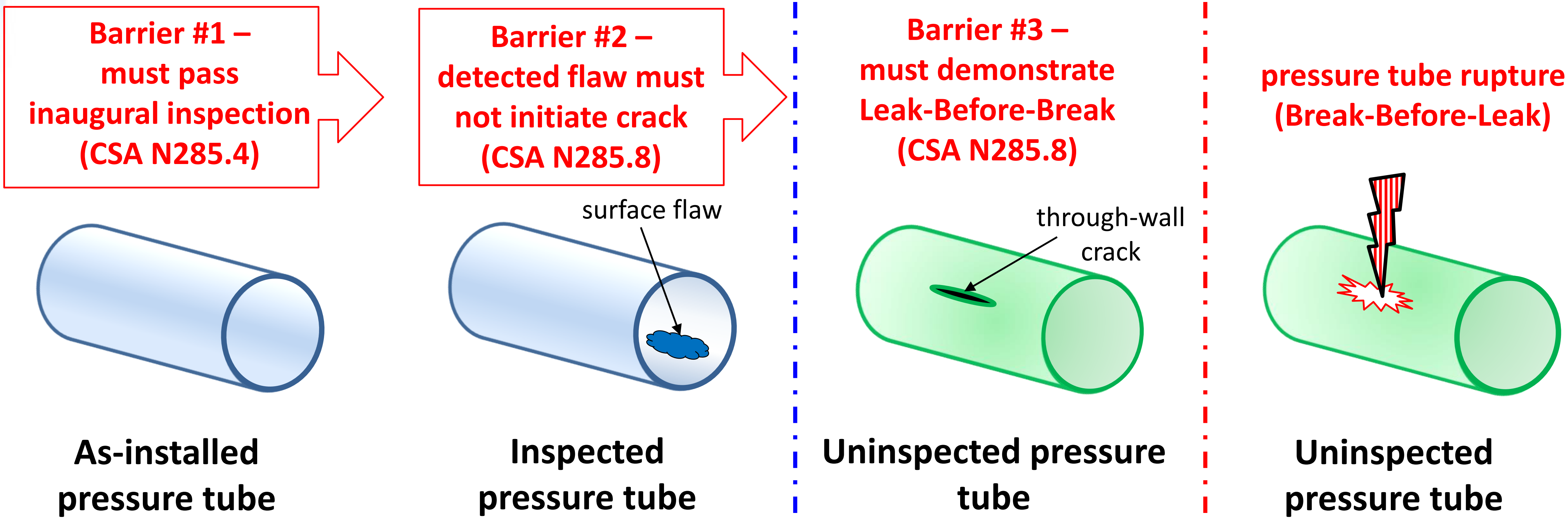
- Flaw initiated in pressure tube
- Flaw develops into crack (e.g. Delayed Hydride Cracking)
- Crack propagates through the PT wall -> primary coolant leakage
- Crack extends axially along PT (predictable rate, by design)

– Leak-Before-Break: reactor cooled and shut-down before PT crack reaches "Critical Length" (point of instability)

– Break-Before-Leak: crack reaches Critical Length before reactor can be shut-down

The example is from CNSC CMD 18-M4

Pressure Tube Case



Pressure Tube Aging Management

Requirement	Regulations	Licensee actions to address requirements
Understand	REGDOC-2.6.3	Industry research and development; fuel channel Condition Assessments
Plan	CSA N285.4	Periodic Inspection Program (PIP); fuel channel Life-Cycle Management Plan
Perform	CSA N285.4, CSA N285.8	Periodic inspections; PT material surveillance; research and development
Demonstrate acceptance criteria met	CSA N285.4, CSA N285.8, REGDOC-2.6.3	Fitness-for-service assessments; follow-up inspections; research and development

The example is from CNSC CMD 18-M4

Summary

- All structures, systems and components (SSCs) degrade over time, which must be managed for short and long term operation
- CNSC requires licensees have aging management programs
- Not all of SSCs can be treated the same- A graded approach must be used
- Aging management of pressure boundary components are governed by REGDOC 2.6.3, CSA N285.4, CSA N285.8, etc.

Initiatives of Japan on standardizing new findings on ageing management

Pr Kenta NURAKAMI

University of Tokyo

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OECD/NEA Workshop
Ageing management considerations in mechanical codes and standards
28 June 2023 @ Hitotsubashi Hall, Tokyo

Initiatives of Japan on standardizing new findings on ageing management

Kenta Murakami

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THE UNIVERSITY OF TOKYO

Disclosure of conflicts of interest

- The author is involved in the development of codes and standards at following organization:
  - ✓Standard Committee at Atomic Energy Society of Japan (AESJ)
    - ✓Taskforce for basic strategy
    - ✓Taskforce for long-term operation standards
    - ✓Subcommittee on Integrated safety improvement
  - ✓Nuclear Standard Committee at Japan Electrical Association (JEA)
    - ✓Subcommittee on safety design

- This presentation does not represent the official opinion of the authors affiliation or any of above organizations

Role of <u>AESJ</u> on the C&S for Ageing Management (AM)

- An only **<u>comprehensive academic society</u>** in Japan on the peaceful uses of nuclear energy and radiation in safe

- Standard Committee of the AESJ
  - ✓Managing approximately 89 codes and guides (including new drafts)
  - ✓Establish and revise C&S based on the latest technical knowledge
  - ✓Monitoring gaps from the best practices and improve standards **<u>ahead of industry and/or regulatory requests</u>**

AM framework in Japan

Regular Activities

✓ Quality Management System (QMS) for Safety
  - **JEA QMS codes/ guide** envelope the requirements of Quality Management Rule

✓ Maintaining the facility and activities to fulfill the Technical Standard and Operational Rule
  - **JEA maintenance code/guide** covers maintenance program
  - **JSME rules on fitness-for-service** covers in-service inspection, evaluation and repairing
  - **Inspection program**, Corrective Action Program, Monitoring of Performance Indicators
  - Periodic inspections along with refueling outage are regulatory required for safety-important SSCs

✓ **Safety improvement review report** (SIRR) is regulatory required by 6 mount after periodic inspection

Regulatory-required Long-Term Evaluations

✓ Final Safety Assessment Report (FSAR): every 5 years with the SIRR

✓ Periodic Safety Review: every 10 years with the SIRR → **AESJ PSR^{+} guide**

✓ **AM technical evaluation**: every 10 years after 30-years operation → **AESJ Plant life Management (PLM) codes**

