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Development of In-Service Inspection Rules for Sodium-Cooled Fast Reactors Using the System Based Code Concept

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Meet the Presenter

Dr. Shigeru Takaya is a principal researcher of Fast Reactor Cycle System R&D Center of Japan Atomic Energy Agency. He earned his Doctor of Engineering degree from the University of Tokyo, Japan, in 2003.

His main areas of research interests are structural integrity evaluation at elevated temperatures, maintenance technologies for SFRs, and optimization of design and ISI requirements on SFRs through plant life cycle based on the System Based Code (SBC) concept.

He also works on development of codes and standards for these fields and participates in several committees of ASME as well as JSME. He serves as the chair of Subgroup on Elevated Temperature Design in JSME, and also the chair of ASME/JSME Joint Working Group on RIM (Reliability and Integrity Management) Processes and SBC in ASME.

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Importance of In-Service Inspection Rules

- In-Service Inspection (ISI) rules provide requirements for periodic inspections of passive components of nuclear power plants during the service, which is important for safety and stable operation.
- Effective and efficient ISI is crucial to suppress operation costs which is one of major components of power generation cost.
- ISI rules also affect design of nuclear power plants because the accessibility to the components where ISI is required needs to be considered appropriately in the design.

ISI rules need to be developed rationally by considering relevant features of reactor type and design of an individual nuclear power plant.

Features of Sodium-cooled Fast Reactors

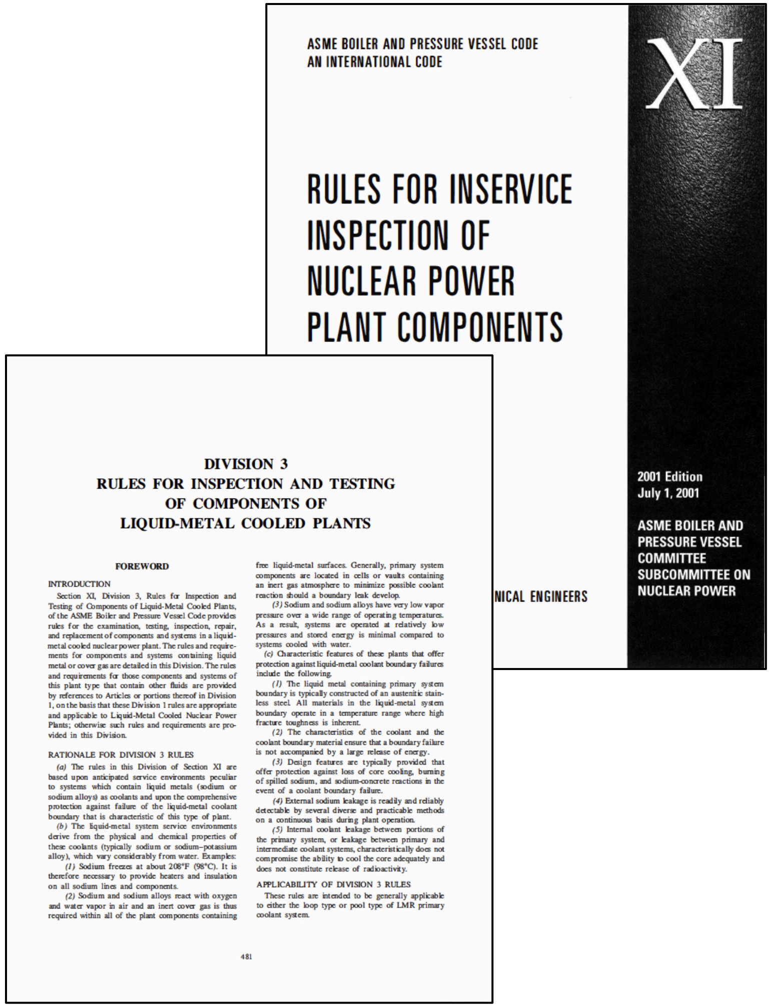
- Sodium-cooled fast reactors (SFRs) have several different features from the conventional Light Water Reactors (LWRs); thus it is not reasonable to apply the ISI rules of conventional LWRs to SFRs directly.

		LWR（PWR）	SFR（Monju）	Features of SFRs
Operating Conditions				
	Coolant	Water	Sodium	・Opaque and chemically active ・Excellent compatibility with structural materials
	Reactor Outlet Temp.	~320℃	~530℃	・Operation in Creep regime
	Difference in Temp. between Reactor Outlet and Inlet	~30℃	~130℃	・High thermal stress
	Operating Pressure	~16 MPa	~1 MPa	・Low pressure
Dimensions（Reactor Vessel）				
	Internal Diameter (ID)	~4 m	~7 m	・Large diameter
	Thickness	~200 mm	~50 mm	・Thin wall thickness
	ID／Thickness	~20	~140	・High ratio

Conventional Standards for ISI of SFRs

- Historically, the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Division 3 had provided ISI rules for liquid-metal cooled plants.
- It was developed as part of the Clinch River Breeder Reactor Plant Project in the U.S..
- The code revision was suspended due to the cancellation of the project, thus several parts, including acceptance standards for examinations of Class 1 Components, were left as being in the course of preparation.

It was practically difficult to apply it to SFR plants.



ASME BOILER AND PRESSURE VESSEL CODE
AN INTERNATIONAL CODE

XI

RULES FOR INSERVICE INSPECTION OF NUCLEAR POWER PLANT COMPONENTS

2001 Edition
July 1, 2001

ASME BOILER AND PRESSURE VESSEL COMMITTEE
SUBCOMMITTEE ON NUCLEAR POWER

...NICAL ENGINEERS

DIVISION 3
RULES FOR INSPECTION AND TESTING OF COMPONENTS OF LIQUID-METAL COOLED PLANTS

FOREWORD

INTRODUCTION

Section XI, Division 3, Rules for Inspection and Testing of Components of Liquid-Metal Cooled Plants, of the ASME Boiler and Pressure Vessel Code provides rules for the examination, testing, inspection, repair, and replacement of components and systems in a liquid-metal cooled nuclear power plant. The rules and requirements for components and systems containing liquid metal or cover gas are detailed in this Division. The rules and requirements for those components and systems of this plant type that contain other fluids are provided by references to Articles or portions thereof in Division 1, on the basis that these Division 1 rules are appropriate and applicable to Liquid-Metal Cooled Nuclear Power Plants; otherwise such rules and requirements are provided in this Division.

RATIONALE FOR DIVISION 3 RULES

(a) The rules in this Division of Section XI are based upon anticipated service environments peculiar to systems which contain liquid metals (sodium or sodium alloys) as coolants and upon the comprehensive protection against failure of the liquid-metal coolant boundary that is characteristic of this type of plant.

(b) The liquid-metal system service environments derive from the physical and chemical properties of these coolants (typically sodium or sodium–potassium alloy), which vary considerably from water. Examples:

(1) Sodium freezes at about 208°F (98°C). It is therefore necessary to provide heaters and insulation on all sodium lines and components.

(2) Sodium and sodium alloys react with oxygen and water vapor in air and an inert cover gas is thus required within all of the plant components containing free liquid-metal surfaces. Generally, primary system components are located in cells or vaults containing an inert gas atmosphere to minimize possible coolant reaction should a boundary leak develop.

(3) Sodium and sodium alloys have very low vapor pressure over a wide range of operating temperatures. As a result, systems are operated at relatively low pressures and stored energy is minimal compared to systems cooled with water.

(c) Characteristic features of these plants that offer protection against liquid-metal coolant boundary failures include the following.

(1) The liquid metal containing primary system boundary is typically constructed of an austenitic stainless steel. All materials in the liquid-metal system boundary operate in a temperature range where high fracture toughness is inherent.

(2) The characteristics of the coolant and the coolant boundary material ensure that a boundary failure is not accompanied by a large release of energy.

(3) Design features are typically provided that offer protection against loss of core cooling, burning of spilled sodium, and sodium-concrete reactions in the event of a coolant boundary failure.

(4) External sodium leakage is readily and reliably detectable by several diverse and practicable methods on a continuous basis during plant operation.

(5) Internal coolant leakage between portions of the primary system, or leakage between primary and intermediate coolant systems, characteristically does not compromise the ability to cool the core adequately and does not constitute release of radioactivity.

APPLICABILITY OF DIVISION 3 RULES

These rules are intended to be generally applicable to either the loop type or pool type of LMR primary coolant system.

