REDUCTION OF RADIOLOGICAL ACCIDENT CONSEQUENCES

Title	**The Steam Generator Tube Rupture accident (SGTR)**
Speaker:	François Parmentier
Affiliation:	Bel V
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Commission européenne

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Table of contents

Acronyms

BWR : Boiling Water Reactor

CVCS : Chemical and Volumetric Control System

DBA : Design Basis Accident

FW : Feedwater

LOCA : Loss of Coolant Accident

NPP : Nuclear Power Plant

PWR : Pressurized Water Reactor

SAR : Safety Analysis Report

SGTR : Steam Generator Tube Rupture

Introduction and some reminders

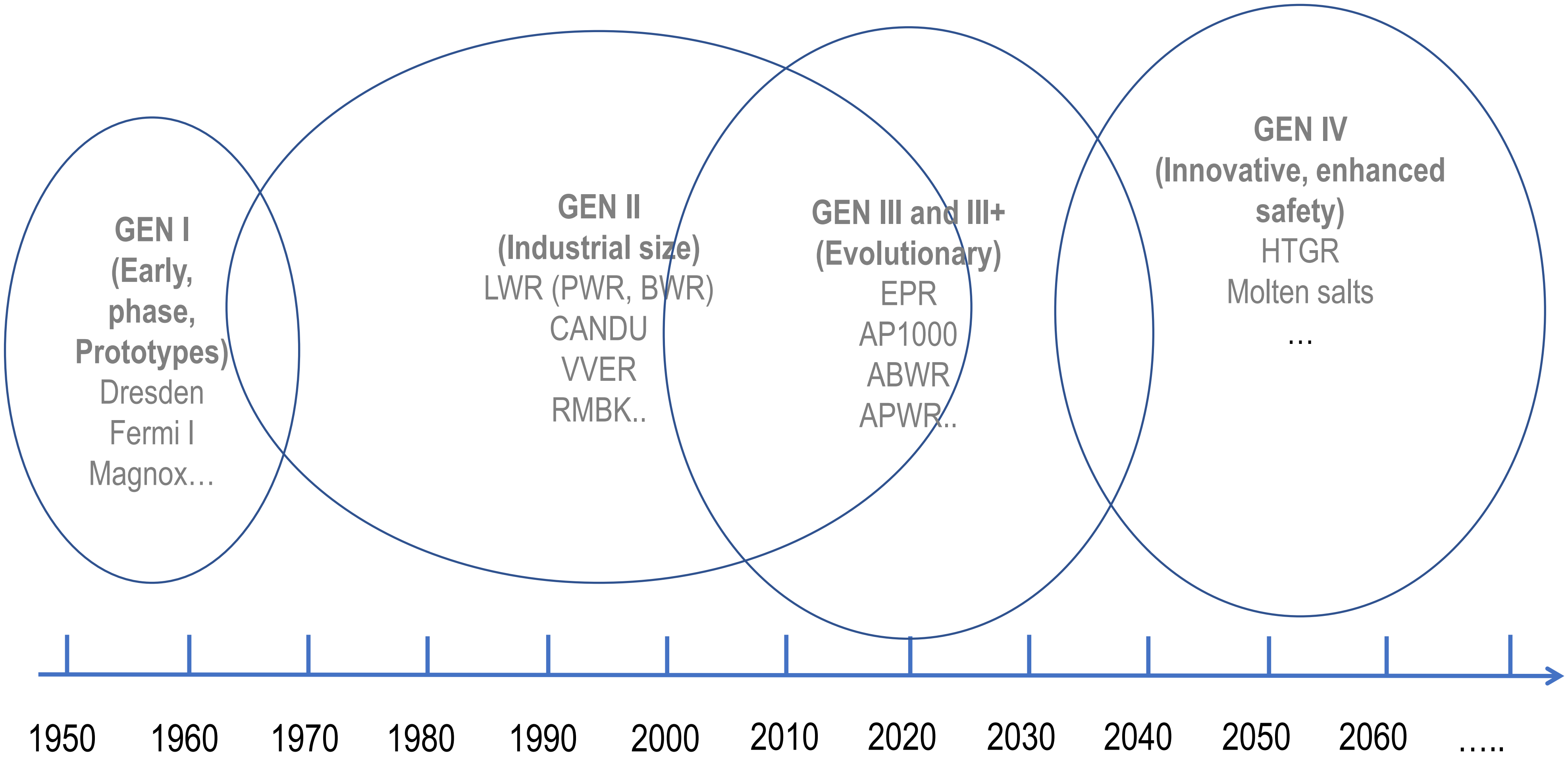
Aim of the presentation (30 min) : providing overview of the SGTR accident for PWR NPP's.

Not exhaustive → gives keys of developments to many aspects

The presentation begins with some reminders about NPP's.

History of NPP's (generations)

