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Nuclear Power Plant Issues in IAEA Draft Technical Guidelines on Fluid-Structure Interactions

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■ IAEA draft technical guidelines on FSIs include plant issues.

In order to avoid future FSI problems, it is important to learn from past FSI events that have occurred at actual plants.

Contents of IAEA Draft Technical Guidelines

2 Plant Issues in IAEA Draft Technical Guidelines on FSIs (2)

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■ Tendencies have been seen in plant issues on FSIs.

(1) Many incidents were caused by vibration which were significantly increased by self-excited vibration when fluid velocity exceeded the critical velocity.

(2) Failures of components were often caused by a combination of FSIs and other factors, such as welding defects, stress concentration, support wear and stress corrosion cracking (SCC).

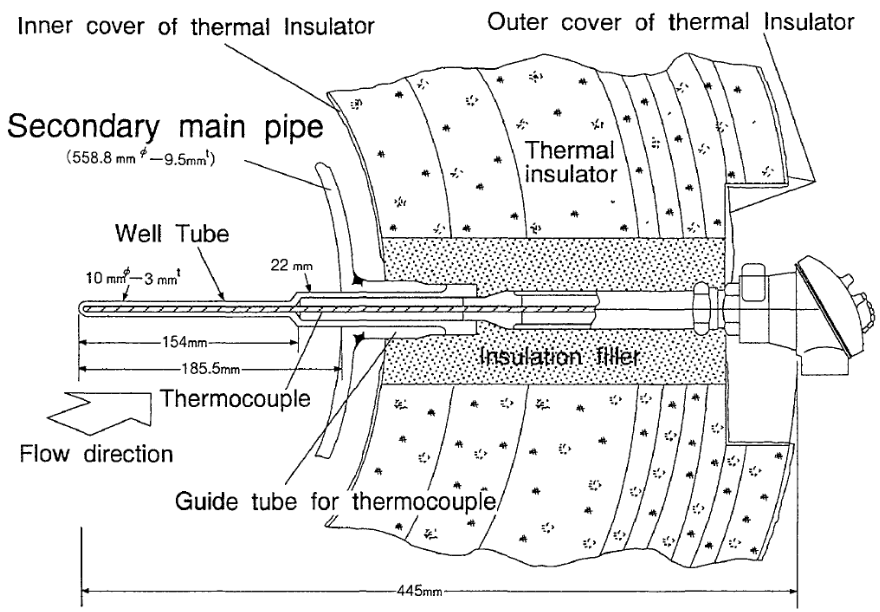
(3) Many events occurred during pre-operation and start-up tests in which a transient operation was implemented. It is possible that intense fluctuating pressure occurred due to valve closures, and pump and air blower start-ups.

(4) ~~Changes~~ Modification of the design and operating conditions for rationalization and performance upgrades often caused FSI problems.

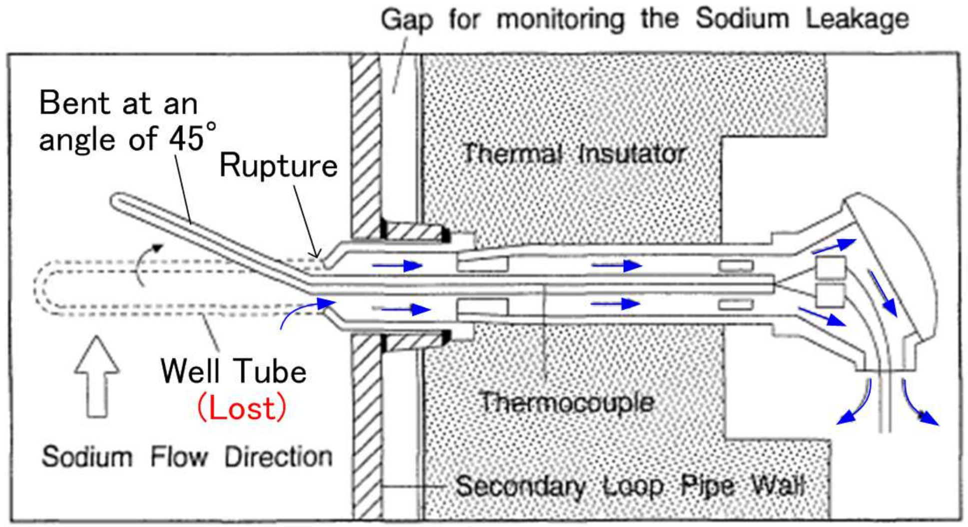
(5) Acoustic waves were propagated over long distances and components were subsequently damaged in a wide range of systems.

3 Failure of Monju Thermometer Well (1)

■ A sodium leakage incident occurred due to breakage of a thermometer well in Monju.



Thermometer Well



Sodium Leak Flow Path

Reference:
A. Miyakawa, et al., 1998, “Technical committee meeting on unusual occurrences during LMFR operation,” Vienna (Austria) 9-13 Nov, *IAEA-TECDOC-1180*, pp. 49-62.