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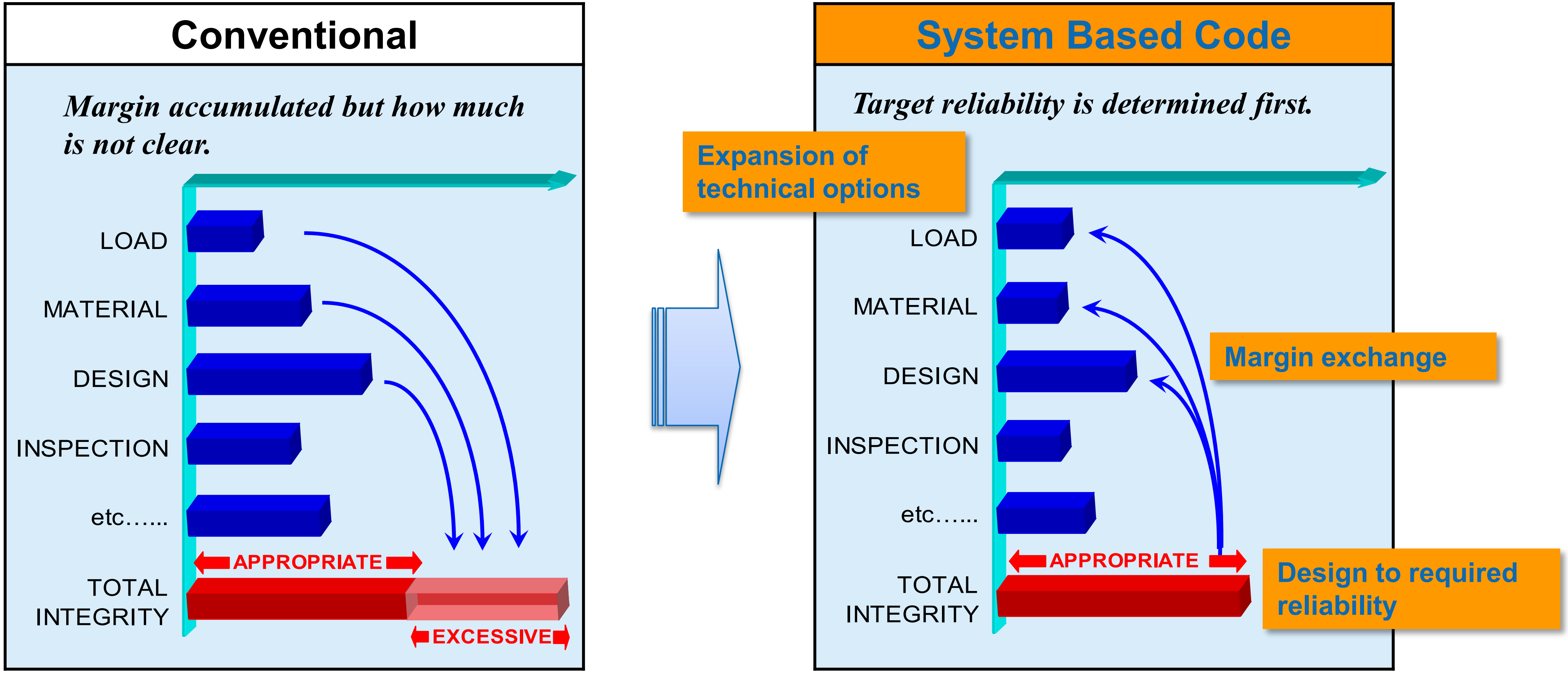
Development of ISI Rules for SFRs based on SBC Concept

To address this situation,

- ASME/JSME Joint Task Group for System Based Code was established in 2012 in the ASME B&PV Code Committee.
- As a result of collaboration between experts from ASME and JSME, Code Case N-875 that provides the alternative ISI requirements for Liquid-metal cooled plants to Sec. XI, Div. 3 was developed based on the SBC concept, and was issued in 2017.
- A fitness-for-service code for sodium-cooled fast reactors was concurrently being developed in JSME, and the first edition was approved in 2021.

System Based Code Concept



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Asada, Y., Tashimo, M. and Ueta, M., 2002, "System Based Code—Principal Concept," Proc. 10th International Conference on Nuclear Engineering, ICONE10-22730.

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Overview of Code Case (CC) N-875

- Approved in 2017
- CC that provides alternative ISI requirements to Sec. XI, Div. 3, using the SBC concept
- Compositions
  - Main body: alternative provisions and a logic flow to establish criterion for application
  - Mandatory Appendices
    - Appendix I: Derivation of Component Target Reliabilities from Plant Safety Requirements
    - Appendix II: Procedure for Structural Reliability Evaluation for Passive Components of Liquid Metal Reactors
- Several key parts of CC N-875 have been incorporated in the new Sec. XI, Div. 2, "Requirements for RIM Programs for Nuclear Power Plants".

Alternative ISI Requirements to Sec. XI, Div. 3

Conditions identified by using the SBC concept

Examination category	Section XI, Division 3	Code Case N-875
Liquid-metal-retaining welds in Class 1 vessels protected by guard vessels	▪ Continuous monitoring ▪ VTM-2	▪ Continuous monitoring*
Liquid-metal-retaining welds in Class 1 vessels not protected by guard vessels	▪ Continuous monitoring ▪ VTM-2	▪ Continuous monitoring*
Liquid-metal-retaining welds in Class 1 piping protected by guard pipe or tank (Heat transport loop piping)	▪ Continuous monitoring ▪ VTM-2	▪ Continuous monitoring*
Liquid-metal-retaining welds in Class 1 piping not protected by guard pipe or tank (Heat transport loop piping)	▪ Continuous monitoring ▪ VTM-2	▪ Continuous monitoring*
Internal components	▪ VTM-3	▪ None

* Acceptance standards were newly prepared.

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Alternative ISI Requirements to Sec. XI, Div. 3 (Cont'd)

- **Acceptance Standard for Continuous Monitoring**

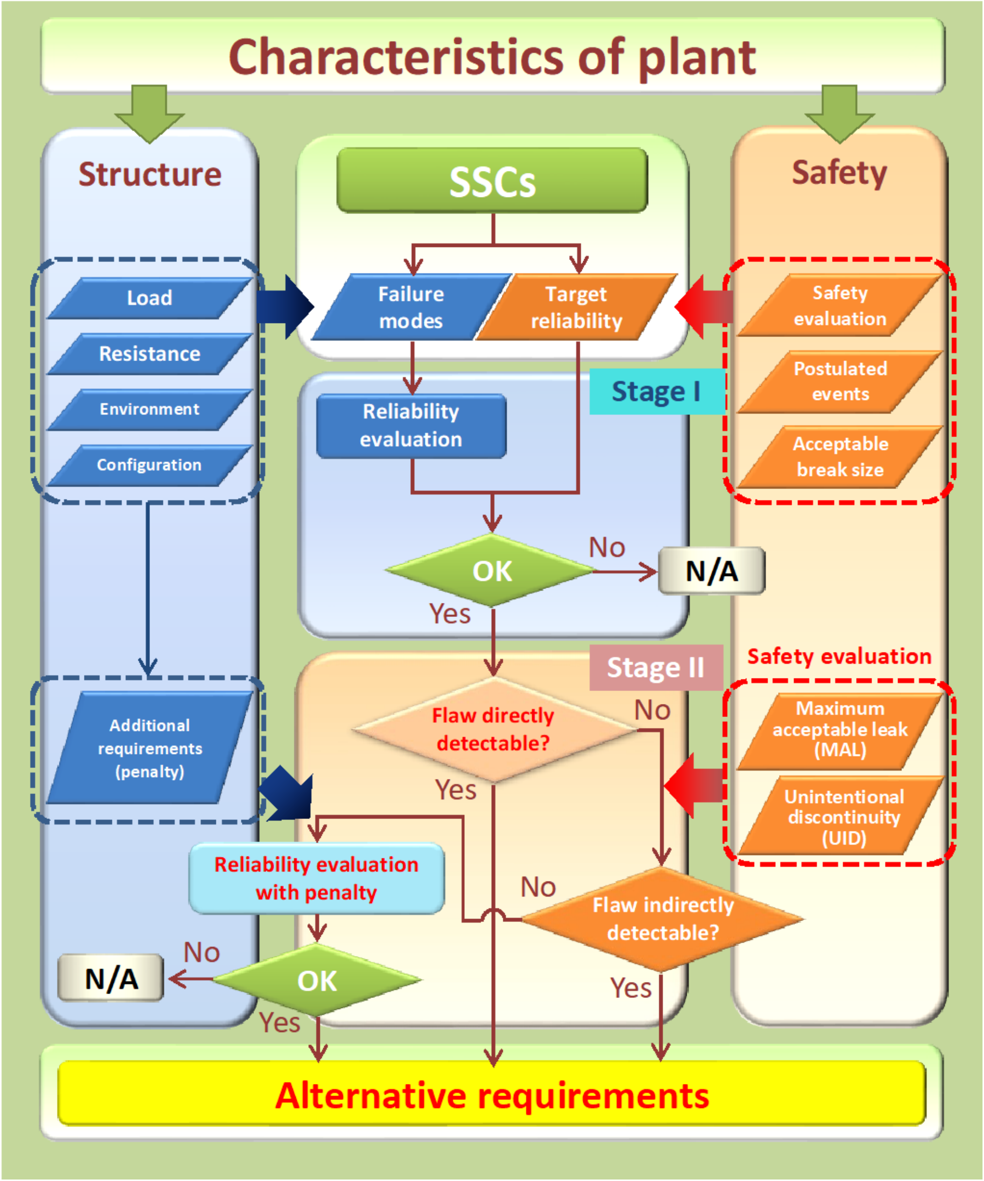
  Once leakage is indicated, it is required to conduct a confirmation of leakage in accordance with the procedure predetermined by the Owner. If the confirmation takes longer time than the determined time, it is conservatively evaluated that the leakage is confirmed.

  – **In case of confirmed:** Immediate shutdown of the system

  – **In case of unconfirmed:** Repair of the leak detectors to meet the minimum percentage of required working leak detectors

Logic Flow (1/3): Overview

- Criteria for application of the alternative ISI requirements based on the SBC concept
- The logic flow consists of
  - Stage I: Structural reliability evaluation
  - Stage II: Safety related evaluation
- Both probabilistic and deterministic approaches are applicable.
- The logic flow could be for general use. It has been incorporated in Appendix for alternate requirements for NDE and monitoring in the new Sec. XI, Div. 2.



