ASME 2014 PVP: FSI-9-2 2.3I IAEA Activities on Fluid Structure Interaction for Nuclear Power Plant

Overview of draft technical guidelines on Fluid-Structure Interaction

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Outline of Presentation

1. Background for developing document
2. Overview of draft technical guidelines

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Background

- In nuclear power plants, **fluid–structure interactions (FSI) is being occurred in operating systems, structures, components (SSCs) can cause excessive force or stress to the structures resulting in mechanical damages** that may eventually threaten the structural integrity of components because all SSCs filled with fluid.
  - SSCs are determined by the dynamic behaviour of both the fluid and the structure in the design.

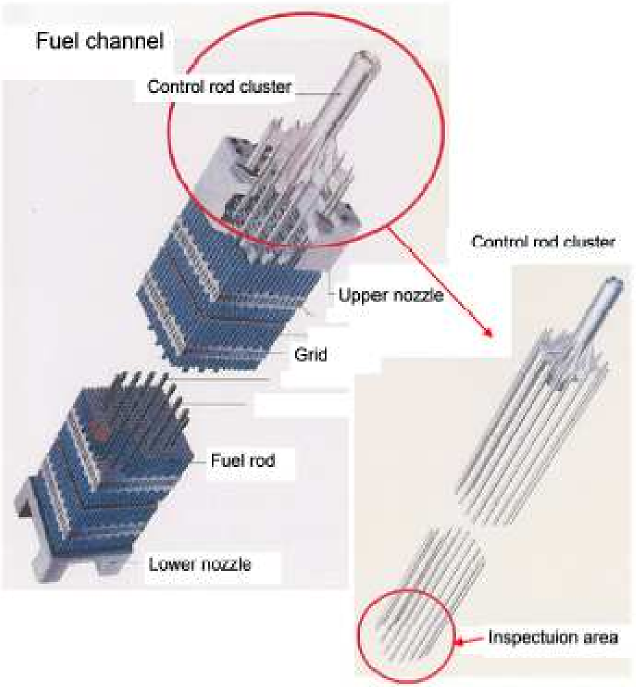
- Fluid-structure interaction (FSI) problems characterize **depending on the type of the external excitation, the shape and modal behaviour of** the structure, and fluid.

The basic problem of fluid-structure interaction involves the evaluation of the hydrodynamic pressure distribution, forces, moments and natural frequencies of the free-liquid surface.

Example: Mechanical Damages due to FSI (1)

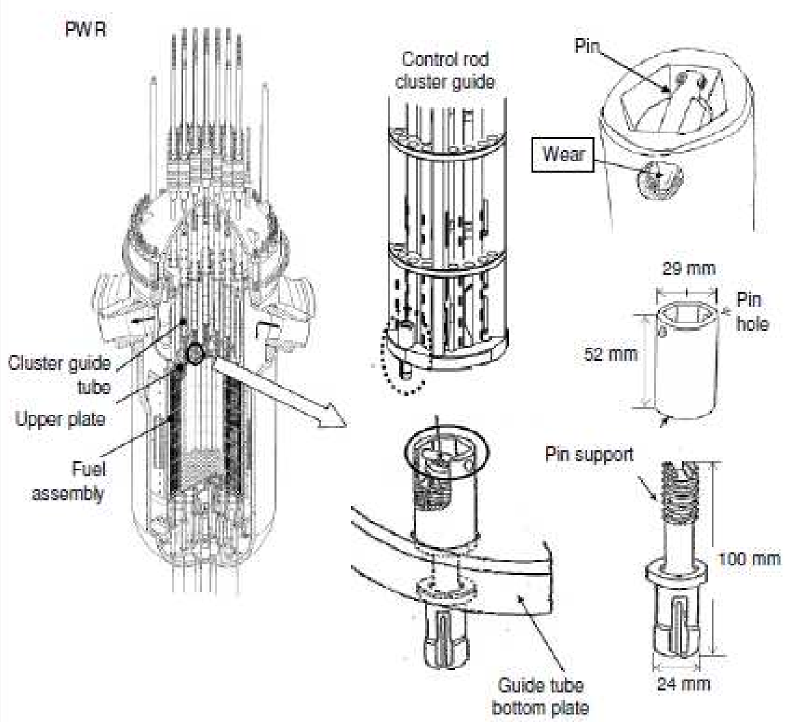
No. A-XX	Control Rod Cluster		
Plant	Genkai- 4 (PWR)	Date	September 19, 2010
Plant condition	Periodic in-service inspection		
Event	Wear at the tip of a control rod cluster		
Component	Control rod cluster in a fuel channel		
Summary	At the examination of wear measurement for control rods during periodic in-service inspection, one control rod cluster not satisfying standard dimensions was detected.		
Cause	The tip of the control rod cluster made contact with the control rod guide tube by small vibration occurred by water flow in reactor vessel. It is estimated that the wear at the tip occurred by the small vibration.		
Countermeasure	Replace the control rod cluster.		
References	NUCIA, September 19, 2010		