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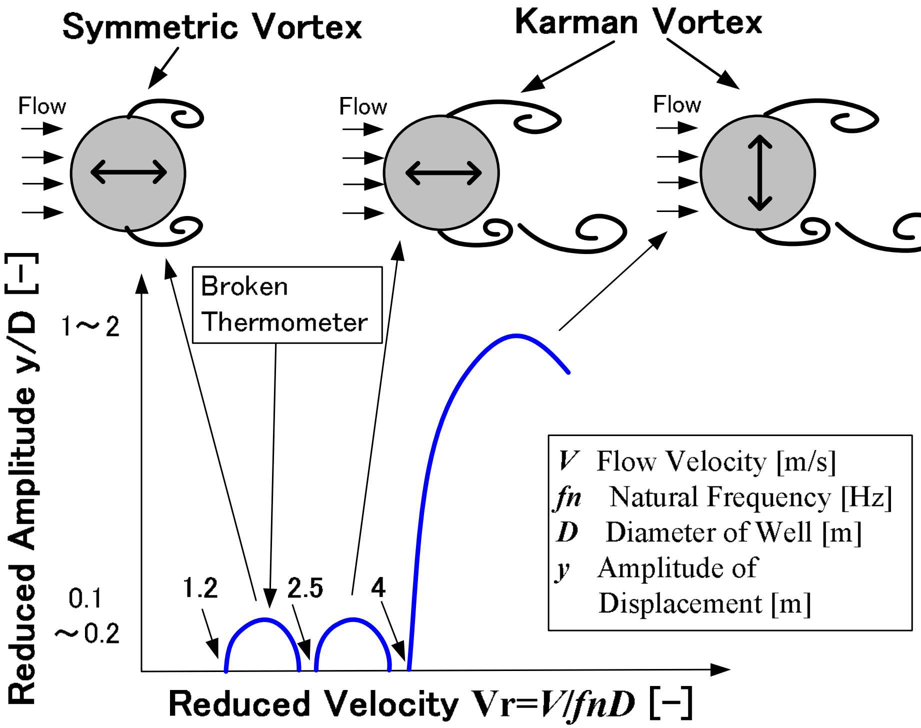
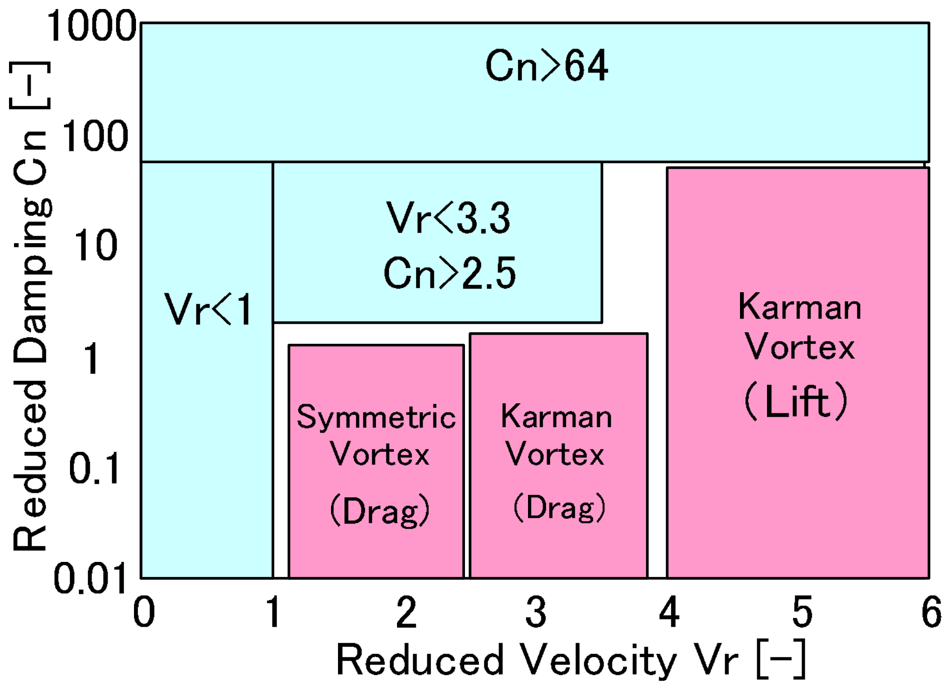
■ Avoiding self-excited vibration is important for guidelines.



Explanation of Vortex Shedding



Conditions for Avoiding Lock-in Synchronization

Reference:
A. Miyakawa, et al., 1998, "Technical committee meeting on unusual occurrences during LMFR operation", Vienna (Austria) 9-13 Nov, IAEA-TECDOC-1180, pp. 49-62.

Reference:
JSME, 1998, "Guideline for Evaluation of Flow-Induced Vibration of a Cylindrical Structure in a Pipe", JSMES012 pp. A25.

