Safety Design Criteria (SDC) Development for Generation-IV Sodium-cooled Fast Reactor System

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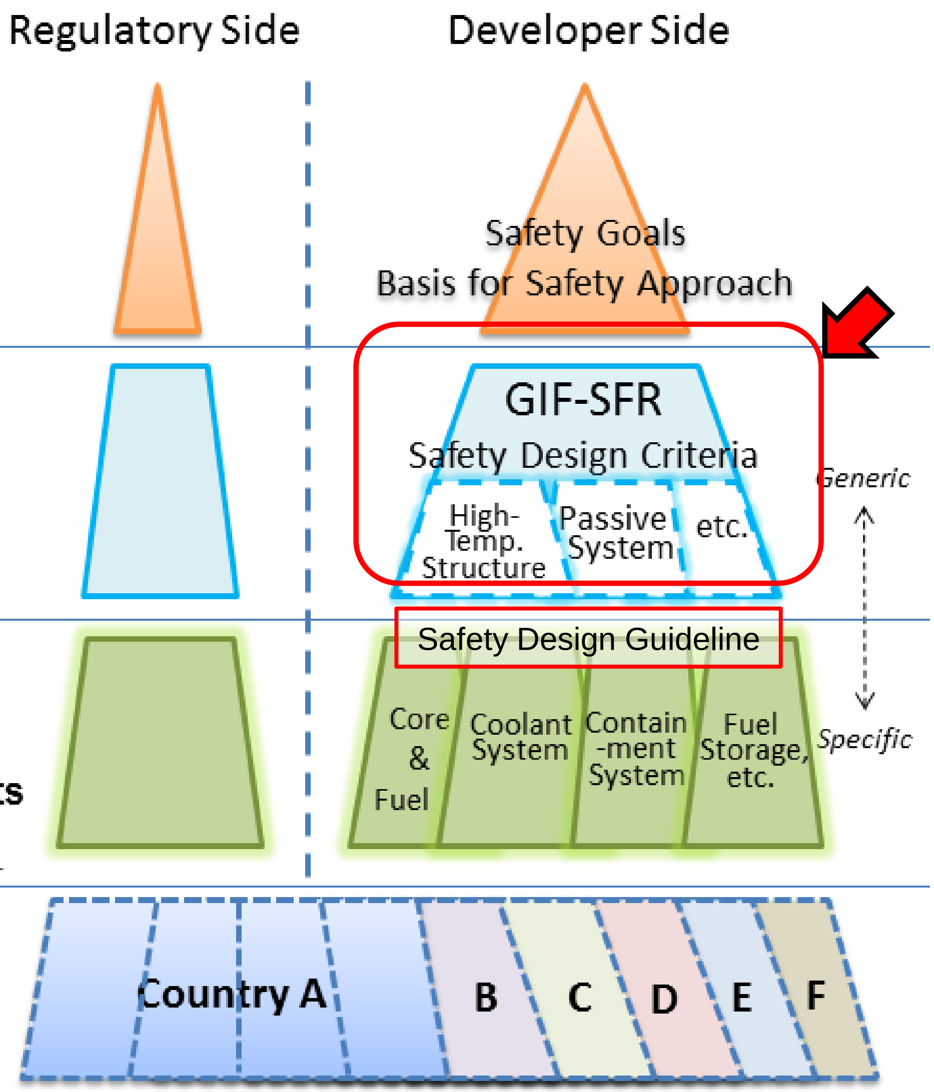
Background

- *Safety Design Criteria (SDC) development for Generation-IV systems was proposed at GIF Policy Group meeting in October 2010*
  - *SFR system was selected as the initial application since it represents one of the more mature next generation nuclear energy concepts*
    - » *Several prototypes being pursued by GIF member states*
- *Task Force (TF) started work in 2011 and completed SDC in 2013*
  - *Establish reference criteria for safety design of structures, systems and components*
  - *Achieve harmonization of safety approaches among GIF member states*
    - » *Realization of enhanced safety designs common to Gen-IV SFRs*
    - » *Preparation for upcoming licensing efforts*

Hierarchy of Safety Standards



Safety Fundamentals

e.g. IAEA SF-1: Fundamental Safety Principles

Safety Requirements

e.g. IAEA SSR 2/1*:
Safety of Nuclear Power Plants: Design

Safety Guides

e.g. IAEA NS-G-1.9*:
Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants

*These are for Light Water Reactors; but not for Gen-IV reactor systems.