Figure



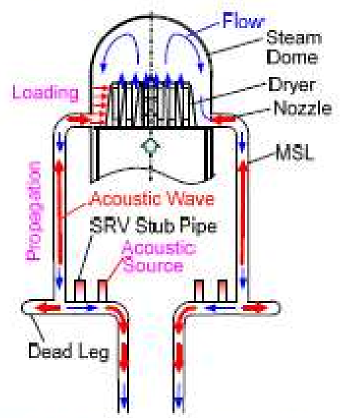
No. A-XX	Cluster Guide Tube in Core Internals		
Plant	Tomari - 1 (PWR)	Date	January 22, 2010
Plant condition	Periodic in-service inspection		
Event	Wear by flow of primary coolant water		
Component	Stopper pin and pin hole for control rod cluster guide tube in PWR core internals		
Summary	During periodic in-service inspection at control rods in reactor vessel, wear was detected at hole for pin and pin itself by underwater camera.		
Cause	Stopper pin was vibrated by flow of primary coolant water and wear had occurred.		
Countermeasure	Improved pin was replaced.		
References	NUCIA, 2010		

Figure

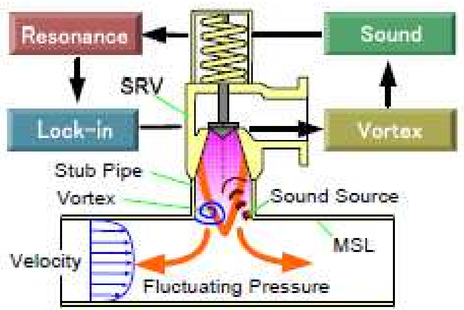


Example: Mechanical Damages due to FSI (2)

No. A-XX	Failure of steam dryer
Plant	Quad City Unit 2 (BWR)
Plant condition	Extended power up-rate — Date: needed
Event	A significant increase in steam moisture in the main steam lines (MSLs). Failure of the steam dryers and valves
Component	BWR main steam systems
Summary	The steam dryers had been damaged by high cycle fatigue due to acoustic-induced vibration in safety relief valve (SRV) stub pipes..
Cause	The principal cause of the dryer failure was flow-induced acoustic resonance in SRV stub pipes. Fluctuating pressures generated in the SRV stub pipes propagated from the SRVs to the dryer through the MSLs and caused dryer loading.
Countermeasure	Acoustic side branches were installed for silencer on the SRV stub pipes.
Reference	G. DeBoo, K. Ramsden and R. Gesior, "Quad Cities Unit 2 Main Steam Line Acoustic Source Identification and Load Reduction", *ICONE*14-89903, 2006.



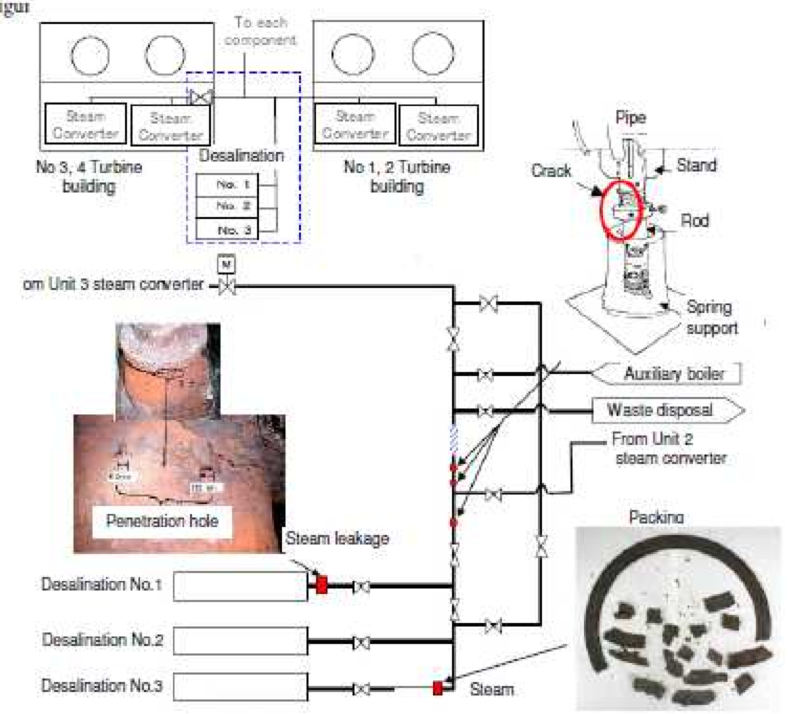
Propagation of fluctuating pressures to BWR dryer.



