Introduction of session 3

Dr Yinsheng LI

Division Head

Materials and Structural Integrity Research Division, Japan Atomic Energy Agency

Introduction of Session 3:
OPEX related to ageing phenomena on RCPB

Haruko Sasaki (NRA, Japan)
Yinsheng Li (JAEA, Japan)

Objective and Scope

Objectives of the workshop:

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.

Scope of session 3

Ageing phenomena such as irradiation embrittlement, SCC, fatigue, …, have been experienced in reactor coolant pressure boundary (RCPB). So, the important ageing phenomena have been taken into consideration in mechanical codes and standards. Recently, flaws due to SCC in stainless steel piping lines were found in PWR primary water environments in some countries. The investigation on the root cause and studies related to crack initiation mechanism and crack growth evaluation are being conducted.

In Session 3, the operation experiences related to ageing phenomena on RCPB and efforts related to reflection in mechanical codes and standards will be discussed.

Presentations in Session 3

There are 5 presentations in Session 3.

1. SCC experience in France, Dr. Rachel Vaucher (ASN, France)

2. SCC in Japan, Dr. Takumi Terachi – (ATENA (KEPCO), Japan)

3. SCC Growth Rate Behavior of Stainless Steels in PWR Primary Water Environments, Dr. Do Jun Shim (EPRI, USA)

4. Fatigue Phenomena Experience, Mr. Paul R. Donavin (Becht Engineering, USA)

5. Integrity Assessment of Reactor Pressure Vessel Against Irradiation Embrittlement in Japan, Dr. Takatoshi Hirota (MHI, Japan)

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

SCC in France

Dr Rachel VAUCHER

Autorité de Sûreté Nucléaire, France

AUTORITÉ DE SÛRETÉ NUCLÉAIRE

29 June 2023

STRESS CORROSION CRACKING EXPERIENCE IN FRANCE:

SAFETY INJECTION AND RESIDUAL HEAT REMOVAL SYSTEMS

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

Rachel VAUCHER
ASN – Nuclear pressure equipment department

CONTENTS

FRENCH FLEET

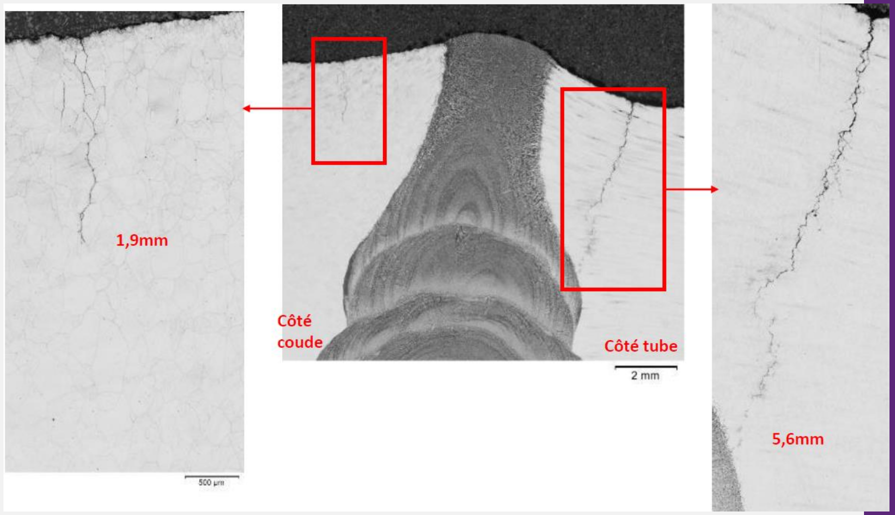
56 reactors in operation:

- A standardized fleet (3 or 4 loops PWR)
- One licensee (EDF)
- One vendor (Framatome)
- 32 of 900 MWe (CP0 and CPY types)
- 20 of 1300 MWe (P4 and P’4 types)
- 4 of 1450 MWe (N4 type)
- Nuclear energy = 75 % of electricity production in France

Reactors	900 MWe CP0 / CPY	1300 MWe P4 / P’4	1450 MWe N4
56	32 4 / 28	20 8 / 12	4

DISCOVERY OF STRESS CORROSION CRACKING (SCC) ON AUXILIARY LINES OF THE MAIN PRIMARY SYSTEM OF THE FRENCH REACTORS

- **Context:**
  - SCC known to affect stainless steel, but not expected on the auxiliary lines of the main primary system.



- **Discovery of the first cracks:**
  - Cracks were incidentally discovered during ten yearly inspections of some reactors
  - SCC currently affects the emergency core cooling systems (ECCS) and residual heat removal systems (RHRS) of 1300 MWe and 1450 MWe reactors; oldest reactors (900 MWe) seem less susceptible
  - 200 welds inspected since December 2021: cracks found are between 1 mm and 23 mm deep

DESTRUCTIVE AND NON DESTRUCTIVE TESTS

- Destructive tests performed on the welds concluded that the flaws found are cracks due to IGSCC

- Non destructive and destructive tests have been performed on whole lines on safety injection and parts of residual heat removal systems

- About 200 welds have been investigated (NDT, expertises, both)

- IGSCC confirmed on :
  - many welds on 1450 MWe reactors,
  - fewer and smaller flaws on P'4 type reactors (1300 MWe)
  - very few and small flaws on P4 type reactors (1300 MWe)
  - nearly none on 900 MWe reactors
  - more investigations are ongoing using the advanced ultrasonic testing

CRACKS ANALYSES

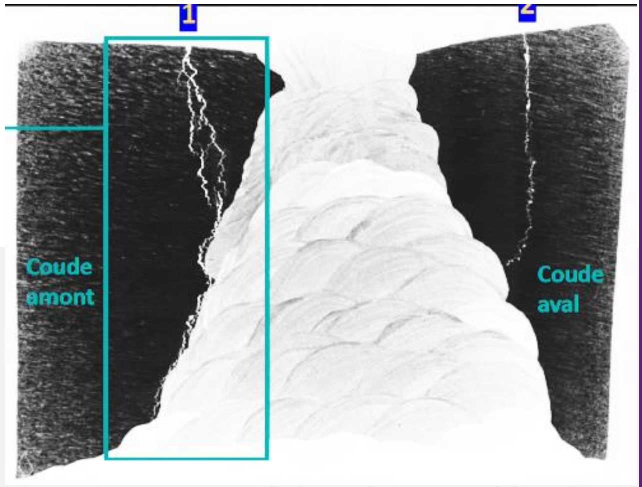
- **Observations**

➢ *Origin of the flaws:* Hardness > 240 HV and fully intergranular cracking lead to characterize the degradation as stress corrosion cracking of stainless steel in a high hardness area induced by welding

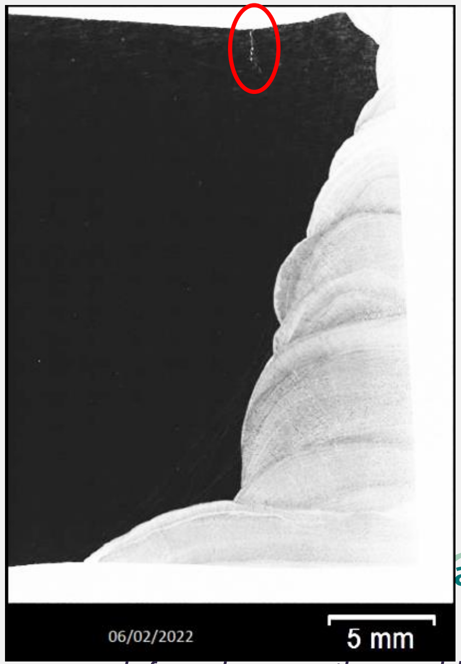
➢ *Location*: in the base metal, in the “thermo-mechanically” affected area of the welds (possibly on both sides of the weld)

➢ *Angular extension* of the crack can reach 360°

➢ IGSCC was *assumed to be limited in depth* (presence of a compression stress area within the wall of the pipe, due to the welding residual stress distribution) until a 23-mm crack was detected on a repaired weld



23mm crack found on a weld (pipe 27mm thick)

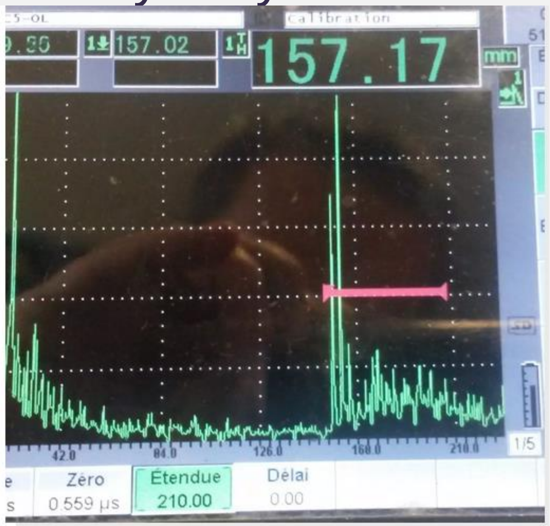


2mm crack found on another weld (same reactor)

FOCUS ON NDT – ULTRASONIC TESTING (UT) (1/2)

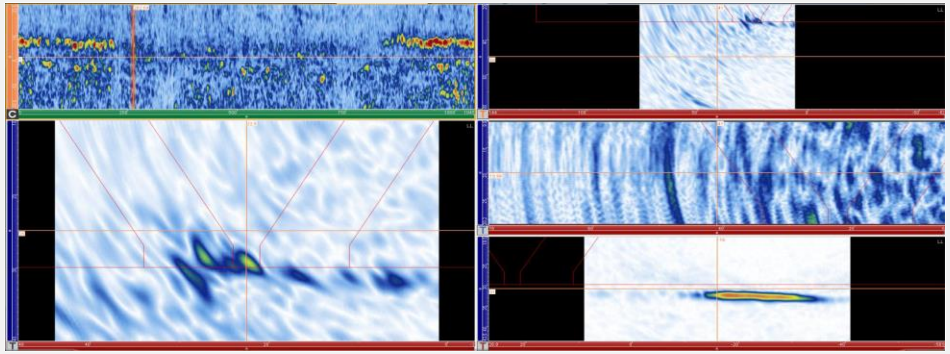
- **Initial NDT procedure**
  - The initial UT procedure, designed to detect thermal fatigue cracks, could not reliably detect IGSCC, even if large IGSCC cracks have been detected using it
  - In order to check if IGSCC could be present in other reactors, EDF has proofread all the NDT reports already drawn up of all the operating reactors, to ensure that no indication has been wrongly considered as non-relevant. Among the 6 reactors were identified with a significant risk of IGSCC and has been inspected, only 1 had significant IGSCC.
  - During spring and summer 2022, destructive analyses were the only way to confirm the existence of IGSCC flaws.

« classic UT » screen



FOCUS ON NDT – ULTRASONIC TESTING (UT) (2/2)

- **Advanced UT**
  - EDF developed a new NDT method in order to be able to detect and size the flaws with accuracy (detection threshold of 2mm and measurement uncertainty of +/-1,1 mm).
  - This technique is based on Phased Array Ultrasonic Testing (PAUT) combined with the Total Focusing Method (TFM). The TFM involves reconstructing an image based on an A-scan matrix.
  - Analysis requires a dedicated training and a significant time.
  - Formal qualification based on RSE-M code in 2023



Example showing the complexity of « advanced UT» analyses

1 - Background

2 - Expertises and Analyses

- Laboratory expertises
- Cracks analyses
- Focus on NDT

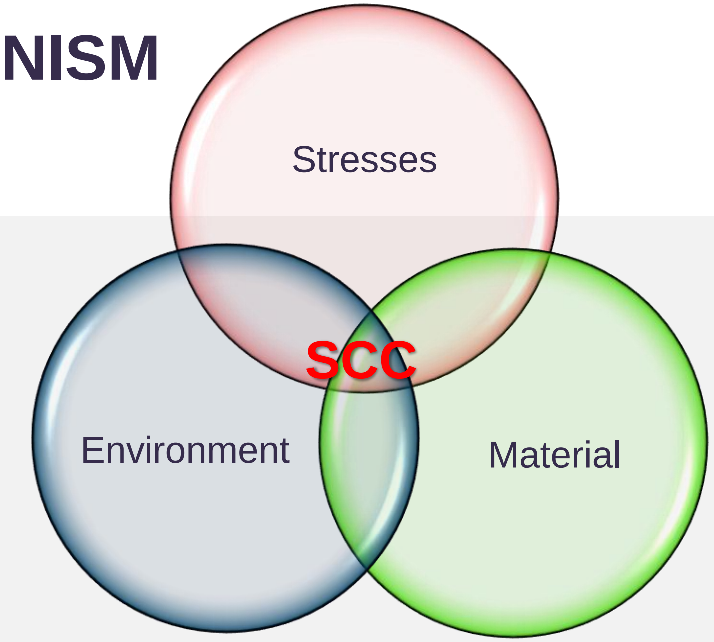
3 - Understanding of the phenomenon

4 - Safety issues

5 - Ongoing challenges

6 - Conclusions and in service inspection programme

SCC = A MULTI-PARAMETER DEGRADATION MECHANISM

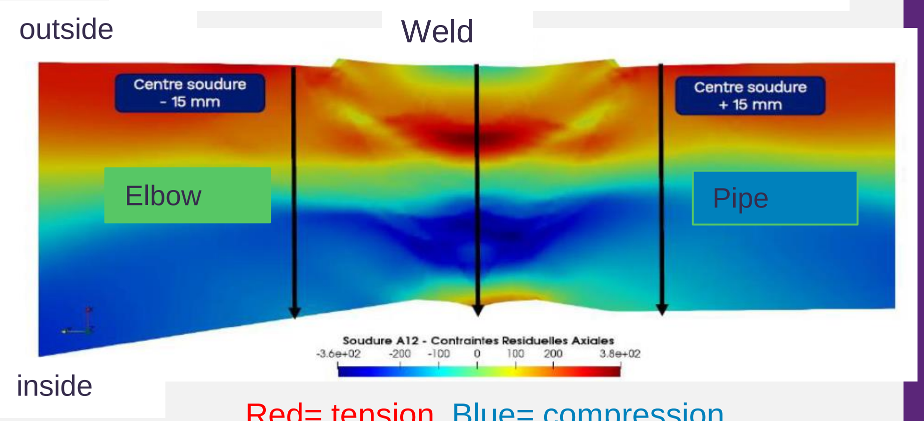


- **Environment**
  - SCC is known as thermally activated
  - No specificity has been identified regarding chemistry or the operation of the reactors. However, discussions are still ongoing about oxygen.
- **Material**
  - High steel hardness induced by welding and surface preparation has been observed but hardness levels remain mostly typical of what can be expected in the mechanically affected area of the welds.
  - End of manufacturing reports analysis → no correlation has been found between the cracks and the welding technologies.
- **Stresses**
  - Influence of manufacturing
  - Operating loads

FOCUS ON THE STRESSES (1/2)

- **Manufacturing stresses**
  - ***Residual stresses*** can be induced by welding and surface preparation (stainless steel welds not subject to any post weld heat treatment)
    - A compression area within the volume of the weld could significantly slow down the propagation kinetic of the cracks according to EDF computations

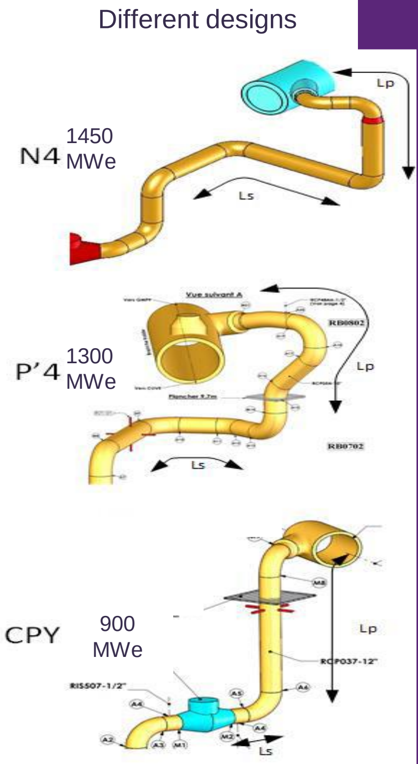
Welding stress simulation



Red= tension Blue= compression

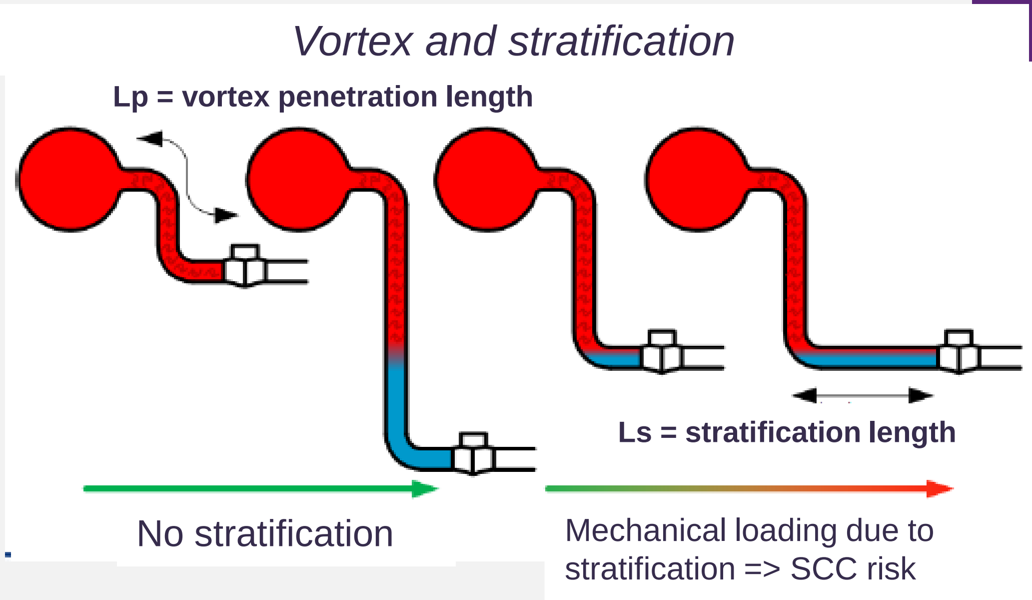
- ***Case of repaired welds***
  - Repairs of welds during the manufacturing of the plant may affect both hardening and residual stresses, and therefore increase the risk of SCC: repaired welds may become susceptible to SCC if they have been repaired, although the line is not susceptible
  - **Bigger cracks detected in some repaired welds** than in similar non repaired welds
  - EDF proposed a first assessment of the influence of the type of repairs performed, but there is still no complete understanding of the impact of repair on SCC

FOCUS ON THE STRESSES (2/2)



- **Operating loads**
  - Computer simulation shows that in service stresses are needed to initiate and propagate IGSCC (≃180MPa)
  - In some configurations, there can a layer of cool fluid under a layer of fluid at the temperature of the main primary circuit, that generates a bending load on the pipe, which can fluctuate, and increase the risk of SCC
  - **Vortex and thermal stratification** are now considered by EDF as the **main causes of in service stresses**: susceptibility ranking carried out for all systems and reactors seems consistent with the detected IGSCC (up to now…)

➢ These **phenomena can also induce thermal fatigue cracks**: some pipes subject to thermal stratification (and susceptible to SCC) showed some thermal fatigue cracks (up to 12 mm depth), even once combined with SCC



1 - Background

2 - Expertises and Analyses

- Laboratory expertises
- Cracks analyses
- Focus on NDT

3 - Understanding of the phenomenon

4 - Safety issues

5 - Ongoing challenges

6 - Conclusions and in service inspection programme

- **Reactors stops**
  - All reactors considered as the most susceptible ones (N4 type, reactors with indications that may have been considered as non relevant) have been stopped to carry out inspection and repairs where needed

- **Safety and mechanical studies**
  - Critical defects calculated through mechanical analysis on the most critical areas are bigger than the observed defects
  - Safety study showed the ability to bring back reactor to safe conditions even with the complete loss of 2 (out of 3 (900MWe)/4 (1300 and 1450 MWe)) safety injection lines

- **Impact on operation**
  - Reinforced reactor oversight to secure the ability to quickly detect any leak has been implemented by EDF on all reactors. Inspections are carried out by ASN on all NPPs

- **Crack Growth Rate**
  - Discussion about the CGR to be considered in mechanical stability calculations of a crack (justification in case the crack is not removed)
- **In service inspection programme**
  - Implementation of the programme between 2023 and 2025 on all reactors
- **Advanced UT: from development to industrial practice**
  - Training of inspectors and supply of inspection devices
- **Further investigations**
  - Similar reactors and identical lines within a same reactor can have IGSCC or not
  - Need of further investigations to assess the impact of various influent parameters
- **Alternative repair methods**
  - Investigations on mitigating/preventive methods (e.g. MSIP)

THE EFFECT OF THE WELDING PROCEDURE – THE CASE OF WELD REPAIRS

- **Current conclusions on repaired welds on stainless steel auxiliary lines**
  - Further technical investigations still need to be carried on the reach a better understanding of the influence of weld repairs on SCC
  - The discovery of a deep SCC crack (23 mm) on a line that was not considered susceptible to the thermal stratification effect, showed that repaired welds on non-susceptible lines can still be highly susceptible to SCC

➔ A specific control programme will have to be carried out be by EDF on the welds that have been repaired during the manufacturing.