Codes and Standards

Domestic Codes & Standards

GIF's Safety & Reliability Goals

SR-1: Excel in Operational Safety and Reliability

Safety and reliability during normal operation, and likely kinds of operational events that set forced outage rate

SR-2: Very low likelihood & degree of reactor core damage

Minimizing frequency of initiating events, and design features for controlling & mitigating any initiating events w/o causing core damage

SR-3: Eliminate the need for offsite emergency response

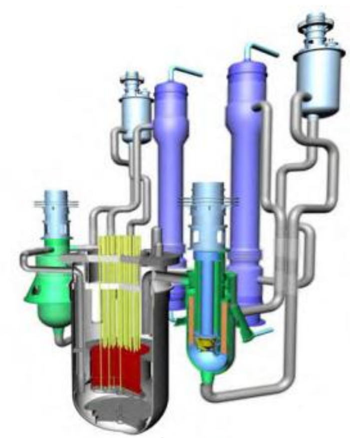
Safety architecture to manage & mitigate severe plant conditions, for making small the possibility of releases of radiation

GIF's Basic Safety Approach

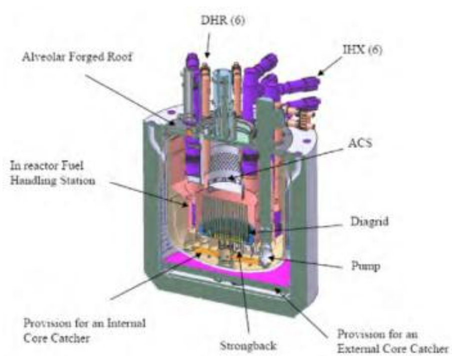
- ***Defence-in-depth***
- ***A combination of deterministic and risk-informed safety approach***
- ***Safety to be built-in to the design, not added-on***
- ***Emphasis on utilization of inherent and passive safety features***

SFR Design Tracks under GIF

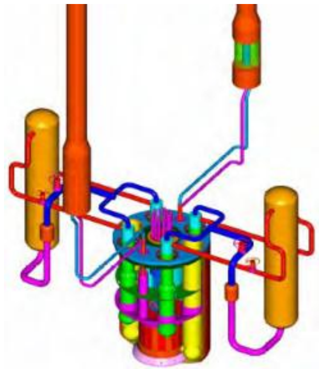
System structure	Loop-type, Pool-type, Small modular
Electric output	50 - 2,000MWe
Coolant system	Primary and secondary [intermediate] coolant system utilizing sodium coolant
BOP system	Water/Steam cycle (alternative concept: Supercritical CO_2 cycle)
Fuel	MOX, Metal, others



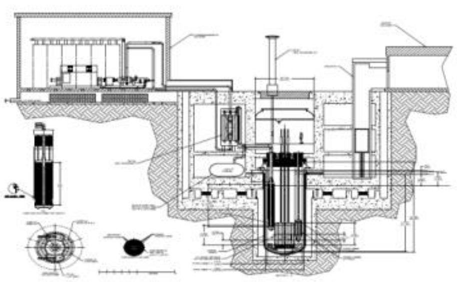
JSFR
[Large Loop]



ESFR
[Large Pool]



KALIMER
[Pool]



SMFR
[Small Modular]