Flow-induced acoustic resonance at the SRV stub pipe

No. B-	Auxiliary Steam Line Pipe in Seawater Desalination Unit (PWR)
Plant	Ooi-1, 2 (PWR) — Date: December 10, 2005
Plant condition	Unit 1 Periodic inservice inspection, Unit 2 Rated load operation
Event	Damage of pipe support and steam leakage at flange of auxiliary steam line
Component	Auxiliary Steam Line Carbon Steel Pipe
Summary	Inspector heard abnormal noise and found steam leakage around desalination facilities located outside building. External wall thinning and small hole were found at desalination No.1 and flange packing were broken near desalination No.3.
Cause	When drain trap valve was closed and drain water was remained in auxiliary steam pipe, auxiliary steam was operated to pass through. Low temperature drain water entered high temperature auxiliary steam pipe. Water hammer was occurred by rapidly condensed steam slot. Due to the condense, water hammer damages support, and pressure decreased in auxiliary steam line. Steam supplied from Units 2 and 3 increased and pressure increased. Due to the pressure increase, steam leakage occurred.
Countermeasure	Check valves open/close before operation
Reference	NUCIA, February 3, 2006

Figur



Background (cont'd)

- Nuclear plants should be designed to be highly secure and capable to withstand a wide range of **internal and external extreme accident l**oads, such as earthquakes, tsunamis, etc., whose intensity exceed the design level one.

- However, in Fukushima Daiichi accident, it was experienced that the external and internal flooding after Tsunami propagates to the SSCs , where they induced **failure of all operable SSCs** at that moment.

- After Fukushima accident, the question was raised : **NPP response in beyond design earthquake conditions** is acceptable ??

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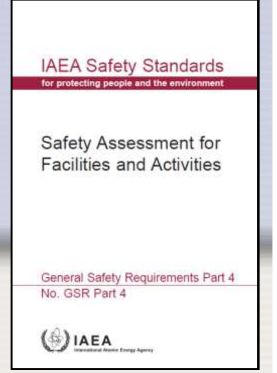
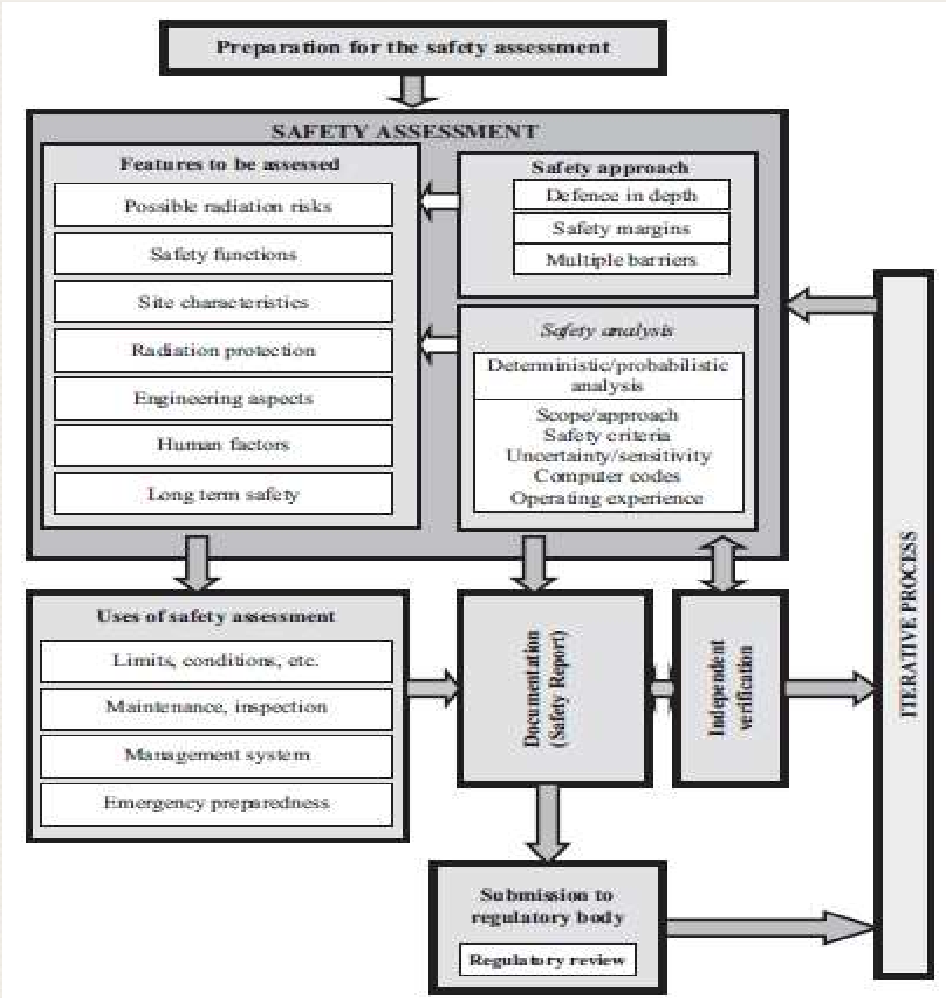
IAEA GSR Part 4Safety Assessment for Facilities and Activities