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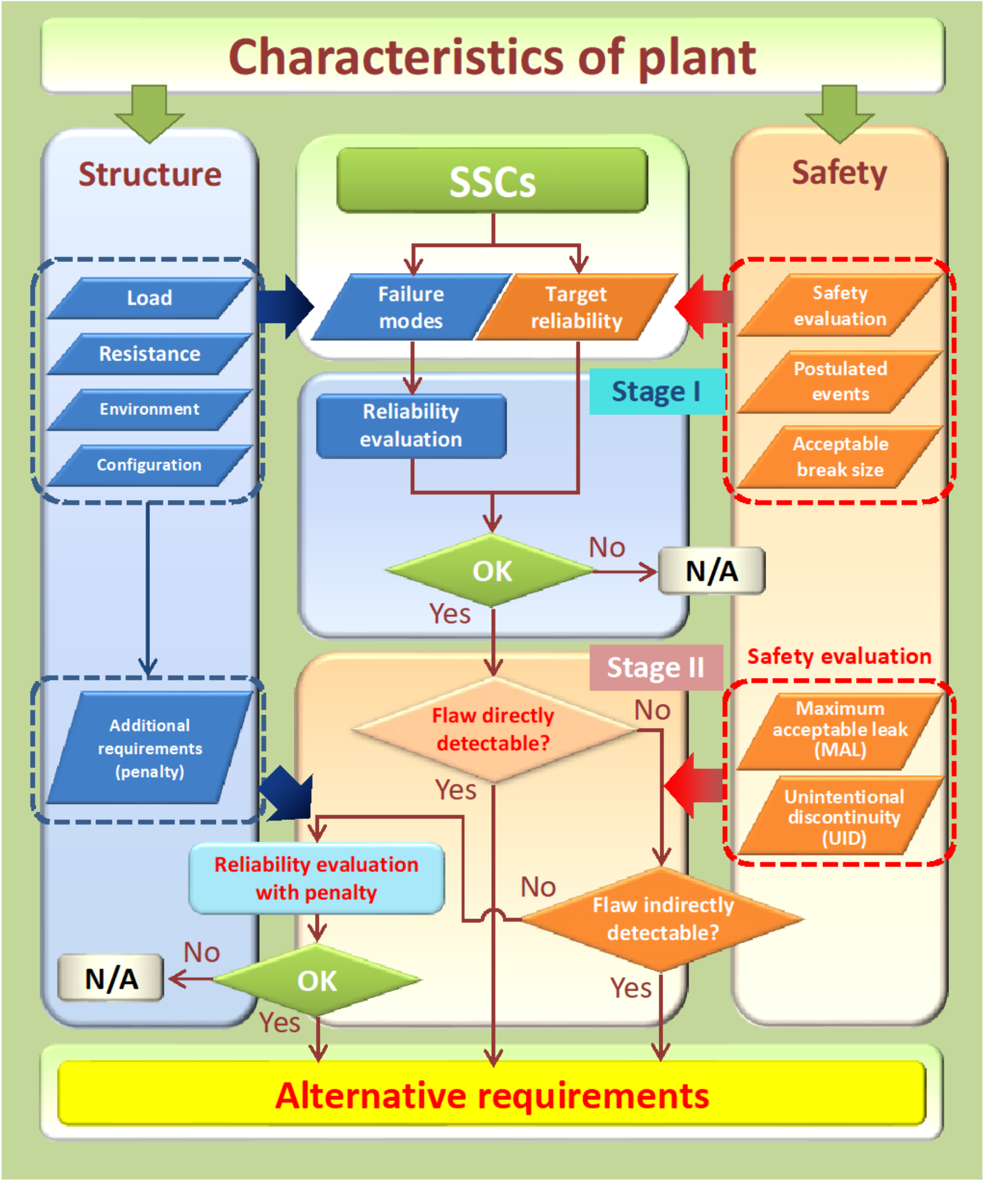
Takaya, S. et al., 2015, ASME J of Nuclear Rad Sci, Vol. 1, Paper #011004.

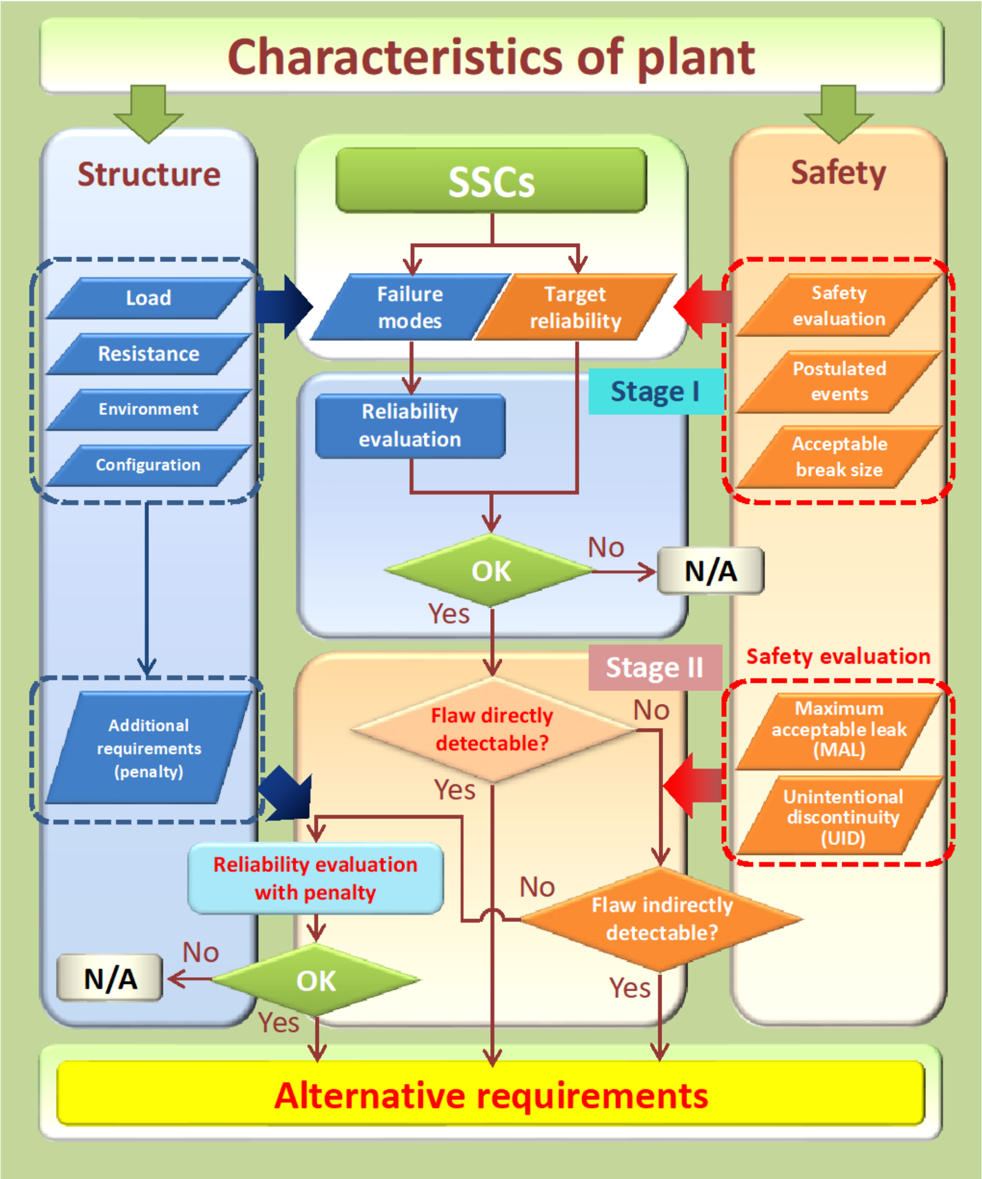
Logic Flow (2/3): Stage I

- The component level structural integrity under design basis conditions is considered.
- Potential failure modes are determined based on degradation mechanisms.
- **Component level requirements (CLRs)** are determined in deterministic or probabilistic manner based on input related to safety evaluation.
- The contribution of ISI is not taken into account.
- If the evaluated reliability meet the CLR, the user may proceed to the Stage II evaluation.



Logic Flow (3/3): Stage II

- The ability to detect flaws that ensures that the plant can be safely shut down before the flaw reaches the maximum acceptable size is considered.
- Either direct or indirect detection is allowable. → **Flexible selection of suitable ISI technologies according to the plant features**
- In case postulated flaws are not detectable, if the additional margin is demonstrated by imposing penalty in structural reliability evaluation, examination for flaw detection is not required. → **Margin exchange between different technical areas via the penalty**



Key Technical Elements: Determination of Failure Modes

- Degradation mechanisms that can potentially produce flaws during service is evaluated based on the list of potential degradation mechanisms provided in the CC as well as operating and research experience.
- Failure modes are determined based on the identified degradation mechanisms.
- Failure modes not addressed in the design code are also considered, if necessary.