STRATEGY FOR THE NEXT YEARS (2023-2025)

- **Pipes considered as highly susceptible**
  - All the pipes will be replaced before the end of 2023 and monitored afterwards (they are repaired without changes in the design)

- **Pipes not considered as susceptible or with a "low" susceptibility (thermal stratification influence)**
  - Low susceptibility lines: controls will be implemented on a selection of welds of each line
  - Lines that are not considered susceptible : controls will be implemented on a selection of representative lines according to a "defence in depth" strategy
  - This strategy will mainly focuses on ECCS and RHRS lines, but will also be extended to the pressurizer line (which has a similar diameter) and on other stainless steel auxiliary lines of a lesser diameter to confirm that they are not susceptible to SCC

STRATEGY FOR THE NEXT YEARS (2023-2025)

- **Welds repaired during the manufacturing**
  - These welds represent the most important safety issue: it is one of these that showed a 23 mm crack, whereas the cracks found on other welds were below 6 mm
  - EDF programme of controls will include the control of all the repaired welds of the ECCS and RHRS lines on all French reactors before the end of 2025, and of all the repaired welds that are the most susceptible according to EDF before the end of 2023.

- **Conclusions**
  - The SCC detected on the auxiliary lines of the main primary system is not an “age-related” event (older reactors less concerned)
  - Investigations have been carried on many reactors to identify the factors triggering stress corrosion cracking
  - So far, the stresses produced by thermal stratification are considered as the most likely root causes
  - Great efforts have been accomplished for the development of an effective NDT and for the replacement of welds in a very limited time
  - However, further investigations are still indispensable to improve the understanding of IGSCC (and thermal fatigue)

Thank you for your attention!

Any question?

AUTORITÉ
DE SÛRETÉ
NUCLÉAIRE

Stress Corrosion Cracking in Japan

Dr Takumi TERACHI

ATENA (The Kansai Electric Power Co.)

SCC in Japan

29th June 2023

ATENA (The Kansai Electric Power Co.)

Takumi TERACHI

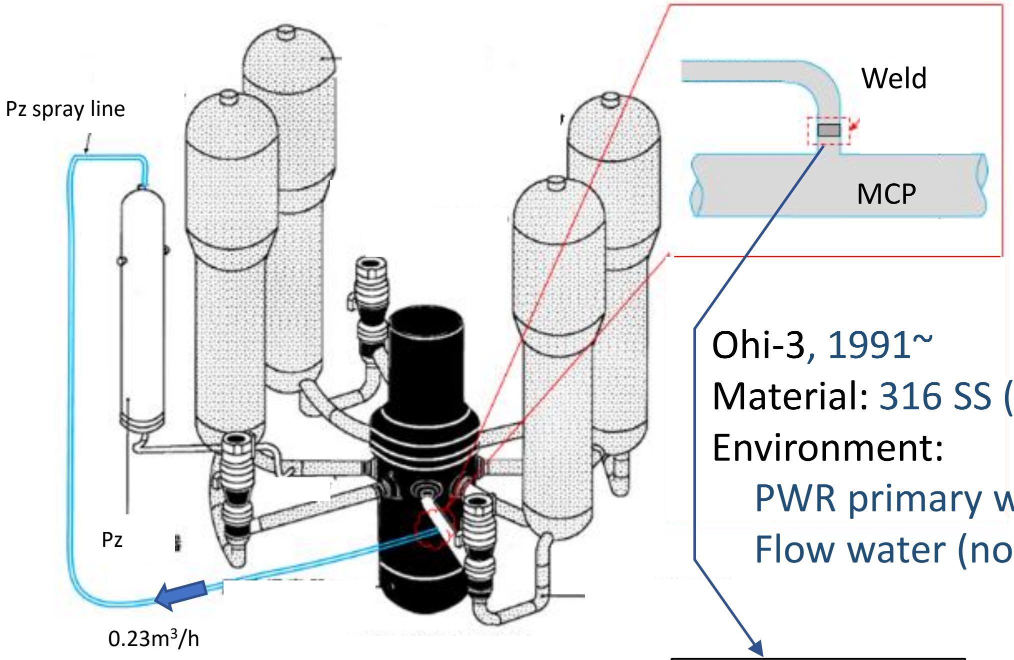
- **Summary of Ohi-3 spray line crack**
  - Observation results
  - Hardness measurement
  - Mock-up test to understand the phenomenon
  - Inspection as a countermeasure

- **Ongoing Study project**
  - Scheme of study project
  - New finding for crack initiation
  - Assumed mechanism

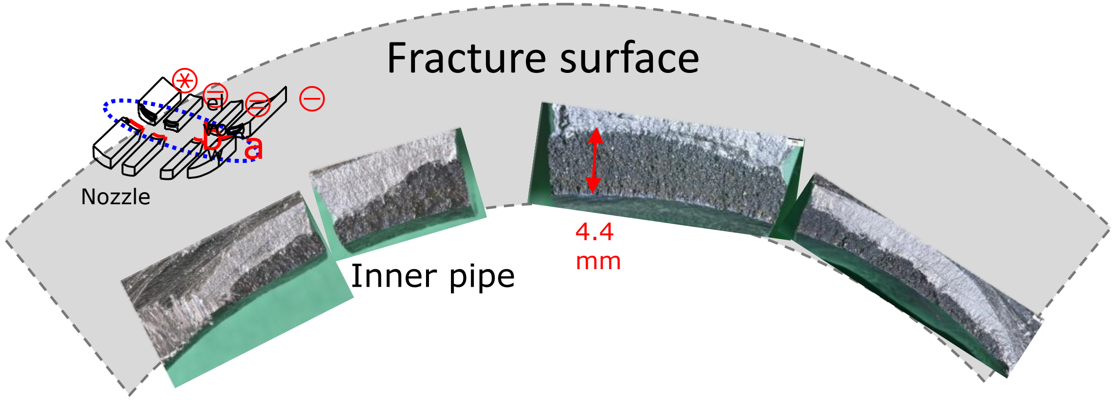
- **Crack growth evaluation**
  - Evaluation method using MRP-458 CGR curve
  - Reproduction of Ohi-3 using MRP-458
  - Influence of diameters

- What was the differences of French and Japan?

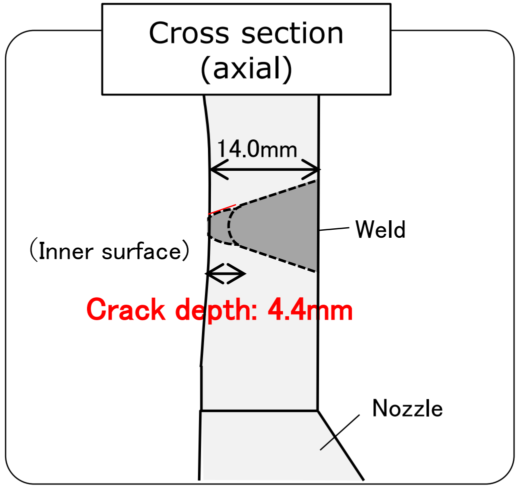
Summary of Ohi-3 spray line SCC

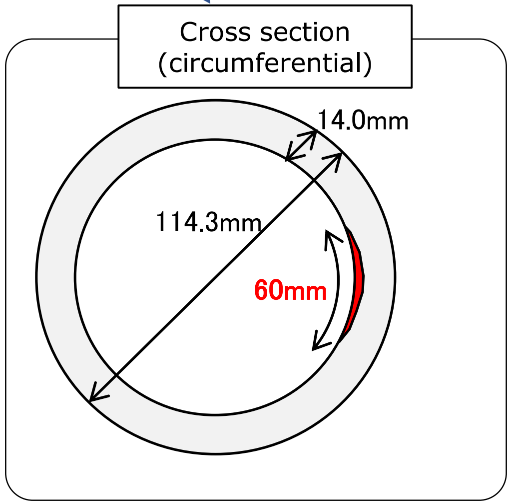


Ohi-3, 1991~
Material: 316 SS (0.04%C)
Environment:
PWR primary water(290℃)
Flow water (not stagnant)



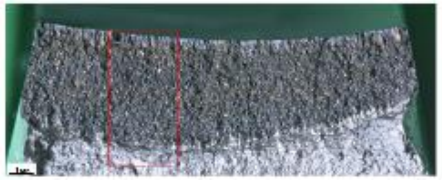
- Typical arch shaped single crack
- Corrosion products covered the fracture surface.
- The deepest length was 4.4 mm

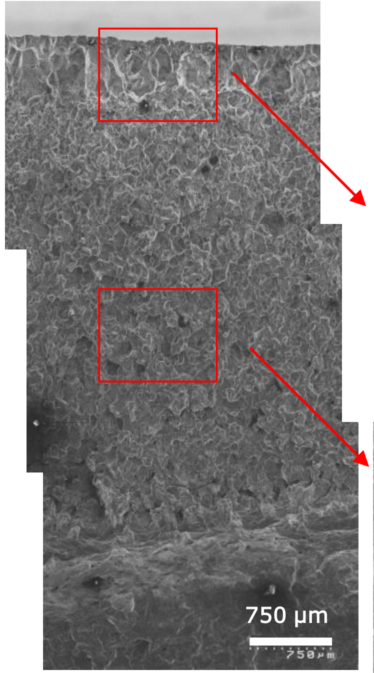


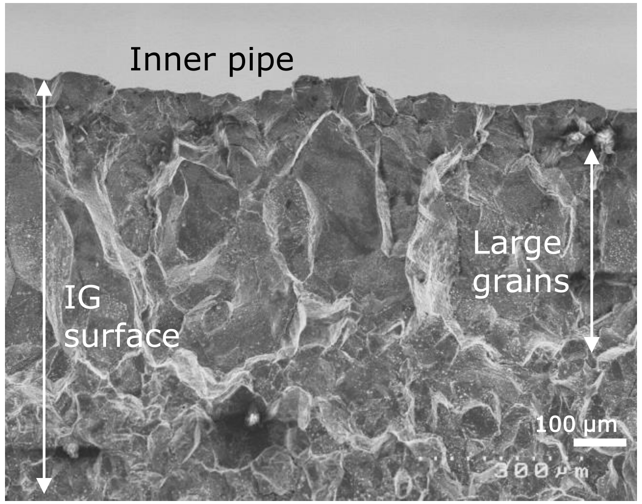


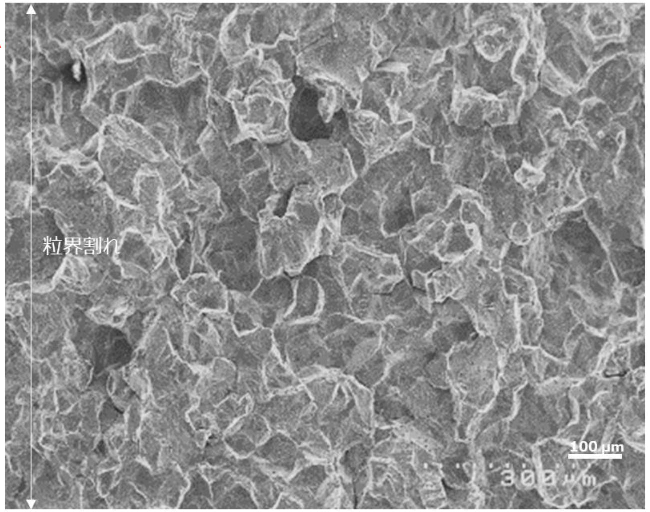
- During the In-service Inspection (ISI) UT indication was found in the pressurizer spray line.
- Circumferential single IGSCC crack was found in Heat affected zone of 316 SS.
- No involvement of oxygen and fatigue stress were eliminated as a cause of cracking.

Crack morphology

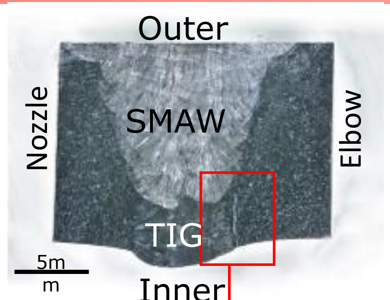


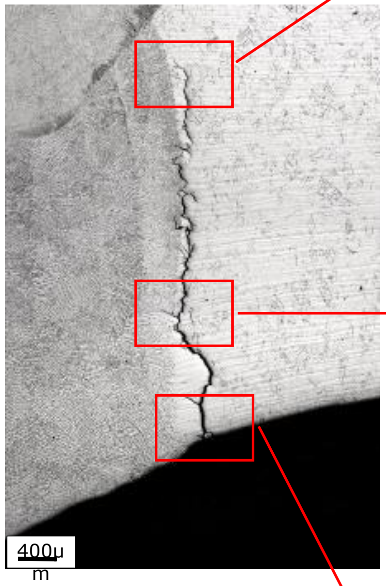






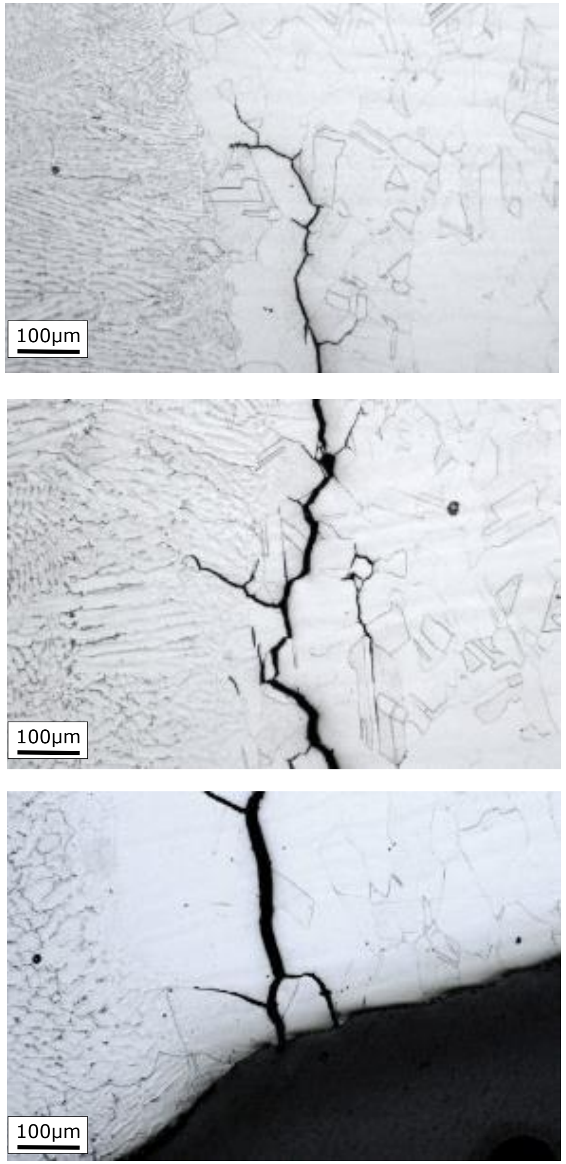
SEM fracture surface images





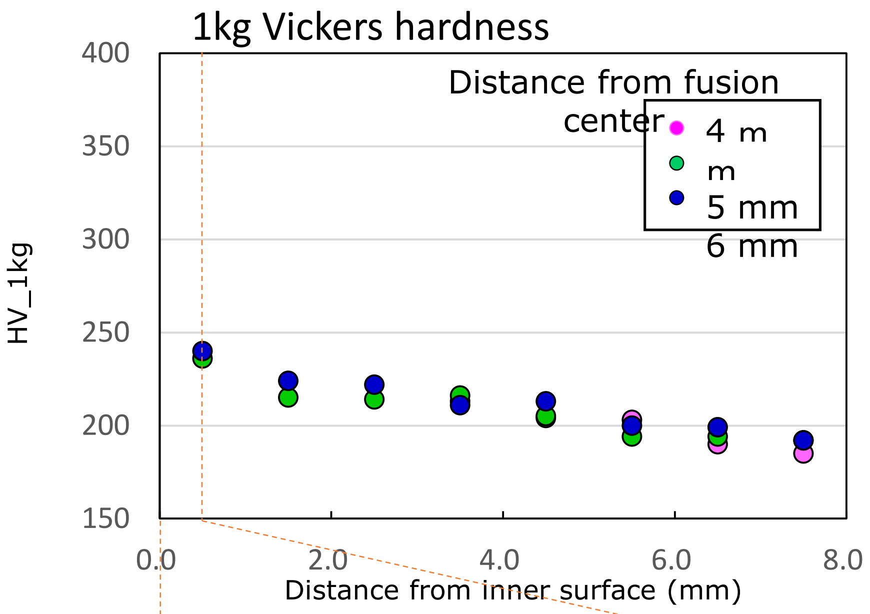
Cross-section

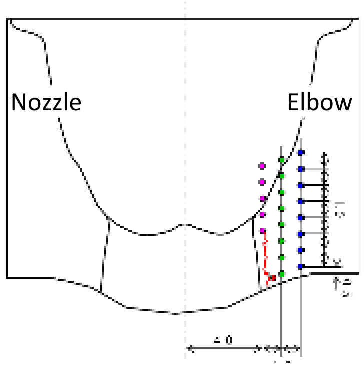
- Inter Granular crack
- Large grains were observed near the surface.
- No weld defect was confirmed.



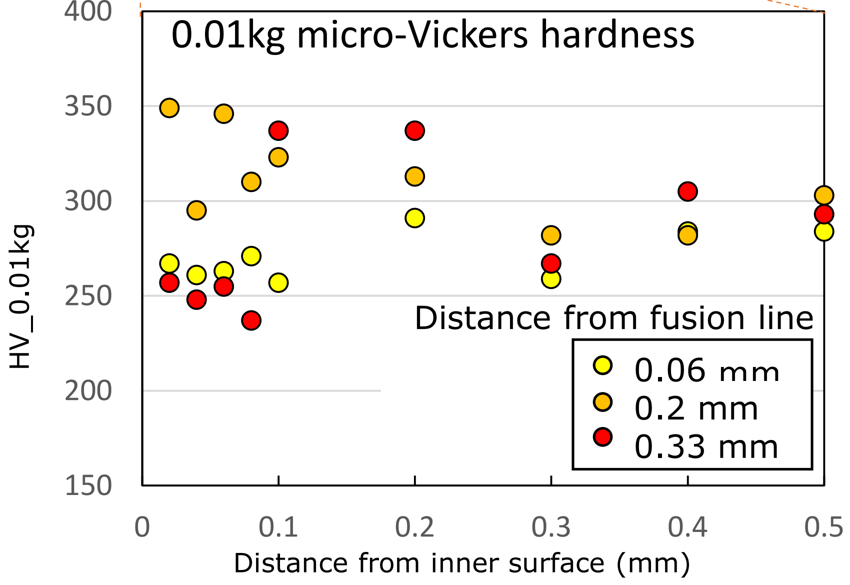
Cross-sectional Images

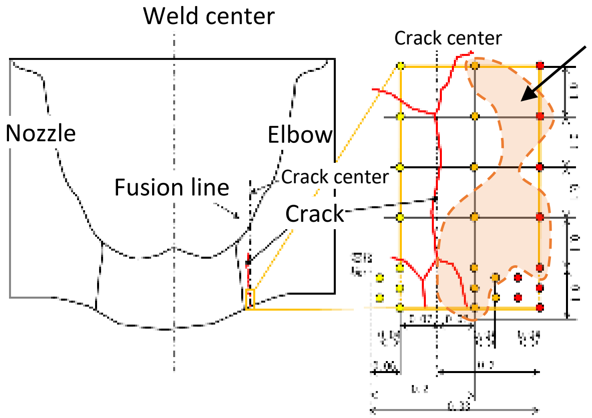
Hardness measurement





- Vickers hardness (1kg) were 180-240 HV1 in HAZ.
- The hardness without any influences of HAZ was approximately 150 HV1.