SPECIFIC REQUIREMENTS

- **Safety assessment and verification requires that a systematic evaluation of all features of the facility or activity relevant to safety be carried out, and includes:**



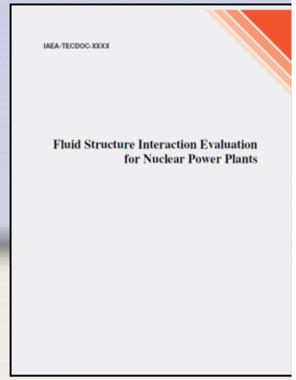
IAEA SSR-2/1 Safety of Nuclear Power Plants: Design

Requirement 10: Assessment of engineering aspects

It shall be determined in the safety assessment whether a facility or activity uses, **to the extent practicable, structures, systems and components of robust and proven design.**

- This could **include consideration of specific loads and load combinations**, and environmental conditions (e.g. temperature, pressure, humidity and radiation levels) **imposed on structures and components as a result of internal events, such as pipe breaks, impingement forces, internal flooding** and spraying, internal missiles, load drop, internal explosions and fire.

Developing IAEA Technical Guideline on FSI


IAEA-TECDOC-XXXX

Fluid Structure Interaction Evaluation for Nuclear Power Plants

- The objective of is focused on developing **technical guidance on analysis for FSI phenomena and evaluation for NPPs**.
- The following are considered within **the scope** of the guidelines:
  - **definition of fluid structure interaction** and related concepts;
  - **state-of-the-art technology** of fluid structure interaction assessment;
  - **fluid structure interaction evaluation methods**; and
  - **guidance to predict and analysis** the FSI phenomena.
  - covers **all reactor types**, at least in generic descriptions of FSI and the document summarizes available guidance.

Outline of Presentation

1. Background for developing document
2. Overview of IAEA technical guidelines (draft version)

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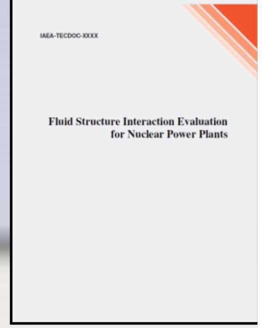
Process Followed to Produce Guidelines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Fluid Structure Interaction Evaluation
for Nuclear Power Plants

- **At an initial meeting on September 2011 in Vienna**
  - **the structure and initial contents of the document were discussed**, potential contributors to the document were identified and a very preliminary draft document was produced.
- **2nd meeting was held on May 2012 in Tokyo**
  - where the structure and initial contents of the guidelines were reviewed and drafted.
- **3rd meeting was held on July 2013 in Vienna**
  - a near-complete draft of the proposed guidelines
  - independent peer review in mid-2014 and production processes to be applied by IAEA in 2015.