5 Dryer Failure at Quad Cites Unit 2 (1)

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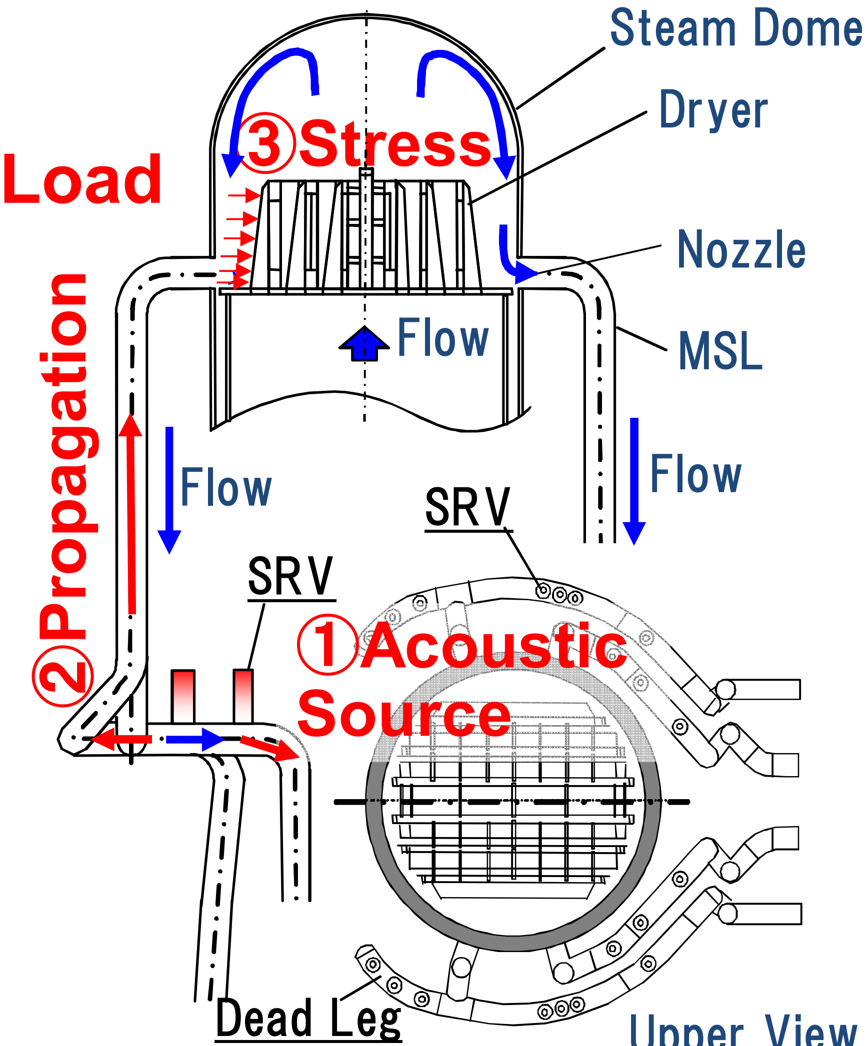
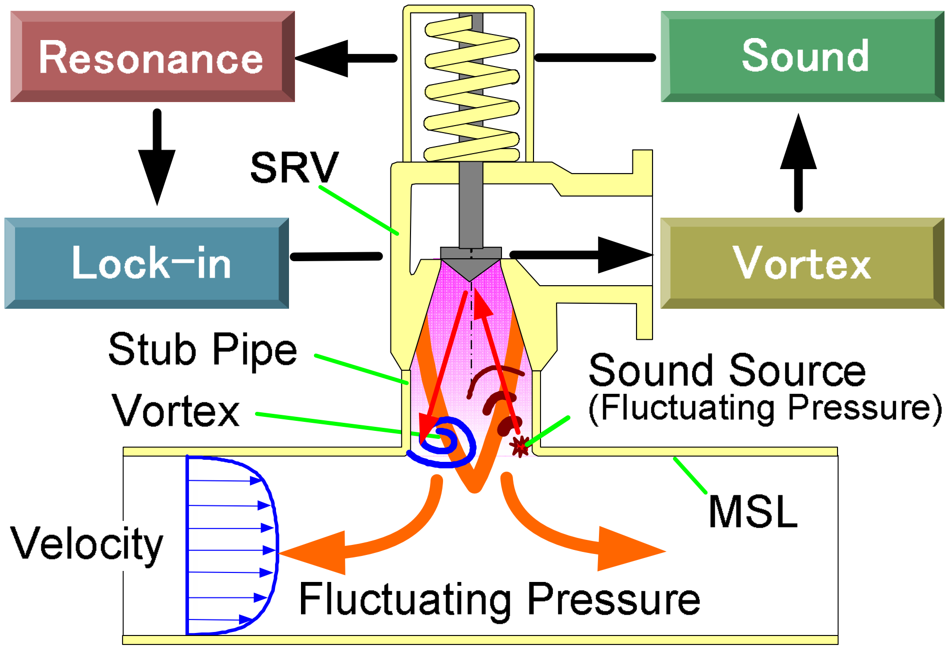
■ Cause of dryer failure was fluctuating pressure at SRVs.



BWR MSLs

▪U.S. BWR-3 dryer failed (117.8% EPU Conditions)

MSL: Main Steam Line
SRV: Safety Relief Valve
EPU: Extended Power Up-rate



Flow-induced Acoustic Resonance at SRVs

Dryer Failure at Quad Cites Unit 2 (2)

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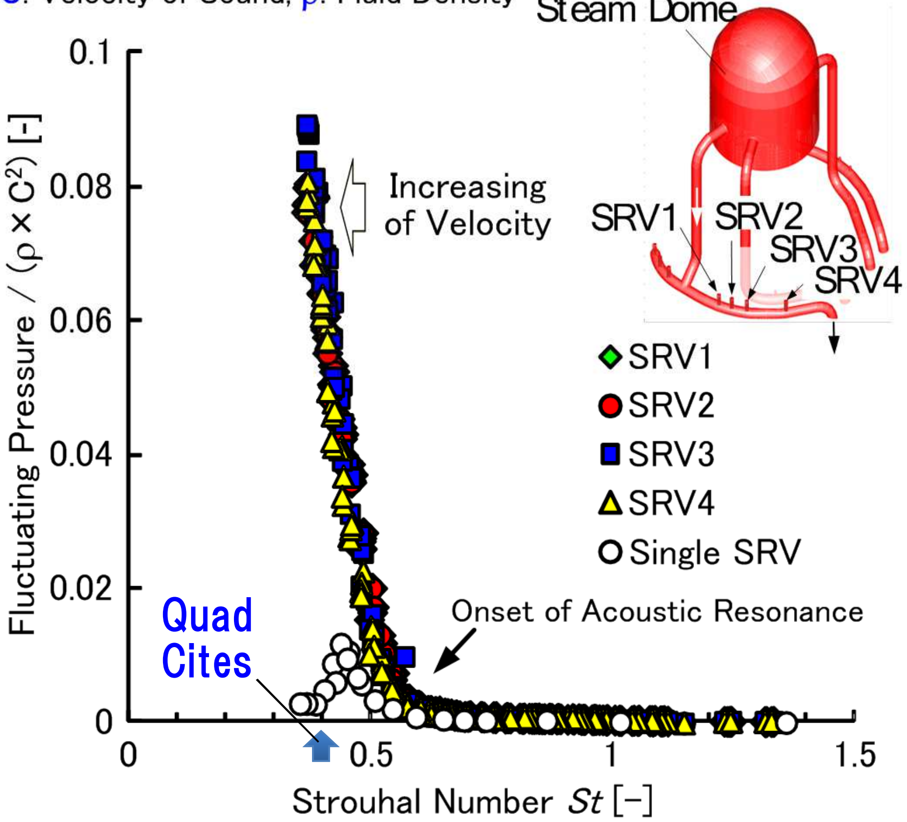
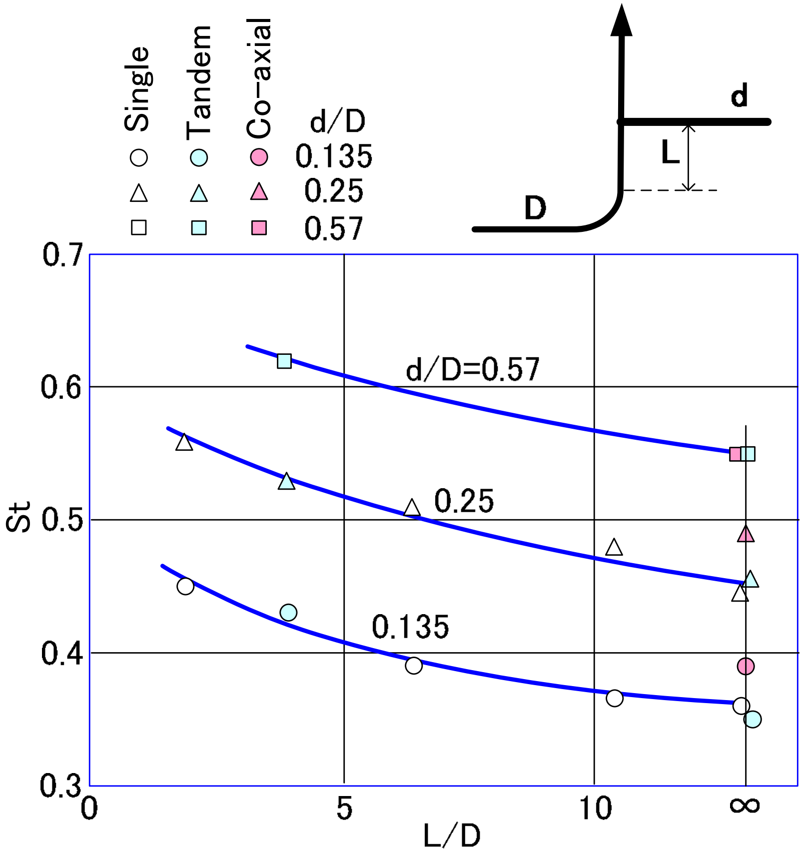
■ Evaluation of acoustic resonance is included in guidelines.