✓ Special Inspection and Renewing operational license at 40-years operation

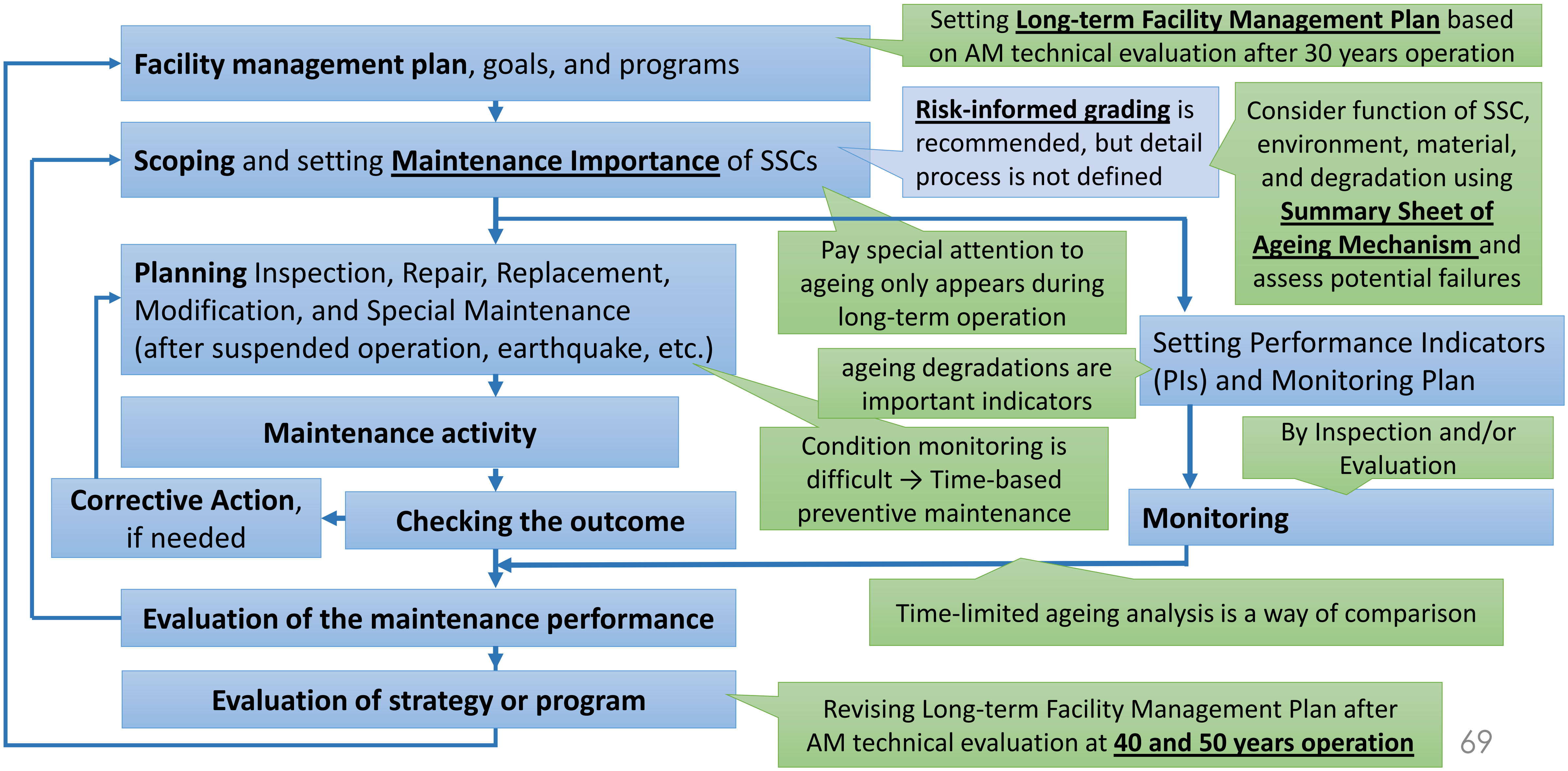
Some Features on Japanese AM practices

- **Time-based inspections for safety-important SCCs is required** by Reactor Establishment and Operation Rule
  - ➢Preventive maintenance for equipment is more likely to be chosen.
  - ➢Reassessments of qualified life is less important.

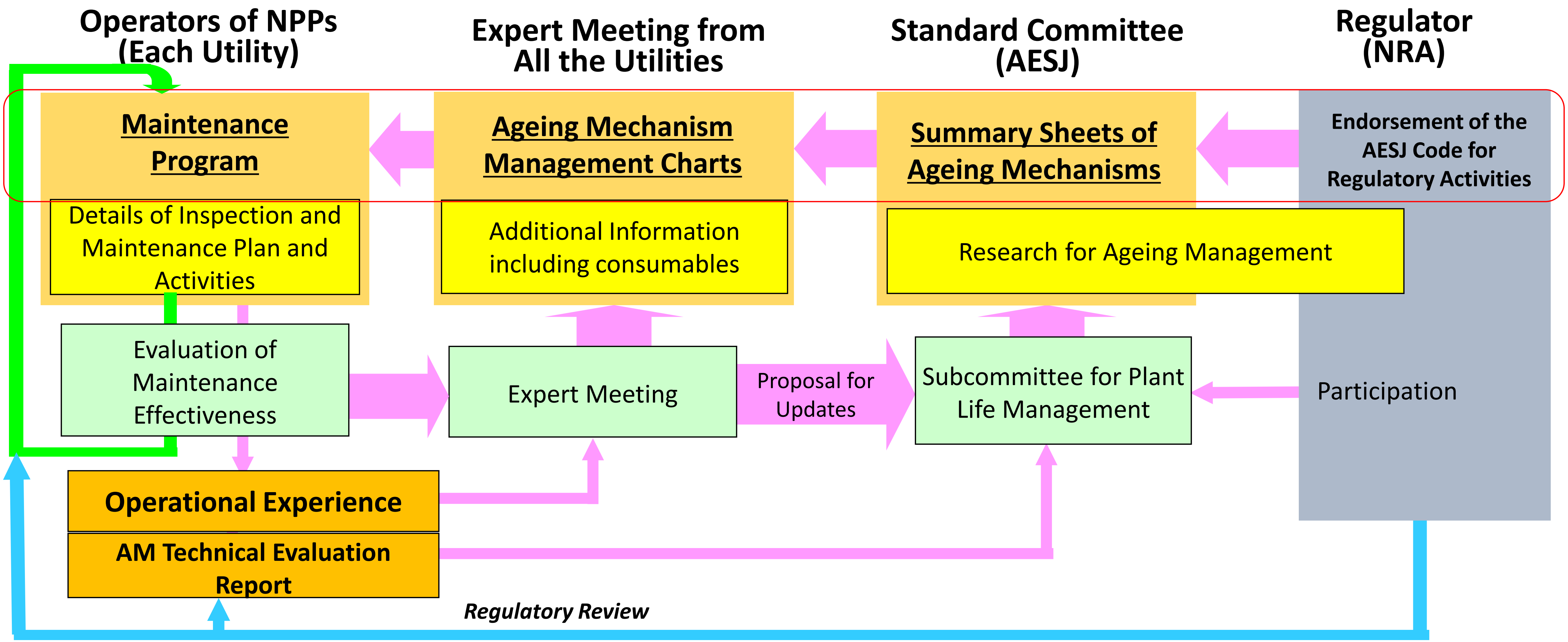
- Maintenance and AM reviews are conducted on each **equipment-by-equipment basis**
- The timing that NPPs started operating is more scattered than in the U.S. and France
  - ➢Regulatory-reviewed ageing management programs (AMPs) are not defined for each detailed aging process,
  - ➢nor are they applied for AM technical PLM review

➢Japanese AM system focuses on **forecasting potential degradation** in safety-important SSCs from broader perspective and **preventing their appearance** in advance

Relation of JEA maintenance code and AESJ PLM code



Continuous Improvement of AM Knowledge Basis



After Sekimura, PLiM-5 (2022)

Example of Ageing Mechanisms in High Pressure Injection Pumps in Takahama unit-1

Required Function	Location	Easy to Replace?	Material	Ageing Degradation							Remarks
				reduce wall thickness		cracking		material degradation		others	
				wear	corrosion	fatigue	SCC	thermal ageing	deterioration		
Pump Capacity and Head	main shaft		stainless steel	▲		●△					Fretting fatigue cracking and possible high cycle fatigue cracking in main shaft
	vane wheel		cast S.S.		△						
	casing ring		stainless steel	▲							
	bearing box		cast iron		▲						
	bearing (sliding)		carbon steel white metal	▲							
	gasket	◎	—								
	shaft joint		low alloy steel								
	speed increasing gear		low alloy steel	●							
	Bearing of speed increasing gear (sliding)		white metal	▲							
	casing of speed increasing gear		cast iron		▲						
Boundary	casing		low alloy steel								Possible cavitation in vane wheel
	casing cover		low alloy steel								
	casing bolt		low alloy steel		△						
	mechanical seal	◎	—								
	gasket	◎	—								
	O-ring	◎	—								
	casing drain tube		stainless steel								
	balancing tube		stainless steel								
	mechanical seal cooler		stainless steel								
Support of components	base plate		carbon steel		▲						
	mounting bolt		carbon steel		▲						
	foundation bolt		carbon steel		●						

[Speed increasing gear]
Although wearing of gear is prevented with lubrication oil at inside of the unit, wearing is possible when it is operated for a long time.
Accordingly, evaluation is required on aging change.

[Corrosion of speed increasing gear casing]
Inside of the unit is exposed to atmosphere of lubrication oil. It is possible to say that no technical problem will occur by the operating condition and comparison with material testing data.

●Important mechanism for LTO, △▲ Possible mechanism for LTO

After Sekimura, PLiM-5 (2022)

Examples of the Summary Sheet of Ageing Mechanism

P01-02 Turbo pump, Horizontal, Vortex

Environment: primary coolant/ Materials: low-alloy steels & S. S.