Safety Advantages of SFRs

- *Low pressure primary and intermediate coolant system*
  - *Guard vessel and guard pipes to "maintain" coolant inventory*
  - *No LOCA concern, no ECCS, no risk for control-rod ejection*
- *Liquid-metal coolant with excellent natural circulation characteristics and a wide margin (~400 degC) to boiling*
- *Inherent safety with "net" negative reactivity feedback during accidents that lead to elevated core/coolant temperatures*
- *Dedicated systems for decay heat removal to an ultimate heat sink*
  - *Large difference between core outlet and inlet temperatures to facilitate reliance on passive systems*
- *Low pressure (~0.5 bar) design pressure for containment (mostly against heat from sodium fires)*
- *Much simpler operation and accident management (long grace period for corrective action)*

Challenges with SFRs

- ***High temperature (>500 degC core outlet temperature) and high core power density***
- ***Liquid sodium coolant that reacts with air, water and concrete***
  - ***These reactions have to be prevented and/or mitigated to avoid their effect on SSCs important to safety***
- ***Fast reactor cores are not in their most reactive configuration***
  - ***Relocation of core materials may lead to a hypothetical core disruptive accident (HCDA)***
- ***For large cores, sodium void worth can be positive***
- ***Opaque sodium coolant could pose in-service inspection and maintenance challenges***

SFR Safety Principals

- *Like LWRs, SFR safety is first based on utilization of multiple redundant engineered protection systems to lower the probability of accident occurrence and to limit its consequences:*
  - *independent scram systems,*
  - *multiple coolant pumps and heat transport loops, and*
  - *multiple barriers to release of radioactive materials.*
- *SFR safety analyses traditionally focus on ATWS during which the reactor scram system is assumed to fail.*
  - *Because HCDAs could potentially result in re-criticalities.*
  - *The safety design features that enhance inherent negative reactivity feedback and passive decay heat removal capabilities provide additional measures to prevent/mitigate HCDAs even during these very low probability accidents.*

Defence-in-depth (DiD) & Plant States

based on IAEA INSAG-12 & SSR-2/1

DiD Levels				
Level 1	Level 2	Level 3	Level 4	Level 5
plant states (considered in design)				Off-site emergency response (out of the design)
Normal operation	AOO	DBA	DEC	
Operational states		Accident conditions		
Normal operation	Anticipated operational occurrences	Design basis accidents	Design extension conditions (including Severe Accident conditions)	

Basic Scheme to outline the SDC

High level safety fundamentals, and safety design goals

- ■ *GIF's Goals for safety & reliability*
- ■ *Basis for safety approach for design & assessment*
- ■ *Requirements in SFR System Research Plan*

1) Particular issues for SFR

- ■ Characteristic of Sodium-cooled Fast Reactor
  - ➢ Reactivity (void) …
  - ➢ Sodium fire & Sodium-water reaction…
- ■ Consideration on Severe Accident
  - ➢ Re-criticality during Core Disruptive Accident
- ■ High Temperature & Low pressure system
  - ➢ Creep property, Leak-Before-Break…
  - ➢ No LOCA and no need of ECCS…
- ■ Enhanced Safety Approach
  - ➢ Passive system for shutdown & cooling

2) Reference of SDC Structure

IAEA SSR 2/1

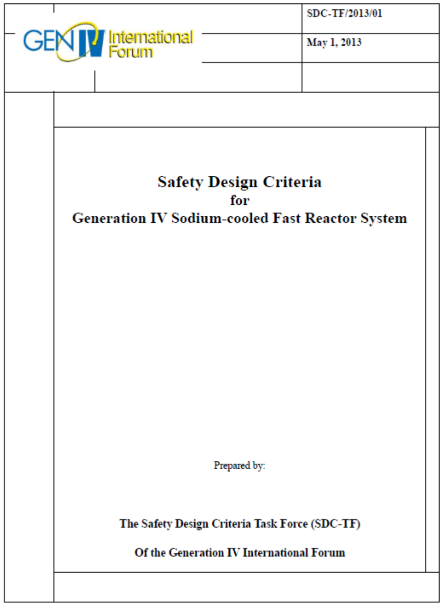
- ■ Management of safety in design
- ■ Principal technical requirement
- ■ General Plant design
- ■ Design of specific plant system

3) Lessons learned from Fukushima Dai-ichi NPPs accident

- ■ Common cause failure by external event
- ■ Loss of power for longer period
  - ➢ Decay heat removal, Fuel pool cooling
- ■ Containment function on spent fuel in the pool
- ■ Preparing multiple AMs, e.t.c.