Table 1
Degradation Mechanisms Criteria

Degradation Mechanism		Degradation Features	Attribute Criteria	Susceptible Regions
TF		crack initiation and propagation	– potential for mixing of hot and cold fluids at core outlet, – potential for liquid level turbulence, – potential for inflow of cold fluid to hot outlet plenum during a scram transient, OR ...tential for relatively rapid temperature change	base metal, heat affected zone (HAZ), and welds

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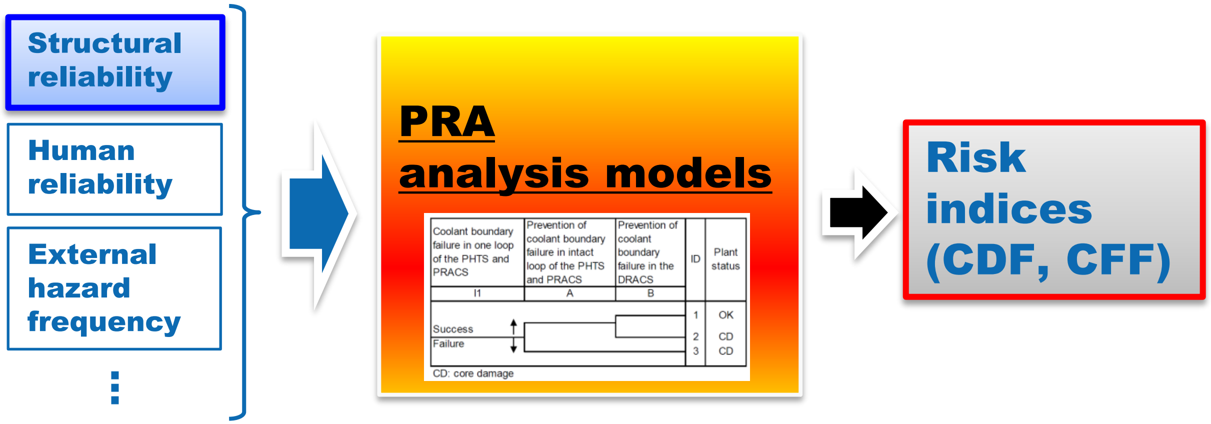
Key Technical Elements: Determination of CLRs

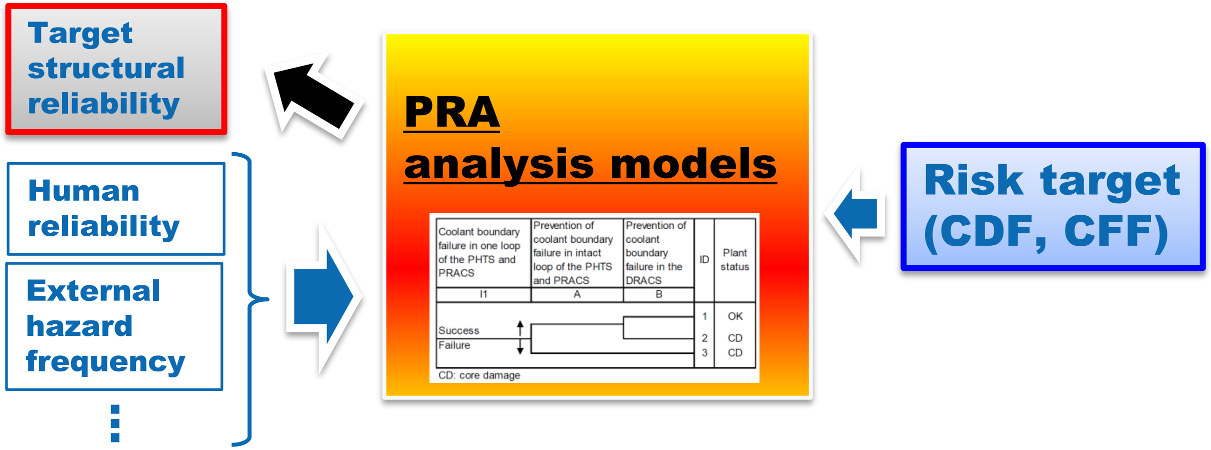
- Component level requirements (CLRs) are established either deterministically or probabilistically based on input related to safety evaluation.
- **Deterministically-established CLRs:**
  - Quantities such as the break size postulated in an accident scenario that define the allowable limits from a safety point of view
- **Probabilistically-established CLRs:**
  - Component level target reliabilities derived from quantitative plant level requirements available in quantities such as core damage frequency (CDF) , Containment Failure Frequency (CFF) or Large Early Release Frequency (LERF).
  - A method for derivation of component level target reliability is provided by Appendix I.

Key Technical Elements:
Derivation of Component Level Target Reliability (App. I)

- Probabilistic Risk Assessment (PRA) is usually used to integrate the individual reliabilities into the risk index.



- The developed method uses PRA **in a reverse way** to derive component level structural reliability from the plant level risk target.



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Kurisaka, K., Nakai, R., Asayama, T., and Takaya, S., 2011, "Development of System Based Code (1) Reliability Target Derivation of Structures and Components", J. Power Energy Syst., Vol. 5, pp.19-32. DOI: 10.1299/jpes.5.19

Key Technical Elements: Calculation of Structural Reliability (App. II)

- Calculated structural reliability could vary depending on which and how uncertainties are considered.
- To narrow scatters of calculation, **App. II uniquely provides procedures for structural reliability evaluation for passive components.**
- The procedures consist of “Failure scenario setting”, “Modeling”, and “Failure probability calculation”.
- JSME developed “Guidelines on Reliability of Fast Reactor Components”, and App. II in the CC was developed based on the Guidelines.

Takaya, S., Machida, H., and Kamishima, Y., 2014, “Elaboration of the System Based Code Concept – Activities in JSME and ASME– (3) Guidelines on Structural Reliability Evaluation for FBR,” Proc., 22nd International Conference on Nuclear Engineering, ICONE22-30570.

Key Technical Elements: Evaluation of Detectability of Flaws

- The basis of Stage II evaluation is detectability of postulated flaws.
- Not only conventional direct detection such as ultrasonic examination but also **indirect detection is allowable.**
- New indices for indirect detection were introduced;
  - **Maximum acceptable leak (MAL)**
    - Applied for coolant boundary items
    - A leak that would not lead to an increase in the CDF or CFF/LERF that has been calculated in the safety evaluation of the plant
  - **Unintentional discontinuity (UID)**
    - Applied for non-coolant boundary items
    - Change in plant parameters, such as temperature and velocity of coolant, that indicates that flaws exist before they lead to an increase in the CDF or CFF/LERF that has been calculated in the safety evaluation of the plant.

A suitable ISI method could be selected according to plant features.

Key Technical Elements: Reliability Evaluation with Penalty

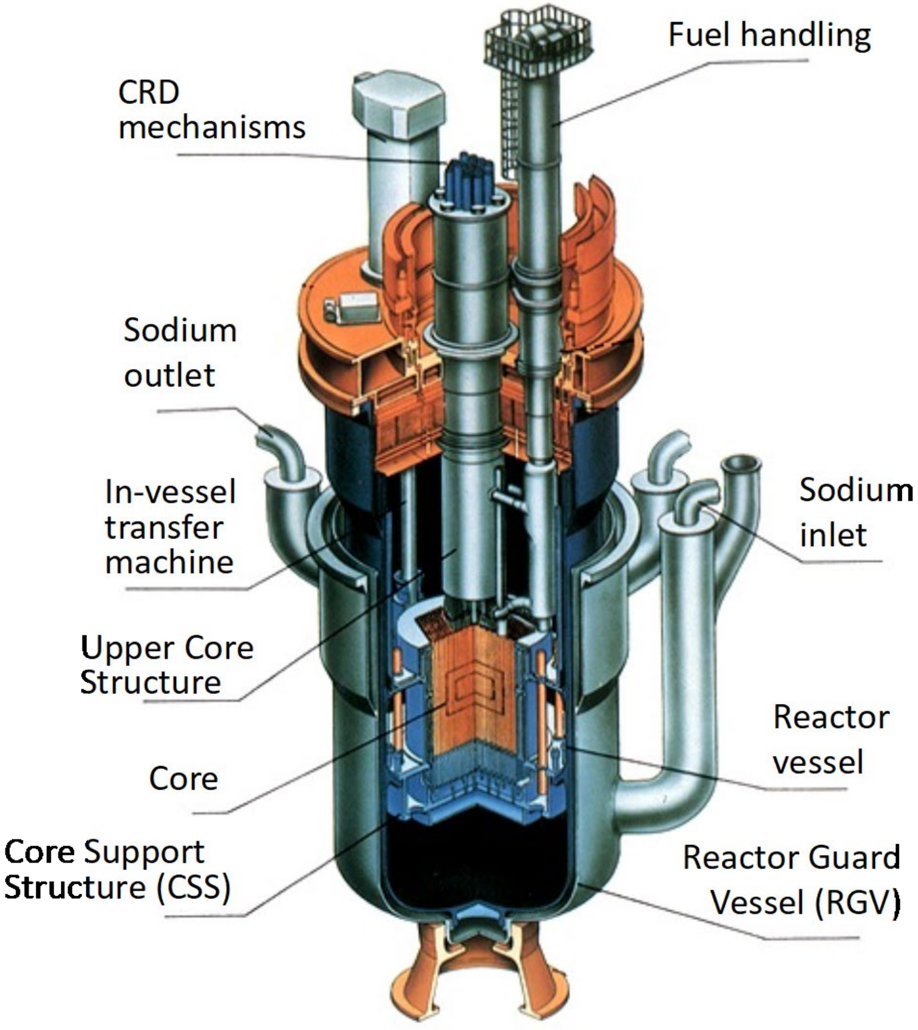
- If postulated flaws are not detectable, reliability evaluation with a penalty may be conducted. If the result meets the target reliability, the examination is exempt.
- An **"unrealistically conservative yet logically imaginable"** penalty which is correlated to the highest consequence failure mode is determined from the following 4 categories:
  - **Load:** An additional load caused by failure of an adjacent component, or part of the component that reduces the loads on the critical portion of the component
  - **Resistance:** A decrease in resistance caused by loss of strength-enhancement mechanisms or metallurgical stability
  - **Environment:** A presumption that the component is subject to the most harmful environment postulated at the component location
  - **Configuration and initial flaws:** Unanticipated flaws or distortions

Examples

- Several trial evaluations have been conducted for the prototype SFR in Japan, Monju, to illustrate the developed logic flow:
  - Upper Core Structure*
  - Core Support Structure**
  - Reactor Guard Vessel**