	Required Function	***Location***	***Material***	***Ageing Degradation***
1	Pump Capacity and Head	main shaft	stainless steel	wear
2				fretting
3				high-cycle fatigue
4		vane wheel	cast S.S.	cavitation
5		vane ring	—	wear
6		casing ring	—	expendable
7		bearing box	cast iron, cast carbon steel	Corrosion (all)
8		bearing (sliding)	—	expendable
9		bearing (rolling)	—	expendable
10		gasket	—	expendable
11		shaft joint	low alloy steel, carbon steel	wear
12		lubrication oil unit	carbon steel, cast iron	corrosion (all)
13		speed increasing gear	low alloy steel	wear
14		Bearing of speed increasing gear (sliding)	white metal	Expendable
15		casing of speed increasing gear	cast iron	corrosion (all)
16	Boundary	casing	low alloy steel with S.S. liner	Not expected
17			Cast S.S.	fatigue
18				SCC
19		casing cover	low alloy steel with S.S, liner	Not expected

- ✓ **[Qualitative risk-informed approach](#) to make cross-links between degradation modes and safety-related components**
- ✓ **Major input to IAEA knowledge-base; IGALL (SRS-82)**

Ageing analysis for long-term defined by AESJ-PLM code

Timing	Every 10 years	Every 10 years after the 30th year
Target SSCs	Primary Coolant Boundary Core Internals	All safety-important SSCs
Target Ageing Phenomena	Low Cycle Fatigue	Low Cycle Fatigue
	Irradiation Embrittlement	Irradiation Embrittlement
	IASCC (irradiation creep/swelling)	IASCC (irradiation creep/swelling)
	High Cycle Thermal Fatigue	High Cycle Thermal Fatigue
		Fretting
		Insulation degradation (electric SSCs)
		Concrete Degradation
External Event	Degradation above + Seismic Motion	Degradation above + Seismic Motion
		Tsunami + ageing of SSCs for tsunami-measures
Target Period for Analysis	From start operation to present	By 60 years of operation

Management of Knowledge Obsolescence on Ageing Case: Reflecting Microscopic-mechanism to Embrittlement Trend Curve for RPV

~1990s 2000s 2010s 2020s

Irradiation test using JMTR (1983-91)

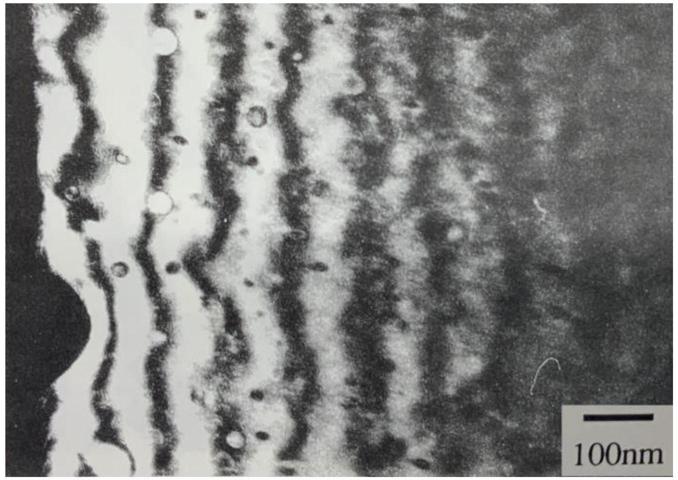
Ishino NED (1990) doi.org/10.1016/0029-5493(90)90157-S

- ✓ Effect of chemical composition of steels
- ✓ High fluence irradiation data

Development of nano-scale analysis technique

- ✓ TEM (round robin), Positron, AP-FIM, etc.
- ✓ Understanding the **embrittlement mechanism**

Cu-rich clusters in a RPV model alloy measured by TEM



Sekimura, AESJ round robin report (1996)

ETC for Japanese NPPs was first referred the JEA surveillance program for RPV code (JEAC-4201-1991) and then included in the code (JEAC-4201-2000 appendix)

Management of Knowledge Obsolescence on Ageing Case: Reflecting Microscopic-mechanism to Embrittlement Trend Curve for RPV

~1990s | **2000s** | 2010s | 2020s

Relatively large embrittlement in old BWRs

Evaluation of PWR reference materials and their additional irradiation in Halden

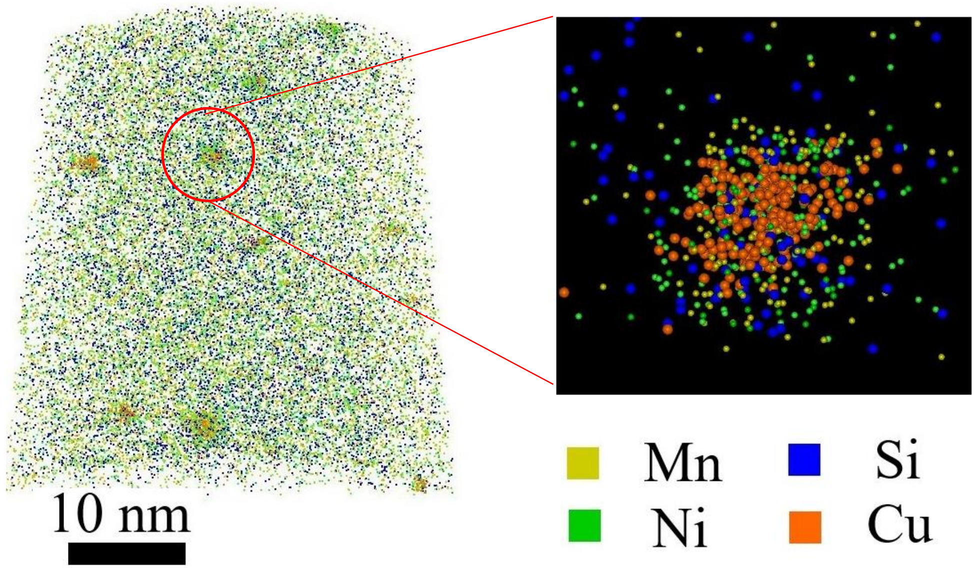
✓ Flux effect

Nano-scale embrittlement mechanism

✓ 3D-atom probe tomography (APT)

✓ Atomic-scale simulation technique

Cu-rich clusters in a BWR surveillance specimen measured by APT



Murakami, JNM (2020)

<u>Mechanism-guided ETC for JEAC-4201-2007 Appendix</u>

Detail information is available from Soneda (2014), ISBN-978-1845699673

Management of Knowledge Obsolescence on Ageing Case: Reflecting Microscopic-mechanism to Embrittlement Trend Curve for RPV

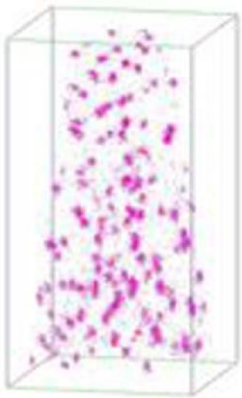
~1990s | 2000s | **2010s** | 2020s

Increasing the number of PWR surveillance data
- ✓ Better understanding of high fluence effect

Nano-scale embrittlement mechanism
- ✓ APT improvement, STEM-EDS
- ✓ CALPHAD

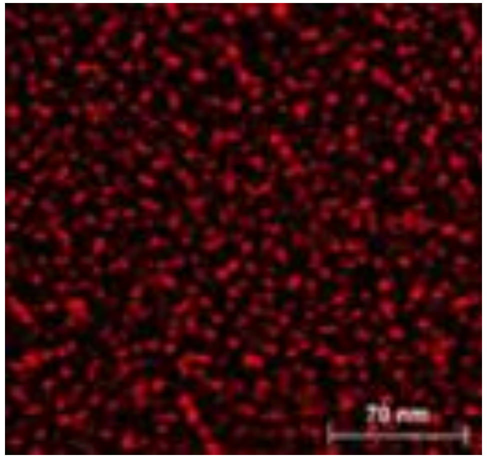
Comparison of APT and STEM-EDS in RPV model alloy irradiated by Research Reactor

APT



Almirall, Scripta (2020)

STEM-EDS



<u>Mechanism-guided ETC is revised in 2013, and will be revised again soon</u>

Preliminary Information is available from Hashimoto JNM (2021) doi.org/10.1016/j.jnucmat.2021.153007

Management of Knowledge Obsolescence on Ageing Case: Reflecting Microscopic-mechanism to Embrittlement Trend Curve for RPV

~1990s 2000s 2010s **2020s**

We need material testing reactor

- ✓ Material test using JRR-3 (beam reactor) is scheduled until 2025 for vilification of Master-curve method using Mini-CT
- ✓ But, irradiation opportunities are still not enough

Data can be harvested from decommissioned NPPs

- ✓ But, in many cases, the magnitude of ageing and the distribution of materials are not sufficient for the future needs
- ✓ **Additional processing for the harvested materials** is important