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS



Introduction and some reminders

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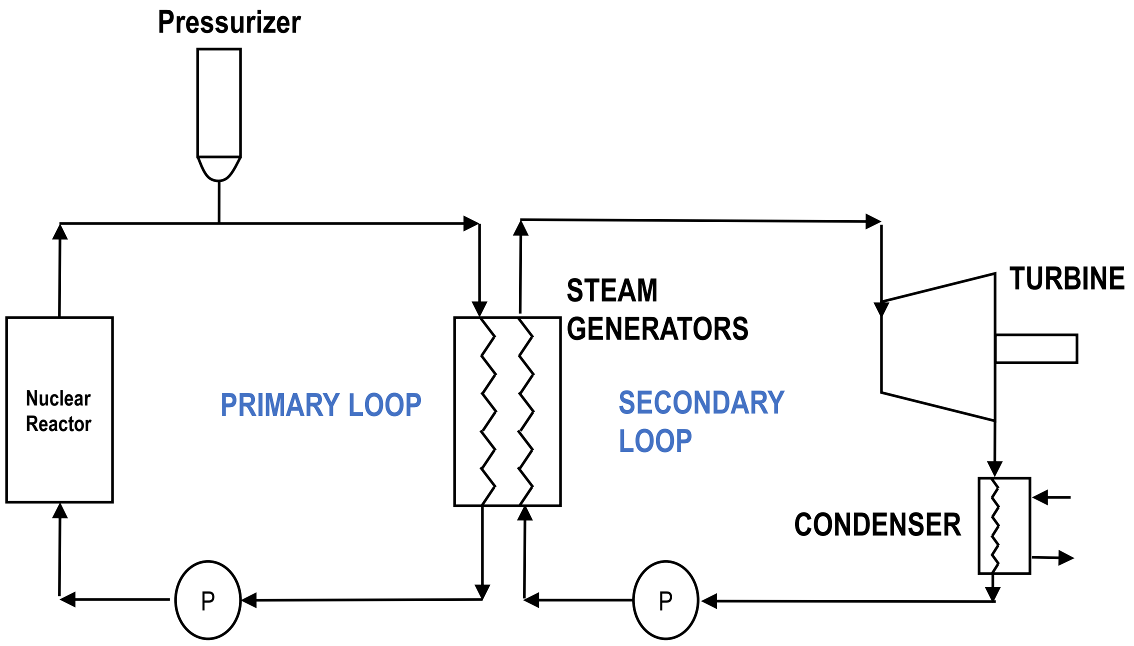
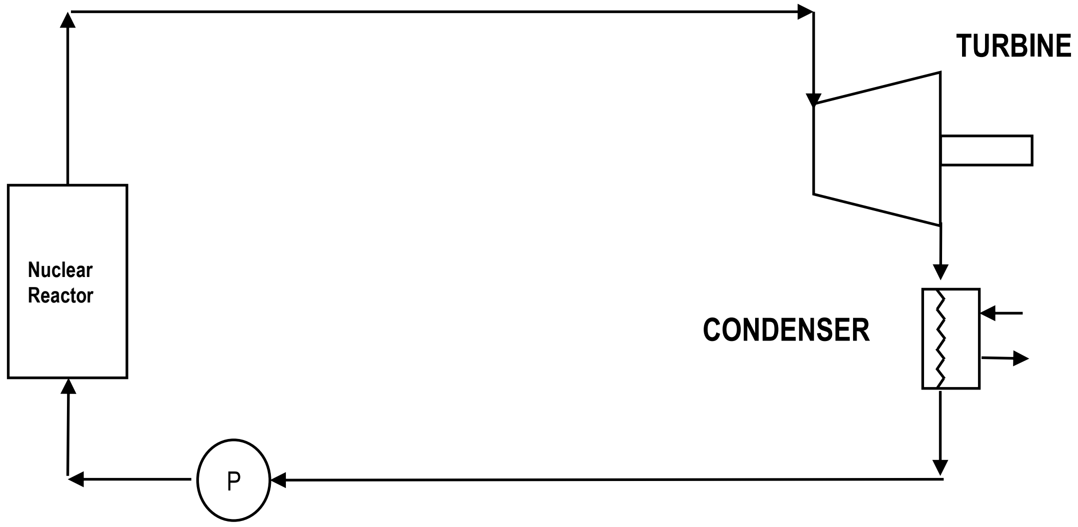
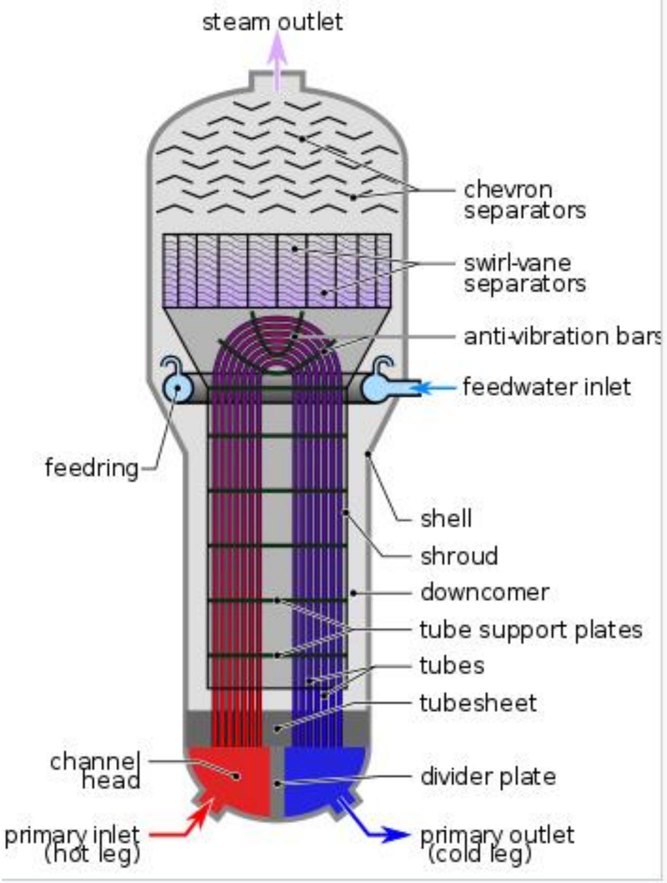
PWR's



BWR's



The steam generators of PWR's



2, 3, 4 SG's per NPP ; ~5000 tubes per SG ; Average tube length = 20 m ;

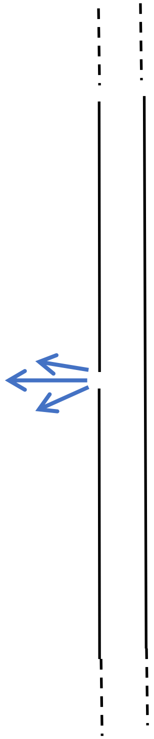
Tube ext. diam = 19 mm (3/4") ; Thickness tube = 1 mm ;

Material = Inconel 690 or incoloy 800

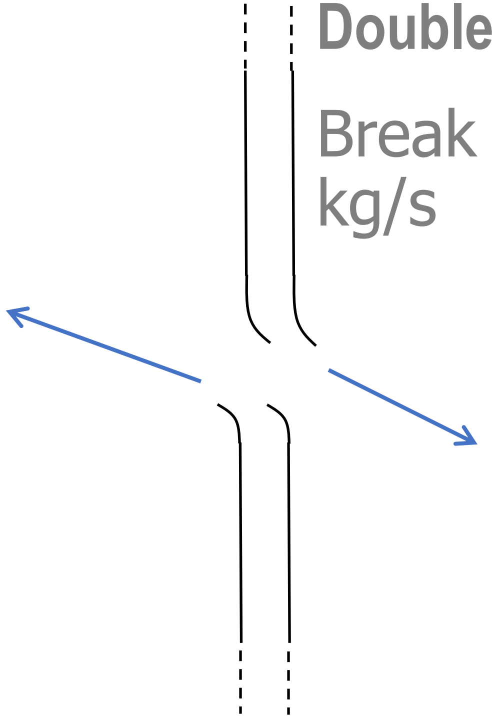
DP primary (tube)/secondary (sheet) : 75 to 95 bar

Leaks, partial break, double ended break

SGTR = a rupture of 1 tube. Can be partial or double ended (in this case both parts of the break are independent).