GIF SFR SDC

Table-Of-Contents of "SDC Phase 1 Report"*


SDC-TF/2013/01

May 1, 2013

GEN IV International Forum

Safety Design Criteria
for
Generation IV Sodium-cooled Fast Reactor System

Prepared by:

The Safety Design Criteria Task Force (SDC-TF)

Of the Generation IV International Forum

*SDC-TF/2013/01, May 1, 2013

6. DESIGN OF SPECIFIC PLANT SYSTEMS

6.1 Overall Plant System	*Criterion 42bis*
6.2 Reactor Core and Associated Features	*Criteria 43-46*
6.3 Reactor Coolant Systems	*Criteria 47-53*
6.4 Containment Structure and Containment System	*Cri.54-58*
6.5 Instrumentation and Control Systems	*Criteria 59-67*
6.6 Emergency Power Supply	*Criterion 68*
6.7 Supporting Systems and Auxiliary Systems	*Cri.69-76bis*
6.8 Other Power Conversion Systems	*Criterion 77*
6.9 Treatment of Radioactive Effluents and Radioactive Waste	*Cri.78-79*
6.10 Fuel Handling and Storage Systems	*Criterion 80*
6.11 Radiation Protection	*Criteria 81-82*

GLOSSARY

APPENDIX:

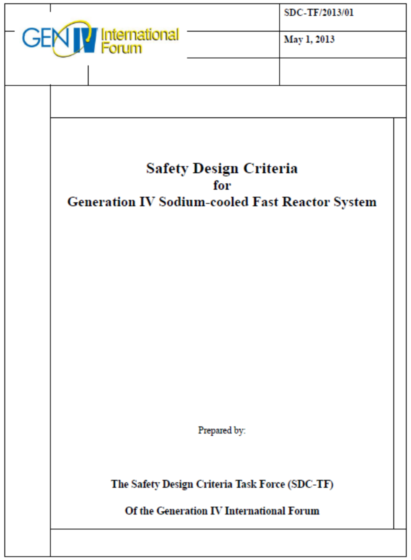
(A) Definitions of Boundaries of SFR systems

(B) Guide to Design Extension Conditions

(C) Guide to Practical Elimination of accident situations

(D) Guide to Utilisation of Passive/Inherent Features

(E) Approach to Extreme External Events


GEN IV International Forum

SDC-TF/2013/01

May 1, 2013

Safety Design Criteria
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Status of International reviews on SDC

- ***GIF SFR "SDC Phase 1 Report"***
  - ***Review requests for the SDC Report***
    - » ***For "Review by external organizations" and***
    - » ***For "Enhancing interaction with regulatory bodies"***
  - ***Sent the report (ca. July 2013) to***
    - » ***International organizations***
      IAEA, MDEP, OECD/NEA/CNRA
    - » ***Regulatory authorities at national level***
      China (NNSA), Euratom (ENSREG), France (ASN), Japan (NRA), Republic of Korea (NSSC), Russia (Rostechnadzor), USA (NRC)

Status of international reviews on "SDC"