  } ← Probabilistic approach

  - Class 1 Piping*** ← Deterministic approach



Main reactor components of Monju

* Takaya, S. et al., 2015, ASME J of Nuclear Rad Sci, 1, #011004
** Takaya, S. et al., 2016, Nucl. Eng. Des., 305, pp. 270–276
*** Takaya, S. et al., 2020, ASME J of Pressure Vessel Technol., 142, #021601

Basic Information on Core Support Structure (CSS)

Function		To maintain core configuration
Material		Type 304 Stainless Steel
Environment		In sodium
Max. temp. (Normal operation)		400 deg. C
Main load (Normal operation)		Cyclic load by reactor start-ups and shutdowns
Category		Internal Components
ISI	Sec. XI, Div.3	VTM-3*: Visual (e.g., periscope and light) or combination of undersodium scanning and dimensional gaging
	CC N-875	None

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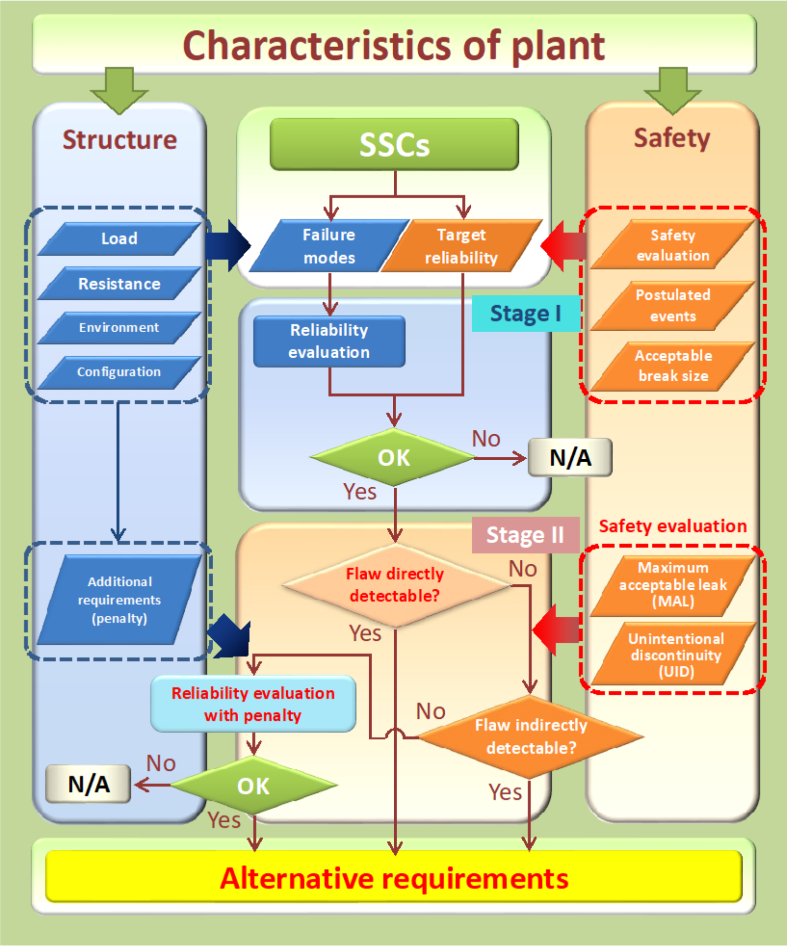
*: Visual examination intending to determine the general mechanical and structural conditions of components and their supports and to detect discontinuities and imperfections.

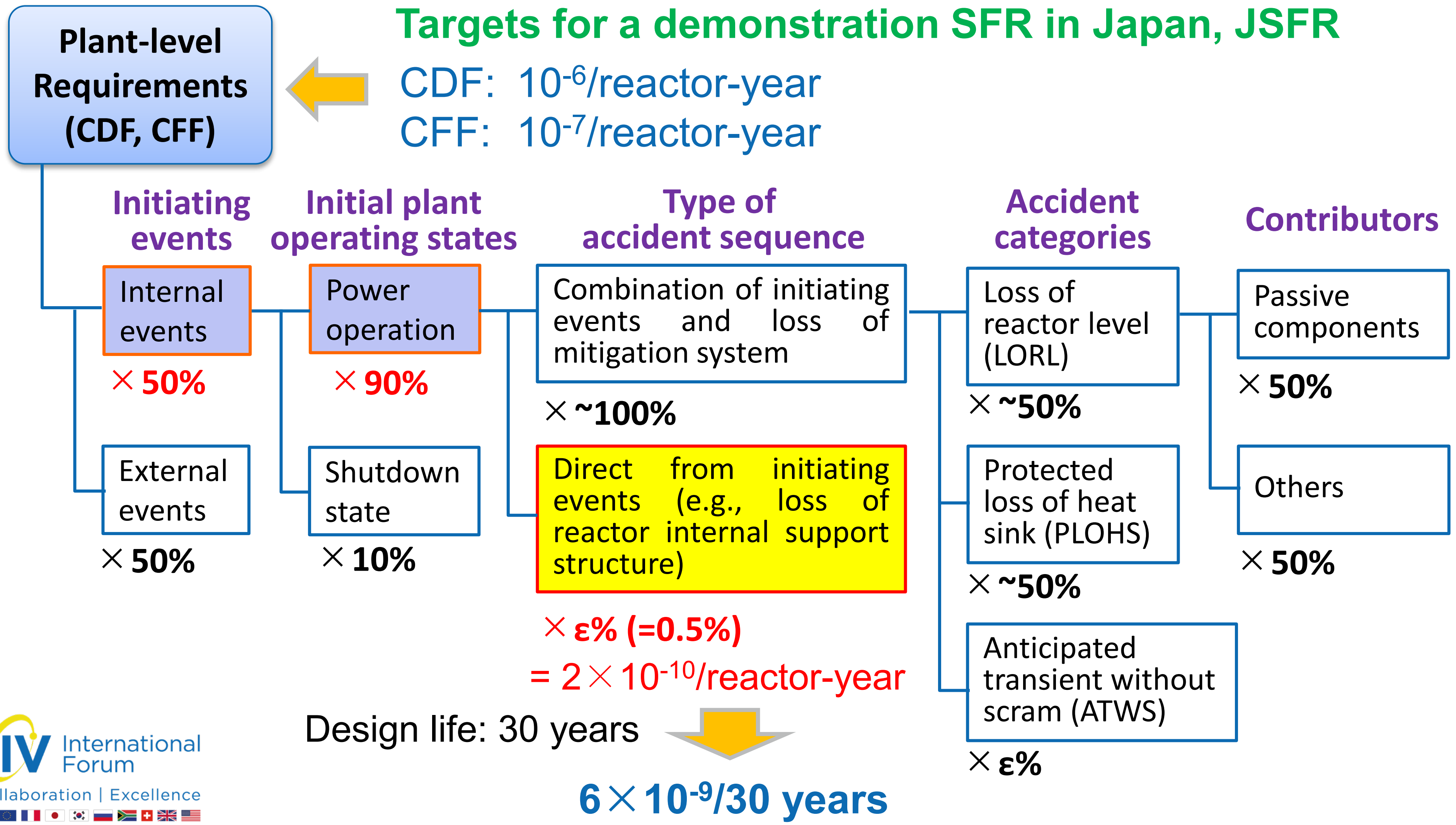
Failure Modes of CSS

- Potential failure modes have to be analysed exhaustively even if they are not addressed explicitly in the design code.
  - Corrosion in purity-controlled sodium is negligible.
  - The temperature is low enough to neglect creep damage.
  - Material properties might change due to neutron irradiation. For example, decrease in ductility is one of concerns, but a surveillance program during the operation is available to confirm neutron irradiation effects.
  - The looseness of the fixing bolts may be another concern, but preventive measures against the rotation of bolts were taken.

Just **fatigue damage** due to cyclic loads by reactor start-ups and shutdowns is left.



Probabilistic CLR(Target Reliability) of CSS



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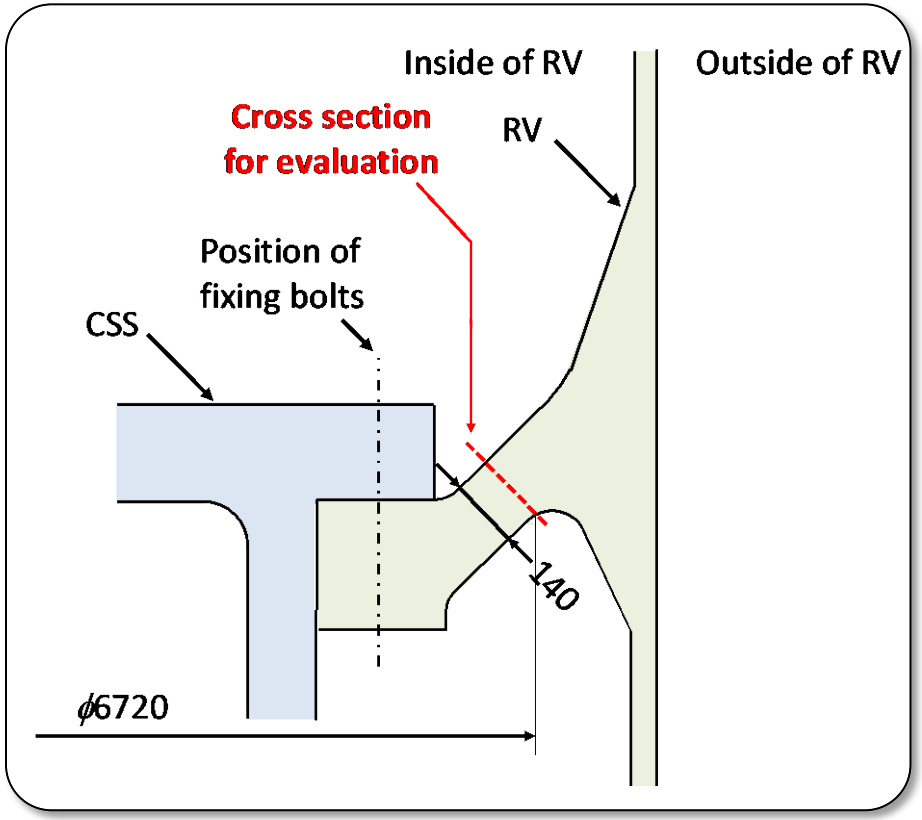
Stage I Evaluation