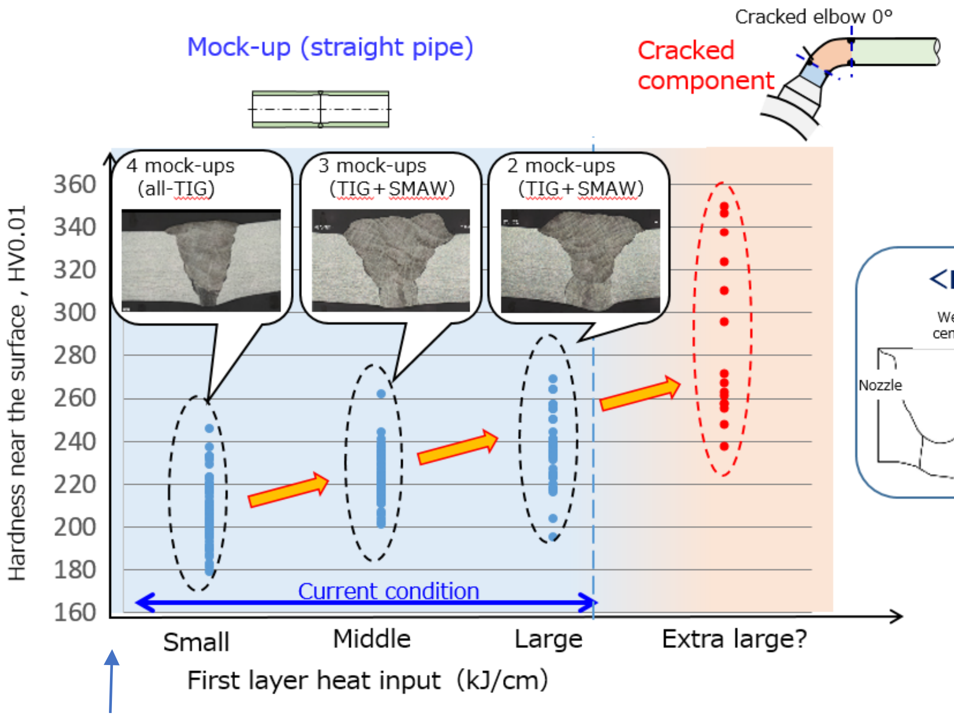


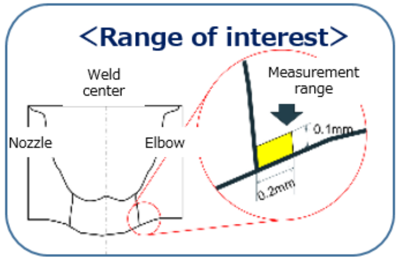
Location over 300HV

- The results were relatively scattered from 230 HV to 350 HV.
- ***Over 300 HV (extremely high hardness) were confirmed*** near the surface.

- **The cause of cracking was regarded as extremely high hardness.**

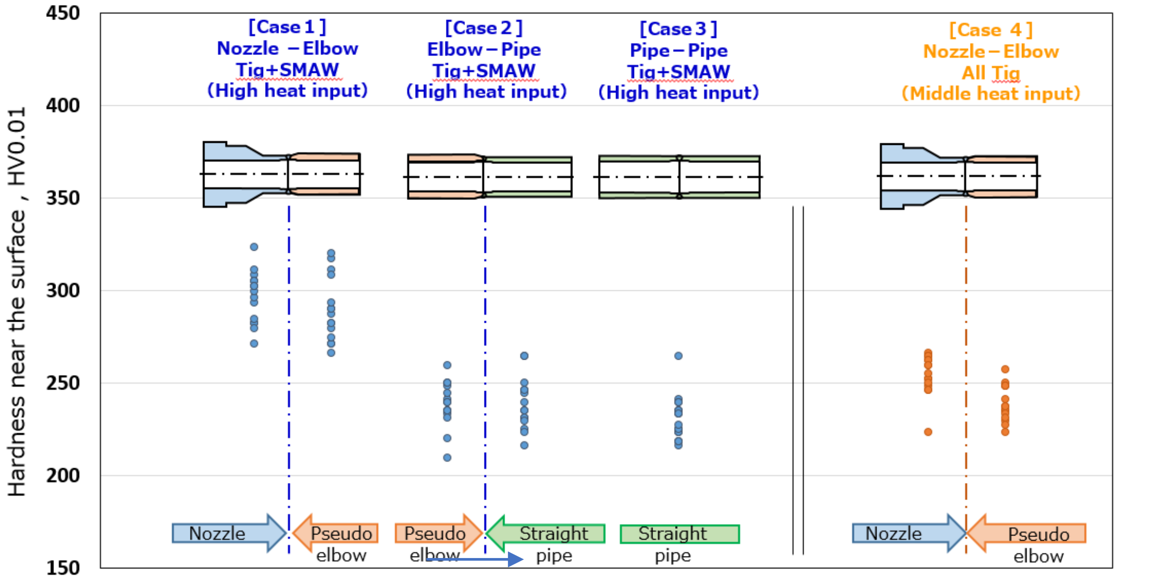
Mock-up tests to confirm the cause of hardness





Influence of Shape factor

- Nozzle affect the hardness
- Tig + SMAW is higher hardness than All Tig weld



*Pseudo elbow: Pipe with the simulated thickness of elbow.

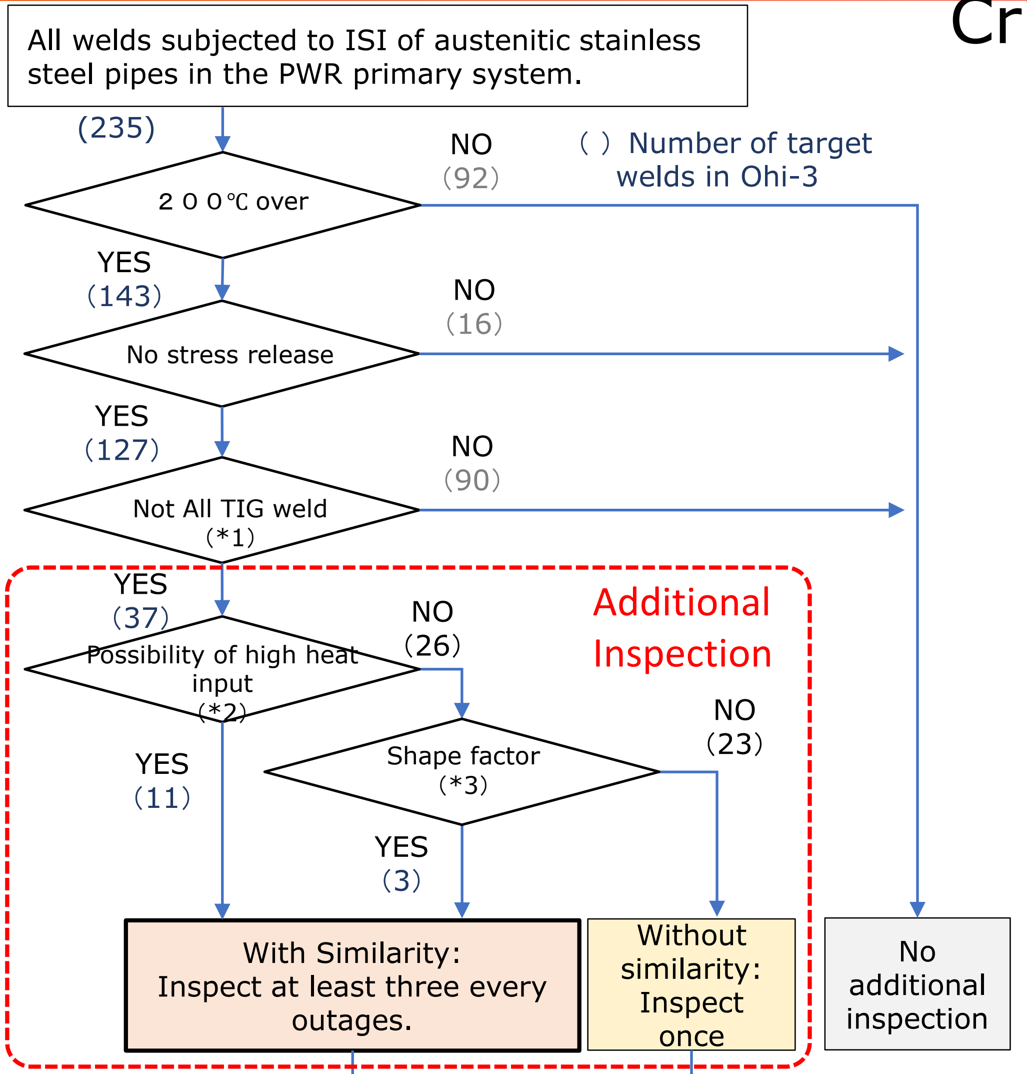
Influence of heat input on hardness

- High heat input increases hardness

Cause of high hardness were

- **High heat input**
- **Attached to Nozzle**
- **Tig + SMAW**

Countermeasures: Additional Inspection Criteria



Number of additional inspections in each NPP

M３	T１	T２	T３	T４	O３	O４
98	94	132	54	44	**37**	43

M: Mihama, T: Takahama, O: Ohi

- We have carried out over **500** additional inspections.
- There are **no cracks** found exclude for the Ohi-3 spray piping.

⇩

- To determine the inspection intervals in the future:

  1. Crack initiation mechanism
  2. Crack growth evaluation

⇩

- ATENA project started for 5 years program

(*1) Weld with All TIG excludes from additional inspection due to the low hardness.
(*2) Exclude factory weld, Include repair weld and welder with three years experience.
(*3) Welded with Nozzle

Source: http://www.atom.pref.fukui.jp/senmon/dai98kai/no.4.pdf

- Summary of Ohi-3 spray line crack
  - Observation results
  - Hardness measurement
  - Mock-up test to understand the phenomenon
  - Inspection as a countermeasure

- Ongoing Study project
  - Scheme of study project
  - New finding for crack initiation
  - Assumed mechanism

- Crack growth evaluation
  - Evaluation method using MRP-458 CGR curve
  - Reproduction of Ohi-3 using MRP-458
  - Influence of diameters

- What was the differences of French and Japan?

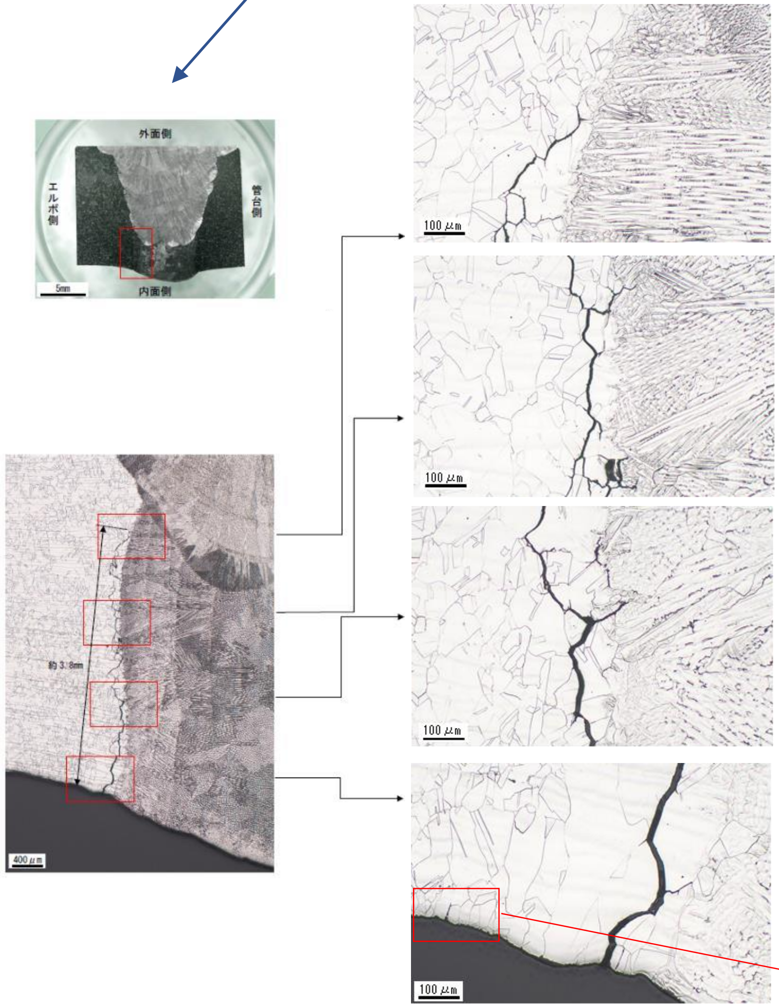
Research schedule in the ATENA project

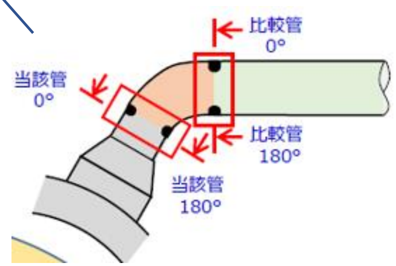
Targets	Research Items	2022	2023	2024	2025
General	1. Investigate the latest knowledge				
Prioritization of Inspection	2. Investigate the cracked material	**Cracked Material**	**Mock up**		
	(1)-①Local strain (SEM/EBSD)				
	(1)-②Hardness investigation				
	(2)-①Welding flaw: Frature surface				
	(2)-②Welding flaw: Cross-section				
	(3)Oxide firlm, Crack tip, TEM microstructure				
	3. Investigation on Crack initiation conditions				
Establishment of structural integrity assessment	5.(1) CGR knowledge investigation				
	5.(2) CGR measurement for insufficient condition				
	5.(3) Establish the CGR curve				
	6. Weld residual stress evaluation				
	7.(1) Structural Integrity assesment method				
	7.(2) Confirmation on LBB margin				
Development of technology base	8. Development of technology base				

- **Mostly Implemented by the CRIEPI and MHI**
- **METI fully support the research budget**

Source: https://www.nra.go.jp/data/000394508.pdf

Tentative finding by ATENA project: Crack initiation

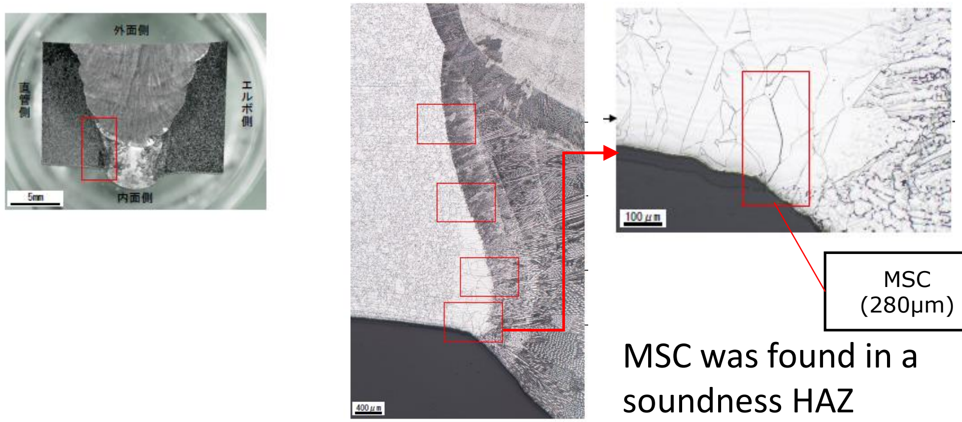




*MSC: Microstructurally Small Crack.

It is defined as a fine crack of about one crystal grain, which cannot be detected by non-destructive inspection.

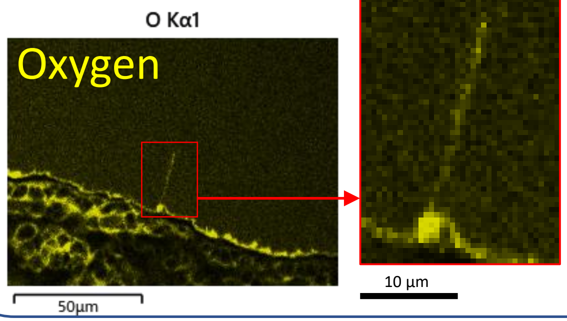
MSC (50μm)



MSC (280μm)

MSC was found in a soundness HAZ

EDS mapping of MSC



MSC was filled with corrosion products

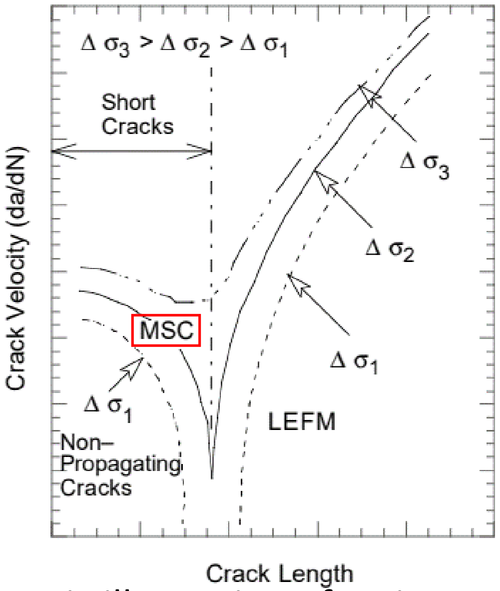
- **Microstructurally small crack (MSC) was found near the primary crack.**
- **A similar MSC was observed in a soundness specimen.**
- **Due to the filled with corrosion products, the MSC was thought to be arrested for a long time. (we are doing further investigation)**

Source: https://www2.nra.go.jp/data/000428959.pdf

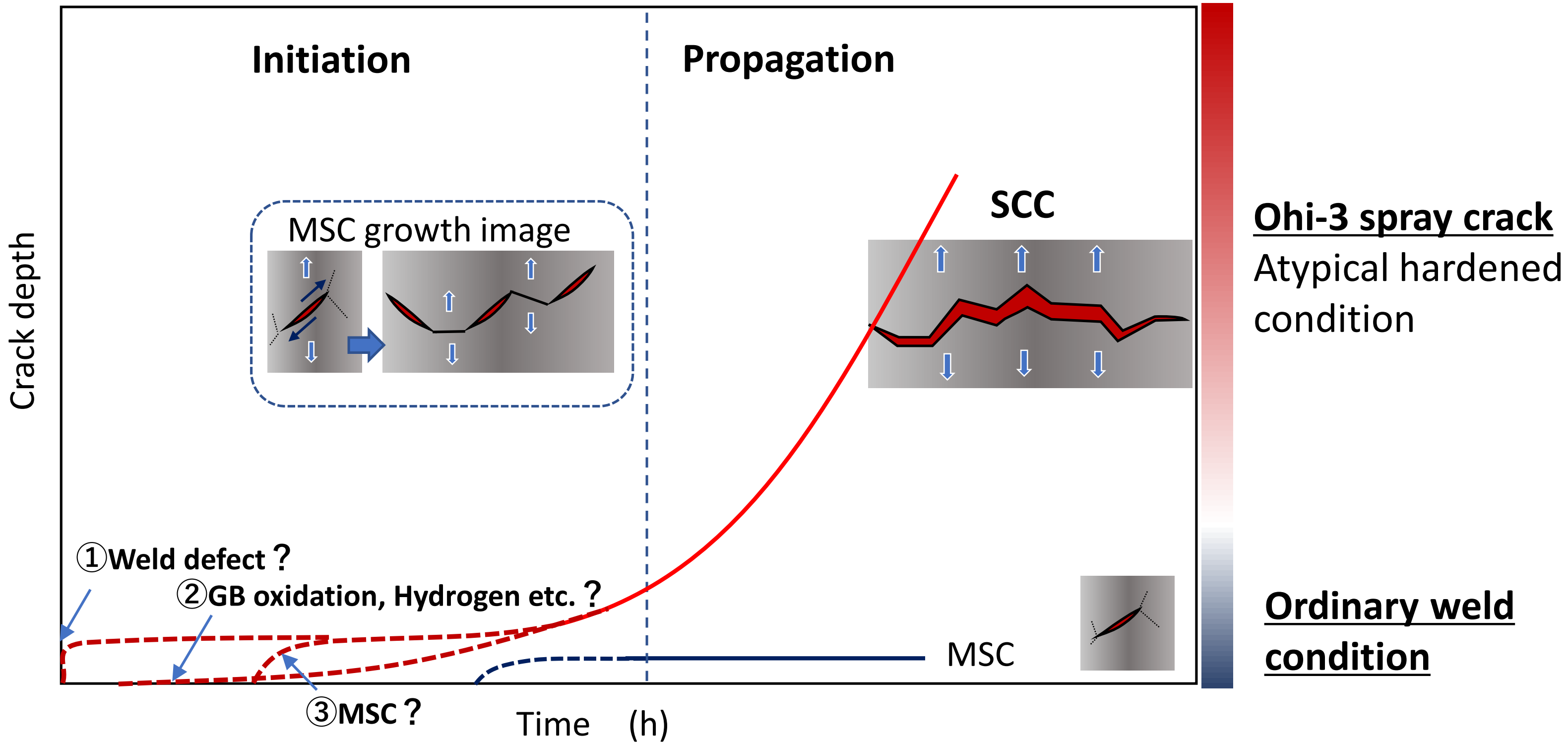
Assumed Crack Initiation Stage (Hypotheses)

Reference information on Fatigue cracks

- MSC is a common knowledge for the crack initiation of **fatigue.**
- It is sensitive to microstructure and shear stress.