Plant-level Obsolescence Management by AESJ PSR$^+$ guide

- Defined a comprehensive review for proactively improving safety, with reference to IAEA-SSG-25
  - ✓ Gap analysis of fourteen safety factors
  - ✓ Global assessment and planning of long-term safety measures

- The guide will be renewed to be “AESJ-PSR$^+$ code” by the end of 2023
  - ✓ Further clarifying the requirements and the acceptable criteria
  - ✓ Defining the requirements for multi-units NPS
  - ✓ Reflecting the experiences of submitting FSAR in restarted PWRs

- List-up numerous findings/gaps, and then consider reasonable measures
  - ✓ **Courage** is required, for breaking the bad conventions in Japan

Other C&Ss for risk-informed decision-making by AESJ

Risk Management

- Integrated risk-informed decision-making Code (AESJ-SC-S012: 2019)
    - English translation is available: https://www.aesj.net/publish-2205

Risk Assessment

- Level-1 PRA Codes/Guides (AESJ-SC-RK010-2022 and AESJ-SC-RK011-2022)
- Level 2 PRA for Outage period (AESJ-SC-P009: 2021)
- Level 3 PRA (AESJ-SC-P010: 2018)
- Tsunami PRA (AESJ-SC-RK004: 2016)
- Seismic PRA (AESJ-C-P006: 2015)
    - English translation is available: https://www.aesj.net/publish-1727
- Ground Fault Displacement PRA (AESJ-SC-RK009:2021)
- PRA for Nuclear Fuel Facilities (AESJ-SC-P011: 2018)

In Lieu of Conclusion

Five goals for improving Japanese C&S for AM by AESJ-LTO Taskforce Phase 1

1. In order to manage issues related to LTO in an **integrated manner**, there is a need for a standardized approach that can be referred to by a variety of stakeholders
2. To **qualify the equipment's functions throughout the plant life**, it is effective to bind standards for important SSCs and visualize their relationship
3. Decisions for LTO should **be risk-informed** and be made in an integrated manner
4. It is effective to identify events that may cause long-term plant outage, and to prepare in advance **basic measures for early restart**.
5. Active efforts is needed to make complementary use of the codes and standards developed by academic societies and the **guides prepared independently by the industry or regulatory body**.

Coffee break

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Experiences on Structural Integrity Evaluation with Various Technical Standards and Analytical Method Improvement

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Experiences on Structural Integrity Evaluation with Various Technical Standards and Analytical Method Improvement

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June 28th, 2023

Nuclear Business Group

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Table of Contents

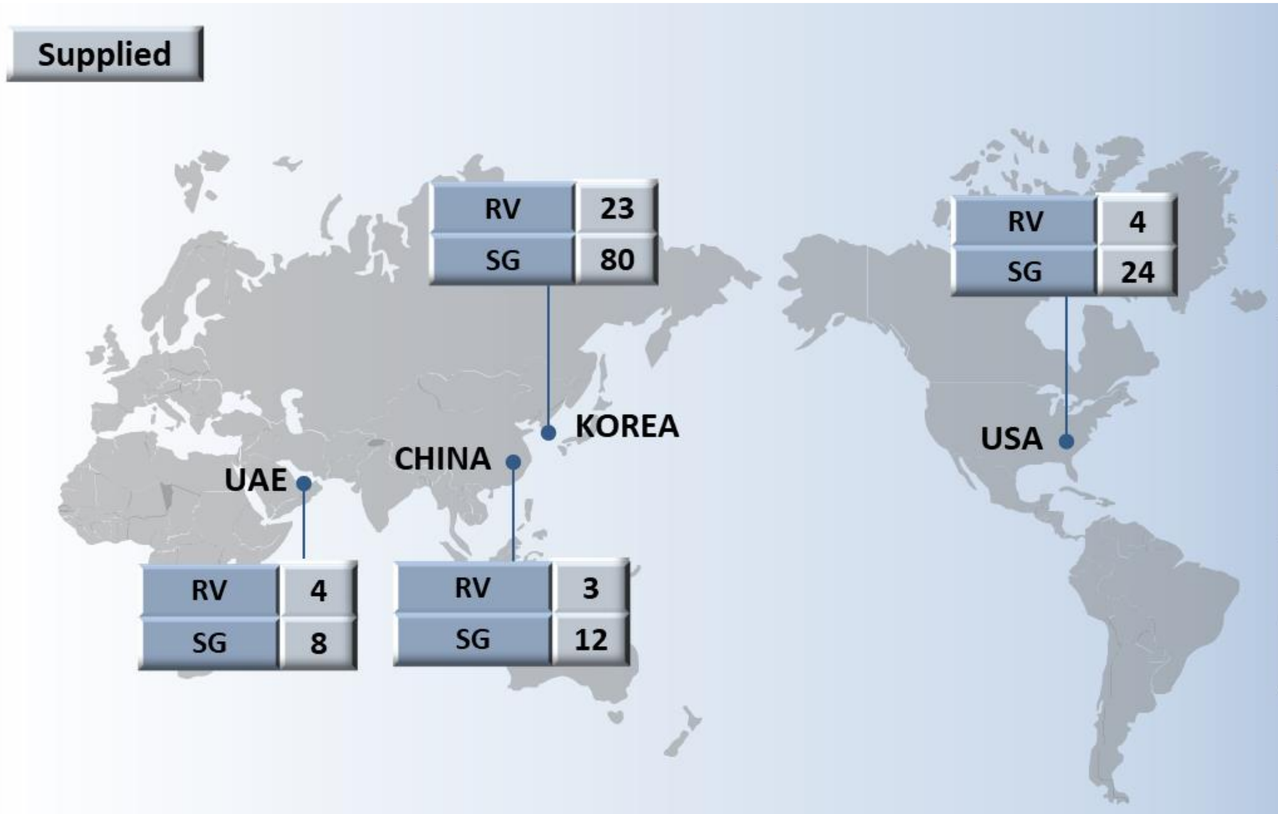
Introduction to Doosan Enerbility

DOOSAN Enerbility

- New Energy Solutions
- Power Plant Equipment / Services
- Plant EPC / Construction
- Material Manufacturing

❖ Nuclear Experiences

- Component Design / Manufacture and Supply
  - 34 RVs & 124 SGs have been manufactured and supplied by Doosan globally.



U.S Project	Operator	Equipment (Unit)
Sequoyah #1 RSG	TVA	RSG (4)
Watts Bar #1 RSG	TVA	RSG (4)
Sequoyah #2 RSG	TVA	RSG (4)
Watts Bar #2 RSG	TVA	RSG (4)
Vogtle #3&4	Southern	RV(2), SG(4)
VC Summer #2&3	SCANA	RV(2), SG(4)

* Including Steam Generators for Replacement

- ❖ Nuclear Experiences
  - Component Design / Manufacture and Supply
    - Doosan supplied major equipment of Barakah #1~4 including RV and SG.

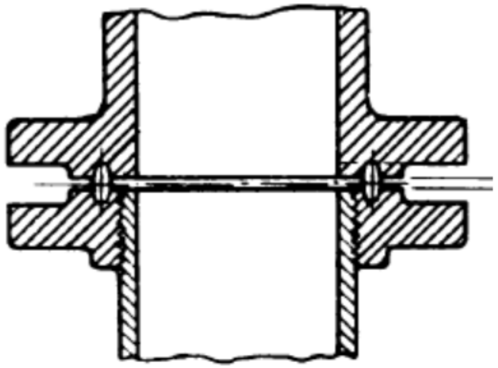
Barakah Nuclear Power Plant

- Experience in Applying PED with ASME
  - Compliance ASME and PED
    - In order to carry out a project in Europe where the PED was applied, the technical standard shall be compliance with PED. The project was performed using ASME.
    - For each requirement in PED, the corresponding ASME requirement was identified.
    - If there are no corresponding requirements in ASME, additional requirements were established to satisfy the PED requirements.
    - In case where a requirement exists in either PED or ASME, a conservative approach was taken by applying all the requirements.