- **Crack initiation evaluation**
  - Assuming crack initiation when fatigue damage, D_f, equals to 1.

$$D_f = n / N_f(\varepsilon_t)$$

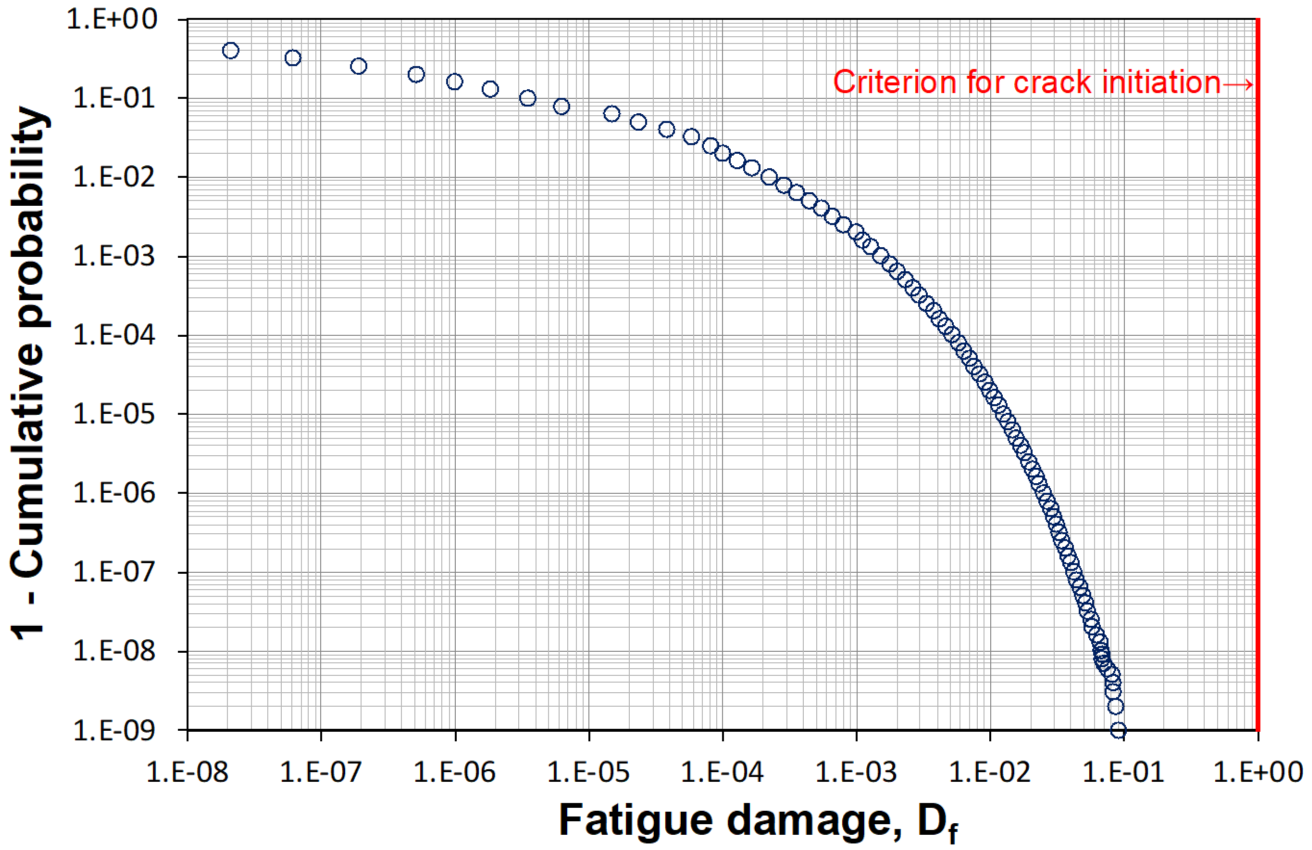
- **Evaluation position**
  - The mount arm from the RV where there is no other load transfer path to maintain core structure

- **Calculation method**
  - Direct Monte-Carlo method
    - Number of Samples: 10^9 samples
    - Random variables: Thermal stress and fatigue life



Stage I Evaluation (Cont'd)

- The number of crack initiation samples was 0 out of 10^9.



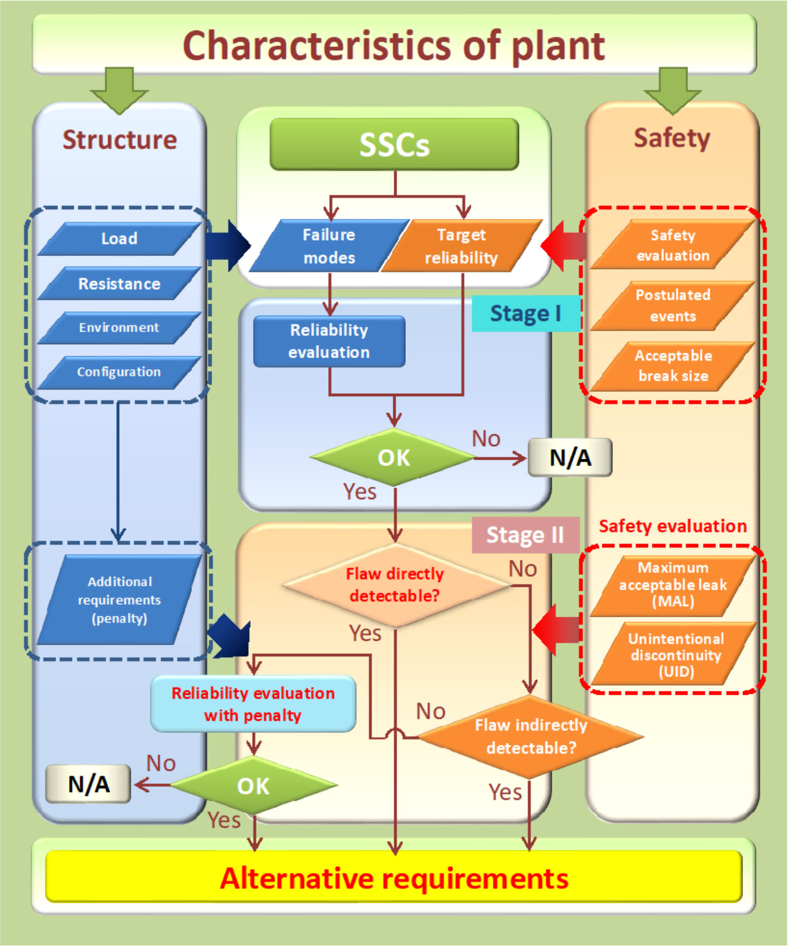
Sufficient reliability

Proceed to Stage II evaluation

Stage II Evaluation

- **Detectability of flaws**
  - The appendix for the under sodium scanning in Sec. XI, Div. 3 was "in the course of preparation". It was assumed that under-sodium scanning systems were not available yet in this evaluation.
  - It was also assumed that there were no monitoring methods to detect flaws in CSS.

- **Reliability evaluation with penalty**
  - **A fully circumferential crack with depth of 10% of the thickness** was assumed as **an initial defect**.



Stage II Evaluation (Cont'd)