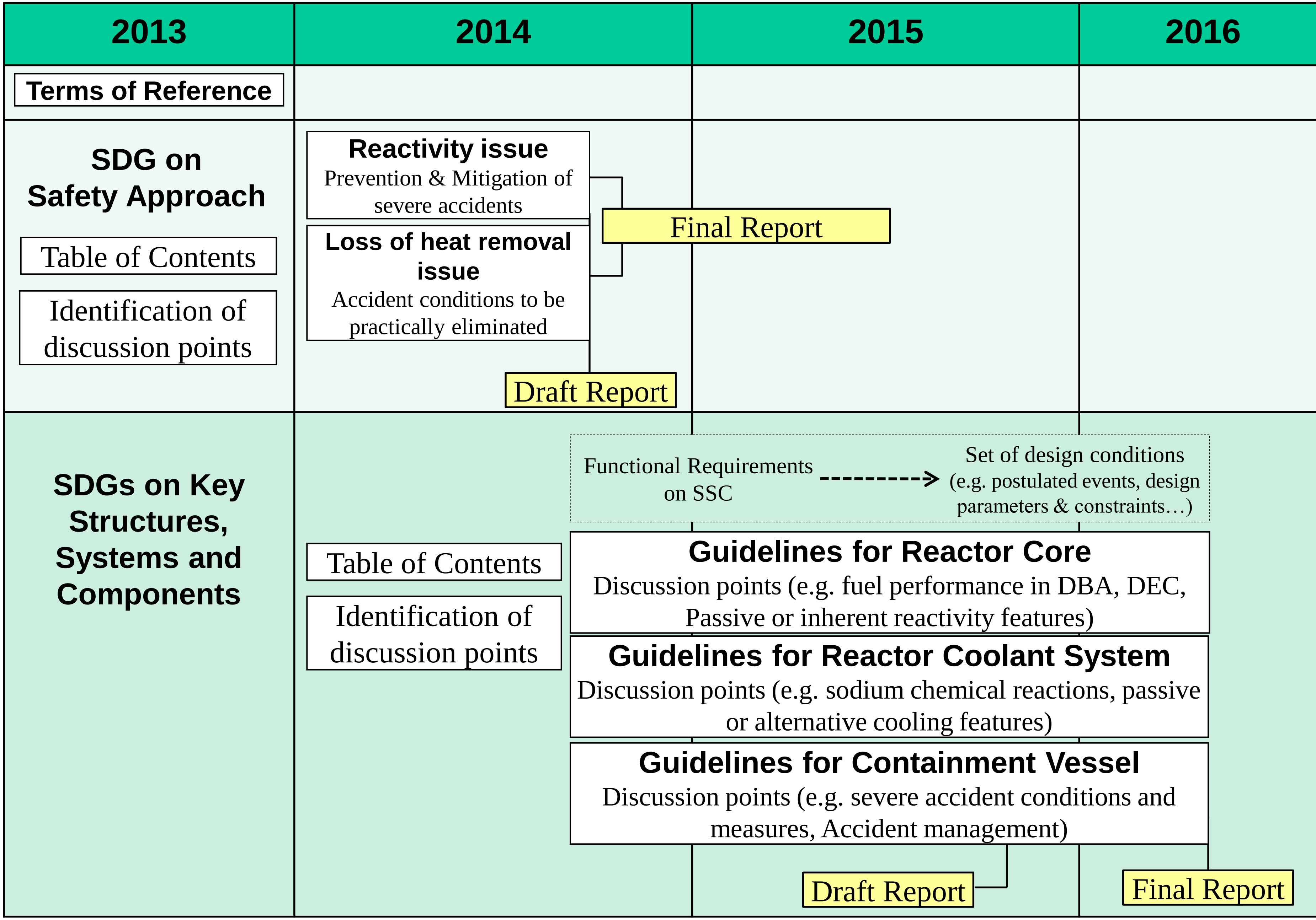
- ***NRC (USA)***
  - ***Comprehensive & detailed review, with proposals (Jan. 2014)***
  - ***GIF SDC Task Force prepared the resolutions to incorporate***
- ***NNSA (China)***
  - ***Review results (Oct. 2013 & Jan. 2014)***
  - ***GIF SDC TF resolution replied (Aug. 2014)***
- ***IRSN (France)***
  - ***Comments on interim version at the GIF-IAEA Safety Workshop (Feb. 2012), resolutions already included in Phase I report.***
- ***IAEA***
  - ***General and technically specific reviews (April 2014)***
  - ***GIF SDC Task Force prepared the resolutions to incorporate***