Schematic illustration of Fatigue crack growth (NUREG/CR-6768)
https://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6787/cr6787.pdf



Hypotheses

① Initiated from Weld Defects.
② Gradually initiated by localized oxidation or hydrogen involvement or Cavity formation etc.
③ MSCs formation and coalesced.

- **Further evaluations are needed to understand the involvement of MSC in the crack initiation.**
- **Crack growth evaluation is important to understand the phenomena.**

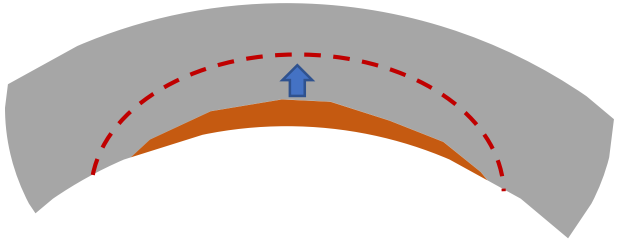
- Summary of Ohi-3 spray line crack
  - Observation results
  - Hardness measurement
  - Mock-up test to understand the phenomenon
  - Inspection as a countermeasure

- Ongoing Study project
  - Scheme of study project
  - New finding for crack initiation
  - Assumed mechanism

- Crack growth evaluation
  - Evaluation method using MRP-458 CGR curve
  - Reproduction of Ohi-3 using MRP-458
  - Influence of diameters

- What was the differences of French and Japan?

Crack growth evaluation method



EPRI provided
crack growth curve
as MRP-458

Before a crack propagated to the acceptable limit, we need to inspect the pipes.

MHI calculated the
Stress

Actual cracked component
Hardness

$CGR = 3.19 \times 10^{-18}\ K^{2.5}\ HV^{6.0} \exp(-85000 / RT)$

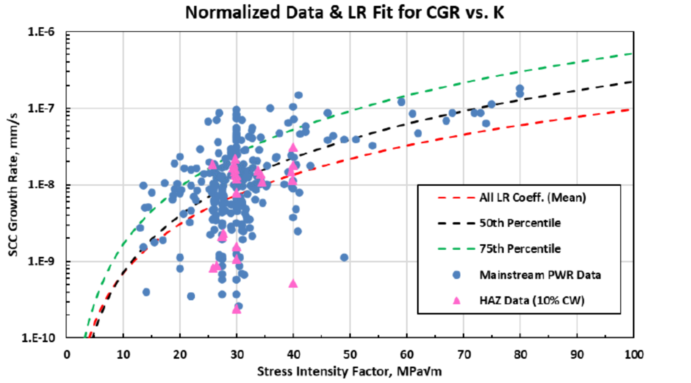
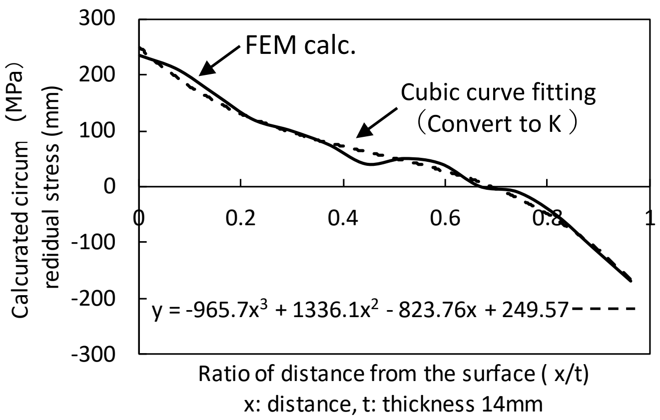


Figure 3-20
SCC growth rate data (mm/s) vs. K with the 50th and 75th percentile curves. The individual data points are normalized to, and the curves are drawn for H_V = 220 and 290°C.

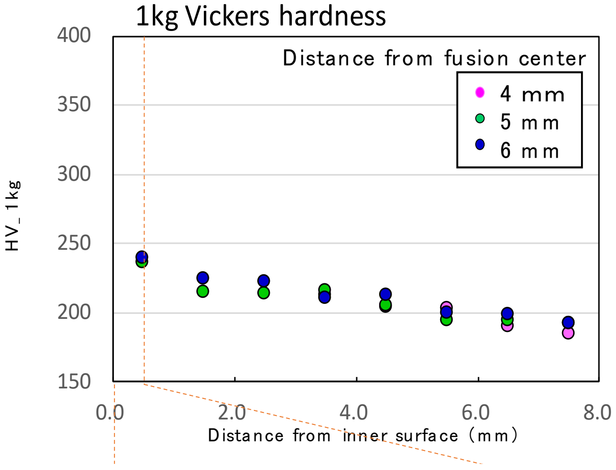
EPRI, SCC crack growth rates in stainless steels in PWR environments, MRP-458 (2022).

\+



Ohi-3 spray pipe simulated (calculated)
Weld residual stress (Tig + SMAW: FEM calculation) + Operating stress (100MPa)
（https://www2.nra.go.jp/data/000339736.pdf ）

\+

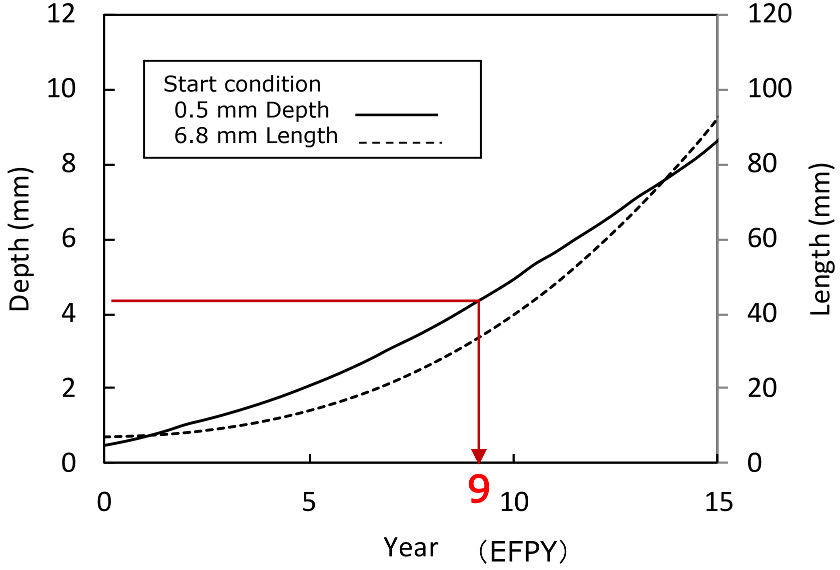


Ohi-3 spray pipe measured value
・Depth direction: Measured at 1 kg
・Longitudinal direction: maximum surface load of 10 g (350$HV_{0.01}$)

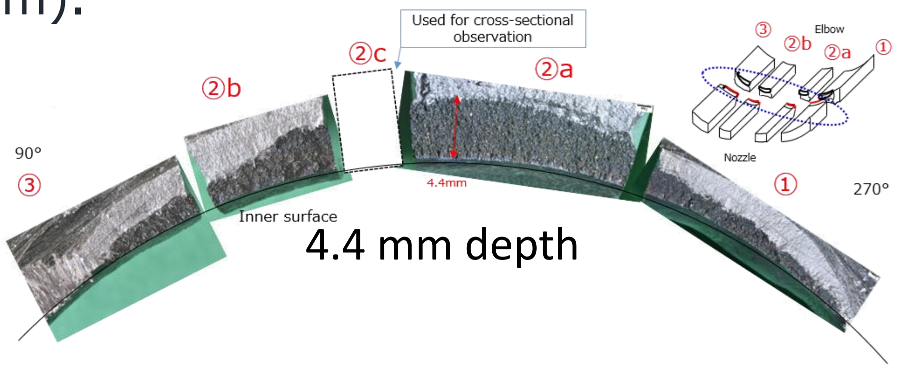
Verification of crack evaluation using MRP-458

We calculated the period for crack growth from a small crack (0.5 mm) to the observed Ohi-3 crack (4.4 mm).

Initial crack size:
(Largest grain size of Ohi-3 OE)



Crack evaluation conditions
EPRI MRP-458
Hardness: Ohi-3 measured
Stress: simulated Ohi-3 FEM result



- It took 9 years for a microcrack to grow into an Ohi-3 crack.

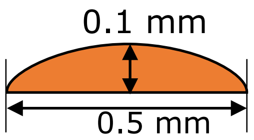
- Although the initial crack size is unclear, the crack evaluation using MRP-458 curve seems reasonable.

Source: Terachi et al., ICONE30, Kyoto (2023).

Influence of Pipe Diameter (Calculated using conventional conditions)

Using the stress and hardness conditions obtained in the past study, a comparison of diameters was carried out.

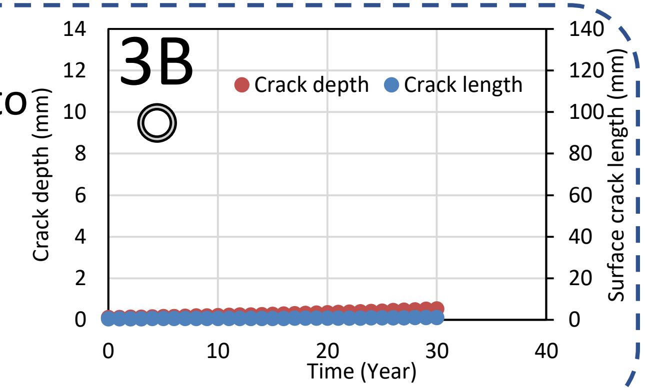
Defined Initial crack size:
Typical grain size (0.1 mm)

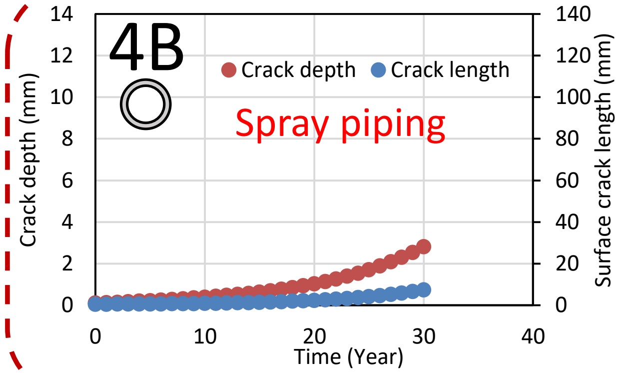


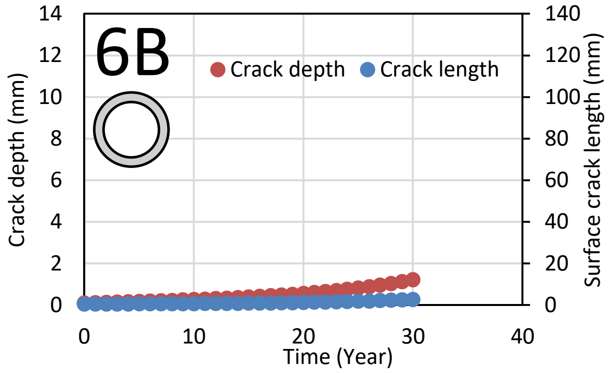
Hardness: Conventional Mock-up with All-Tig weld*1
Stress: Conventional stress: All Tig condition*1 + operating stress (100 MPa)

(*1 Results of Joint research on Electric power companies in Japan and MHI)

Low CGR due to low hardness

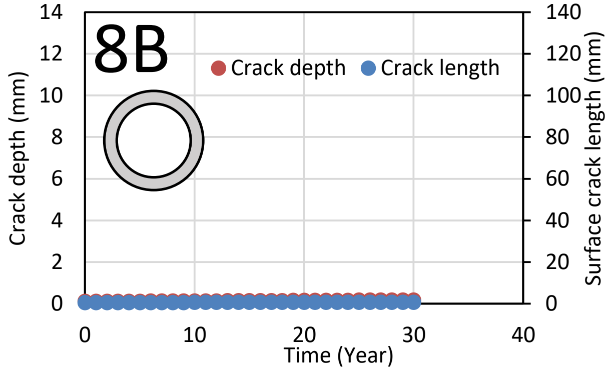


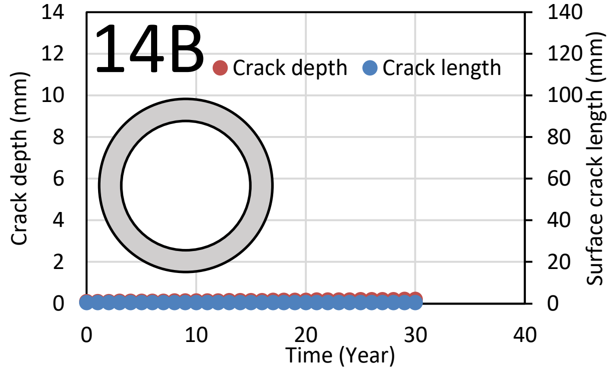




Relatively high CGR

Low CGR due to low stress near the inner surface





- **Hardness and residual stress depend on welding conditions.**
- **Standardization of these is desirable to examine the inspection frequency of components.**

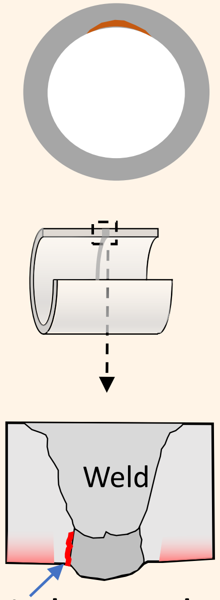
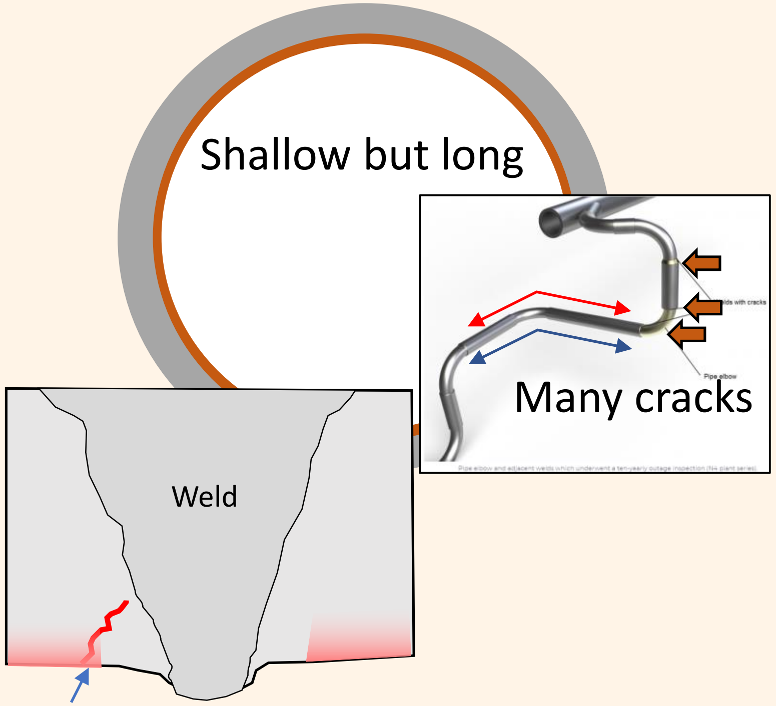
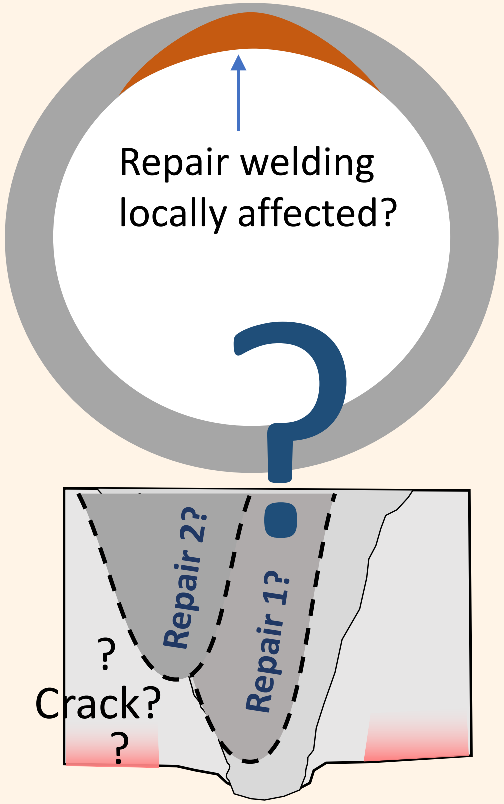
- **Summary of Ohi-3 spray line crack**
  - Observation results
  - Hardness measurement
  - Mock-up test to understand the phenomenon
  - Inspection as a countermeasure

- **Ongoing Study project**
  - Scheme of study project
  - New finding for crack initiation
  - Assumed mechanism

- **Crack growth evaluation**
  - Evaluation method using MRP-458 CGR curve
  - Reproduction of Ohi-3 using MRP-458
  - Influence of diameters

- **What was the differences of French and Japan?**

Differences between Japan and France

	Japan (Ohi-3)	**France 1 (Safety injection line)**	**France 2 (Recently found)**
Assumed Influenced Factor	High hardness caused by atypical welding?	Thermal stratification or vortex flow caused atypical stress?	Locally twice repair welding?
Estimated crack geometry (not confirmed)	4B piping Shallow and Local  Crack is located on the **boundary**	10B~12B piping  Cracks are started from the **surface-damaged layer** (estimation)	Deep (85%) but Local 

These schematic images are estimated from published information.

Summary

- **Summary of Ohi-3 spray line crack**
  - A circumferential IG crack was found on the spray line of Ohi-3 during ISI
  - 4.4 mm depth crack seems propagated in heat affected zone as IGSCC.
  - High heat input and Nozzle shape is considered the primary factor of casing high hardness.

- **Ongoing Study project**
  - ATENA (Atomic energy association) launched five years study project.
  - Microstructurally small cracks (MSC) were observed around the crack initiation site. Due to the filled with corrosion products, the MSC is thought to be arrested for a long time.

- **Crack growth evaluation**
  - EPRI MRP-458 provides a reasonable crack growth rate.
  - The effect of pipe diameter, 4B which is Ohi-3 condition, was assessed as having the highest risk.

- **What was the differences of French and Japan?**
  - Differences were found in the initiation site and the influence of residual stresses.

Thank you for your attention.