* PED: Pressure Equipment Directive

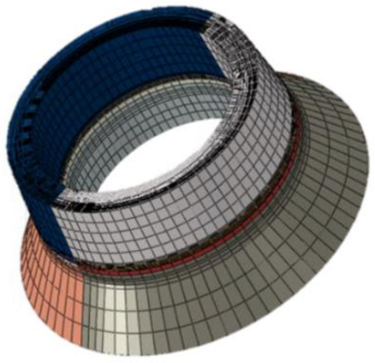
❖ Experience in Applying PED Standards

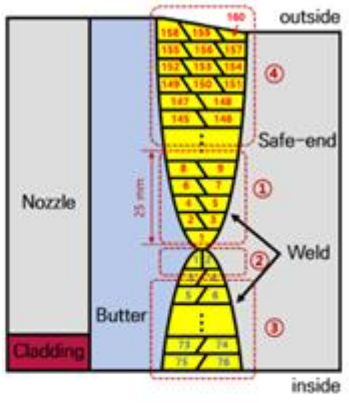
- Test Pressure
  - The test pressure requirements of PED and ASME are different.
  - The more conservative(higher value) test pressure requirement of PED was applied out of two requirements.
- Allowable Limit
  - The stress limit of PED and the stress limit of ASME are different.
  - The more conservative(lower value) stress limit of ASME was applied, and the allowable limit was calculated accordingly.
- Remarks
  - Applying the conservatism of both codes with different approaches could pose difficulties in design due to excessive conservatism.

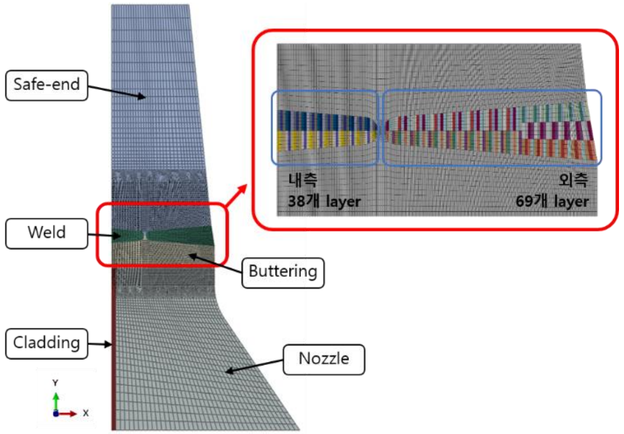


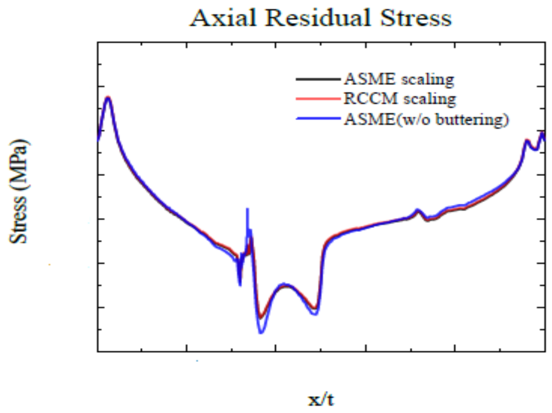
❖ Residual Stress and Deformation Evaluation using ASME & RCC-M Materials

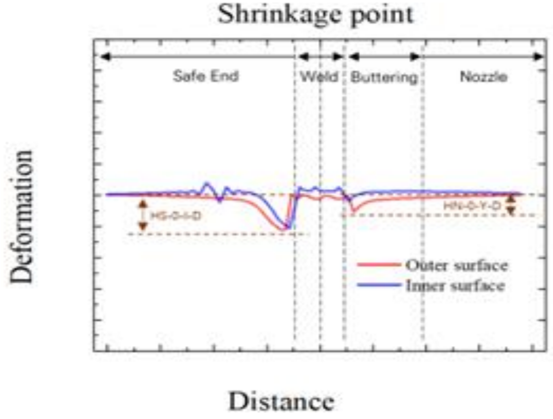
- Nozzle to Safe End Weld (Dissimilar Metal Weld)
  - Dissimilar metal weld between nozzle to safe end was analyzed using ASME & RCC-M materials.
  - Residual stresses and deformation were compared and analyzed.







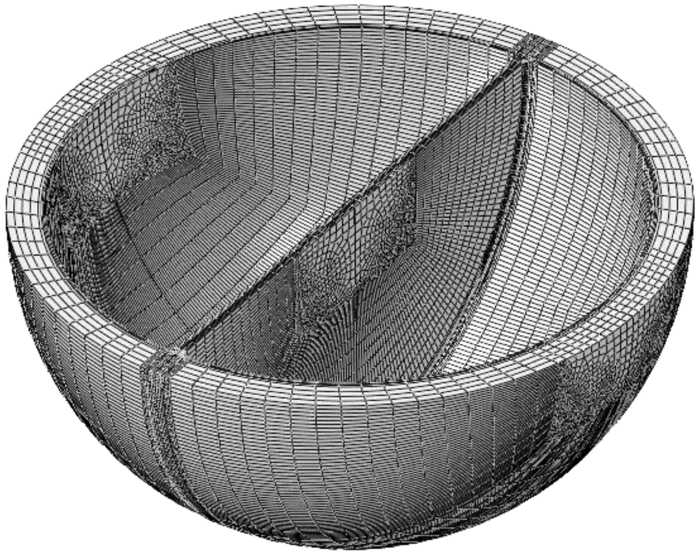


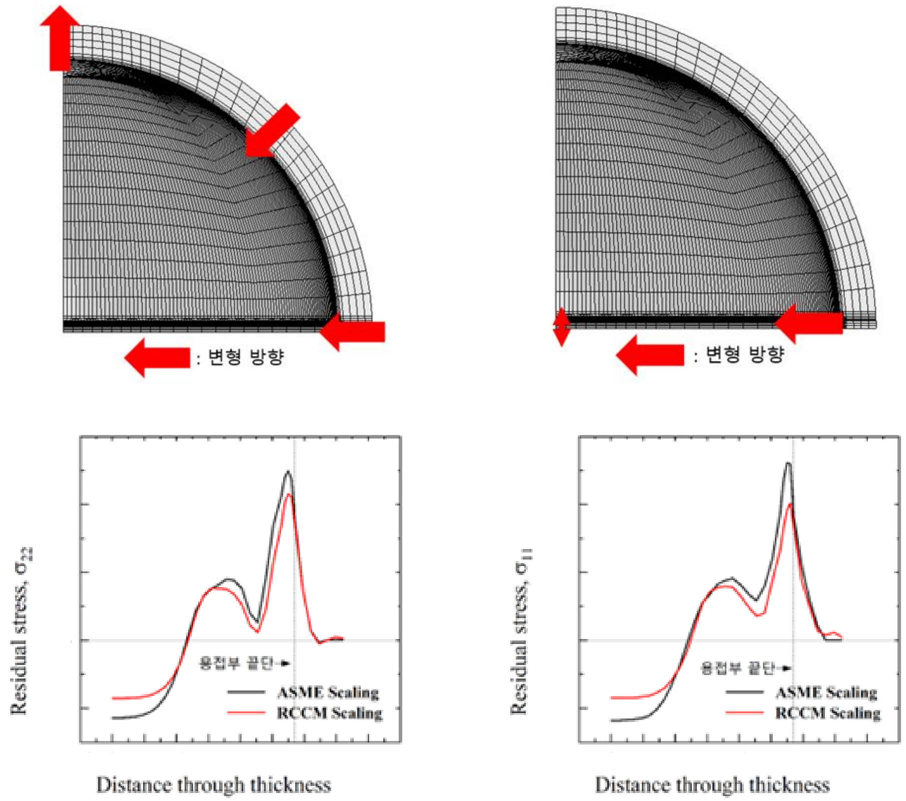


❖ Residual Stress and Deformation Evaluation using ASME & RCC-M Materials

▪ Divider Plate to Head Weld

- Divider plate to head weld was analyzed using ASME & RCC-M materials.
- Residual stresses and deformation were compared and analyzed.