External feedbacks have been or are being incorporated

Safety Design Guidelines (SDG) Development

- ***Main objective***
  - ***to support practical application of SDC in design process for improving safety in specific topical areas***
    - » ***including use of inherent/passive safety features***
    - » ***design measures for prevention and mitigation of severe accidents.***
  - ***Initial topical areas are considered:***
    - » ***Particular importance since a fast reactor core is typically not in its most reactive configuration***
    - » ***Quantification of key criteria for safety improvement***

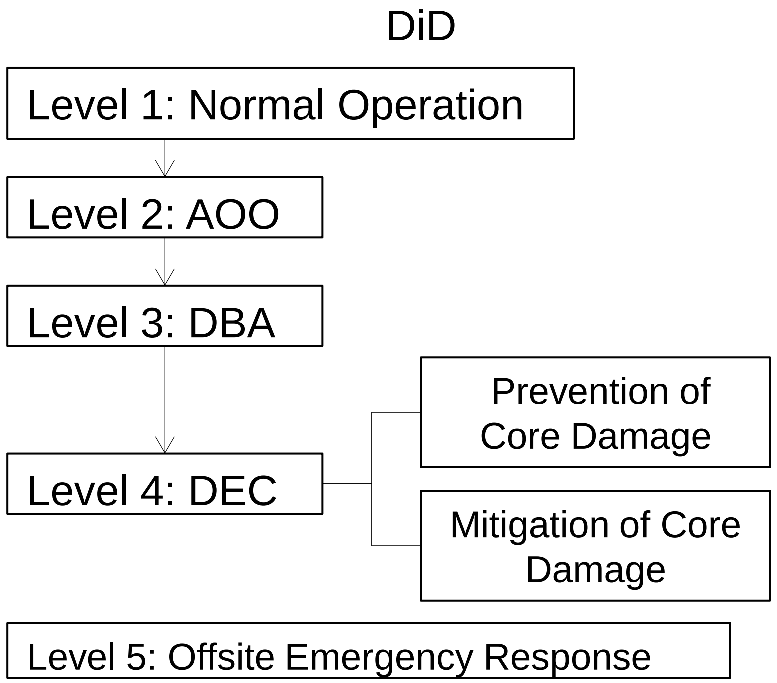
Schematic View of SDG Development Schedule



General Approach to Normal Operation, AOOs, and DBAs

- **Normal Operation**- Stable operation, with controlling reactivity, temperature, flow…

- **AOOs/DBAs**- Shutdown the reactor and maintain decay heat removal sufficient to keep reactor core and system temperatures within the applicable design limits.



General Approach to Design Extension Conditions

- **Prevention of Core Damage**
  - Accident sequences typically caused by failure of one or more systems related to safety
  - Postulated initiating events more severe than those in DBA
- **Mitigation of Core Damage**
  - Mitigation of consequences of postulated accidents where significant core damage may occur, with the objective of maintaining the containment function to limit radioactive releases.

Exploiting SFR Characteristics to Enhance Safety

- ***<u>Passive/Inherent safety for DEC</u>***
  - ***On reactivity***
    - » ***Inherent reactivity feedback** to reduce the power as core temperatures rise or*
    - » ***Passive mechanism** are applicable for shutdown systems, such as SASS, HSR, and GEM*
  - ***On decay heat removal***
    - » ***Natural circulation** of single phase sodium coolant*
    - » ***can be placed in different locations for enhancing diversity***

Exploiting SFR Characteristics to Enhance Safety

- ***<u>In-Vessel Retention</u>***
  - *In the course of core degradation during unprotected transients, measures should be provided to prevent prompt criticality*
  - *Reactor coolant boundary should maintain the boundary function against pressure load including fuel-coolant interaction*
  - *Measures should be provided for ensuring long term cooling of core materials inside the reactor vessel under sub-critical condition*

Practical Elimination of Accident Situations:

- ***Severe accidents with mechanical energy release higher than the containment capability***
  - *Power excursions for intact core situations*
    - » *Large gas flow through the core*
    - » *Large-scale core compaction*
    - » *Collapse of the core support structures*
- ***Situations leading to the failure of the containment with risk of fuel damage***
  - *Complete loss of decay heat removal function that leads to core damage and failure of primary coolant boundary*
  - *Core uncovering due to sodium inventory loss*
- ***Fuel degradation in fuel storage or during when the containment may not be functional due to maintenance***
  - *Core damage during maintenance*
  - *Spent fuel melting in the storage*

Design requirements on reactivity characteristics

Normal Operation, AOO, DBA, DEC w/o Core Damage

» ***Shall require inherent reactor power stability***

» ***Reactor Shutdown System shall prevent sodium boiling and maintain core coolable geometry***

Design Extension Condition with Core Damage

» ***Shall prevent excessive insertion of reactivity by coolant boiling, cladding and fuel relocation after core damage***

Quantification of requirement on reactivity characteristics

- **For Normal operation, AOO and DBA**
  - » *Power reactivity coefficient < 0 (Negative)*
  - » *Reactor shutdown capability with inherent feedback > Postulated reactivity insertion*
- **For Design Extension Condition**
  - » *Before core damage: same as the requirement for DBA,*
    - – *Achieved by passive measures or inherent features*
  - » *After core damage:*
    - – *Total reactor core reactivity < 1$ (below prompt criticality)*
  - » *Sodium void worth can be positive as far as the above conditions are satisfied.*

Concluding Remarks

- ***The "Safety Design Criteria Phase 1 Report"***
  - ***Issued by the GIF on May 2013***
  - ***Disseminated for international review to:***
    - » ***International organizations***
    - » ***National Regulatory Bodies***
  - ***Important feedbacks have been or are being incorporated:***
    - » ***e.g. IAEA, IRSN, USNRC, NNSA …***
- ***The "Safety Design Guidelines" development in Phase II***
  - ***Started from Sept. 2013***
  - ***Two Safety Design Guidelines (SDG):***
    - » ***Safety Approach and Design Conditions SDG in final drafting stage***
    - » ***Key Structures, Systems and Components SDG***

Thank you

for your attention !!

Difference between "GIF SDC Criteria" and "IAEA SSR 2/1 Requirements"

SDC Criteria (total 83): Modified 20, Added 2, Deleted 1, Un-changed 60
[Added: Overall Plant System & Sodium heating systems / Deleted: ECCS]

*M: Modified	A: Added	D: Deleted	U: Unchanged

Example:

IAEA SSR 2/1 Requirement #	paragraph #	GIF SFR SDC Criterion #	paragraph #	Status* M/A/D/U
6. DESIGN OF SPECIFIC PLANT SYSTEMS				
OVERALL PLANT SYSTEM				
		42bis		A
REACTOR CORE AND ASSOCIATED FEATURES				
43		43		M
	6.1		6.1	M
	6.2-6.3		6.2-6.3	U
44		44		M
			6.3bis	A
			6.3ter	A
			6.3quater	A
45		45		U
	6.4		6.4	M
	6.5		6.5	M
	6.6		6.6	M
			6.6bis	A
46		46		M
	6.7-6.8		6.7-6.8	U
	6.9		6.9	M
	6.10-6.12		6.10-6.12	U
REACTOR COOLANT SYSTEMS				
47		47		U
	6.13		6.13	M
	6.14		6.14	M
			6.14bis	A
			6.14ter	A
	6.15		6.15	M
			6.15bis	A
			6.15ter	A
	6.16		6.16	M
			6.16bis	A
			6.16ter	A
			6.16quater	A
			6.16quinquies	A
48		48		M
49		49		M
50		50		M
	6.17		6.17	M
			6.17bis	A
51		51		M
52				D [incl. in #51]
	6.18		6.18	M
	6.19		6.19	M
			6.19bis	A
53		53		M