This is personal, but I will be transferred to INSS (a subsidiary company of Kansai) as a researcher next week.
My current contact will be closed soon please contact me as follows.
E-mail: terachi@inss.co.jp

Coffee break

SCC Growth Rate Behavior of Stainless Steels in PWR Primary Water Environments

Dr Do Jun SHIM

Technical Executive
EPRI

SCC Growth Rate Behavior of Stainless Steels in PWR Primary Water Environments

Do Jun (DJ) Shim
Technical Executive, Materials Reliability Program

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

Contents

- Background
- SCC Growth Rates in Stainless Steels in PWR Environments
  – MRP-458
- Ongoing Activities
- Summary

Background

Background

- Stress corrosion cracking (SCC) in austenitic stainless steels has been observed in PWR primary water environments
  - MRP-236, Revision 1: Stress Corrosion Cracking of Stainless Steel Components in Primary Water Circuit Environments of Pressurized Water Reactors (2017)
  - MRP-352: Materials Reliability Program: Assessment of the Current Status and Completeness of Work on Inner and Outer Diameter Stress Corrosion Cracking of Austenitic Stainless Steels in PWR Plants (2013)

Background (cont'd)

- In 2020, SCC observed near the pipe-to-nozzle weld in the pressurizer spray line of Ohi Unit 3
  - Cracking due to high heat input during welding process and potential hardness increase due to weld repair
- In 2021, indications were observed in safety injection piping of EDF plants
  - Defects mainly around welded joints
  - High level of hardness detected in the vicinity of the root pass
  - Propagation of crack only in the part of the thickness of the pipe with high hardness

Crack propagation along grain boundary in heat affected zone of the weld

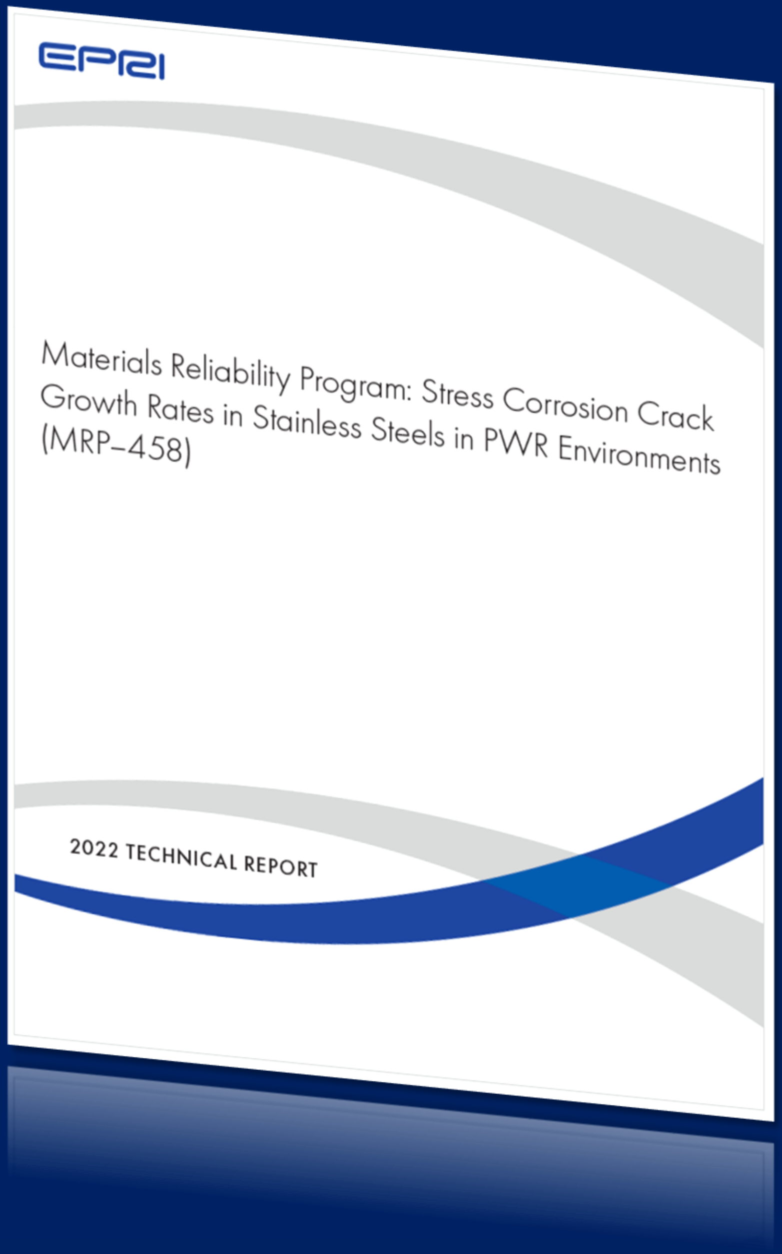
MRP-458

SCC Growth Rates in Stainless Steels in PWR Environments

MRP-458 (3002020451)

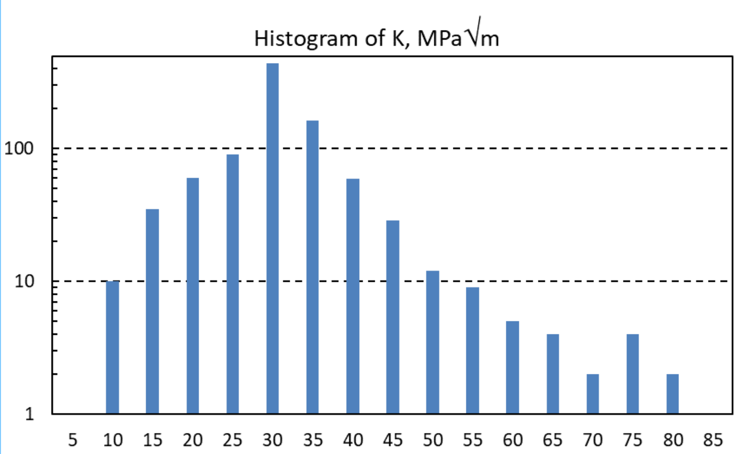
- Stress Corrosion Crack Growth Rates in Stainless Steels in PWR Environments
  - Published in August 2022
  - Includes data used in developing disposition curves
  - Publicly available on EPRI website

Public Website
www.epri.com



Database Development

- Several primary sources of data were surveyed for relevant SCC growth rate data in light water reactor environments from the following sources:
  - Proceedings of the 19 international conferences spanning 40 years
  - Publications from key international laboratories that have contributed significantly to the international efforts in this area
  - Additional data provided by the Institute of Nuclear Safety System (INSS) in Japan
  - Additional published and unpublished data by principal investigator (P. Andresen)



Overall PWR SCC Database

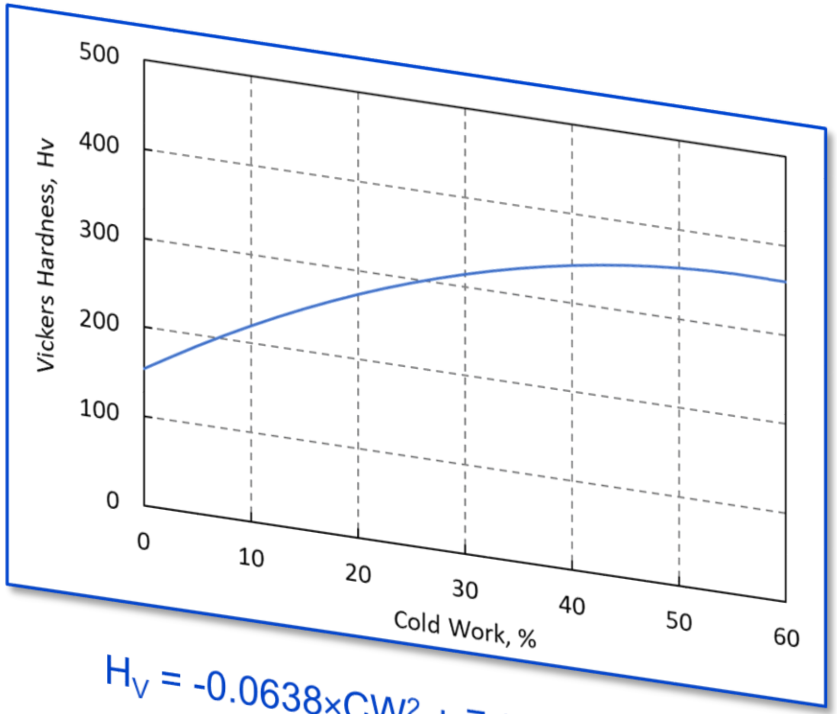
With highlighting for: PWR Mainstream Data in Column A (265 data)

PWR Representative HAZ Data in Column F (33 data)

Index	Lab	CT / ID	Material	Heat	Condition	%CW	ΔTime h	Δa mm	CGR mm/s	K MPa√m	Temp. °C	O_2 ppm	H_2 cc/kg	B ppm	Li ppm	SO4 ppb	Cl ppb	Score 1 to 5
1	Lucideon	L028	316NG			0	0	0.002	6.20E-09	40.0	325	2	0	1200	2	0	0	3
2	Lucideon	L028	316NG			0	948	0.012	1.00E-08	40.0	325	2	0	1200	2	0	0	2
3	Lucideon	L028	316NG			0	873	0.000	1.00E-10	40.0	325	2	0	1200	2	0	100	3
4	Lucideon	L028	316NG			0	60	0.074	2.90E-07	40.0	325	2	0	1200	2	0	100	1
5	Lucideon	L028	316NG			0	1182	0.998	2.30E-07	40.0	325	0	30	1200	2	0	0	1
6	Lucideon	L029	316NG			30	184	1.427	2.30E-06	40.0	325	0	30	1200	2	0	0	1
7	Lucideon	L029	316NG			30	249	1.744	2.00E-06	30.0	325	0	30	1200	2	0	0	1
8	Lucideon	L029	316NG			30	145	1.641	3.50E-06	30.0	325	2	0	1200	2	0	0	1
9	Lucideon	L029	316NG			30	67	0.180	6.40E-07	20.0	325	2	0	1200	2	0	0	1
10	Lucideon	L029	316NG			30	193	1.110	1.90E-06	20.0	325	0	30	1200	2	0	0	1
11	Lucideon	L030	316NG			20	184	0.635	1.10E-06	40.0	325	0	30	1200	2	0	0	1
12	Lucideon	L030	316NG			20	249	0.168	2.00E-07	30.0	325	0	30	1200	2	0	0	1

Database Development (cont'd)

- Database was ultimately comprised of 924 data
- Total of 34 data characteristics were recorded (when available) for each data
  - Data source
  - Material/heat and condition
  - Level of cold work (converted to hardness)
  - Specimen size and test condition (K, temperature, water chemistry, etc.)
  - Crack growth rate
  - Etc.



$H_V = -0.0638 \times CW^2 + 7.268 \times CW + 153.5$

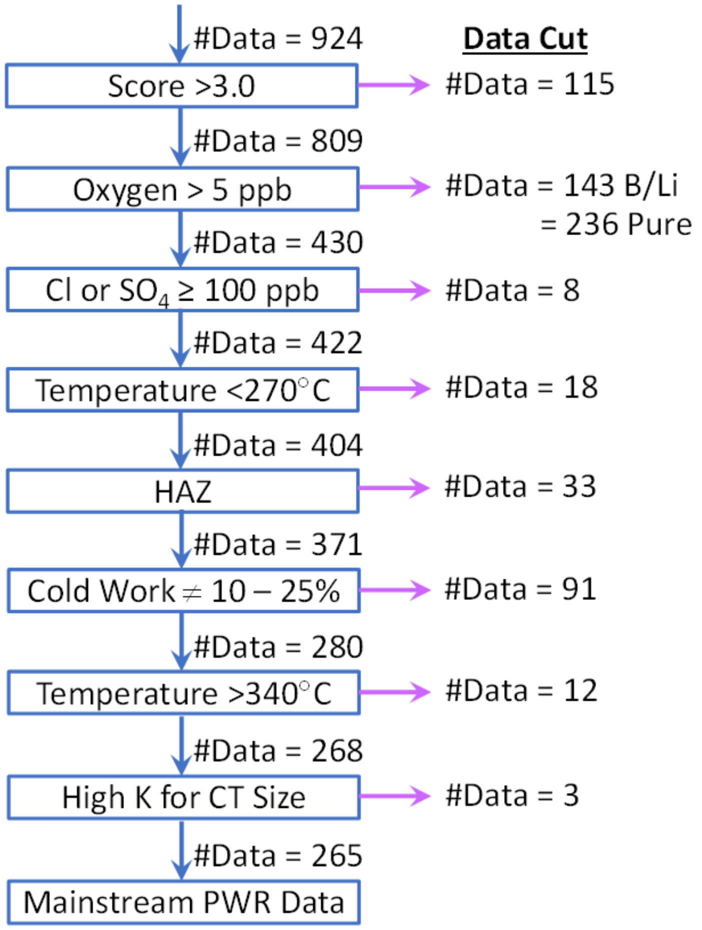
Database Analysis and Dependencies

- Many different multiple linear regression analyses were performed for subsets of the database
- Dependencies considered for disposition curve
  - A power law dependency for stress intensity factor (K)
  - A power law dependency on Vickers hardness (H_v), which was used in preference to cold work (or residual strain measurements by EBSD)
  - An Arrhenius dependency for temperature, expressed as exp(-Q/RT)

$$da/dt = C \times H_v^{\alpha} \times K^{\beta} \times \exp(-Q/RT)$$

Mainstream PWR Data

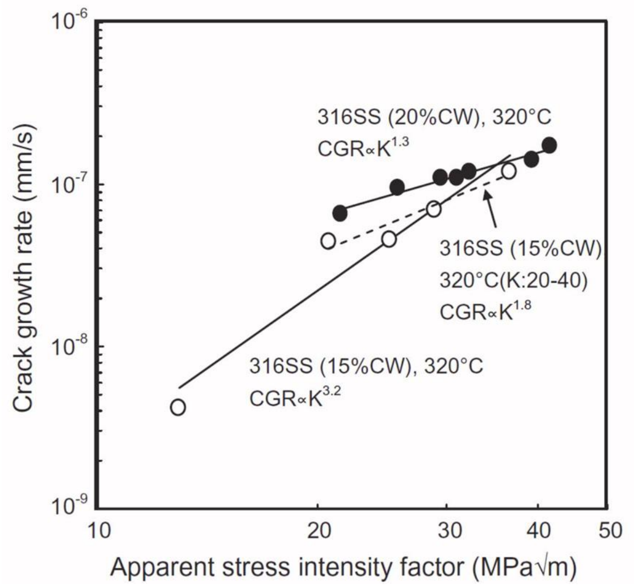
- **HAZ data were set aside** because, while highly relevant to SCC in plant components, little is known about the laboratory testing details
- **Cold work was limited to 10 – 25%** because this is the most relevant range in plant components and welds.
  - Cracks in stainless steel in PWRs typically occur in the HAZ close to the weld fusion line where the weld residual strains are highest
  - Also, in this relevant range, the effect of cold work was well behaved, while at higher levels of cold work, the power law dependency varied
- **Temperature was limited to 270 – 340°C** because this is the most plant-relevant temperature range
  - The data exhibited a different dependency below 270 °C
  - Above 340°C, some data showed a peak in the growth rate (and subsequent lower growth rates with increasing temperature) while other data did not

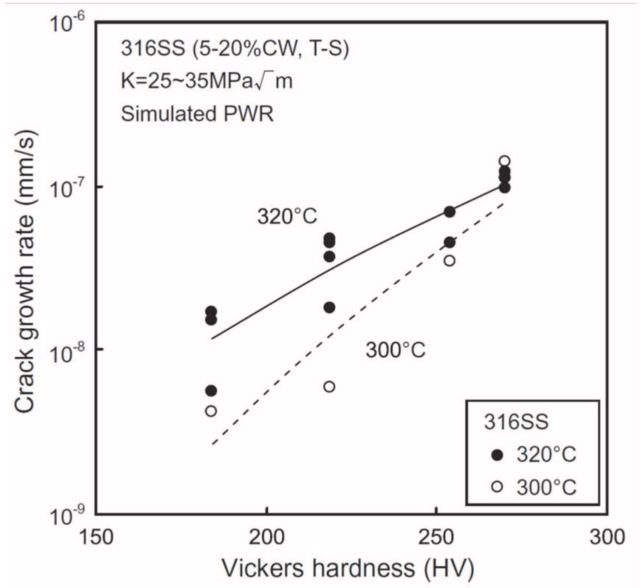


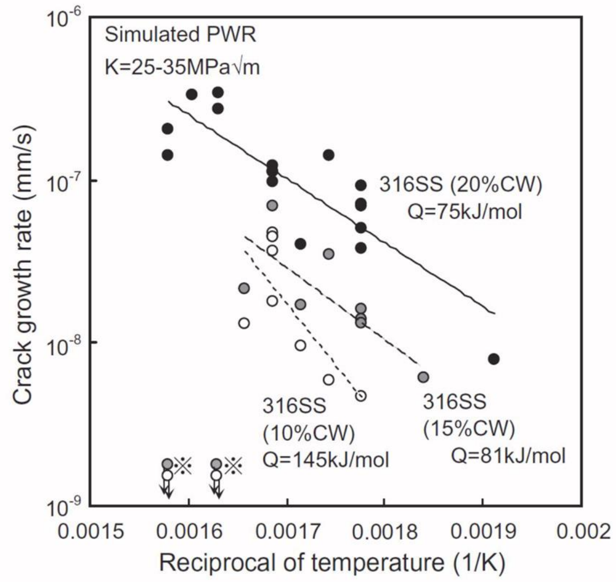
Dependencies from As-Identical-As-Possible (AIAP) Data

- Collection of AIAP data analyzed
  - 30 AIAP data for K dependency
  - 8 AIAP data for hardness dependency
  - 13 AIAP data for temperature dependency

	K exponent	H_v exponent	Q (J/mole)
Median value	2.18	4.53	98500



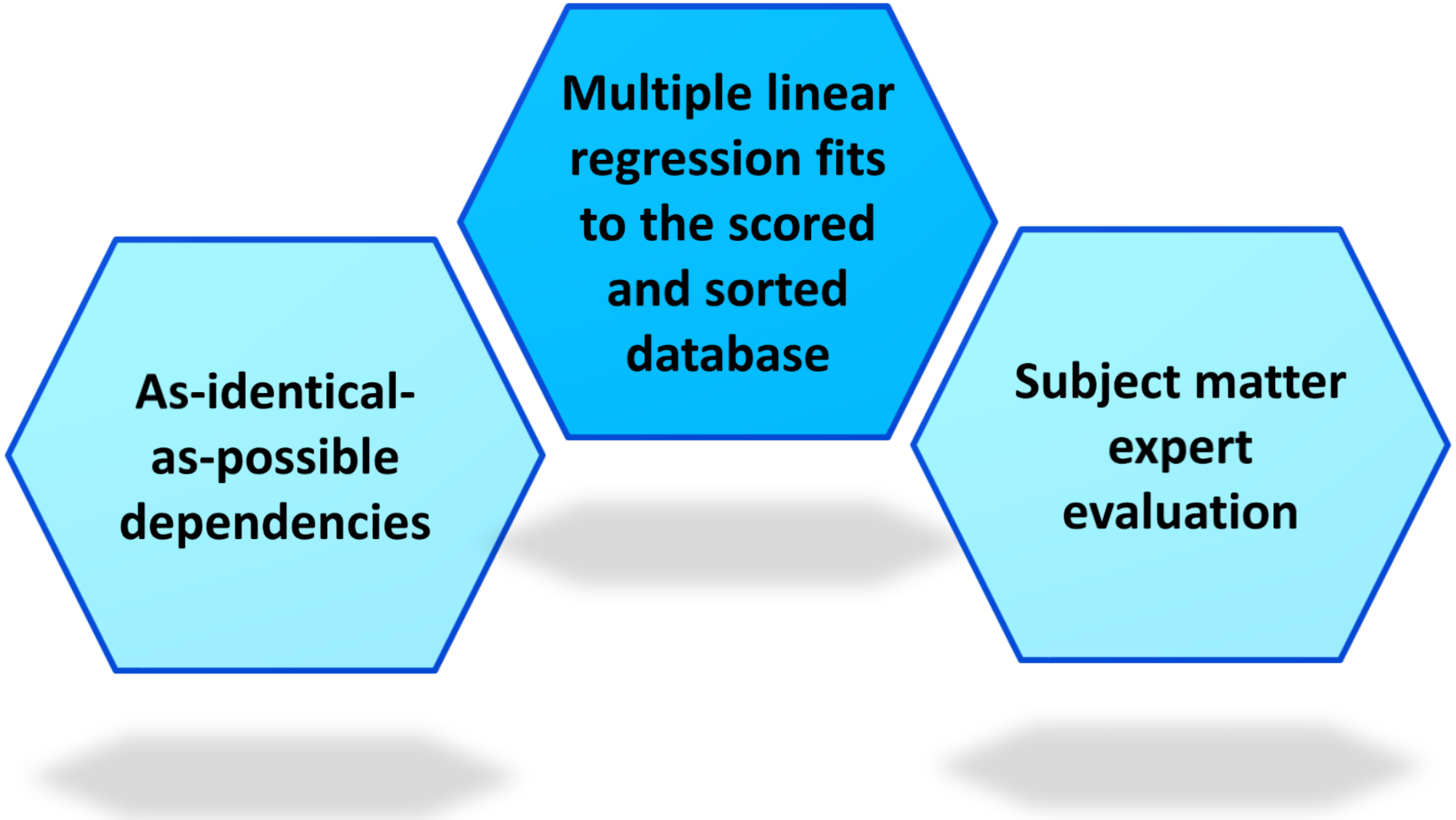




Terachi et al. "SCC Growth Behaviors of Austenitic Stainless Steels in Simulated PWR Primary Water", J Nuclear Materials 426 (2012) 59–70