Leak or partial break



Double ended break

Break flow =~20 kg/s

Table of contents

SGTR : overview of different scenarios and related risks (1)

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Plenty of different SGTR scenario exist as function of :

- Importance of the break (partial, double ended, one tube or several tubes)
- Major features during the event that can affect the result : as behavior of the FW control (can compensate break flow or not)
- Realistic or conservative hypotheses. There are mandatory rules to respect for the safety demonstration which is presented in the SAR. (Ex : considering a single failure is mandatory for DBA's)

Following US regulation RG1.70, SGTR = part of the accidents to be studied in the SAR§15. Usually considered as a condition IV event ("Limiting faults" which is the most rare) → considered as such in the design of GEN II reactors

SGTR : overview of different scenarios and related risks (2)

Several kinds of risks are related to SGTR :

1. Core uncovery : at the origin, the SGTR was studied as a small LOCA → it was mainly studied if the core could be uncovered during the event and so if it was a threat for the fuel integrity
2. Radiological release : SGTR = accident during which **the 3 barriers are by-passed (→ also called a "By-pass accident") : fuel leak postulated + SGTR itself + discharge valves to atmosphere**
3. Affected SG overfilling : the steam lines are not designed to resist to the weight of water → could break

Typical primary and secondary pressures evolution during SGTR

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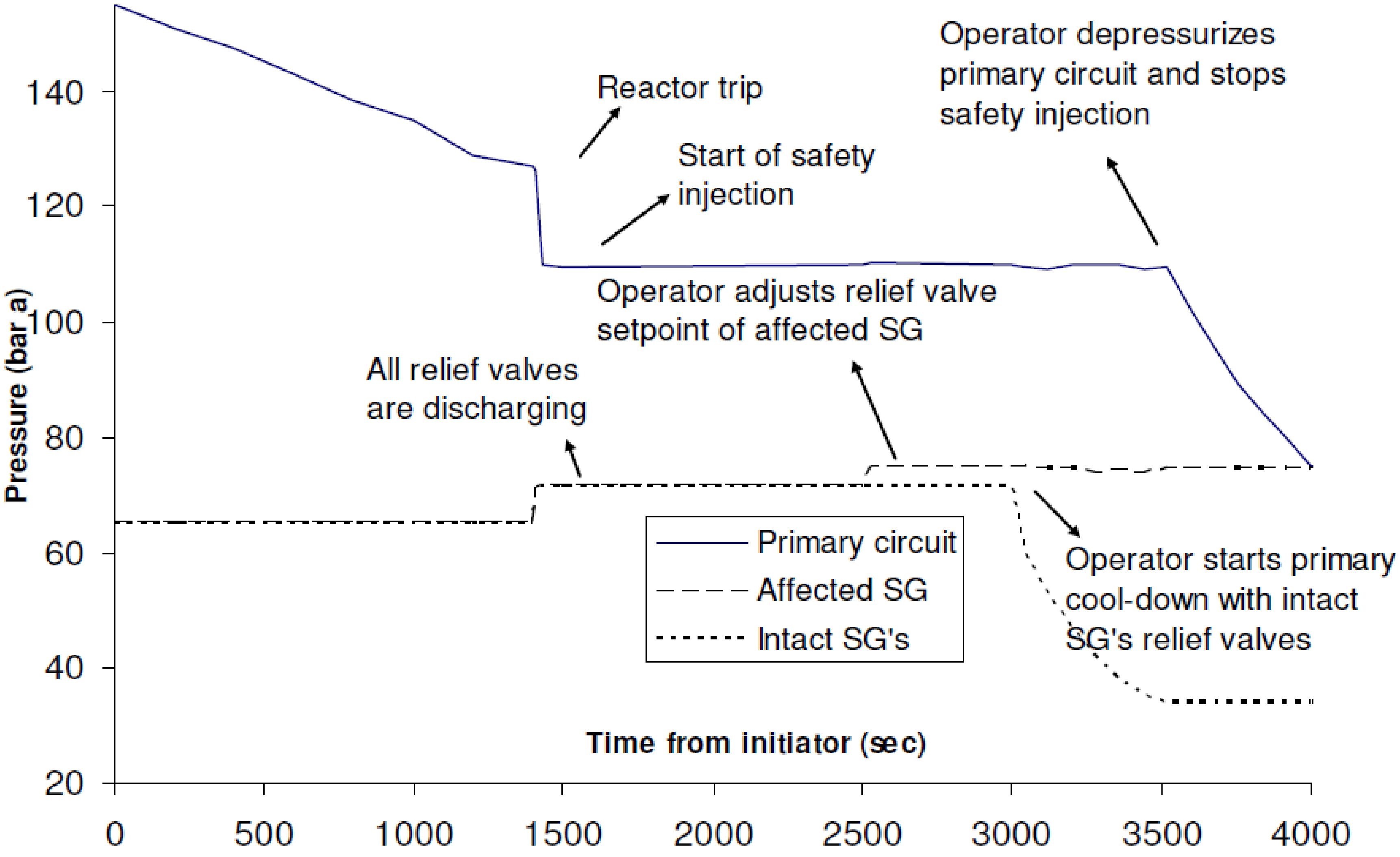


Figure 1 : typical pressures evolution during SGTR event (starting at Hot Full Power)

Table of contents

History of the SGTR since the 70'ies

3 big phases :