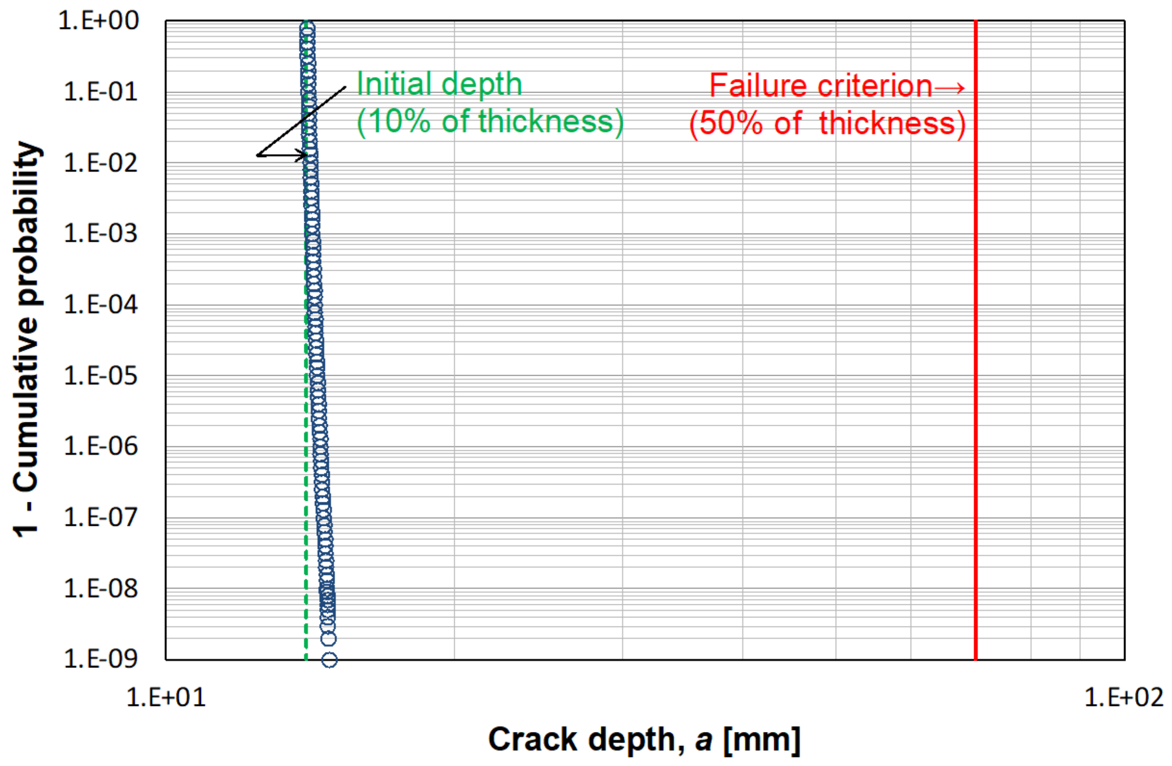
- **Crack growth evaluation:**

$$\frac{da}{dn} = C_f \Delta J_f^{\,m_f}$$

- **Failure criterion:**
  - 50% of wall thickness

- **Direct Monte-Carlo method:**
  - Number of samples: 10^9 samples
  - Random variables: Thermal stress and C_f

➡ The number of failure samples was 0 out of 10^9.



Sufficient reliability even with penalty

Alternative requirement of CC N-875 is applicable;
No NDT is required for CSS.

Basic Information on Primary Heat Transfer System Piping

Function		Primary heat transfer
Material		Type 304 Stainless Steel
Environment		Inside: Sodium, Outside: Inert gas
Max. temp. (Normal operation)		530 deg. C
Main load (Normal operation)		Cyclic load by reactor start-ups and shutdowns
Category		Liquid-metal-retaining Class 1 piping protected /not protected by guard pipe or tank
ISI	Sec. XI, Div.3	Continuous Monitoring, and VTM-2*
	CC N-875	Continuous Monitoring

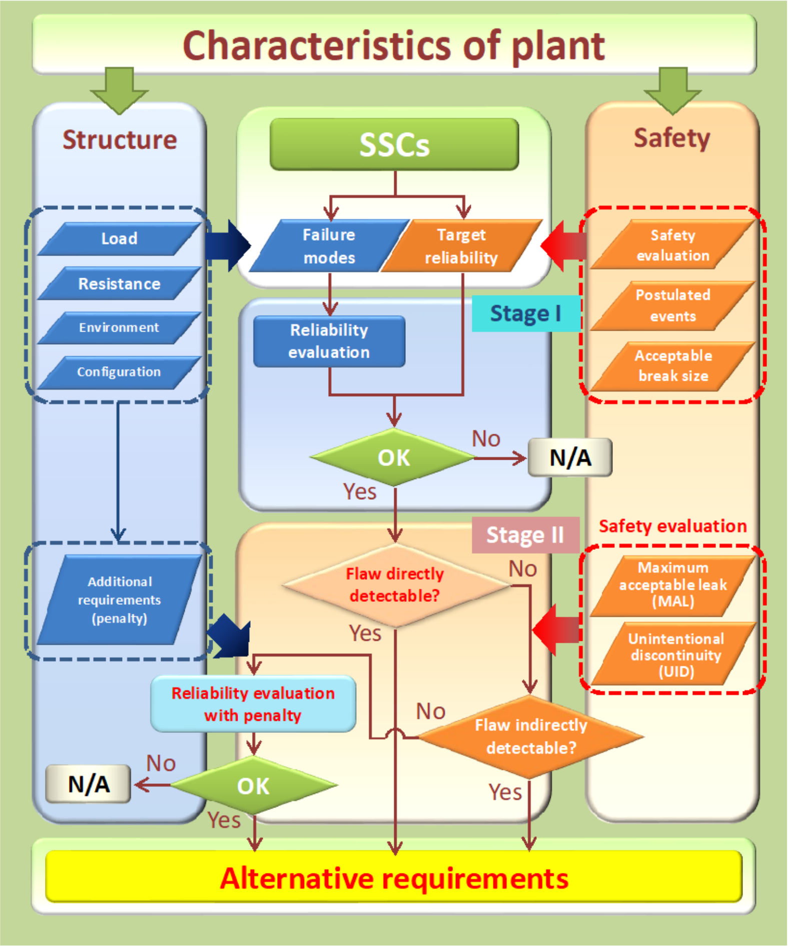
*: Visual examination intending to detect accumulations of liquids, liquid streams, liquid drops, and smoke

Failure Modes and CLR of PHTS Piping

- **Failure mode:**

  Crack initiation and propagation due to **Fatigue-creep interaction damage**

- **Deterministic CLR:**
  - **The break size postulated in the plant safety evaluation** can be used as a deterministic CLR.
  - As for the PHTS of Monju, **a wall-through crack with the opening area of 22 cm^2** was postulated in the plant safety evaluation.



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Stage I Evaluation

- **Deterministic Approach**
  - Fatigue-creep interaction damage is already addressed by the design code.
  - **Fatigue-creep interaction damage** of PHTS piping is restricted **below the design allowable level**.

It is considered that a crack would **NOT** initiate and propagate to the size determined as the CLR, in the conditions of the PHTS.

Proceed to Stage II evaluation

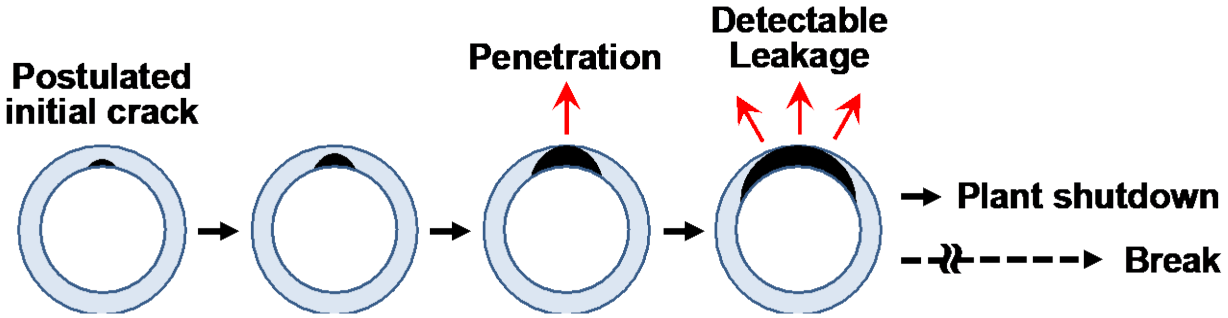
Stage II Evaluation

- **Detectability of Flaws**
  - **Sodium leak detection systems** for small-scale leaks from the PHTS in Monju
    - Sodium ionization detector (SID) : detects a leak by monitoring a change of ion current produced by ionizing sodium aerosol in an inert gas atmosphere
    - Differential pressure detector (DPD) : detects a leak by monitoring a change of difference pressure at the filter installed in the detector
    - These detectors are **designed to detect leaks of 100 g/h.**
  - **MAL:** In the plant safety evaluation of Monju, a through-wall crack with the opening area of 22 cm^2 (CLR) was postulated, and the leakage rate just after the accident was evaluated to be **approximately 80 kg/s**.

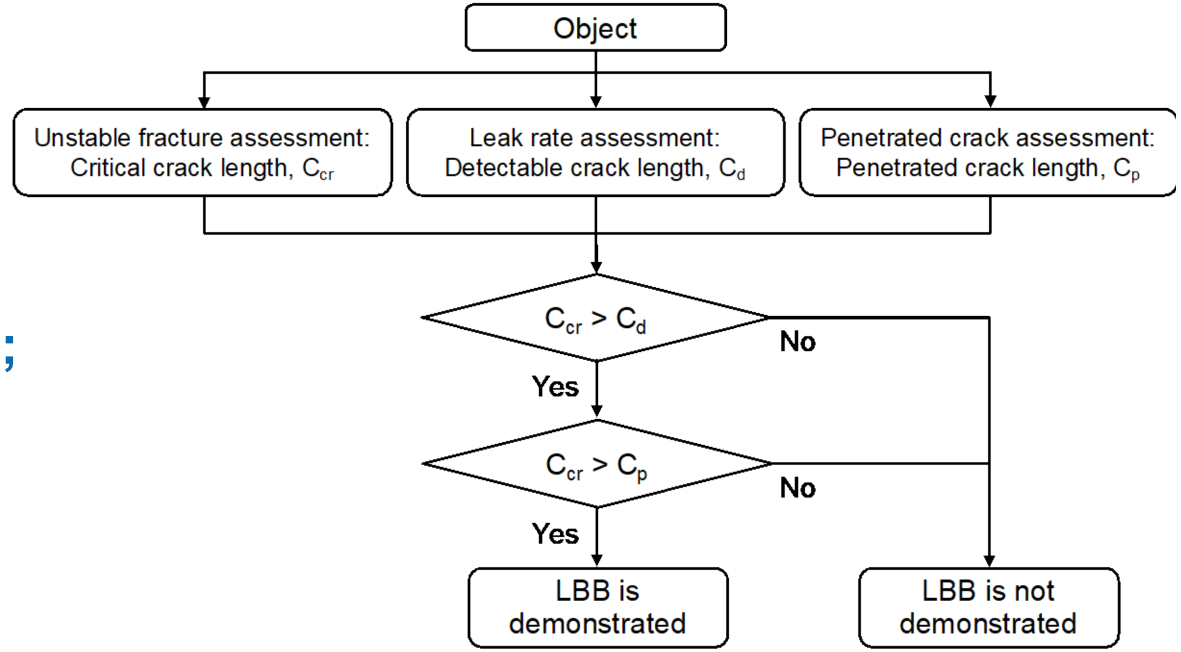
The capability of leak detection of SID and DPD is much greater than the MAL.