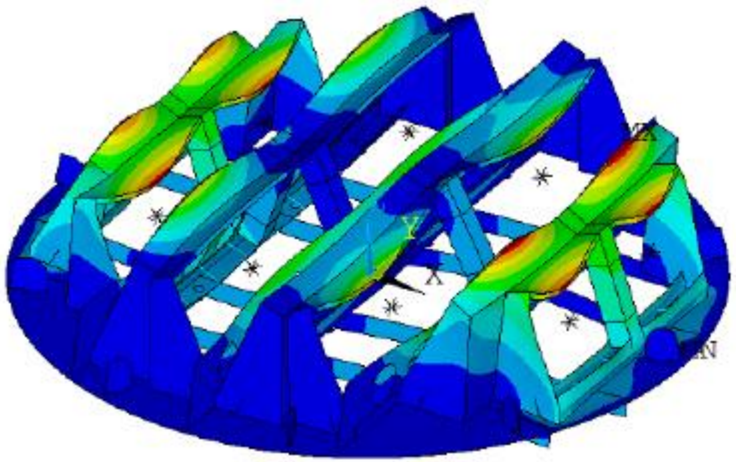




Introduction to Analytical Method Improvement

❖ Steam Generator CVAP

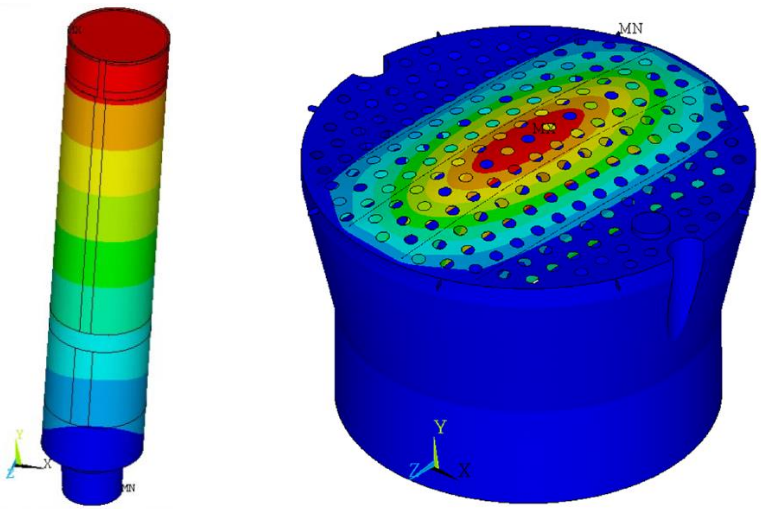
- Vibration and Stress Analysis Program
  - Vibration and stress analysis program was performed according to Reg. Guide 1.20 Rev.3.
  - Calculated stresses were compared to fatigue stress limits in ASME Section III.



Mode Shape of Steam Dryer

Location	Criteria	Safety Margin
Steam Dryer	Fatigue	≥ 10
Moisture Separator	Fatigue	≥ 10
Separator Support	Fatigue	≥ 10
Shroud	Fatigue	≥ 10

Stress Evaluation

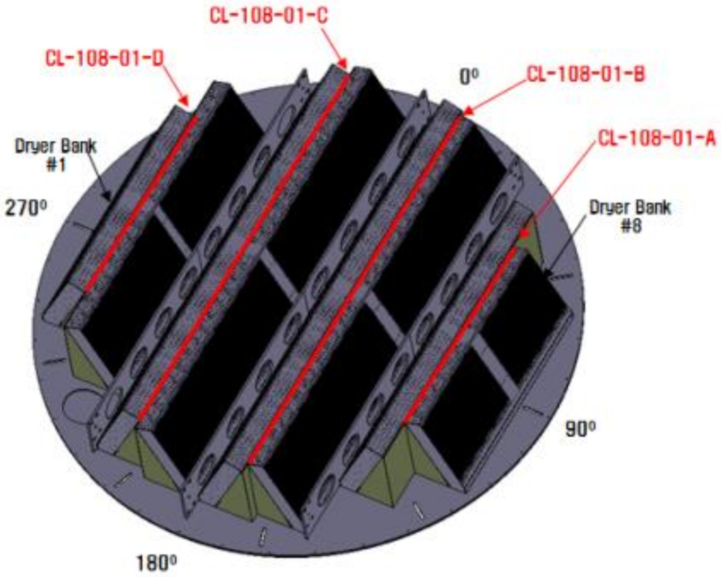


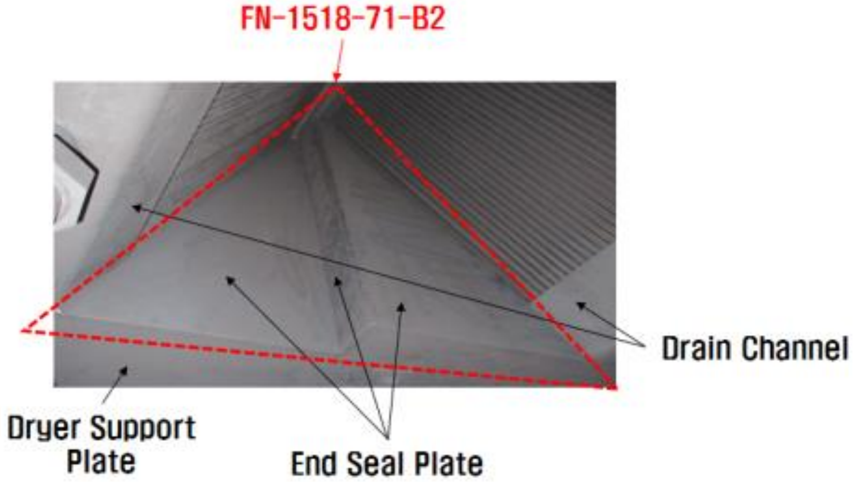
Mode Shape of Moisture Separator

- CVAP : Comprehensive Vibration Assessment Program

- Safety Margin = (Allowable - Calculated) / Calculated

- Steam Generator CVAP
  - Inspection Program
    - Inspection program was performed according to Reg. Guide 1.20 Rev.3.
    - All observations or measurements meet acceptable limits.



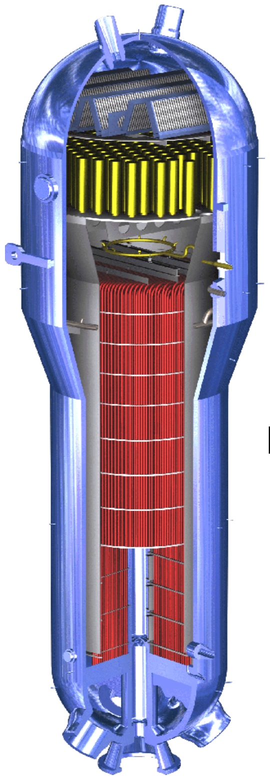


Areas inspected

❖ Blowdown Load Evaluation for Steam Generator FLB

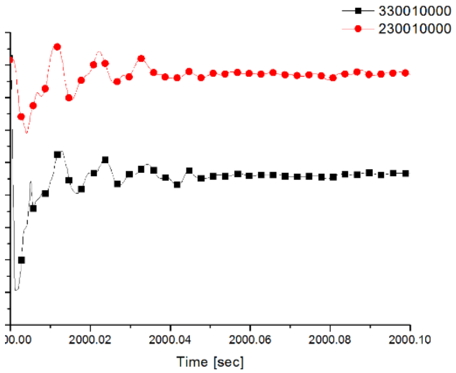
▪ Blowdown Load Evaluation Method

- Blowdown loads (such as pressure, flow rate etc.) are generated for SG Feedwater Line Break (FLB) to evaluate the structural integrity of internal structure.