FIRST phase : design of GEN II PWR's (70'ies) : SGTR considered as a small LOCA, without particular problem because considered as rare event (condition IV) and represents no threat for the fuel integrity

SECOND phase : the operation of GEN II NPP's (REX) illustrated 2 major problems of barriers leak tightness (not considered at the design stage)

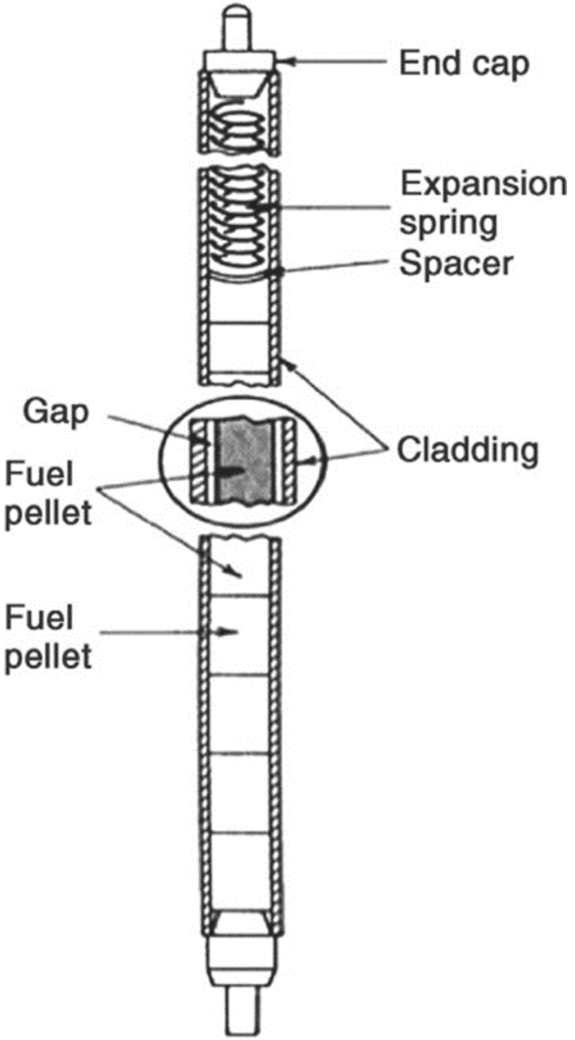
- Several occurrences of SGTR in the world : ex. in US (1982 Ginna, 1987 North Anna), in Belgium (1979 Doel 2) → Belgian safety authority imposed to "re-classified" it from condition IV to III (still today the case) → more stringent radiological limits to respect
- The first barrier (fuel cladding) is often leaking (releasing iodine-131 in primary)

THIRD phase : concrete (hardware) improvements on reactors in operation + improvement in the design of new (GEN III reactors)

Table of contents

Source term : leaking of the first barrier (1)

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS



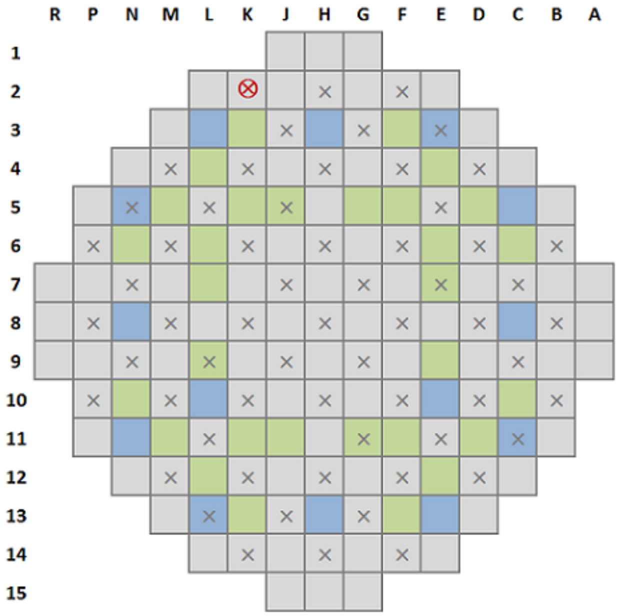
A Leaking rod can have different causes :
- Fabrication (grid to rod fretting)
- Foreign material in the primary
- Chemical reaction
- “Fuel pellet–cladding” interaction

Source term : leaking of the first barrier (2)

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

1 fuel assembly : 17 x 17 – 25 (guide tube + instrumentation) = 264 fuel rods per assembly

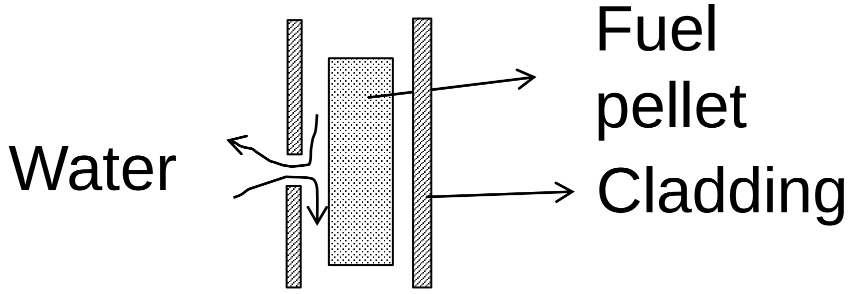
Valid for GEN II plants like Tihange 2, Tihange 3, Doel 3, Doel 4



157 assemblies in a PWR core

→ 157 x 264 = 41448 fuel rods in one core

Source term : leaking of the first barrier (3)



What happens when a fuel rod is leaking ?

Equilibrium between

- water of the primary entering in the rod
- water that goes out carrying fission products : Iodine-131 (I-131)= most significant species for contamination

Technical specification of NPP's imposes max I-131 concentration (GBq/t)

Info : Normal recorded concentration without leak =~ 1 to 10 MBq/t

Limit to respect as fixed in tech. spec. (Belgium) = 0.9 GBq/t

→Factor 1000 between both situations !

→Operating close to tech. spec. limit implies the presence of one or several leaking/damaged rods

Source term : leaking of the first barrier (4)

Actions taken by the Utility in case of detected leaking rods ?