Stage II Evaluation (Cont'd)

- **Detectability of Flaws (Cont'd)**
  - **Demonstration of Leak-Before-Break (LBB)**
    - Essential to show the effectiveness of the continuous monitoring
    - Guidelines for LBB assessment of SFR provided by JSME*



LBB concept



Postulated flaws are detectable indirectly.

Alternative requirement of CC N-875 is applicable; Just continuous monitoring is required for the PHTS piping.

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* Yada, H. et al, 2021, Proc. ASME PVP, Paper# PVP2021-61942.

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JSME Standards Related to ISI of SFRs

Development of Codes and Standards based on the SBC concept

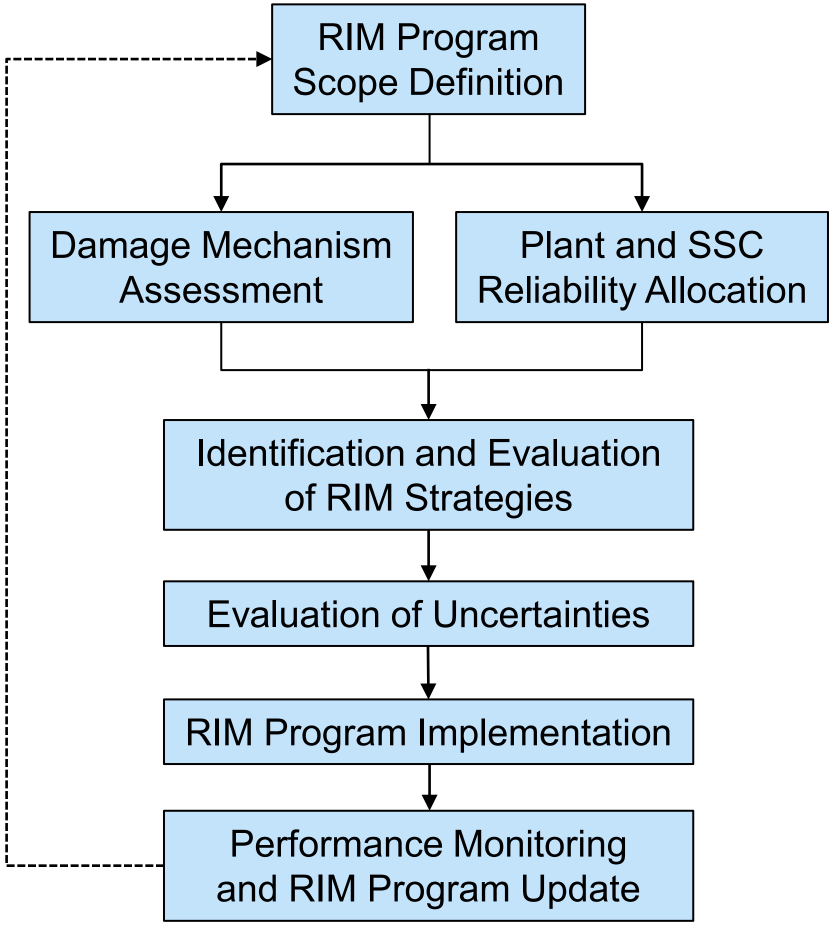
- **Fitness-for-Service Code for Sodium-cooled Fast Reactors**
  - Approved in 2021
  - General Rules, and ISI Requirements of Class 1 Components and their Supports are provided in 1st Edition
  - Continuous monitoring is required to coolant boundaries as CC N-875

- **Guidelines for LBB Assessment of Sodium-cooled Fast Reactors**
  - Approved in 2021
  - Available to determine the sensitivity of leakage detectors

- **Guidelines for Reliability Evaluation of Sodium-cooled Fast Reactor Components**
  - Approved in 2017
  - Unique guidelines providing procedures of reliability evaluation which is important for the SBC concept

ASME Standards for ISI of Advanced Reactors

- **ASME Sec. XI, Div. 2, "Reliability and Integrity Management (RIM)"**
  - **Brand-new fitness-for-service code for all types of nuclear power plants** published in 2019
  - **Technology-neutral requirements** with supplements for specific types of nuclear reactors (currently for LWR and HTGR)
  - **Sharing the basic concept with System Based Code and incorporation of CC N-875**
    - The procedures of deriving target reliability → Mandatory Appendix II
    - The logic flow and reliability evaluation procedures → Non-mandatory Appendix A
  - **Development of the supplement for liquid-metal (Sodium) cooled reactors** is one of the **top priority action items** of Sec. XI committee.
  - ASME/JSME Joint Working Group on RIM Process and SBC is now developing the supplement based on ASME Section XI, Division 3, CC N-875, and JSME Standards including LBB Assessment Guidelines for SFRs.



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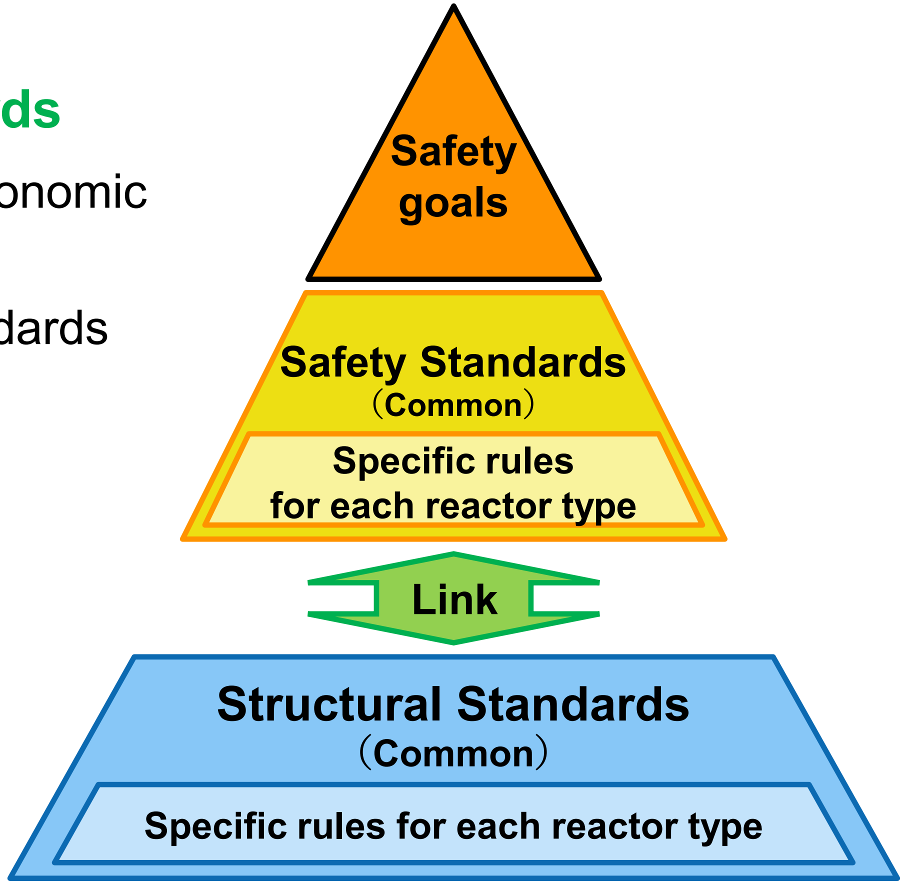
Future Visions

- **Link between Safety Standards and Structural Standards**
  - Essential to develop nuclear plants that balance safety and economic efficiency at high level.
  - "Reliability Target" is a promising key concept to link both standards

- **Technology-inclusively Development of Standards**
  - Essential to develop Standards for various types of advanced reactors efficiently in a timely manner
  - Important from the view of explainability
  - The SBC concept is expected to work as basic principles.



Summary

- Effective and efficient ISI is crucial for safety, stable operation, and economic efficiency of nuclear power plants.
- SFRs have several desirable features such as excellent compatibility between sodium and structural materials while traditional volumetric and surface tests are not as easily performed as in LWRs.
- The SBC concept can be used to develop ISI rules rationally by considering relevant features of reactor type and design of an individual nuclear power plant.
- ASME B&PV CC N-875 is a good and important example of development of Codes and Standards based on the SBC concept, and the details were introduced.
- Recently, the new Sec. XI, Div. 2, RIM, has been developed for various advanced nuclear reactors. RIM shares key concepts with SBC, and the Case has been partially incorporated.
- JSME has also developed a new fitness-for-service code and related guidelines based on the SBC concept.
- Link between safety standards and structural standards as well as development of technology-inclusive standards are important for advanced reactors. The SBC concept is expected to work as basic principles.

Upcoming Webinars

Date	Title	Presenter
26 October 2022	Sodium Integral Effect Test Loop for Safety Simulation and Assessment (STELLA)	Dr. Jewhan LEE , KAERI, Republic of Korea
28 November 2022	Visualization Tool for Comparing Energy Options	Dr. Mark Deinert, Colorado School of Mines, USA
14 December 2022	The Mechanisms Engineering Test Loop (METL) facility at Argonne National Lab	Dr. Derek Kultgen, Argonne National Laboratory, USA