St =Resonance Frequency × Stub Diameter/ Flow Velocity
C: Velocity of Sound, ρ: Fluid Density



Acoustic Resonance in Main Steam Line & Dryer (Air Test)



Onset of Acoustic Resonance

Reference

S. Ziada and S. Shine, "Strouhal Number of Flow-Excited Acoustic Resonance of Closed Side Branches", *J. Fluids and Structures*, Vol. 13, pp.127-142, 1999.

Steam Generator Tube Failure at Mihama (1)

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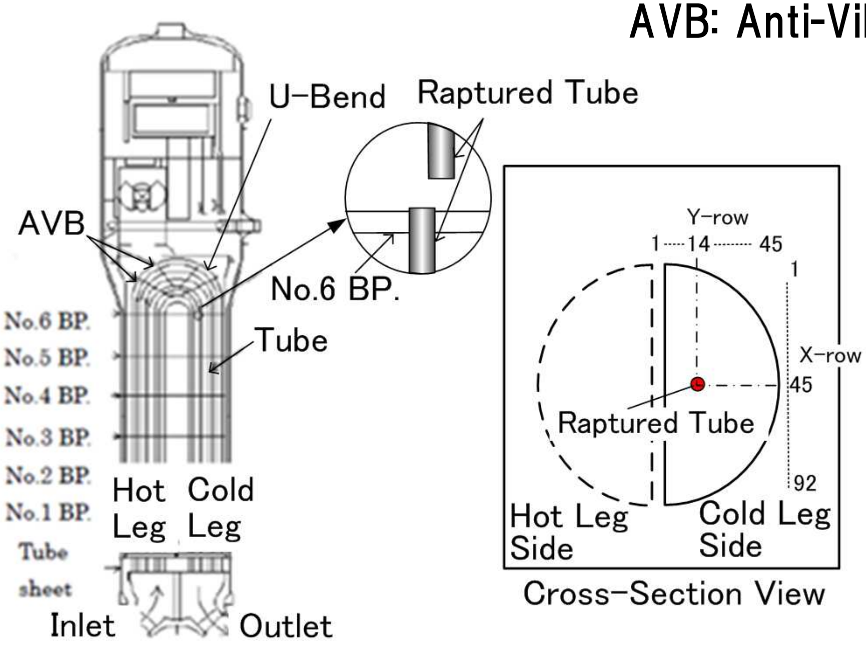
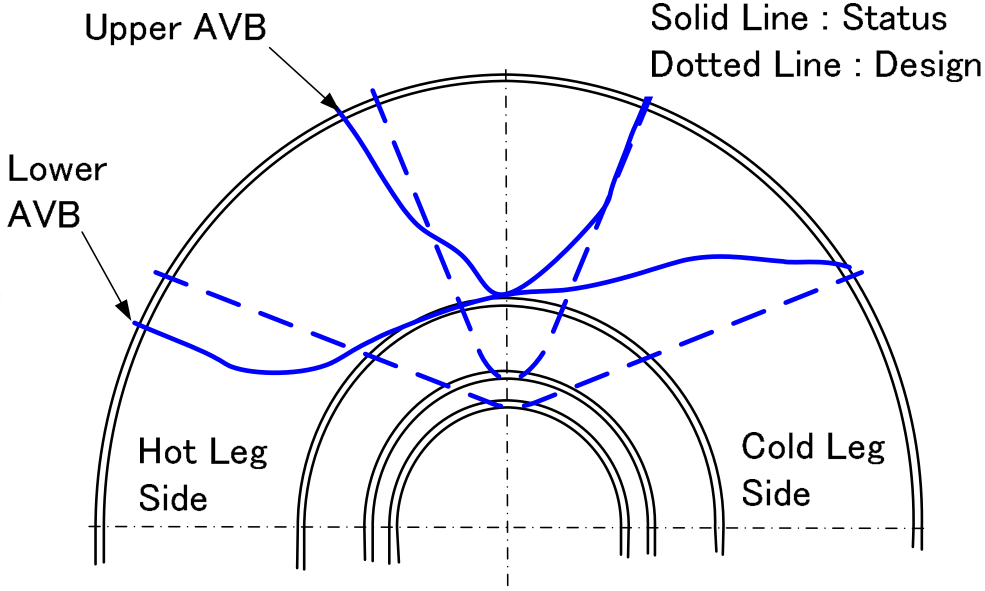
■ Main cause of SG failure was fluid elastic vibration.