- At power : the primary circuit is filtrated (CVCS) to try to respect the tech. spec. limit I-131 if possible. If not → obliged to go to Cold Shutdown.
- Even in case of success to respect tech spec with leaking rod : bad situation for 2 reasons : produces waste (filters to replace) + high primary activity close to tech spec = source term in case of several potential accidents (among them SGTR) → SGTR in itself represents no threat for the fuel integrity !
- During the outage that follows : different techniques are leading to identify the leaking assembly (sipping) → removed from the core (even if still at low burnup) and definitely stored in used assemblies pool

Source term : leaking of the first barrier (5)

Statistics about the subject (can be made for a country, of for a company)

IAEA definition : Fuel Failure Rate (FFR) = proportion of fuel rods that will present a failure (a leak) during one year in the core = r D/N

With :

- r = average number of leaking rods per assembly (=1.3 for 17x17)
- D = number of detected leaking assemblies ***during one year of operation*** that will be discharged from the core
- N = number of fuel rods submitted to irradiation ***also during one year of operation***)

→ The FFR can be calculated for example for a country like France or Belgium during one year of operation.

Source term : leaking of the first barrier (6)

Up to the 90'ies : the FFR was about 5 E-5. One core representing 41448 rods, means that D = FFR . N/r = 5E-5 41448/1.3 = 1.6 → each core had in average 1.6 assemblies with leaking rods → almost each cycle was contaminated by leaking fuel ! → The Utility operated almost always close to tech spec limits values + had to filter the primary continuously, producing wastes…

Similar figures in different countries.

First phase of progress before 2000.

INPO (Institute For Nuclear Power Operation) published guidelines in 2006 "Zero fuel failure by 2010" → Second phase of progress due to improvements in fabrication, anti-debris filter

Source term : leaking of the first barrier (7)

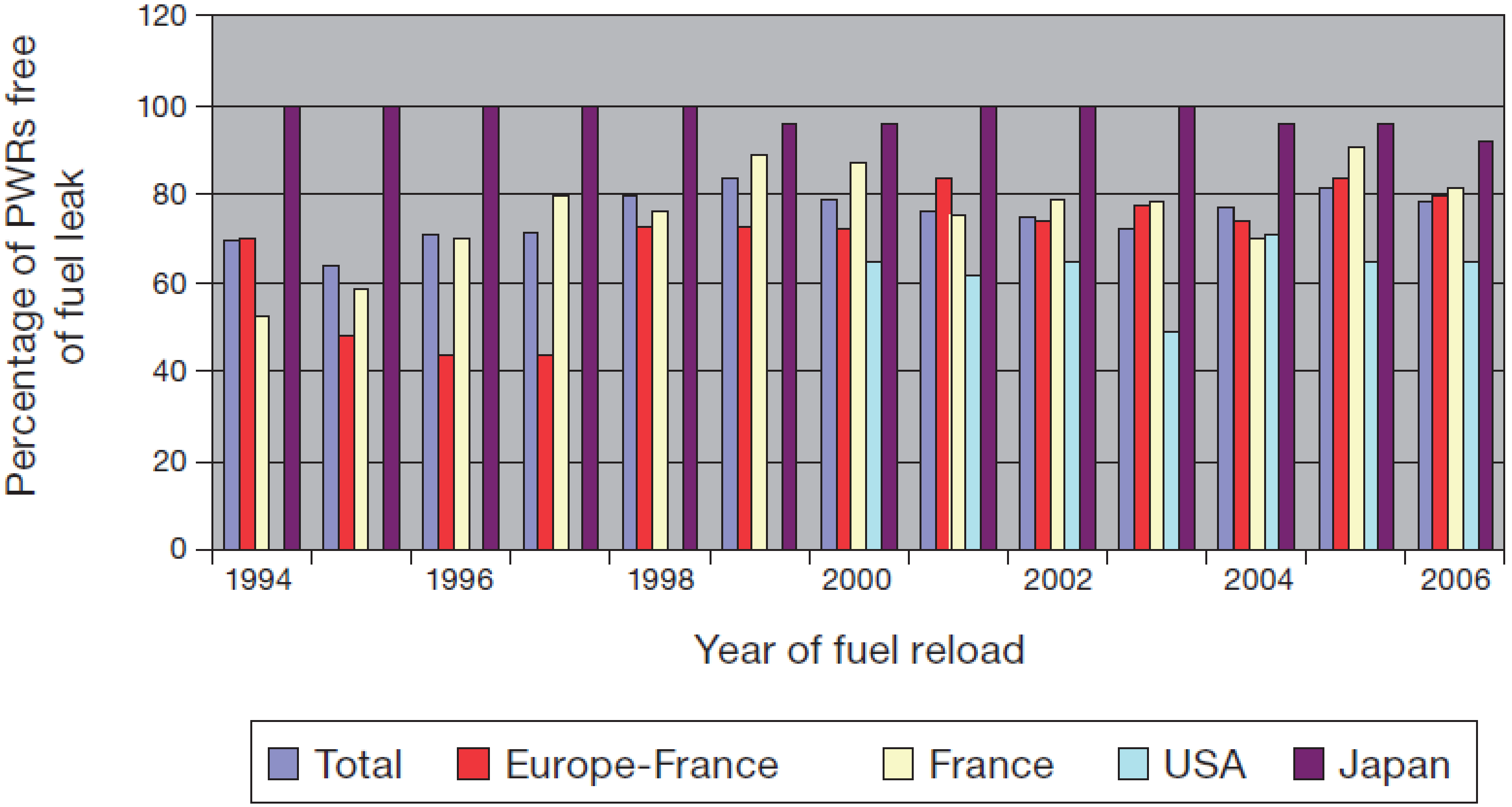


FIG. 3.10. Percentage of PWR units with zero fuel leakers.