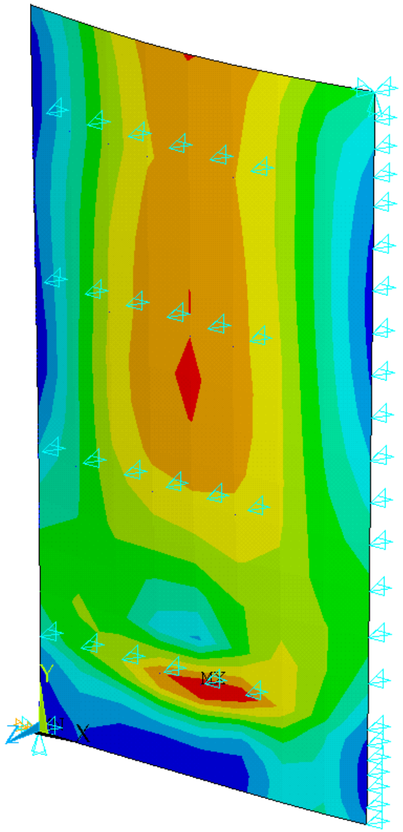


Steam Generator

Blowdown Load from Analysis

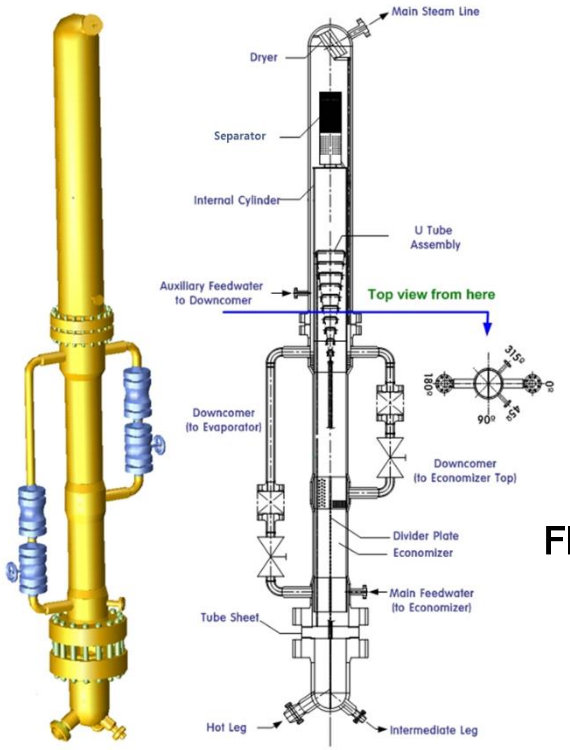


Blowdown Load to Structural Analysis



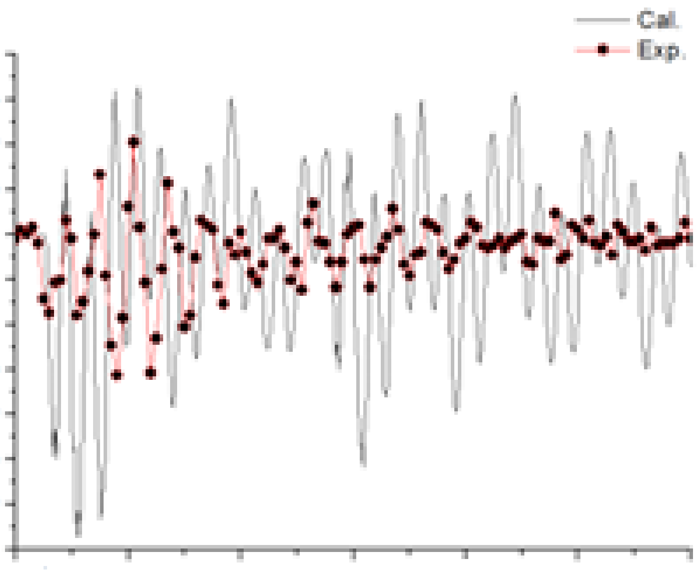
Stress Result for Internal Structure

- ❖ Blowdown Load Evaluation for Steam Generator FLB
  - ▪ Validation of Method
    - To validate the SW, the validation tests and analyses were performed.



KAERI’s Test Facility (ATLAS)

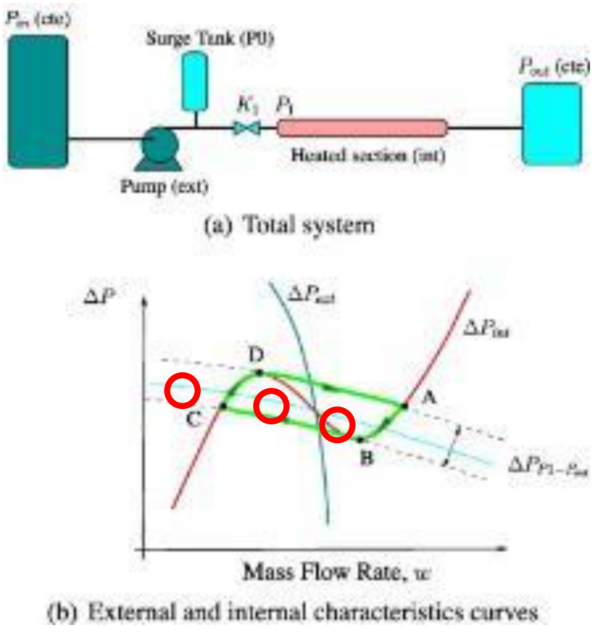
FLB Test and Analysis



Compare between Test and Analysis Results

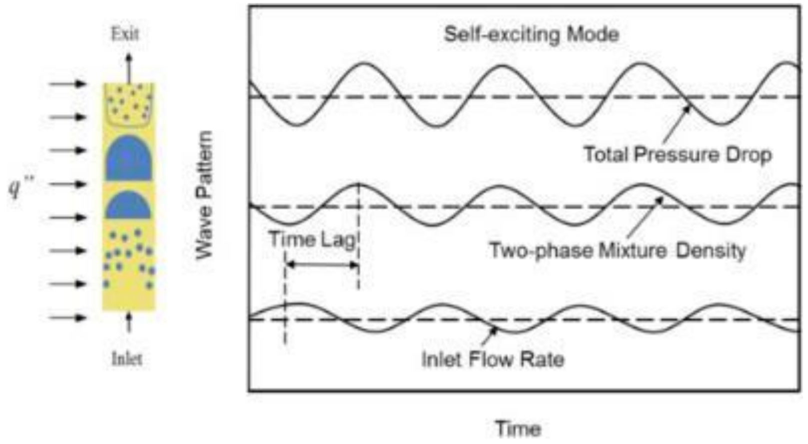
SW Validation !

- ❖ Two-phase Instability in SMR (Static and Dynamic Instability)
  - ▪ Evaluation Program
    - Developing evaluation program for two-phase instability

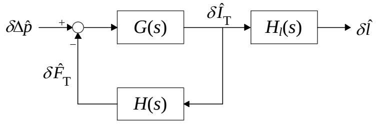


Multiple equilibrium points

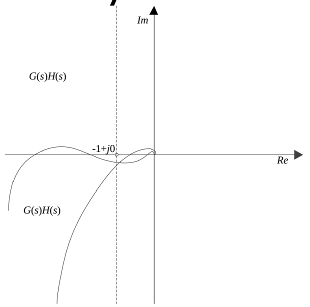
Static Instability



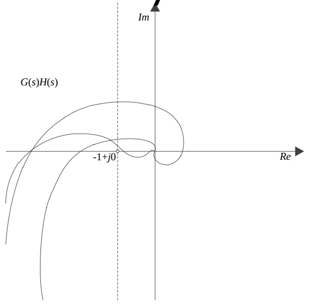
Transfer function



Stable system



Unstable system



Dynamic Instability

❖ Environmental Fatigue

- Environmental Fatigue Evaluation Method
  - US NRC Regulatory Guide 1.207 Revision 1 has been issued in July 2018.
  - Development of environmental fatigue evaluation method to incorporate the changes in Regulatory Guide 1.207 Revision 1

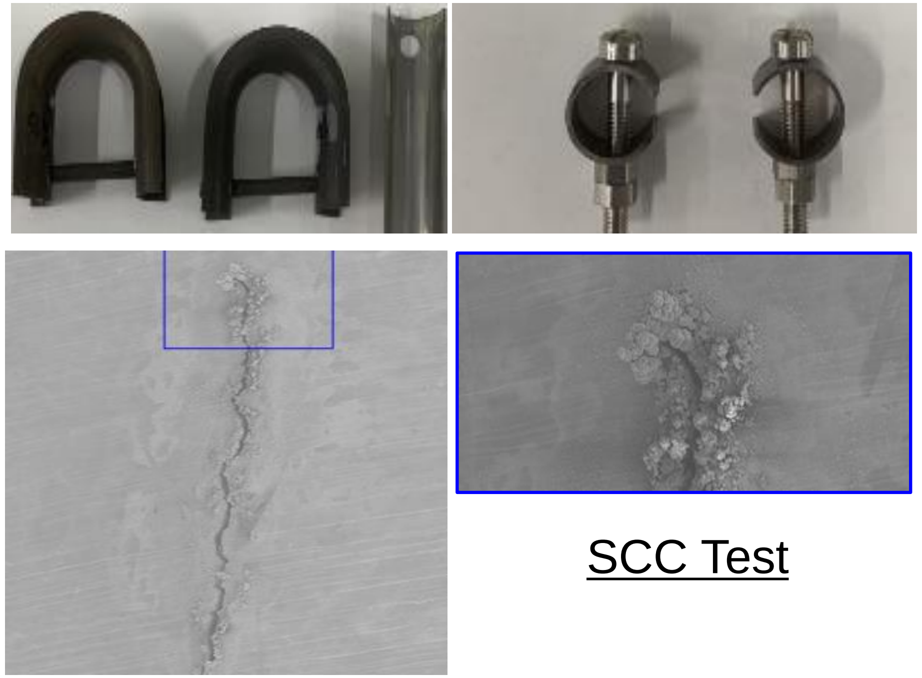
Major Changes in Revision 1

- Calculation method for environmental fatigue correction factor (F_{en})
- Applicable temperature ranges for environmental fatigue correction factor
- Applicable components
- Fatigue design curve in air environment