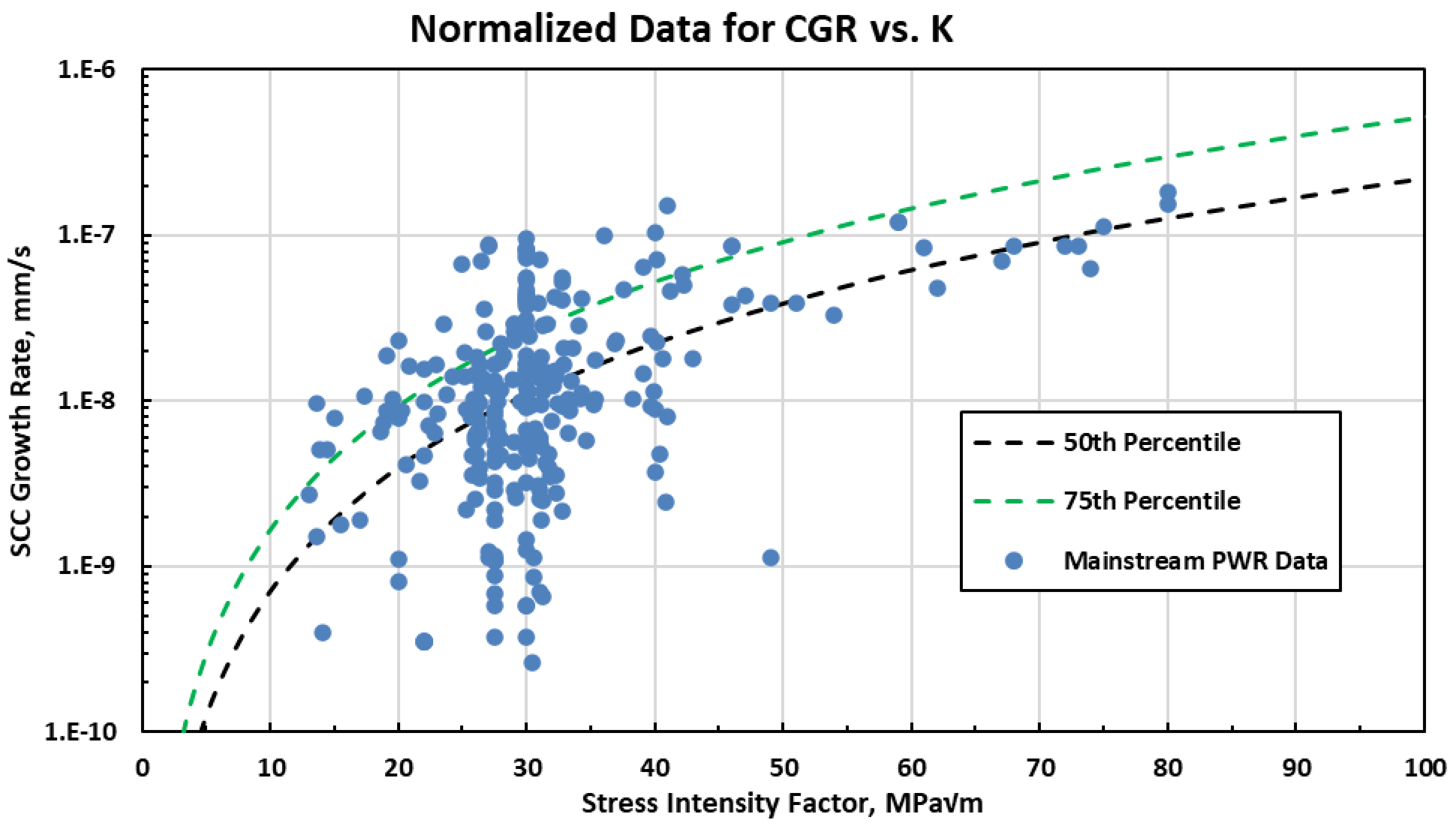
Disposition Curve

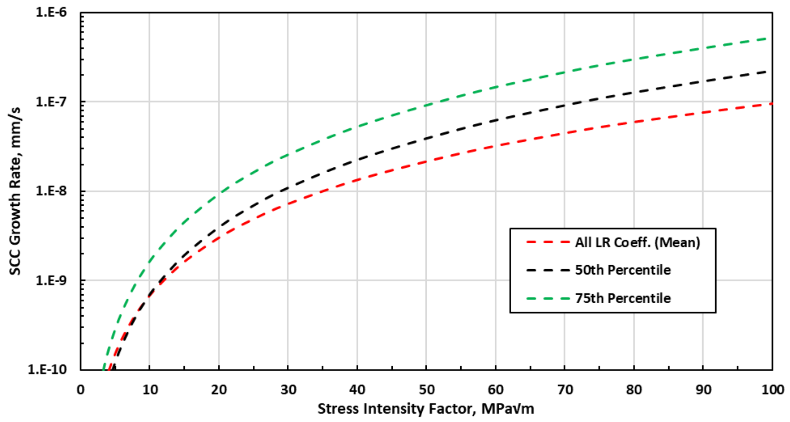
- Dependencies and coefficients were chosen over the range of K, H_v, and temperature based on:



Disposition Curve (cont'd)

- Crack growth rate vs. K





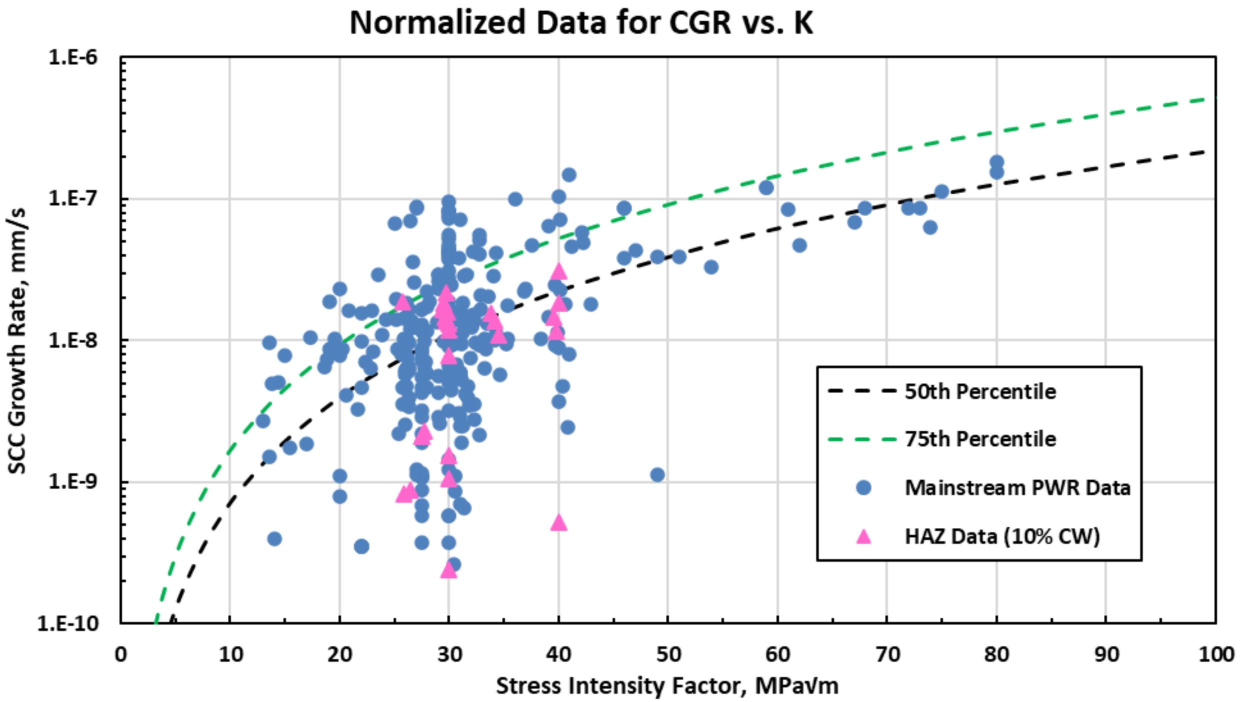
$$da/dt = C \times H_v^{\alpha} \times K^{\beta} \times \exp(-Q/RT)$$

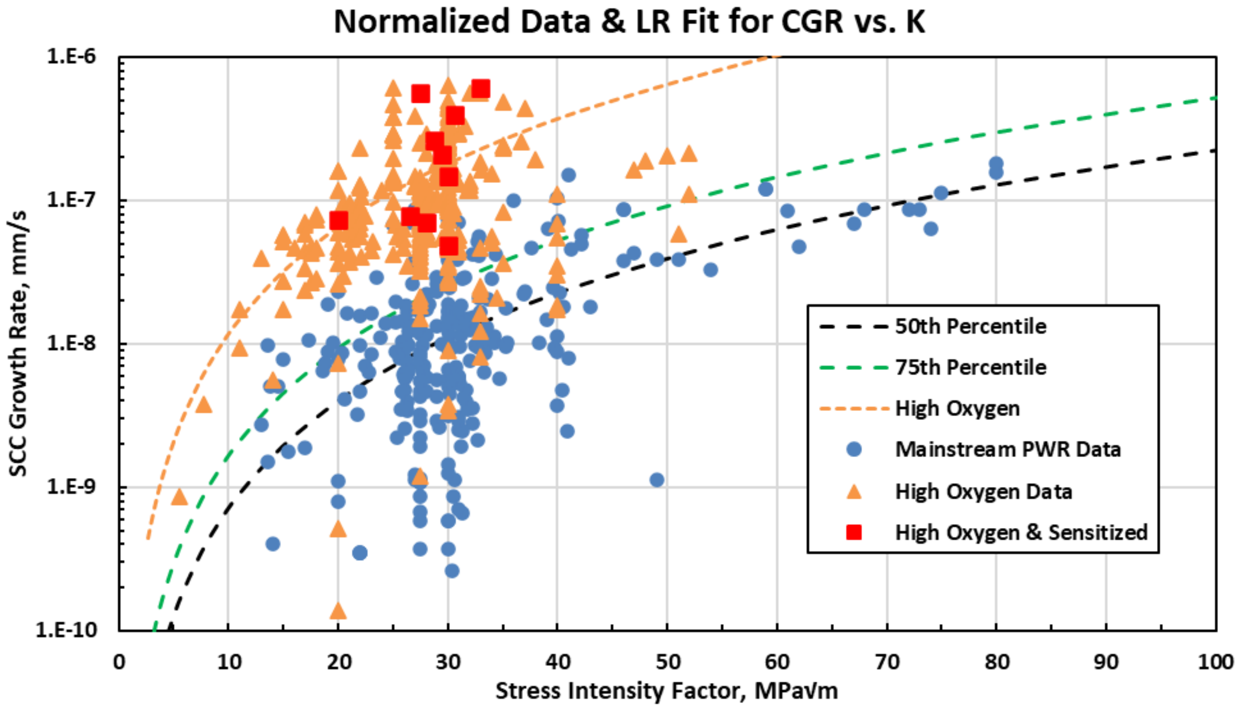
Data Evaluated	C	β	α	Q (J/mole)
All LR Coeff. (Mean)	1.32×10^{-21}	2.15	7.60	88,750
50th Percentile Curve (including AIAP)	1.50×10^{-18}	2.50	6.00	85,000
75th Percentile Curve (including AIAP)	3.19×10^{-18}	2.50	6.00	85,000

75th percentile coefficient (C) is 2.13 time above 50th percentile

Disposition Curve (cont'd)

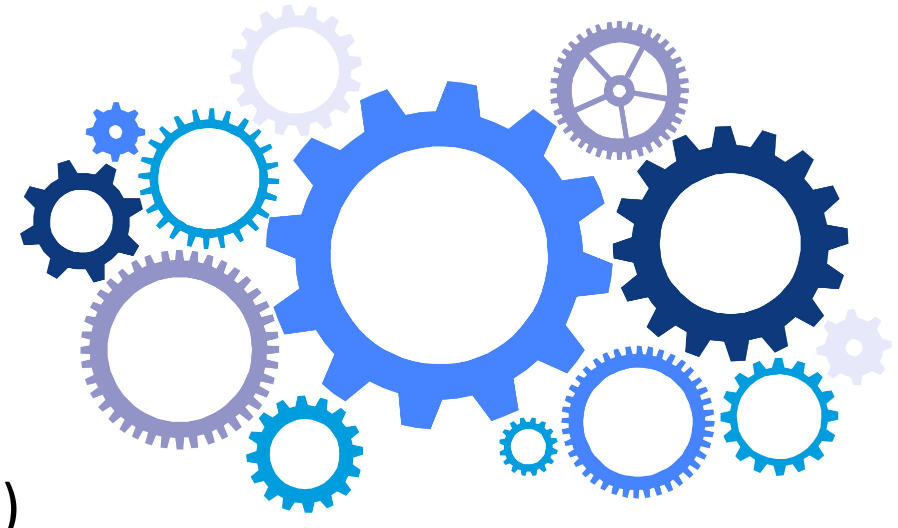
- HAZ can be dispositioned using the 10% CW base metal data
  - Using 10%CW (or H_v=220) in the disposition curve equations
- For high oxygen data (≥100 ppb) a factor of 7 can be applied to the 75th percentile curve





Rationale for 75th Percentile Curve

- There have been several applications where the 75th percentile was used instead of the mean (i.e., the 50th percentile)
  - This approach provides a more conservative estimate of the crack growth rates and is typically selected for use when data scatter is large
- Cracking in service components is more likely to occur in materials with faster crack growth rates
  - 75th percentile may be selected in such cases to represent the mean of the upper half of the crack growth rate variability distribution



- Examples of such applications;
  - PWSCC growth of Alloy 600 (MRP-55, R1 / MRP-420, R1) and Alloy 82/182/132 welds (MRP-115 / MRP-420, R1)

Ongoing Activities

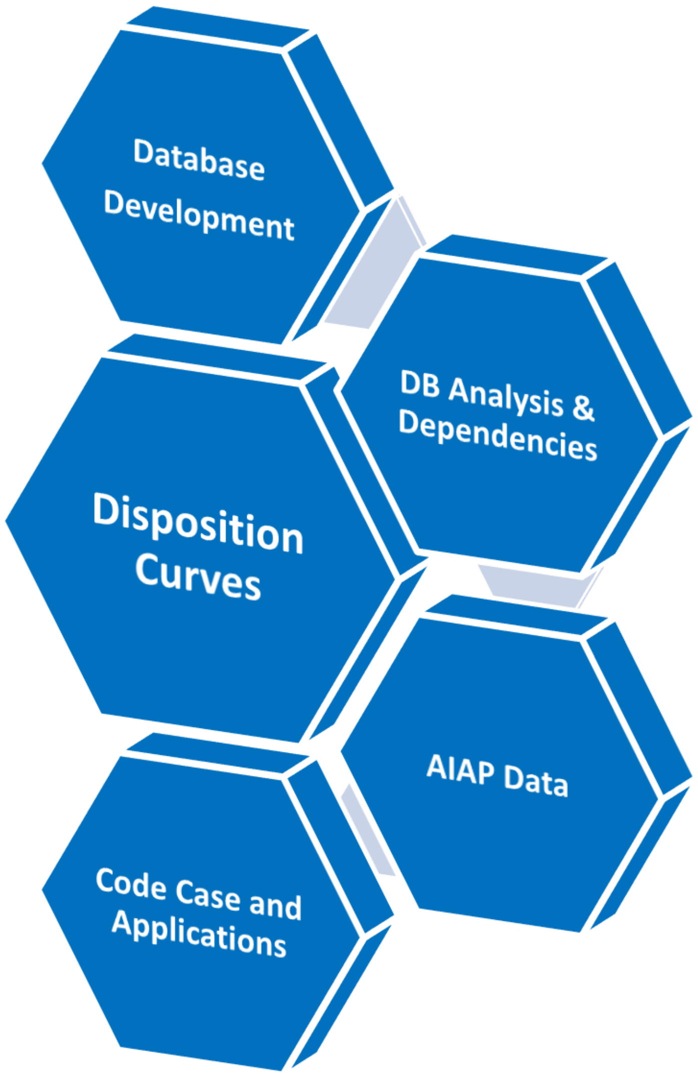
Ongoing Activities Related to MRP-458

- Comparison with other CGR models
  - Models from Japan, France, US
- ASME Code Section XI (WGFERC)
  - Draft Code Case
- Application in flaw tolerance evaluations
  - Support PWROG/MRP Focus Group studies
  - Safety and applicability assessments

Summary

Summary

- Several primary sources of data were surveyed to develop a database of SCC growth rate in light water reactor environments
- Database analysis was performed to guide the selection of key dependency parameters and final sorting/grouping of data
- In addition, dependencies from As-Identical-As-Possible (AIAP) data were investigated
- Disposition curves (50^{th} and 75^{th} percentiles) were developed in MRP-458
- Ongoing activates include model comparison, Code Case development, and applications to flaw tolerance evaluations



Together…Shaping the Future of Energy®

NEA
NUCLEAR ENERGY AGENCY

NRA JAPAN

Fatigue Phenomena Experience

Paul R. Donavin, P.E.

Becht Engineering

Fatigue Phenomena Experience

Paul R. Donavin, P.E.

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

Fatigue Monitoring

- Fatigue monitoring ranges from manually counting operational transients to automated integrated analysis with the plant computer
- The manual system counts the number of heat-ups and cool downs, plant trips, and other defined transients
- The automated system takes inputs from the plant computer and recalculates the cumulative fatigue usage daily.
- Most plants use the automated system, but maintenance is variable

Fatigue Monitoring Experience

- FatiguePro

- <u>PROs</u>
  - Makes counting cycles consistent
  - Identifies transients which may not always be readily apparent (i.e. charging nozzle transients)
  - Provides good traceability and documentation
  - Use of the fatigue calculations in the software is beneficial for high usage locations which experience a variety of transients which are typically less severe than design.

- <u>CONs</u>
  - Knowledge required to run the software.
  - Costs associated with developing and maintaining the plant specific versions and the software required to obtain the input data from the plant computer.
  - May be cumbersome to clean up input data.