Figure from [1]

Source term : leaking of the first barrier (8)

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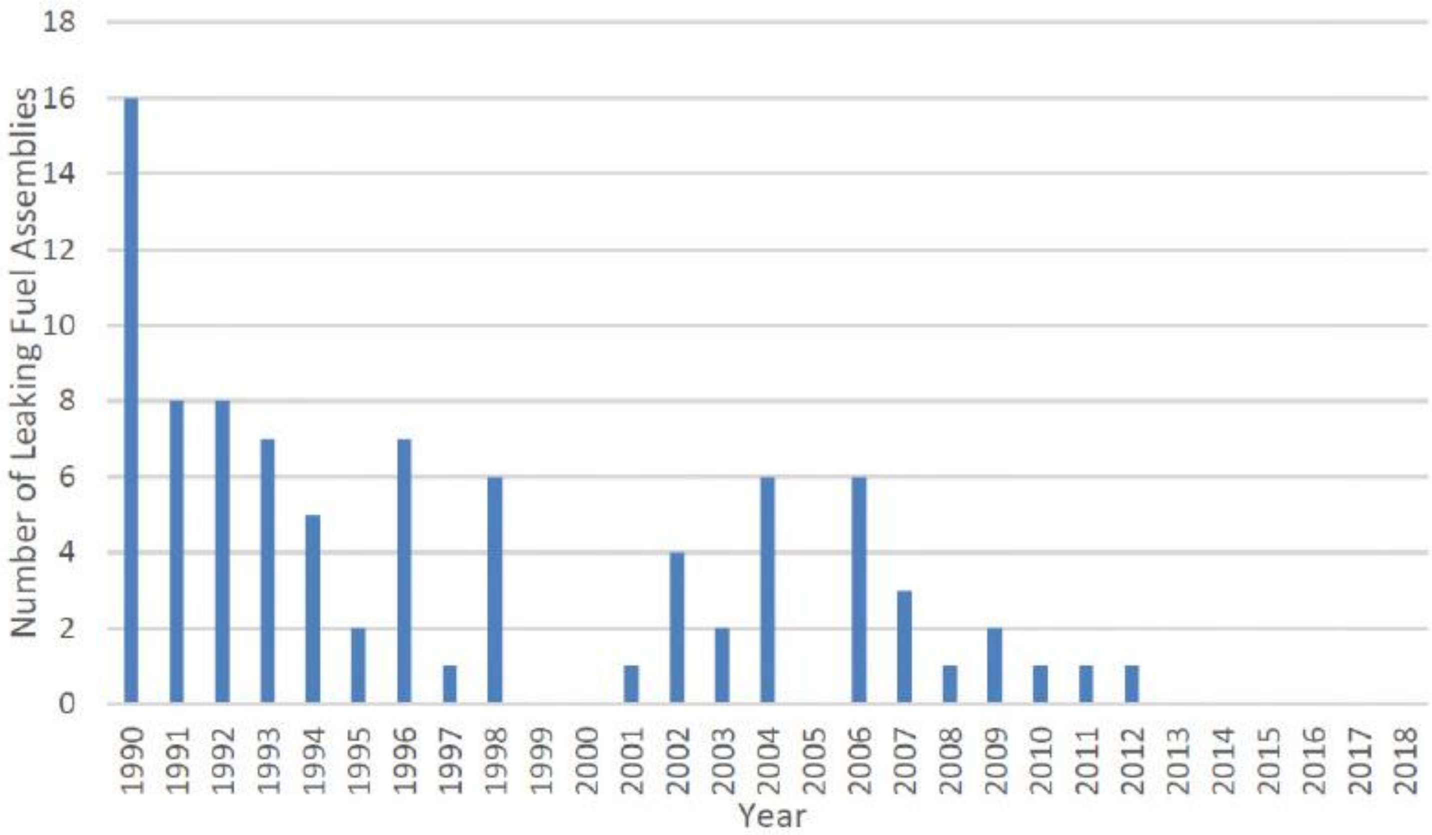
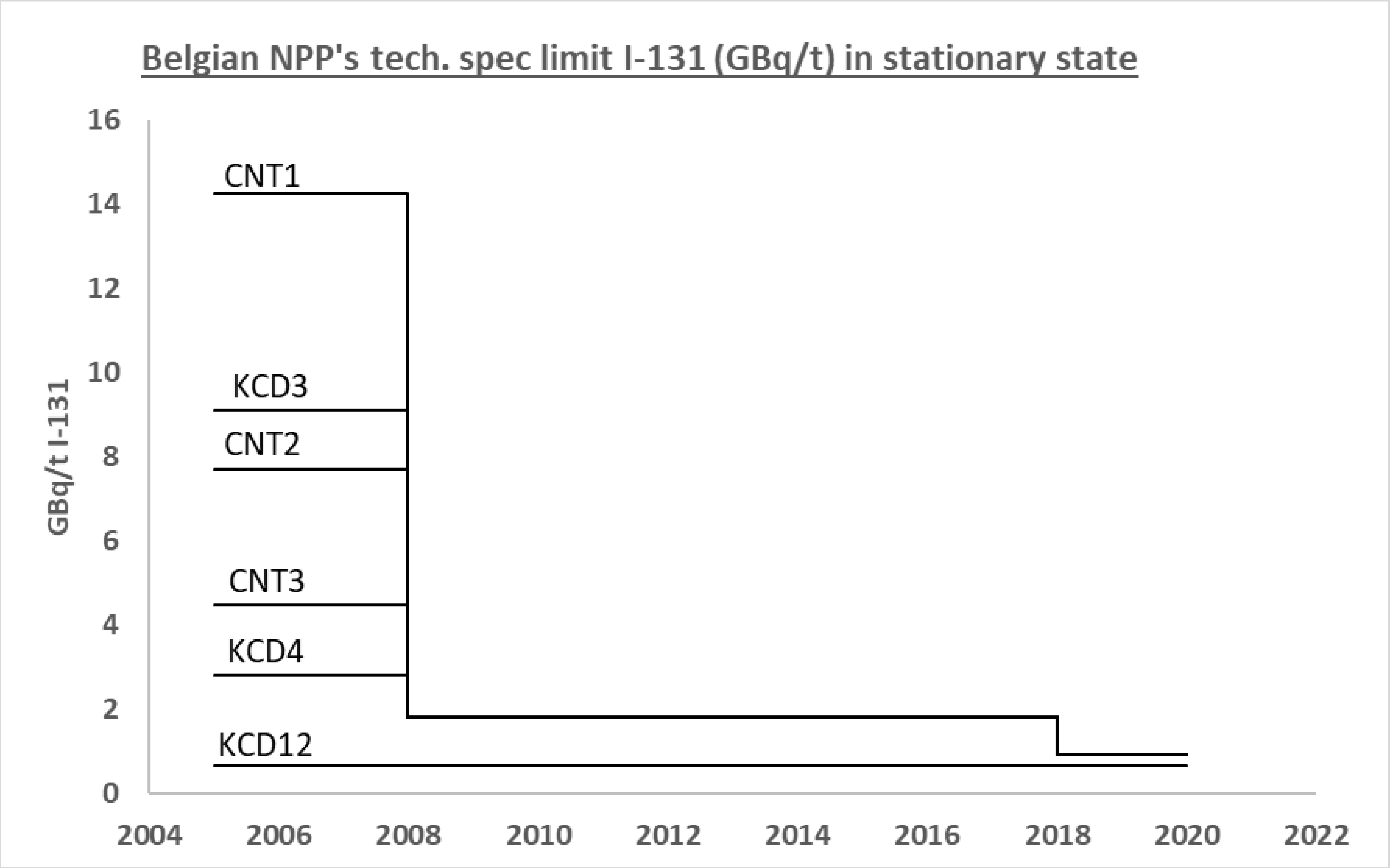
Figure : Detected leaking fuel assemblies per year in Belgium