- Leakage from steam generator in Mihama Unit-2 in Japan.
- A tube was broken at the bottom of the upper U-bend.
- The tube had not been supported at the U-bend with an AVB.

AVB: Anti-Vibration Bar



Steam Generator



U-Bend

Reference
Agency of Natural Resources and Energy, 1991, "Final Report on the Steam Generator Tube Break at Mihama Unit No.2 of Kansai Electric Power Co., Inc. That Occurred on February 9, 1991", Ministry of International Trade and Industry, Japan

Steam Generator Tube Failure at Mihama (2)

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■ Unknown flow distribution was evaluated by peaking factor.

Fluid-Elastic Stability Margin: *FSM*

$$FSM = \frac{V_c}{V_e}$$

Effective Flow Velocity: *Ve*

$$V_e^2 = \frac{\frac{1}{\rho}\int_0^L \rho(x)V^2(x)\phi^2(x)dx}{\frac{1}{m}\int_0^L m(x)\phi^2(x)dx}$$

m mass per unit length [kg/m]
ρ fluid density [kg/m^3]
ϕ vibration mode [-]

Critical Flow Velocity: *Vc*

$$\frac{V_C}{fD} = C \cdot \left[\frac{m(2\pi\zeta)}{\rho D^2}\right]^a$$

D outer diameter of tube [m]
f natural frequency of tube [m]
ζ damping ratio[-]

Consideration of Flow Peaking Factor

STEP 1

・Check *Vc* in the case of row 9 in North Anna considering its irregular AVB pattern.

STEP 2

・Check the target U-bend of another plant considering its AVB pattern with experimental data based on comparison with North Anna.

STEP 3

・Estimate *FSM* of plant.

Water Hammer Events

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■ Water hammer occurred~~s~~ due to valve operation, pump sudden start-up and shutdown.

Type	Date	Components	Event	Category	Cause
Oconee Unit 2	July 1996	Moisture separator re-heater drain line	Pipe rupture	Condensation induced water hammer	Workers opened valve earlier than planned.
Vogtle Units 1,2	Feb. 1996	Nuclear water service system	Pipe weld crack	Hydraulic transients	Impact of discharge flow from the starting pump.
H. B. Robinson Unit 2	Oct. 1996	Cold-leg accumulator injection line	Damage to a seismic restraint	Hydraulic transients	Opening the accumulator isolation valve
Waterford Unit 3	Dec. 1996	Low pressure safety injection system	Water leaked from safety injection tank	Hydraulic transients	Presence of some volume of nitrogen gas in the pump discharge line

Reference
U. S. NRC Information Notice No.91-95: Supplement: Water Hammer Events Since 1991

10 FSIs of BWR Internals

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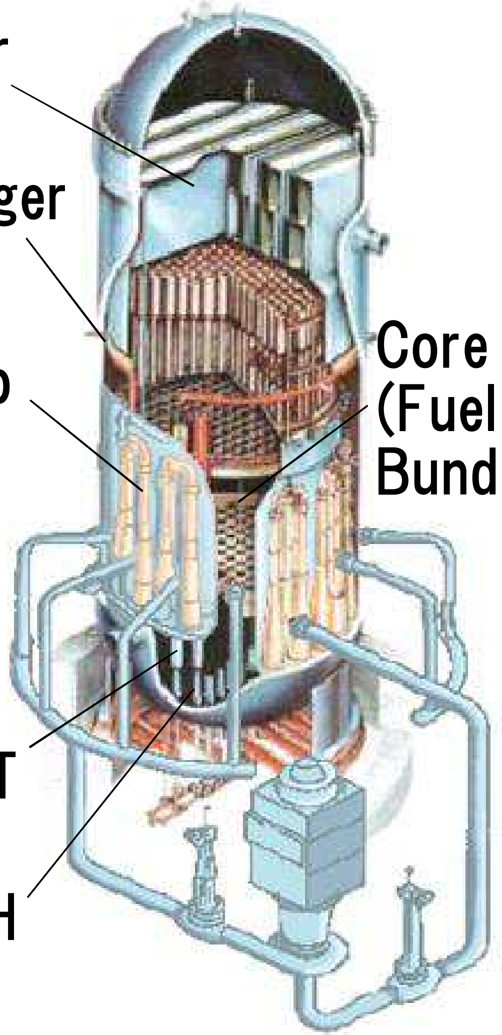
■Evaluation of BWR internals are included in guidelines.



BWR

Designs of the reactor internals are for prototypes or the operating conditions are changed.

Vibration assessments:
- Measurements and inspections in actual plants
- Analyses
- Mock-up tests

The following are important internals regarding FSIs.

(1) Jet pumps
(2) Internals in lower plenum
- Control rod drive housings (CRDH)
- Control rod guide tubes (CRGT)
- Instrumentation tubes (ICMGT)

(3) Spargers and piping
(4) Steam dryers and their assemblies
(5) BWR fuel bundle

11 FSIs of PWR Internals

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Designs of reactor internals or operating conditions are changed.