Fatigue Monitoring Experience

- Westems

- Monitoring system identified high rate of safety injection line transients due to SI valve testing in hot conditions.
- Charging nozzle monitoring has typically shown higher than design cyclic rates, but lower transient severity
- Steam Generator main feedwater nozzle transient issues
- PZR lower head monitoring

ASME
SETTING THE STANDARD

Fatigue Monitoring Experience

- Plant Performance
- As plant performance improves, the fatigue usage projections decrease.
- Certain plant transients are not as severe as originally assumed
- Reported annually or longer intervals
- Past Operating Practices
- Before transitioning to higher mode, all potential holds should be cleared

Fatigue Monitoring Experience

- Unexpected Transients
- Using steam in pressurizer during extended outages caused increased fatigue usage in surge line
- Rapid changes in Pressurizer level causes unexpected indications
- Support for license renewal/life extension

ASME Future Actions

- Revise Appendix I with material specific fatigue curves (In cooperation with JSME)
  - Individual curves based on limited material groups
  - Use Smith-Watson-Topper vs Modified Goodman
- Total Fatigue Life Approach
- Use modern cycle counting methodology
  - For Example, Rainflow analysis
- Use strain-based methodology
  - ASME Record 16-2736
  - CORDEL Fatigue Proposal

Material Specific Fatigue Curves

- Alternative design fatigue curves for Section III Appendices, Mandatory Appendix I, Division 1
  - Methodology for developing design fatigue curves
  - Key attributes uses tensile strength, limited materials for each curve set, 95% confidence versus 2 and 12 data factors

Total Fatigue Life Approach

- Provides fatigue life and re-examination interval
- Based on fatigue crack growth
- Similar approach to API-579/ASME FSS-1

Modern Cycle Counting Methodology

- Rainflow Analysis
- Time of Flight

Use Strain-based Methodology

- Define strain-based definition for used elastic plastic analysis in an Appendix XIII fatigue assessment.
  - Changes to Appendix XIII to include a method of demonstrating fatigue acceptability through the use of Plastic Analysis.
  - ASME Record 16-2736

CORDEL Fatigue Proposal

- Cyclic Plasticity
- Cycle Counting
- Fatigue Life Analysis LWR coolant environment, austenitic stainless steel & nickel based alloys
- Crack initiation (FCI)
- Fatigue crack growth (FCG)

Fatigue Phenomena Experience

- **Any Questions?**

Integrity Assessment of Reactor Pressure Vessel Against Irradiation Embrittlement in Japan

Dr Tokatoshi HIROTA

Mitsubishi Heavy Industries, LTD. Kobe, Japan

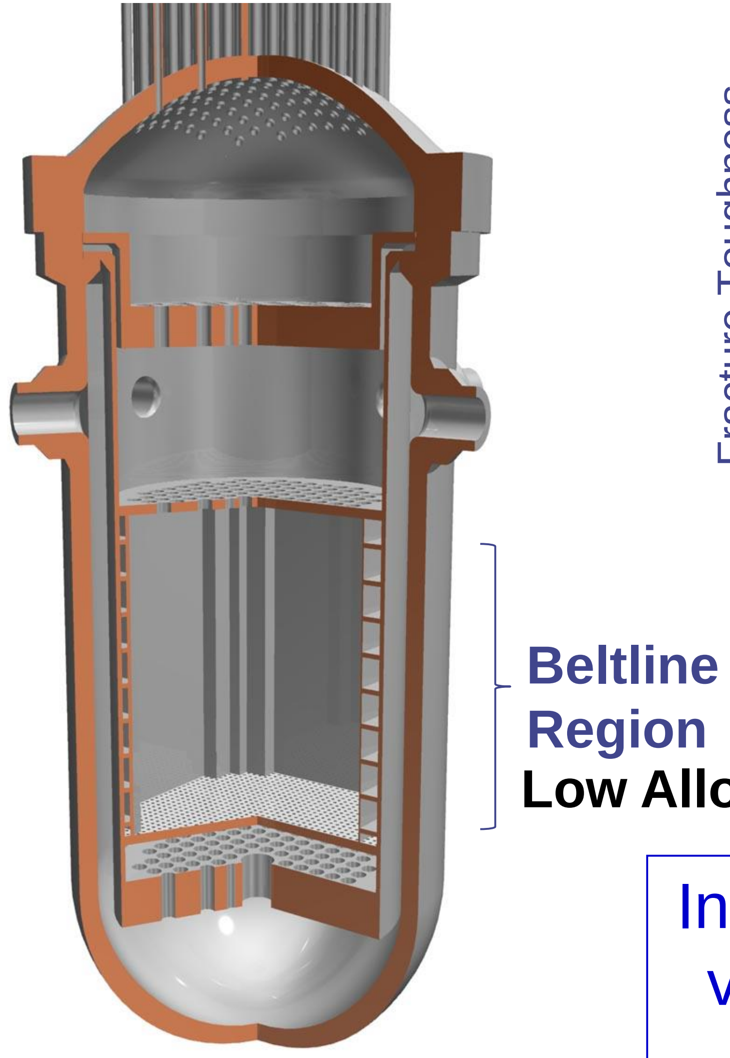
Integrity Assessment of Reactor Pressure Vessel Against Irradiation Embrittlement in Japan

Takatoshi Hirota
Mitsubishi Heavy Industries, LTD. Kobe, Japan

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

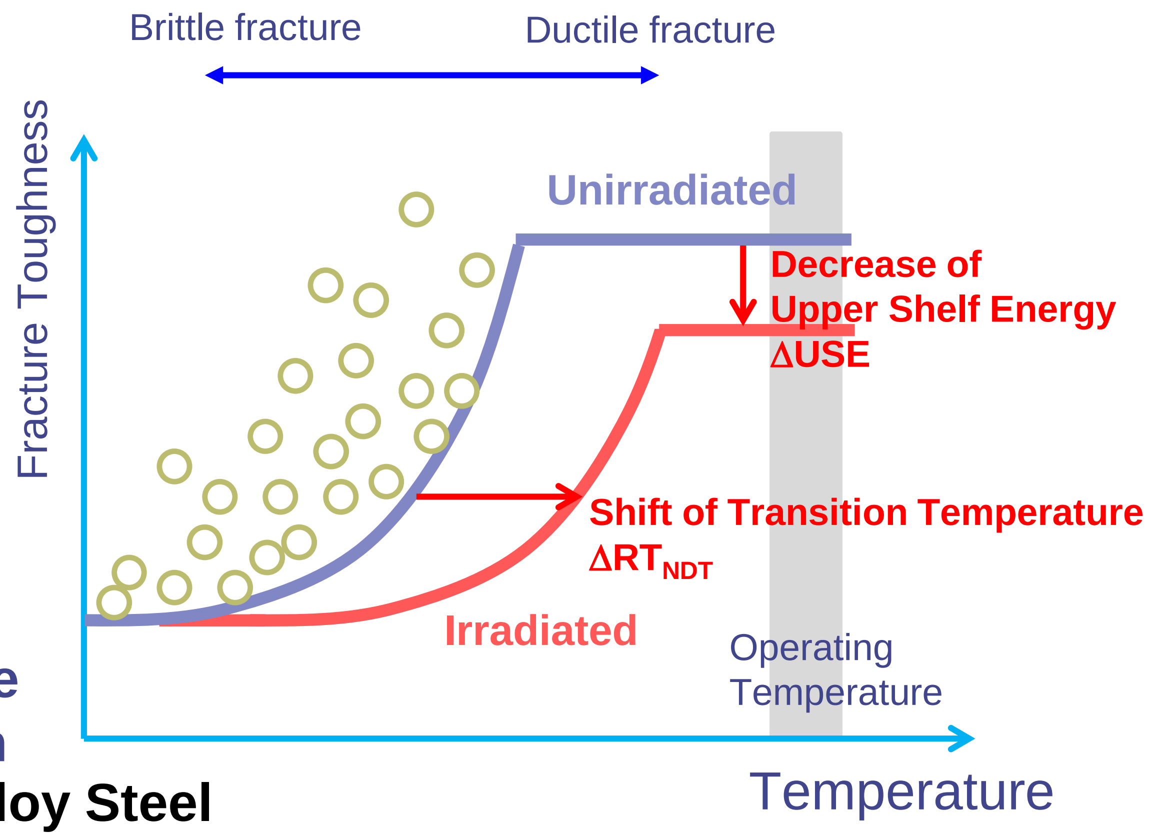
CONTENTS

1. Irradiation Embrittlement of Reactor Pressure Vessel



Low Alloy Steel

Reactor Pressure Vessel (RPV)



Integrity assessment of reactor pressure vessel must be performed considering irradiation embrittlement.

2. SCHEME OF INTEGRITY ASSESSMENT

Integrity assessment rules of RPV against irradiation embrittlement are specified in industrial code (the Japan Electric Association code).

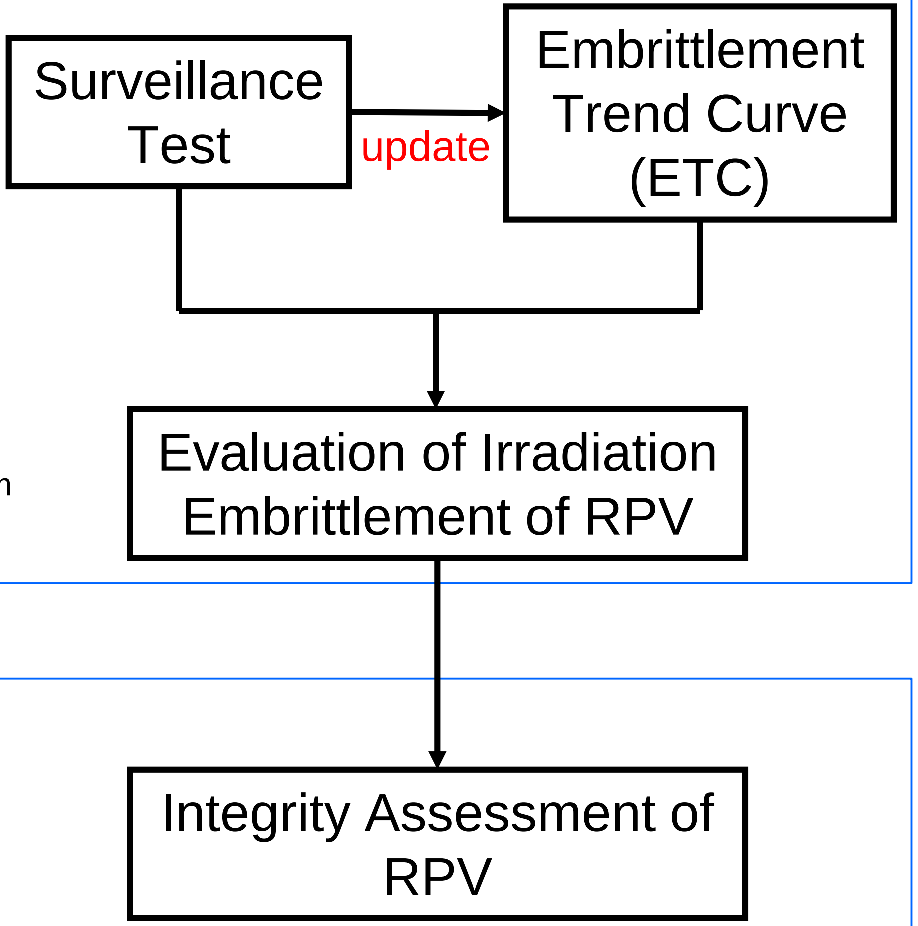
JEAC 4201

- Irradiation embrittlement data of RPV material is obtained by surveillance tests.
- Irradiation embrittlement of RPV is evaluated by ETC and surveillance test data.
- ETC is continuously updated by surveillance data.

Surveillance test : Mechanical test specimens taken from RPV material are installed in RPV and periodically withdrawn and tested throughout plant operation.

JEAC4206

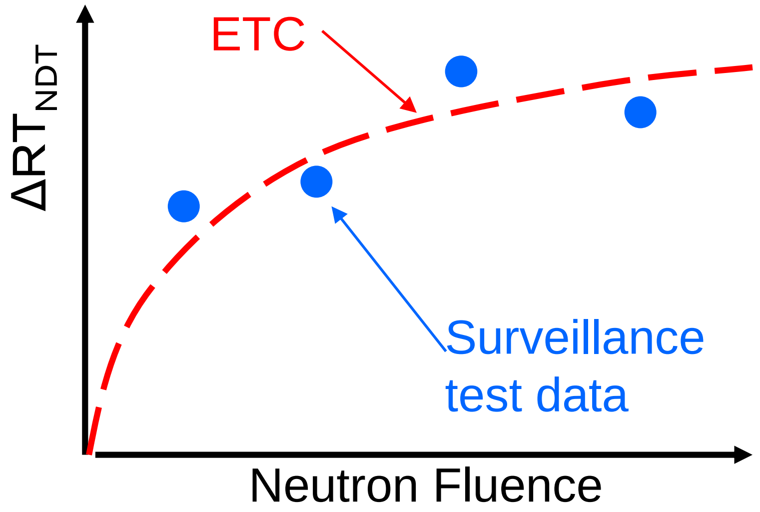
- Integrity of RPV is evaluated considering irradiation embrittlement.



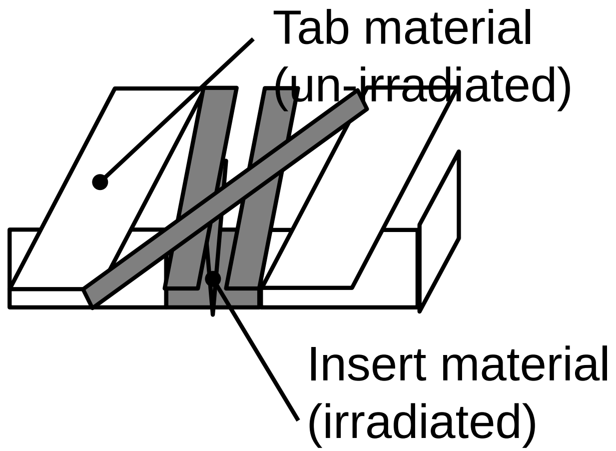
3. CONTENTS of JEAC 4201

JEAC4201: Method of Surveillance Tests for Structural Materials of Nuclear Reactors

- JEAC4201 specifies **surveillance program** and **ETC for ΔRT_{NDT} and ΔUSE**.
- 2007 edition of JEAC4201 (JEAC4201-2007) incorporated **surveillance program for long-term operating period** and **reconstitution of surveillance specimen** etc., in order to accommodate operation periods of more than 40 years.



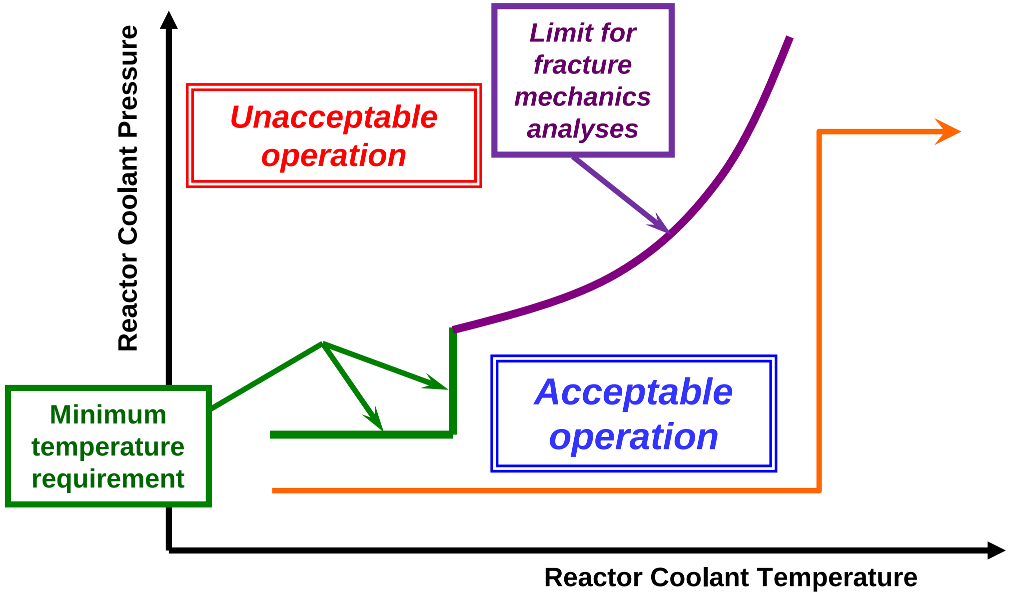
ETC of ΔRT_{NDT}



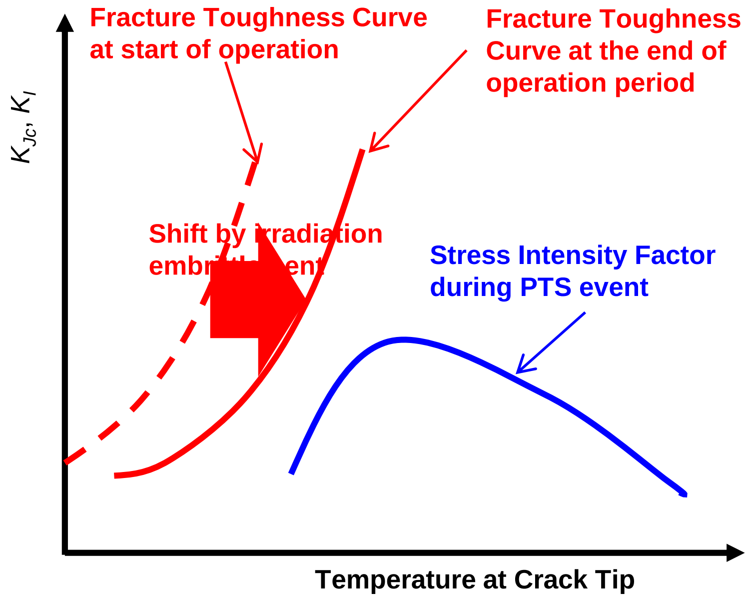
Reconstitution of Charpy specimen

JEAC4206: Verification Method of Fracture Toughness for In-service Reactor Pressure Vessel

- JEAC4206 specifies **fracture toughness requirements**.
  - ✓ Brittle fracture
    - ➢ Reactor coolant pressure - temperature limit for heat up and cool down
    - ➢ Integrity assessment for Pressurized Thermal Shock (PTS) events
  - ✓ Ductile fracture
    - ➢ Screening criteria of USE (> 68J)
    - ➢ Integrity assessment in case of USE <68 J



Reactor Coolant Pressure – Temperature Limit for Heat up and Cool down



Integrity Assessment for PTS Events

5. ETC IN JEAC4201

- The Japanese Embrittlement Trend Curve (ETC) for ΔRT_{NDT} was first introduced in **JEAC4201-1991**.

$$\boxed{\Delta RT_{\mathrm{NDT}} = (\text{Chemistry Factor}) \times (\text{Fluence Factor})}$$

- After the accumulation of surveillance data as well as the progress made on understanding of the embrittlement mechanism, a new ETC was developed and incorporated in **JEAC 4201-2007**.
- This ETC is a **mechanism-guided ETC** assuming that the embrittlement is caused by the hardening of materials due to the **formation of solute atom clusters (ΔT_{SC}) and matrix damage (ΔT_{MD})**.

$$\boxed{\Delta RT_{\mathrm{NDT}} = \sqrt{\Delta T_{SC}^{\;2} + \Delta T_{MD}^{\;2}}}$$

- Recently, same as in Japan, the **US** also employed the mechanism-guided ETCs such as **in ASTM E900***) **and 10CFR50.61a****).

*) ASTM E900-21," Standard Guide for predicting radiation-induced transition temperature shift in reactor vessel materials", (2021).

**) U.S. Nuclear Regulatory Commission, "Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," 10CFR50.61a, (2010).

5. ETC IN JEAC4201

- Because ΔRT_{NDT} of some materials were underestimated by the ETC, **coefficients of the ETC were recalibrated**, while retaining the same equation forms as prescribed in **JEAC4201-2007 (2013 addendum)**.
- JEAC4201-2007 (2013 addendum) was **endorsed by Japanese Regulator (Nuclear Regulation Authority: NRA)**.
- However, **some comments** (review by related organization (experts), appropriateness of weighting data in calibration, tendency of deviation of surveillance data from ETC, etc.) **were provided** based on technical evaluation by NRA.

6. REVISION OF ETC IN JEAC4201

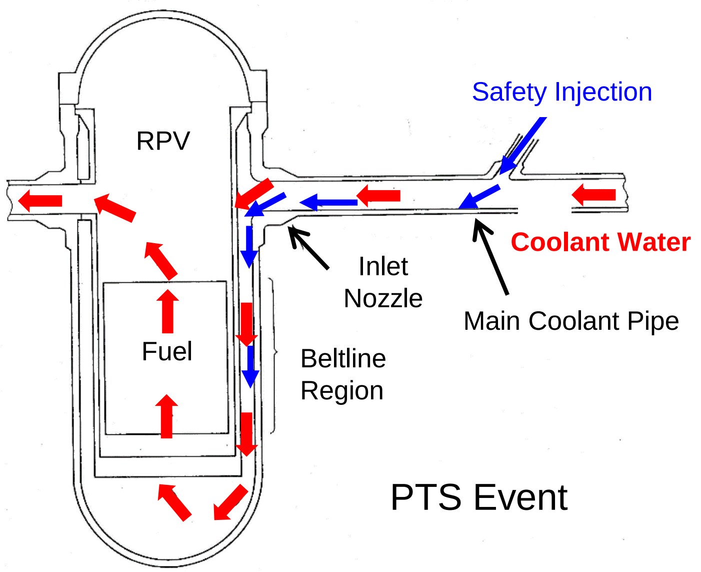
MITSUBISHI HEAVY INDUSTRIES, LTD.