Main fact : not any assembly with leaking rods has been discharged since 2012.



Same since 2018 and today

Source term : leaking of the first barrier (9)

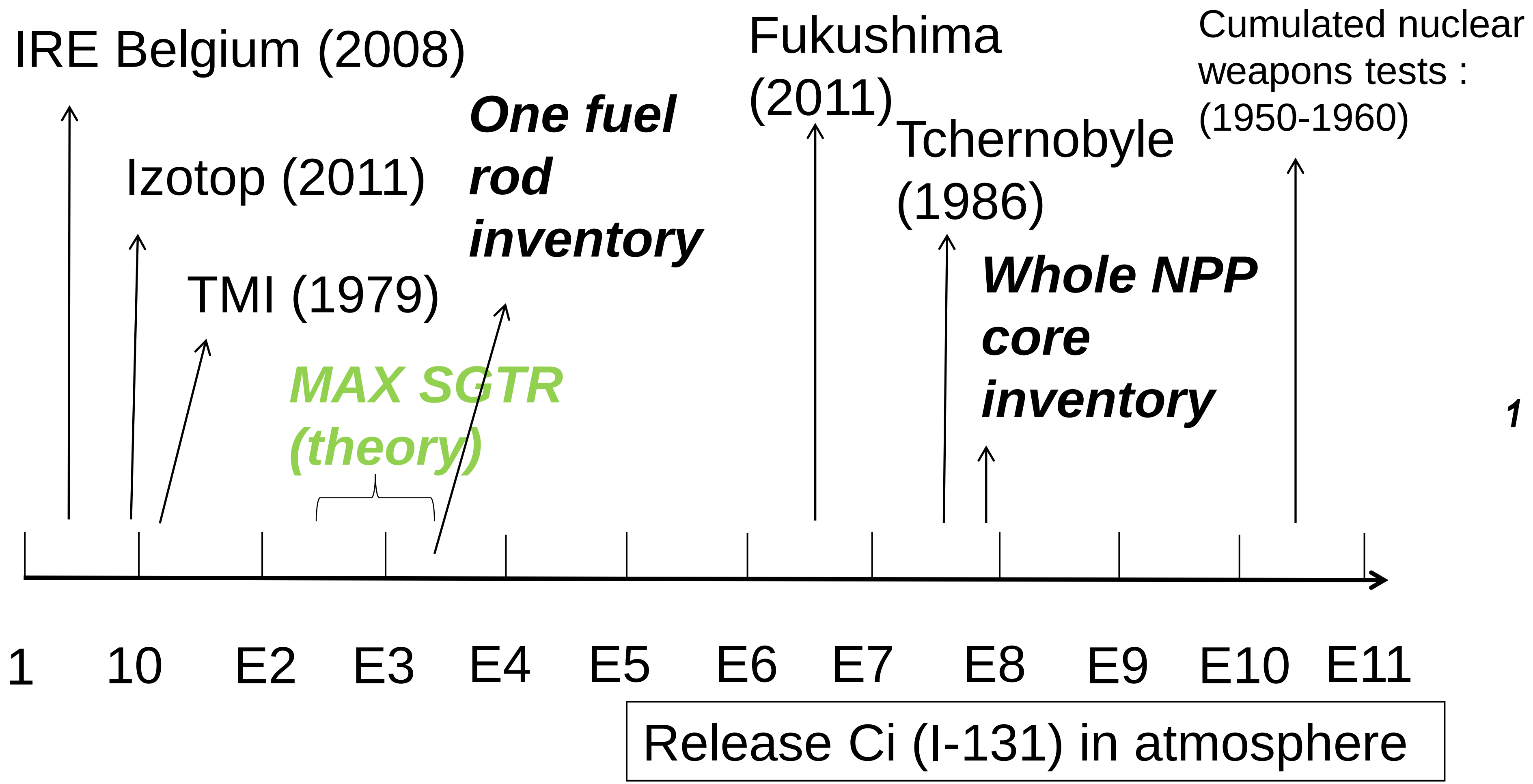


REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

Table of contents

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

SGTR maximal releases to put in perscpective



1 Ci= 37GBq

Table of contents

Evolution of the SG tubes materials (1)

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The material selection of SG's is a key issue due to the following reasons :

- Thickness of the tubes (=~ 1 mm) affects the SG performances (heat transfer resistance)
- SG tubes are part of the primary circuit barrier, and submitted to a high pressure difference : 75 to 95 bars
- Complex phenomena of corrosion can take place during operation → the tube integrity can evolve and is to be continuously monitored; Tubes can be "plugged" if needed
- Foreign objects, if present in primary or secondary, can hit tubes in operation (potential cause of SGTR)

Evolution of the SG tubes materials (2)

Different phases of evolution in the materials :