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Contents of The IAEA Guidelines on FSI

- The contents of the IAEA guidelines on FSI are listed below

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2. General Description of FSI

- Section 2 of the guidelines gives **a general description of FSI**.
- **Section 2.1 covers the added mass, damping and stiffness** which need to be obtained by using appropriate analyses or experiments when evaluating flow-induced vibrations.
- **Section 2.2 addresses vortex induced vibration** considering a single cylinder and a tube bundle in particular.
- **Section 2.3 described self-excited vibration**, which can cause transients of nuclear power plant due to its large amplitude which could be led the pipe break.

2. General Description of FSI (cont'd)

- **Section 2.4 describes forced vibration** through a number of safety cases relevant to nuclear plant:
  - turbulence-induced vibration of a tube in cross flow; turbulence induced vibration of a tube bundle in cross flow; turbulence induced vibration of a tube and a tube bundle in parallel flow; a pipe downstream of a pressure control valve; and periodic fluctuation in a pipe downstream of a pump.
- **Section 2.5 addresses water hammer (or hydraulic shock)** which is the momentary increase in pressure which occurs in a water system when there is a sudden change of velocity of the water.
  - This sub-section of the guidelines describes both causes of water hammer and finding practical solutions.

3. Design and Maintenance against Fluid Structure Interaction

- **Section 3** is a comprehensive section **describing design and maintenance against FSI.**
- **Section 3.1**, which describes **flow induced vibration in nuclear components**, covers industrial codes for flow induced vibration.
  - It addresses **vortex shedding, fluid elastic instability, turbulence induced vibration and axial flow**.
  - It also addresses **how to evaluate the stress and fatigue strength** under flow induced vibration.
- **Section 3.2** addresses **fuel assemblies** and **tubes in steam generators** while **Section 3.3** addresses the **thermowell**, covering how to avoid or suppress vortex-induced vibrations, calculation of bending stress due to drag force and evaluation of stress amplitude by turbulence-induced random vibration.

3. Design and Maintenance against Fluid Structure Interaction (cont'd)

- **Section 3.4** addresses a number of issues related to **Reactor Pressure Vessel (RPV) internals,** such as **flow induced vibration** of reactor internals in PWRs and BWRs, including down-comer regions, control rods, guide tubes and instrumentation pipes.
- **Section 3.5** describes design and maintenance against **acoustic fatigue** and **water hammer** in pipes as well as **the pipewhip jet forces** which occur in the event of **pipe failure**.
- **Section 3.6** addresses **valves**, which are another important component for design against FSI, where **turbulence-induced valve vibration, self-excited vibration of flow-restriction valves** and **cavity tone excitation of relief valves** need to be considered.

4. Plant Issues

- **Section 4** on plant issues provides publicly **available examples** of flow-induced vibration, water hammer, acoustic fatigue and vibration in a range of nuclear plant.
- In addition, the guidelines **contain appendices which summarize events** in a consistent format, each summary providing information on the plant including a schematic figure, the event and its date, the cause of the event, countermeasures and references for more information.

4. Plant Issues (cond't)

Table: Some Events Described in the Guidelines

Plant	Component	Event/cause
PWR	Control rod cluster	Wear from flow-induced vibration
BWR	Steam dryer	Failure due to flow-induced acoustic resonance
BWR	Drain pipe	Leakage due to high cycle fatigue as a result of vibration induced by pulse pressure and pump body motion.
PWR	Cooling water pipe	Leakage due to high cycle fatigue as a result of the natural frequency of a pipe being almost coincided with the pulse frequency of a pump
PWR	Valve at high pressure injection pump	Leakage due to high cycle fatigue as a result of pipe resonance with random vibration during minimum flow operation
PWR	Steam line pipe in seawater desalination unit	Water hammer leading to steam leakage
PWR	Auxiliary steam supply pipe in turbine building	Water hammer leading to pipe support failure

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5. Computational Methods

- **Section5** gives a **general description of computational methods for FSI analysis and this information.**
- **Section 5.1** addresses **mathematical equations and numerical modelling** for solving the governing equations of fluid flow in conjunction with the structure's motion.
  - For example, unsteady simulations can be performed using Large Eddy Simulation or Direct Numerical Simulation at high Reynold's number flow, etc.
- **Section 5.2** describes **Computational Fluid Dynamic** Methods which are used for the numerical simulation of complex fluid flow.