- In the design stage, the margins of safety for abnormal vibrations such as vortex shedding lock-in or self-excited vibration are initially checked.
- Prediction of the vibration response due to flow turbulence and pump pulsation is conducted by analysis and/or flow testing.
- During the pre-operational testing ~~of~~ in actual plants, assessment with vibration measurements is required for a new design plant (prototype) on the first of a kind of engineering (FOAKE) ~~of a new design (prototype).~~
- Inspections after pre-operational testing are conducted as the final check.

<u>The following items are important for assessment of PWR internals regarding FSIs.</u>

A) Flow-induced vibration in downcomer region of reactor
B) Excitation test by using a scale model of the reactor internals
C) Eigen value analysis
D) Flow-induced vibration test using a scale model
E) Verification of turbulence-induced vibration analysis method
F) Evaluation method for the pressure fluctuation caused by a reactor coolant pump
G) Vibration analysis for reactor internals of a prototype
H) Vibration measurement for reactor internals of a prototype
I) Flow induced vibration for reactor control rods

12 Plant Issues (1) Control Rod Cluster

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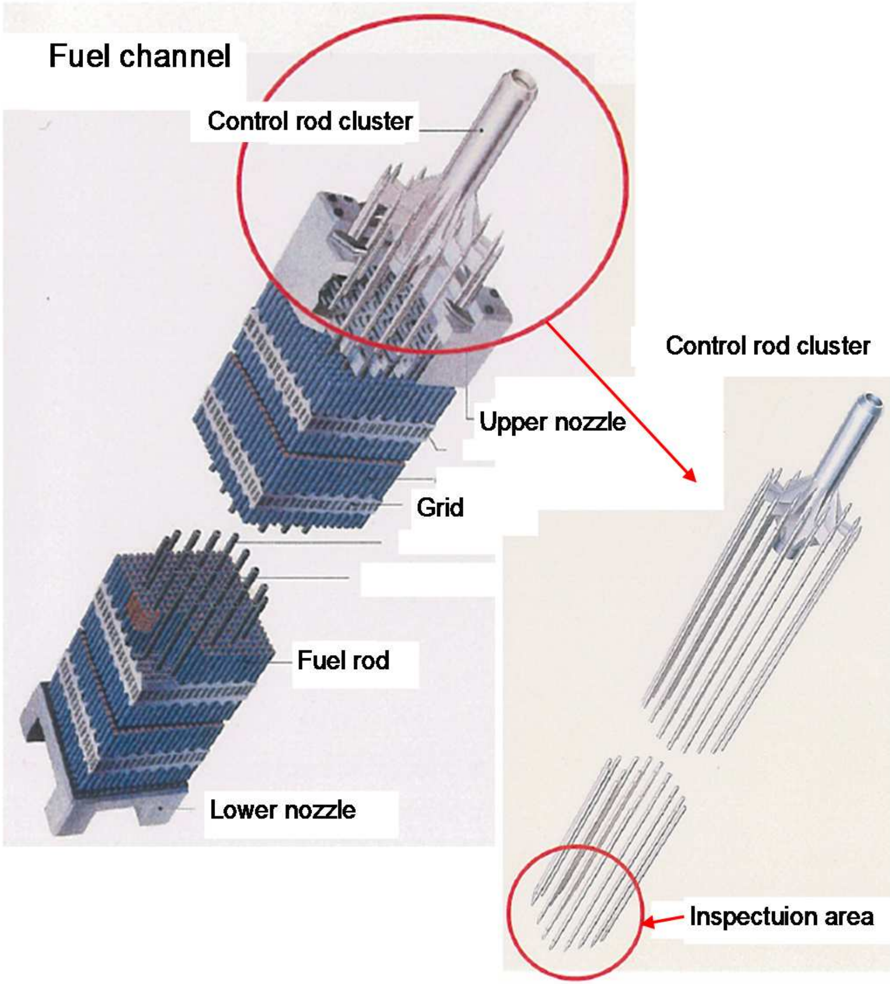
■ Tip of the CR cluster contacted with CR guide tube by small vibration due to water flow.

Components	Control rod (CR) cluster in fuel channel.
Plant	Genkai-4 (PWR)
Date	September 19, 2010
Plant condition	Periodic in-service inspection
Event	Wear at the tip of the control rod cluster
Reference	NUCIA, 2010-Kyusyu-M001 Rev. 1