- Originally (GEN I), austenitic stainless steel → corrosion
- Gen II : various selected material among them Inconel 600 (ex : all original SG's in Belgium in 70'is 80'ies) → also corrosion was observed after few years of operation. Today, all SG's in Belgium have been replaced.
- Currently, all new SG's tube are using 2 materials showing no (or few) problem of corrosion : inconel 690 (selected by France since a long period, all Belgian SG's except Doel 3), and incoloy 800 (used by Germany : current material for Boorsele (Nerderland) and Doel 3 (Belgium))

Other significant improvements reducing the risks of SGTR :

- Optimized chemical treatment of the secondary system (hydrazine)
- "Foreign material exclusion" procedures for workers

Evolution of the SG tubes materials (3)

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Ternary diagram illustrating all used alloys (past and present) for SG tubes

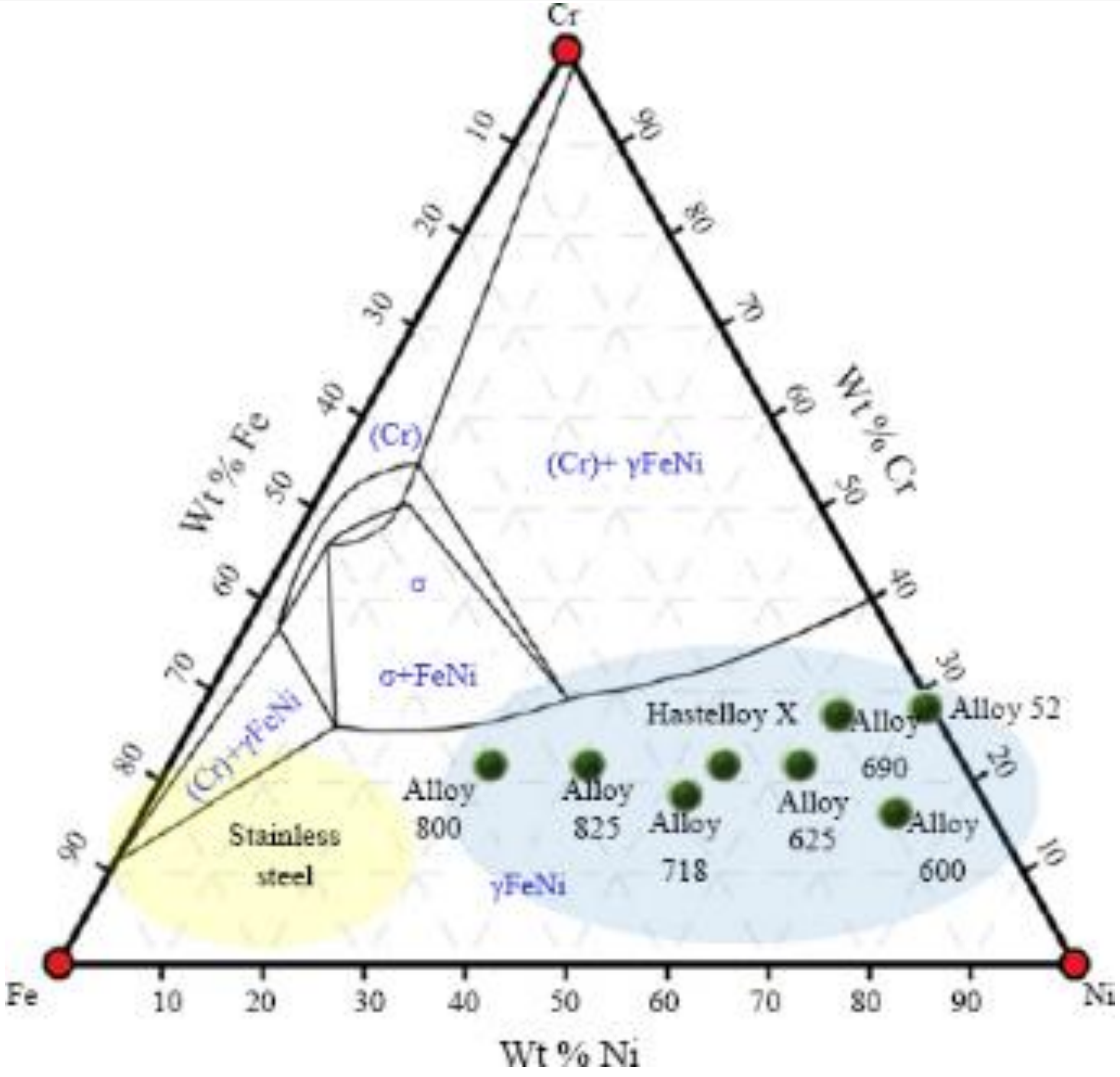

Table of contents

Licensing activities regarding SGTR in Europe

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In countries like France and in Belgium, for existing GEN II reactors, SGTR = historical subject of :

- Intense exchanges/discussions between safety authorities and Utilities (also recently)
- New safety studies
- Specific hardware modifications

Ex : In Belgium,…..even with NPP's today at their end of life, new classified "gamma global" detection chains have been just installed (2021-2022) in several units (Tihange 1, Tihange 3, Doel 4) in order to provide a faster/reliable detection in case of SGTR.

Table of contents

Conclusions

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

- At the design stage of GEN II reactors, SGTR importance was underestimated as this was revealed after some years of NPP's operation
- Since this period, many improvements (worldwide) were performed in order to reduce the risk linked to this accident → improving prevention and mitigation for GEN II + taken directly into account at the design stage of GEN III like EPR
- The way the subject was (is still) treated in different countries can be highly specific (ex : Belgian treatment being unique)….In particular the way to calculate radiological consequences → the aim of project like R2CA is to perform inventory of methodologies and try to uniformize them

Some references (public)

REDUCTION OF RADIOLOGICAL CONSEQUENCES OF DESIGN BASIS & DESIGN EXTENSION ACCIDENTS

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[4] TOP SAFE Dubrovnik conference 2008 A1-026 "The SGTR licensing evolution and associated modifications for nuclear units in Belgium" : François Parmentier, Jean-Charles Delalleau https://www.euronuclear.org/download/topsafe-2008-proceedings/

Thank you!

Francois.parmentier@belv.be

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