Contributors to Drafting and Review

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Closing Remarks

- For the past three years, a small IAEA team has been **active in producing draft guidelines** on fluid-structure interaction for nuclear power plant.
- The guidelines are now **close to being complete**.
- It is very welcome to hear **ASME member's comments/suggestions** for further updating document.

Thank you – Questions/Comments?

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IAEA Safety Standard Framework

Principle 5 and 9

Comprehensive & detailed analysis required

2006

IAEA Safety Standards – Fundamental Safety Principles – Safety Fundamentals No. SF-1

Present the overall objectives, concepts and principles of protection and safety. They are the policy documents of the safety standards

FUNDAMENTALS

IAEA Safety Standards Series – Safety of Nuclear Power Plants: Design – REQUIREMENTS – No. NS-R-1

2000

IAEA Safety Standards – Safety Assessment for Facilities and Activities – General Safety Requirements Part 4 – No. GSR Part 4

2009

IAEA Safety Standards Series – Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport Safety – REQUIREMENTS – No. GS-R-1

Establish requirements that must be met to ensure the protection and safety of people and the environment, both now and in the future

REQUIREMENTS

Guidance on performing DSA and SA and their application

IAEA Safety Standards – Deterministic Safety Analysis for Nuclear Power Plants – Specific Safety Guide No. SSG-2

IAEA Safety Standards – Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants – Specific Safety Guide No. SSG-3

IAEA Safety Standards – Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants – Specific Safety Guide No. SSG-4

2009

2010

IAEA Safety Standards Series – Regulatory Inspection of Nuclear Facilities and Enforcement by the Regulatory Body – SAFETY GUIDE – No. GS-G-1.3

GUIDES

Provide guidance, in the form of more detailed actions, conditions or procedures that can be used to comply with the Requirements

IAEA SAFETY STANDARDS

Safety margins of operating reactors

SAFETY REPORTS SERIES

TECHNICAL DOCUMENTS (TECDOC)

Practical examples and detailed methods for the application of the Safety Standards. Detailed Technical Reports

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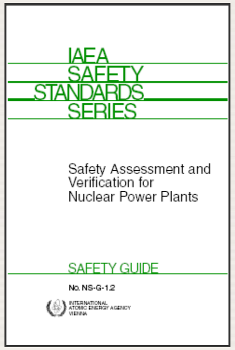
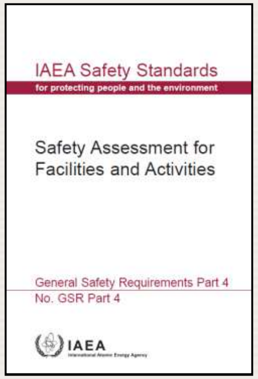
IAEA "Assistance" in Nuclear Safety


IAEA
SAFETY
STANDARDS
SERIES
Safety Assessment and Verification for Nuclear Power Plants
SAFETY GUIDE
No. NS-G-1.2

[Requirement]
Safety of NPPs: Design
No.SSR-2/1


IAEA Safety Standards
for protecting people and the environment
Safety Assessment for Facilities and Activities
General Safety Requirements Part 4
No. GSR Part 4
IAEA

[Requirement]
Safety Assessment and Verification for Nuclear Facilities
No, GSG-4

Other Safety Guides →

[Guide]
Safety Assessment & Verification of NPPs
(NS-G-1.2)

[Guide]
Deterministic Safety Analysis & Applications for Nuclear Reactors
(SSG-2)

[Guide]
Development and Application of Level 1 PSA for Nuclear Reactors
(SSG-3)

[Guide]
Development and Application of Level 2 PSA for NPPs
(SSG-4)

[Guide]
Severe Accident Management Programme for NPPs
(NS-R-2.15)

[Guide]
Safety Goals

[Guide]
Risk Informed Decision Making Process
(DS365)

1) Code V&V (peer review by ASME)
2) FSI

1)Uncertainty
2)Fuel Behaviour
3)Source Term
4)Fuel Quality

1)PSA Quality
2)Plant Specific Data
3)PSA Uncertainty
4)Design Evaluation

1)Risk Monitors
2)SPI (CRP)
3)PSA:VVER1000
4)SPI (Regulatory)
5)PSA Procedure

1)H2 Mitigation
2)SAA:In-Vessel
3)SAA:Ex-Vessel
4)SAA:PHWRs
5)AA for AM