- A new ETC was discussed in **Subcommittee on Validation of Irradiation Embrittlement Trend Curve for Reactor Pressure Vessel Steels (IET)** in the Japan Welding Engineering Society (2017 – 2019FY) with experts on this field.
  - ✓Research for latest knowledge about irradiation embrittlement
  - ✓Review of new ETC
  - ✓Confirmation of calibration process of coefficients in new ETC
- The new ETC was developed as a mechanism-guided ETC mainly by **revising the microstructure evolution model** using an irradiation database of APT (Atom Probe Tomography) data on RPV materials by Central Research Institute of Electric Power Industry (CRIEPI).[*)]
- The new ETC is **being discussed** for incorporation to a revision of JEAC 4201 with other revision items in the Japan Electric Association.

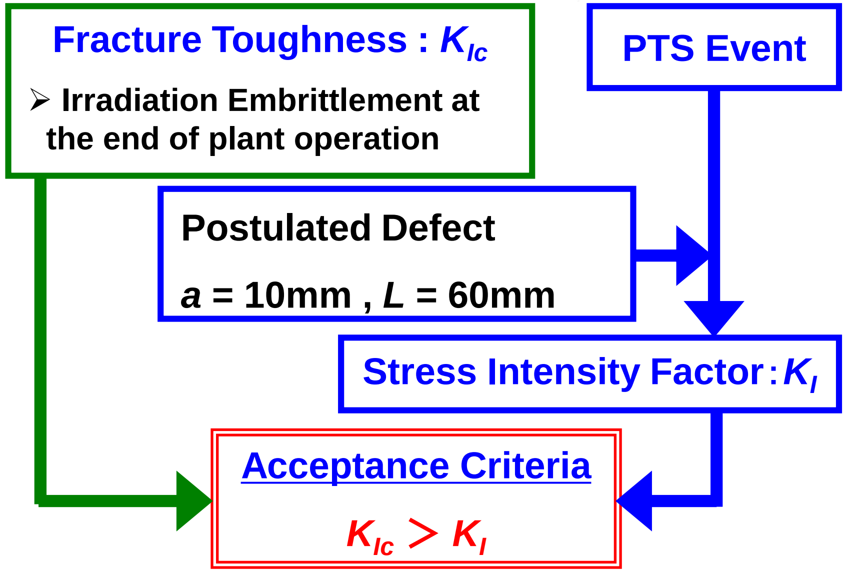
*) Hashimoto, Y., et al., "Development of new embrittlement trend curve based on Japanese surveillance and atom probe tomography data", Journal of Nuclear Materials, Vol. 553, 153007 (2021).

7. PTS ASSESSMENT PROCEDURE

- A **Pressurized Thermal Shock (PTS) event** is a phenomenon in a pressurized reactor pressure vessel in which **rapid cooling** of the vessel occurs due to injection of safety injection water, etc., associated with the activation of the Emergency Core Cooling System (ECCS) corresponding to service level C and D, generating **high tensile stress** on the vessel's inner surface that is superimposed on the membrane stress due **to internal pressure**.
- If the fracture toughness of the vessel decreases considerably due to neutron irradiation and rapid cooling, and if **defects such as cracks initially** existed near the inner surface, it is considered that if the internal tensile stress generated by the PTS event exceeds a certain limit, **cracks may develop and damage the reactor pressure vessel**.
- JEAC4206 specifies a method of integrity assessment based on **fracture mechanics** to prevent damage to the reactor pressure vessel in PTS events.

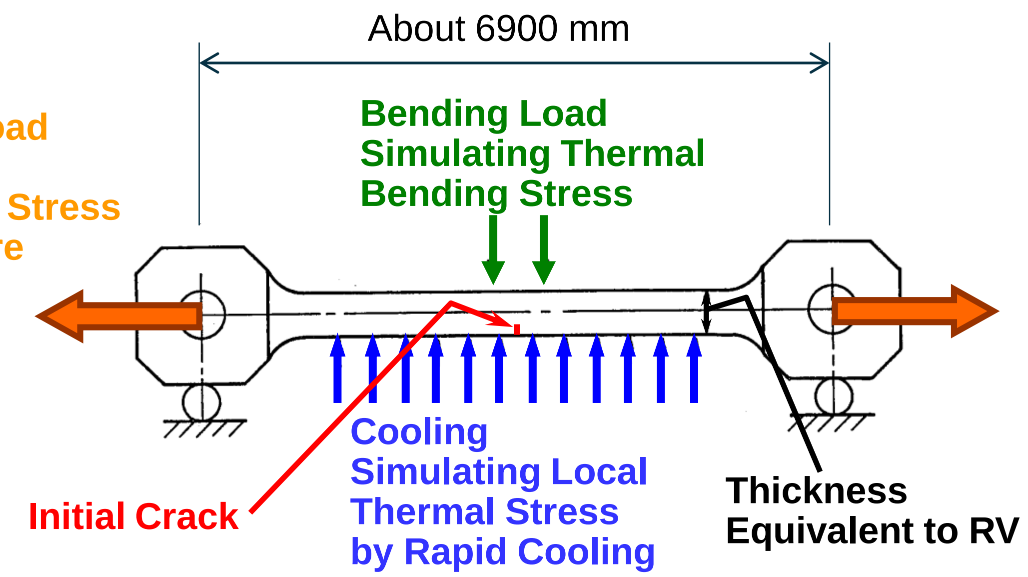


PTS Event



PTS Assessment Procedure

- Assessment procedure against PTS events was developed including **demonstration tests**, in **a Japanese national project** *) and incorporated in **JEAC 4206-1991**.
- In JEAC 4206-2016, significant revisions were made to the PTS assessment procedure **based on the latest knowledges**.
- However, JEAC 4206-2016 was **not endorsed** as a result of the technical evaluation by NRA, so **previous version, JEAC 4206-2007** is applied now.
- The main revisions in PTS assessment procedure of JEAC 4206-2016 are shown on the following pages.



Specimen in demonstration test of national project *)

*) Mishima, Y., et al., " PTS integrity study in Japan", International Journal of Pressure Vessel & Piping, 58 (1994) 91-101.

7. PTS ASSESSMENT PROCEDURE

Change of Calculation of Stress Intensity Factor

◆ JEAC4206-2007

Stress intensity factors, K_I are calculated for a surface defect at inner surface of RPV by **K_I solution for surface defect**.

◆ JEAC 4206-2016

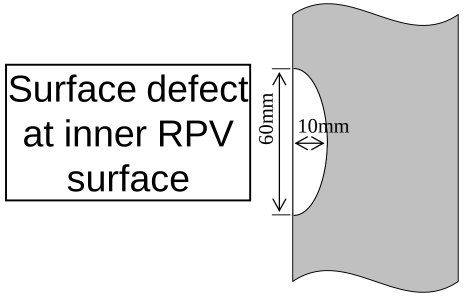
✓ **To accurately evaluate** based on the latest knowledge, a **subsurface defect beneath cladding** at RPV inner surface is postulated and stress intensity factors are calculated by **K_I solution considering cladding effects***) or by **FE analyses with FE model including defect and cladding**.

✓ Smaller defect sizes may be used, if its detectability can be ensured by non-destructive examination.

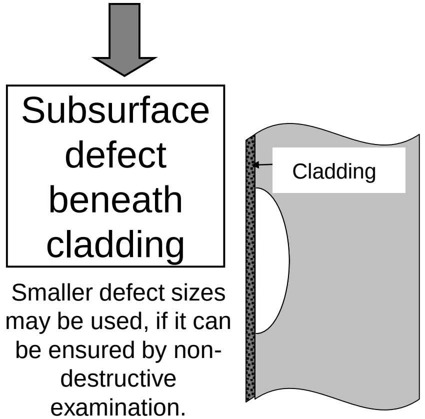
✓ In addition, **residual stress by clad and joint (in case of weld metal) welding** shall be considered.

◆ Issue raised in Technical evaluation by NRA

Material properties of the cladding used for calculation shall be clearly specified in JEAC4206.



JEAC4206-2007



JEAC 4206-2016

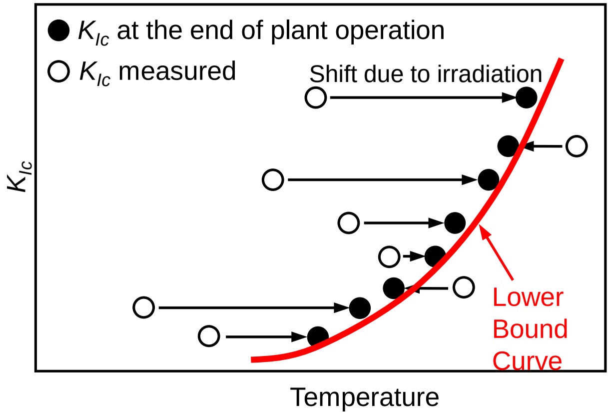
*) Marie, S., Chapuliot, S., "Improvement of the calculation of the stress intensity factors for underclad and through-clad defects in a reactor pressure vessel subjected to a pressurized thermal shock", Int. J. Pressure Vessels Piping, 86[8], 517-531 (2008).
Moinereau, D., Bezdikian, G., Faidy, C., "Methodology for the pressurized thermal shock evaluation: recent improvements in French RPV PTS assessment," Int. J. Pressure Vessels Piping, 78[2], 69-83 (2001).

MITSUBISHI HEAVY INDUSTRIES, LTD.

Change of Fracture toughness curve

- JEAC4206-2007

A lower bound curve of experimentally measured K_{Ic} data, which consider the shift of irradiation embrittlement from measured to the end of operation by JEAC4201 ETC is used.



JEAC4206-2007

- JEAC 4206-2016

✓ A **5% tolerance lower bound Master curve** is used as fracture toughness curve.

✓ In addition, based on **Japanese surveillance fracture toughness data**, 5% tolerance lower bound curve using **Charpy index temperature, T_{r30}**, was developed.

- Issue raised at technical evaluation by NRA

The fracture toughness Master curve have **some unresolved technical issues**, and it was considered **premature to judge their applicability**.

Master curve proposed by Wallin are commonly used for fracture toughness evaluation of ferrite steels. It can express not only the scatter inherent in fracture toughness quantitatively based on weakest link theory but also dependence on temperature in transition region for ferritic materials.

Master curve Equation

$$K_{Jc(0.0x)} = K_{min} + \left[ln\left(\frac{1}{1-0.0x}\right)\right]^{1/b}\{11+77exp[0.019(T-(T_o))]\}$$

b : Shape parameter of Weibull distribution (b=4)

T_o : Reference temperature of Master Curve (Determined by fracture toughness tests or **T_{r30} + ΔT_t (constants for product form)**)

0.0x : Cumulative probability level (0.0x=0.05)

K_{min} : K_{min}=20MPa√m

JEAC4206-2016

*) Yoshimoto, K, et. al., “Applicability of Fracture Toughness Curves Developed for Japanese Pressure Vessel Steels to Structural Integrity Evaluation,” Proceedings of ASME 2015 Pressure Vessel & Piping Division Conference, July 19-23, 2015, Boston, Massachusetts, USA, PVP2015-45275

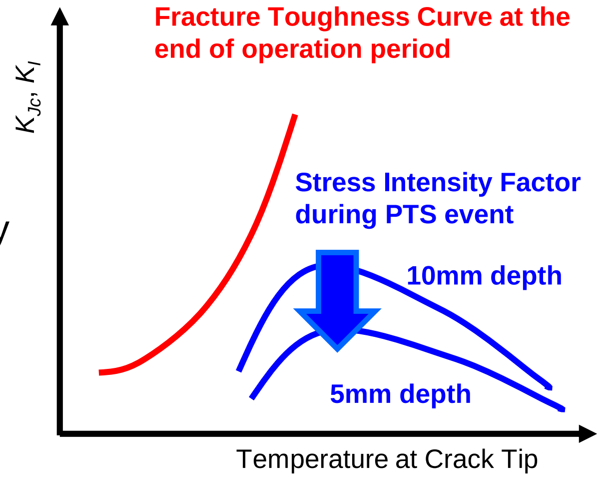
8. REVISION OF PTS ASSESSMENT PROCEDURE

- JEAC4206-2016 was not endorsed and some issues were raised in technical evaluation by NRA.
- Smaller postulated defect sizes ensured by non-destructive examination are considered as **almost appropriate** in technical evaluation by NRA.
- **Addendum of JEAC4206-2007** is being examined in order to reflect NRA's request to have a **more detailed procedure** about determination of smaller postulated defect sizes.
- The other issues raised in technical evaluation by NRA should be reflected in **next edition of JEAC4206**.
- Because there are some progresses in fracture evaluation procedure such as **probabilistic fracture mechanics (PFM)** and **plastic constraint effect**, they should be reflected in **JEAC 4206 in the future**.

1) Smaller defect size

- Postulated flaw size in JEAC4206-2007 was determined based on the knowledge at the time of **more than 30 years ago** about **non-destructive examination during manufacturing and in-service inspection** and **fatigue crack growths** for RPV beltline region.
- **No harmful defects have been confirmed** in RPV beltline region of Japanese actual plants during ultrasonic examinations (UT)
  - **In-service inspections** for weld joints of operating plants
  - **Special inspection** for entire beltline region including base metal required for the application of operation period extension.
- The **detectability of UT during in-service inspection was investigated** by the Japanese national project.
- The option of smaller defect sizes are being examined in **addendum of the JEAC4206-2007**.

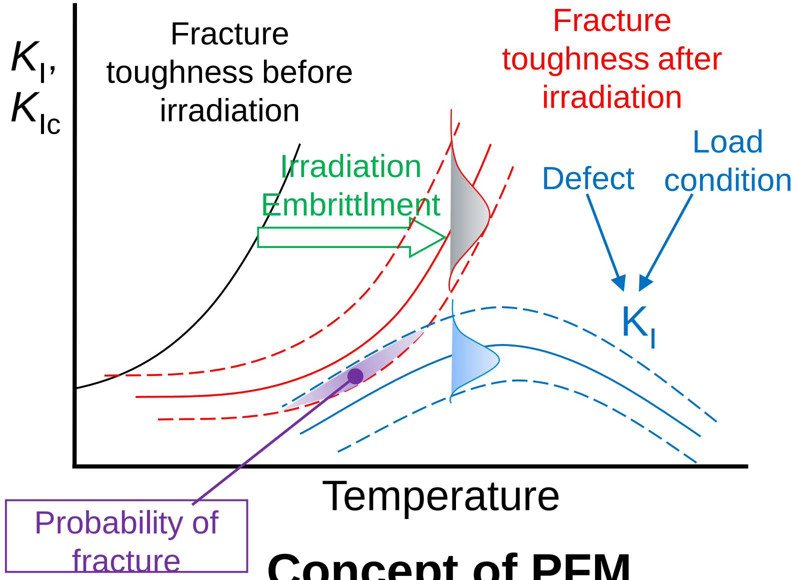
  If there is **no indication of a maximum echo heights DAC 20% or higher** in entire beltline region, smaller defect size may be postulated from **detectable defect size by applied UT procedure** according to the above national project and **subsequent fatigue crack growth till the end of plant operation**.



K_I by smaller postulated defect size

2) Probabilistic Fracture Mechanics (PFM)

- **Deterministic fracture mechanics** evaluates **whether fracture is occurred or not**, by using **conservative evaluation conditions**.
- PFM calculates **failure frequency** considering **variations in influence factors to failure**, such as defects, fracture toughness, etc.
- PFM has long been investigated and already incorporated into **PTS screening criteria in the US** (R.G. 1.154 (1987) *) and 10CFR50.61a (2010)**)).
- In Japan, Japan Atomic Energy Agency, JAEA has developed the **PFM analysis code PASCAL** and application of PFM has been investigated.
- JEAG4640-2018, **"Guideline for calculating failure frequency of reactor pressure vessels based on probabilistic fracture mechanics"**, was issued.
- Discussions between NRA and utilities regarding the application of PFM to actual plants especially for **optimization of in-service inspections** are underway.
- PFM should be included in JEAC 4206 in the future.



Concept of PFM

*) U.S. Nuclear Regulatory Commission, "Format and Content of Plant-specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors", Regulatory Guide 1.1.54, (2021).

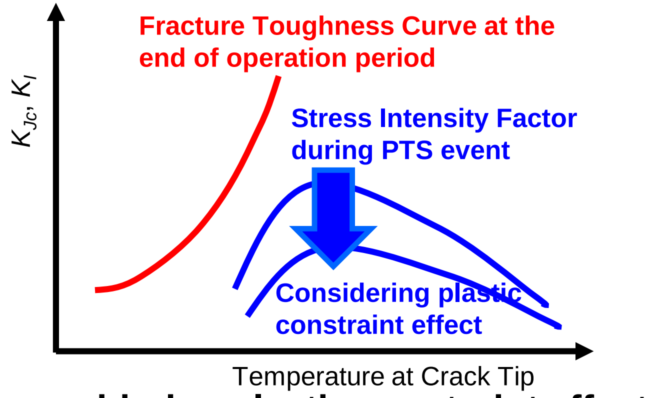
**) U.S. Nuclear Regulatory Commission, "Alternate Fracture Toughness Requirements for Protection against Pressurized Thermal Shock Events," 10CFR50.61a, (2010).

MITSUBISHI HEAVY INDUSTRIES, LTD.

3) Plastic Constraint Effect

- It is considered that evaluation by fracture toughness based on the **standard fracture toughness test specimens** such as Compact Tension specimens includes **a margin for realistic defects in actual structures** from the viewpoint of the plastic constraint effects.
- **Several standards** (ISO 27306, WES 2808, BS7910) were published to address plastic constraint correction for fracture evaluation.
- The subcommittee on Constraint-based assessment of Fracture in DBTT region (CAF) in the Japan Welding Engineering Society (2018-2022FY) studied **assessment methods of fracture considering the plastic constraint effect***).
- In **U.S. (ASME code)** and **France (RCC-M and RSE-M)**, defect assessment procedures considering plastic constraint effects are also being investigated.
- **Assessment procedure considering plastic constraint effects** should be incorporated to JEAC4206 for more realistic assessments in the future.

*) Hojo, K., et. al., "Constraint Effect on Fracture in Ductile-Brittle Transition Temperature Region (Report 3)," Proceedings of ASME 2023 Pressure Vessel & Piping Division Conference, July 16-21, 2023, Atlanta, GA, USA, PVP2023-105965.



K_I considering plastic constraint effect

Conventional fracture mechanics

Fracture toughness ← C(T) or SE(B) — High constraint

- large flaw
- Bending stress distribution

In case of shallow cracks or tensile stress field, stresses around the crack tip are low due to low plastic constraint even at the same stress intensity factor *K*.

Actual structure — Low constraint

- Small flaw
- Dominant tensile stress distribution in thickness

Too conservative evaluation result

9. CONCLUSION

- Based on surveillance tests data and operational experiences such as in-service inspections and special inspections, **knowledges about integrity assessment of RPVs for irradiation embrittlement are accumulated continuously**.
- **Knowledges on evaluation methods based on research results** such as PFM and plastic constraint effects are also obtained continuously.
- It is important to **continuously incorporate the latest knowledges into the integrity assessments of the actual plants** especially as the operation period are lengthened.
- **Industrial standards should reflect the latest knowledge** and should be evaluated by the regulator, so that it can be applied to actual plants, which will also lead to **improved public acceptances**.

THANK YOU FOR YOUR ATTENTION!

MOVE THE WORLD FORWARD

MITSUBISHI
HEAVY
INDUSTRIES
GROUP

Questions ? Answers !

Lunch Break

Workshop will resume at 13:55PM