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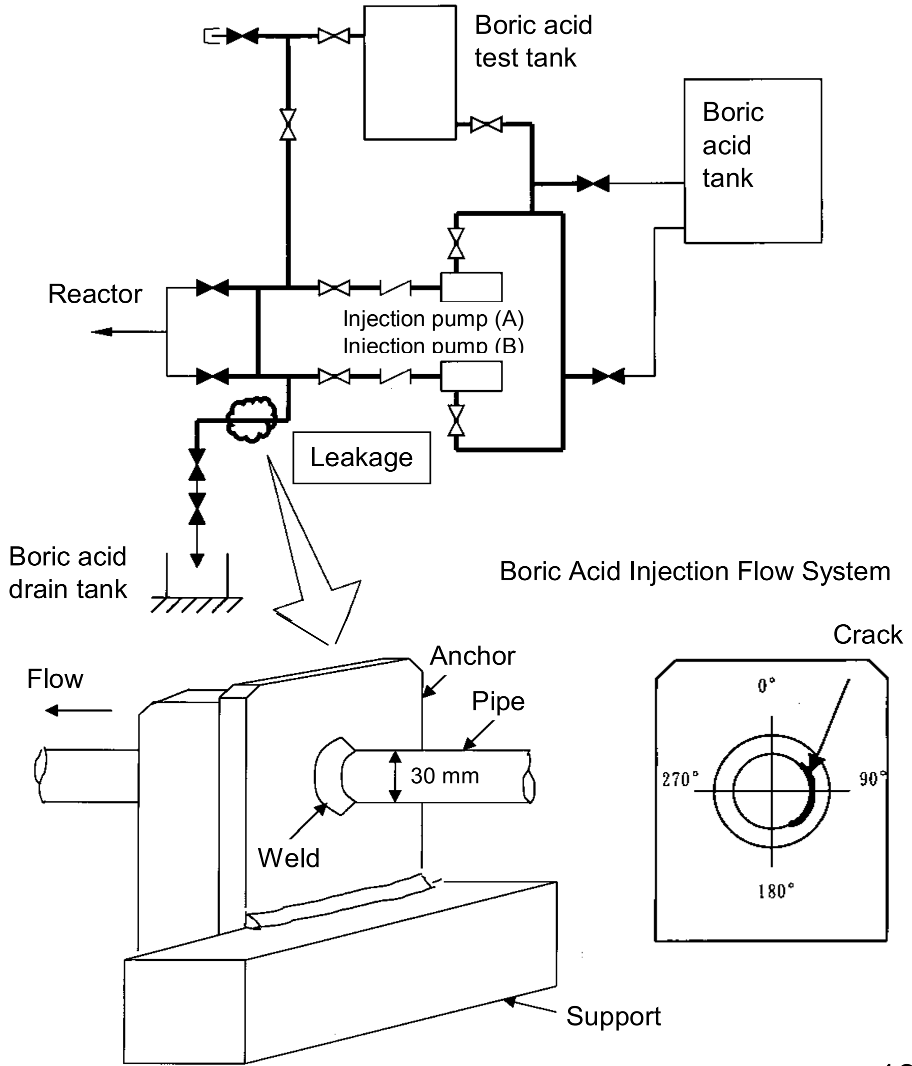
The drain pipe was vibrated by pulse pressure and pump body motion.

Component	Drain pipe for boric acid injection system
Plant	Kashiwazaki-Kariwa -1 (BWR)
Date	September 30, 2003
Plant condition	Periodic in-service inspection
Event	Leakage at welded joint of boric acid drain pipe
Reference	NUCIA, 2003-Tokyo-M020



Plant Issues (3) Pump and Piping System

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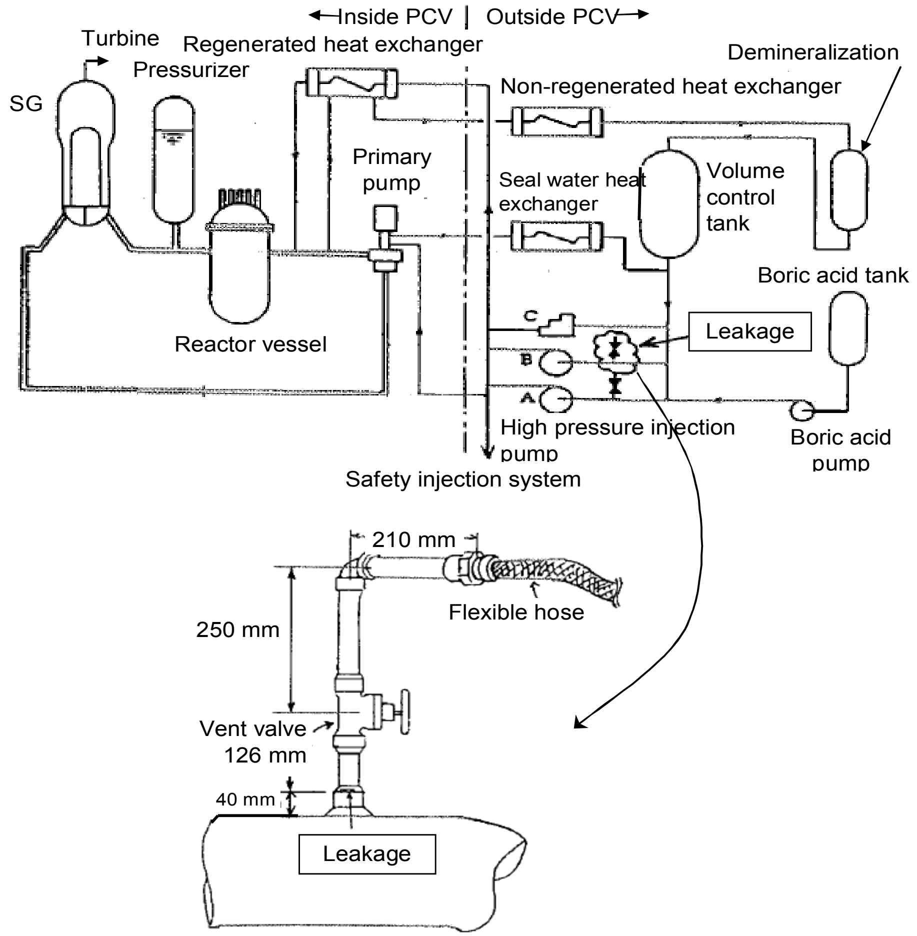
■ Vent pipe resonated with random vibration during minimum flow operation.

Component	Gateway Bent Valve at High Pressure Injection Pump in PWR
Plant	Ooi-2 (PWR)
Date	March 24, 1998
Plant condition	Rated load operation
Event	Crack was observed at welded joint.
Reference	NUCIA, 1998