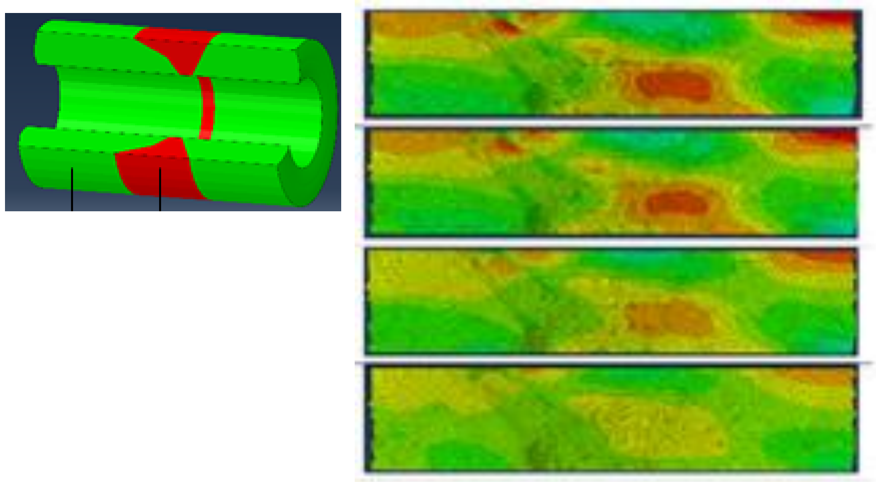
Introduction to Analytical Method Improvement

❖ SCC and Creep

- Stress Corrosion Crack
  - Conducting SCC tests on materials that are susceptible to SCC
- Creep
  - Performing creep analysis in high temperature environment



SCC Test

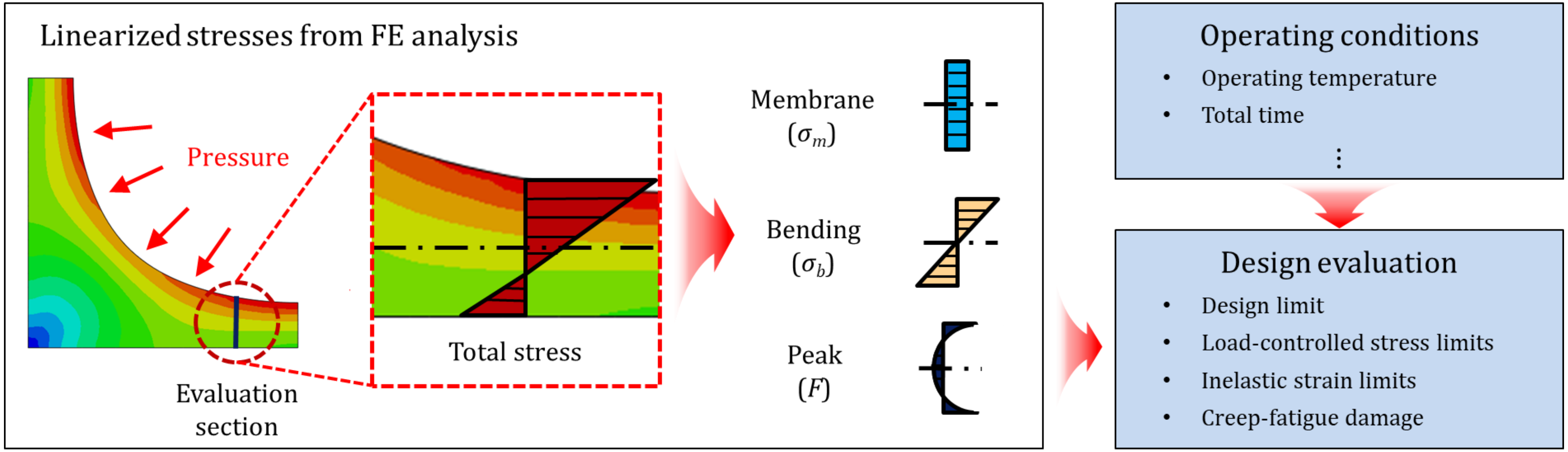


Residual & Creep Analysis

❖ Structural Evaluation for High Temperature Reactor

▪ Structural Evaluation Program

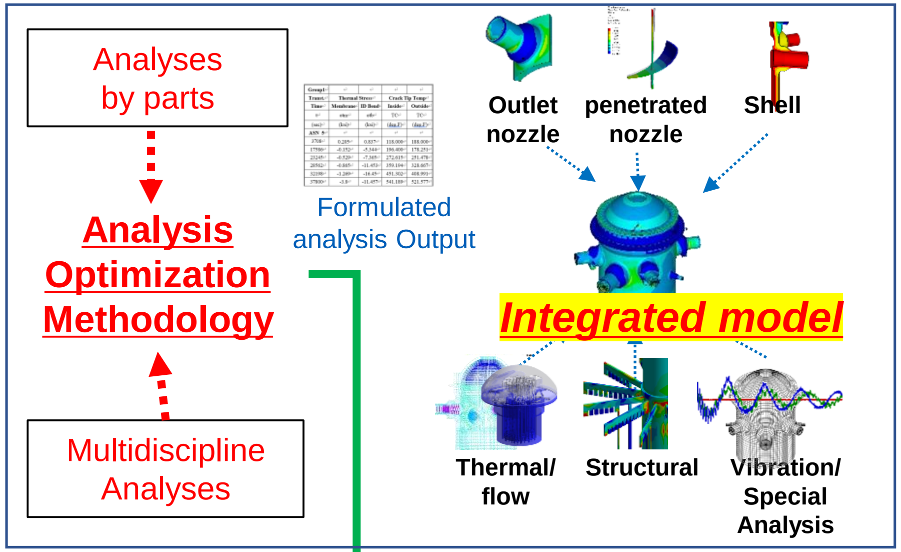
- Development of high-temperature reactor design evaluation program according to ASME Section III, Division 5
- Validation of high-temperature reactor design evaluation program



❖ Analysis Optimization and Evaluation Automation

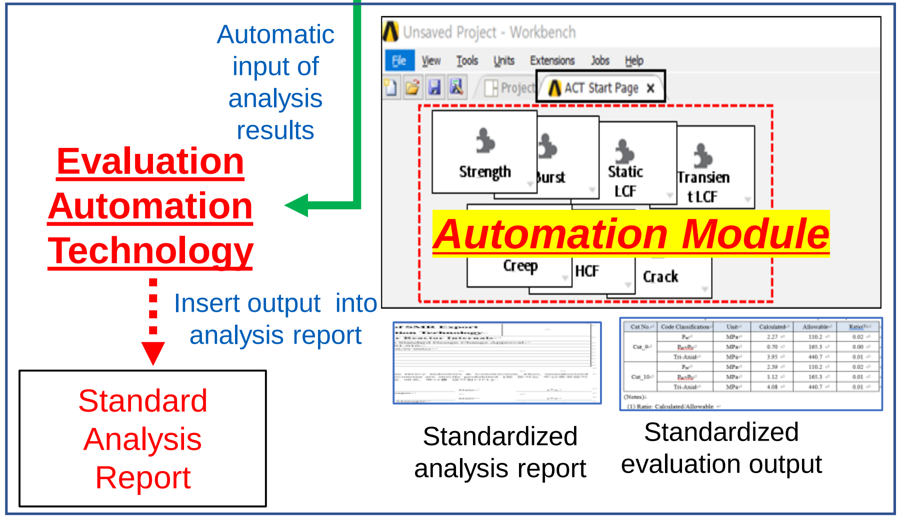
Analysis Optimization methodology

- To improve analysis models for parts
- To develop an integrated model methodology
- To validate and verify the methodology



Evaluation Automation Technology

- Development of evaluation automation program
- Material Properties Database
-
- Automate output for standard analysis report



Conclusion

- Experiences on Analysis for Structural Integrity with Various Tech Standards
  - Doosan designed, manufactured and supplied major equipment globally using various technical standards.
  - Doosan has extensive experience in various ageing management.

- Introduction to Analytical Method Improvement
  - Doosan is developing and improving analytical method
    - to meet the new and revised regulatory requirements and standards
    - to apply technical requirements to new reactors
    - to resolve various issued regarding ageing
    - to standardize, validate, and verify analysis methodology

Thank you

Questions ? Answers !

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Conclusions of day 1

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End of day 1

Workshop will resume tomorrow at 9:00AM

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