Plant Issues (4) Jet Pump Instrumentation Pipe

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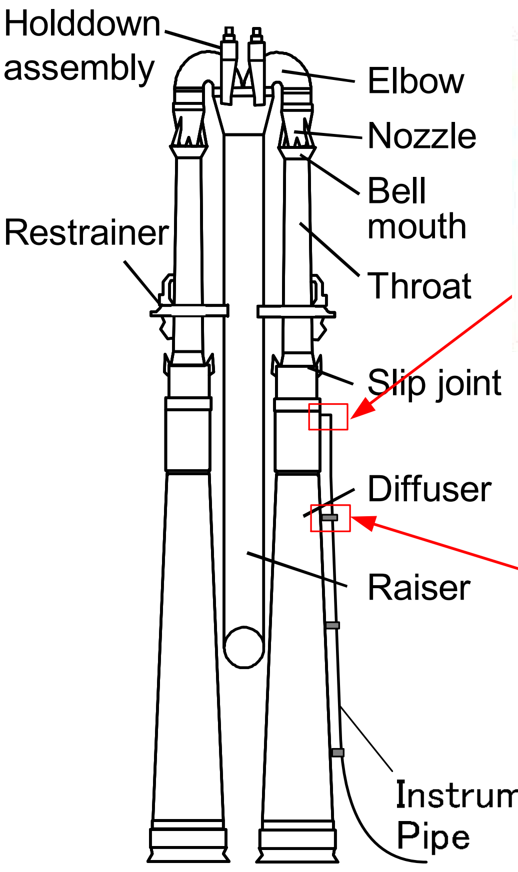
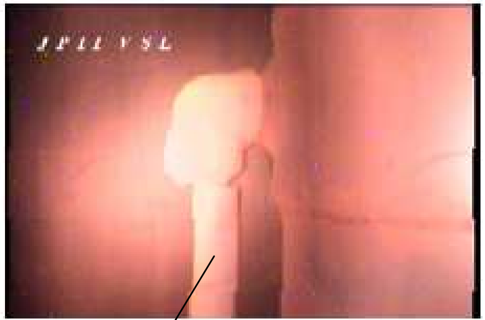
■ PLR pump pulsation resonated with natural frequency of pipes.

Component	Jet pump (JP) instrumentation pipe
Plant	Tokai-2 (BWR)
Date	Nov. 22, 2006
Plant condition	Flow control valve was at low flow gate opening and PLR pumps were at high speed.
Event	Ruptured pipes were observed at welded location by inspection.
Reference	NUCIA, 2006-Genden-M016

PLR: Primary Loop Recirculation



Overview of JP



Instrumentation Pipe

Summary

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This paper introduced nuclear power plant issues described in IAEA draft technical guidelines.

✓ There have been many problems caused by FSIs in nuclear power plants. In order to avoid FSI ~~problems~~ issues for newly designed components, it is important to learn from past FSI events.

✓ IAEA draft technical guidelines provide the design guide to predict the critical velocity at the onset of resonance ~~. The critical velocity~~ for the thermometer well, BWR dryer, PWR steam generator tube and so on. ~~will be shown in~~ The ~~IAEA technical~~ guidelines is also ~~taking~~ included past FSI events ~~into consideration~~.

✓ If the designs of the reactor internals are for prototypes or the operating conditions are changed, new analyses, tests and site measurements should be used for evaluation of FSIs. It is also important for the technical guidelines to show how to implement the analyses, tests and site measurements.

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Nuclear Power Plant Issues in IAEA Technical Guidelines on Fluid-Structure Interactions

2014/07/20

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Plant Issues (2) Cluster Guide Tube

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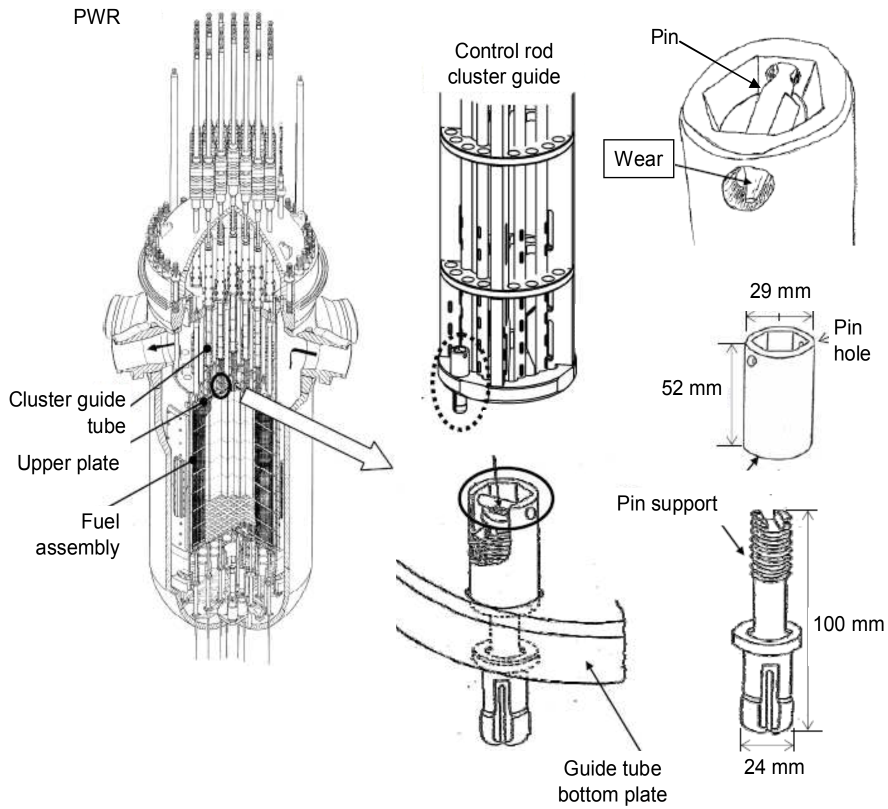
■ Stopper pin was vibrated by flow of primary coolant water and wear had occurred.

Component	Cluster guide tube
Plant	Tomari -1 (PWR)
Date	January 22, 2010
Plant condition	Periodic in-service inspection
Event	Wear by flow of primary coolant water
Reference	NUCIA, 2009-Hokaido-M009



Plant Issues (4) Pump and Piping System

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■ Eigen frequency for flashing pipe almost coincided with pulsation of the pump.

Component	Pump in PWR reactor auxiliary cooling water system
Plant	Ooi-3 (PWR)
Date	February 15, 2002
Plant condition	Rated load operation
Event	Leakage at welded point of flashing pipe
Reference	NUCIA , 2001-Kansai-M009