French Post-Fukushima Complementary Assessments - General Approach and Resulting Safety Improvements for the High Flux Reactor located in Grenoble

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Abstract. Following the accident that occurred on the Fukushima Daiichi nuclear power plant on March 11th 2011, the licensees of French nuclear facilities were asked to engage a safety reassessment of their facilities with the aim of evaluating their capacity to withstand extreme situations beyond design basis assumptions. These specific reassessments, called Complementary Safety Assessments (CSAs), were carried out on the basis of the specifications for the stress tests requested by the European Council. In France, these reassessments included all nuclear power plants in operation but also nuclear cycle facilities and research reactors. This paper presents the analysis performed by French licensees in the framework of CSAs and the opinion of the Institute of Radiological Protection and Nuclear Safety (IRSN) which has been largely involved in the evaluation of the CSAs. Then, the paper introduces the concept of "hardened safety core" firstly defined by IRSN and presents a concrete implementation of the "hardened safety core" based on the example of the High Flux Reactor (RHF) research reactor, operated by the Laue-Langevin Institute (ILL), in Grenoble (France).

Key Words: Complementary Safety Assessment, Hardened Safety Core, Laue-Langevin Institute

1. Introduction

Following the accident that occurred on the Fukushima Daiichi nuclear power plant on March 11th 2011, the French Nuclear Safety Authority (ASN) asked the French nuclear licensees, by the mean of regulatory decisions [1], to carry out a reassessment of their facilities in the light of the Fukushima accident. These reassessments, called Complementary Safety Assessments (CSAs), were based on the specifications attached to the aforementioned decisions and consistent with the specifications for the stress tests requested by the European Council.

The aim of the CSAs carried out in France is to take into account the lessons learned from the events that hit the Fukushima Daiichi nuclear site by evaluating the capacity of nuclear facilities to withstand extreme situations beyond design basis assumptions. The scope of CSAs included nuclear power plants in operation or under construction as well as nuclear facilities considered to be high-priority[1] like the High Flux Reactor (RHF) in Grenoble, the Osiris reactor in Saclay, the Jules Horowitz Reactor (RJH, under construction in Cadarache) and main cycle industries as La Hague (AREVA) facility.

The Institute of Radiological Protection and Nuclear Safety (IRSN), which is the main technical support organization of ASN, has been largely involved in the review of the CSAs carried out by licensees. In that context, IRSN conducted extensive technical discussions with licensees.

[1] The French CSAs were carried out by splitting nuclear facilities into three categories depending on their vulnerability to the phenomena which led to the Fukushima-Daiichi accident and the importance and scale of any consequences of an accident affecting them.

2. The French CSAs general approach

2.1. The nuclear safety approach in France

The general safety demonstration approach for French nuclear facilities consists in constantly seeking for safety improvement. Safety enhancing particularly relies on:

- the taking into account of the operating facilities feedback;
- the periodic safety reviews which are an obligation for all French nuclear facilities since 2006, including compliance review and safety reassessment accordingly to up-to-date safety standards and practices;
- the development and the updating of scientific and technical reference repositories.

Nuclear facilities design and specifications aim at demonstrating that the technical and organizational provisions in effect in the facility allowing the management the operation-related risks (under normal or accidental situations) including the ability of facilities to withstand the hazards (of internal or external origin) which may affect the facility. Specifications are likely to be modified throughout the facilities life, especially thanks to the safety periodic reassessments.

The safety demonstration in France is based on the defense-in-depth principle. This principle aims at designing the facilities in order to avoid incidental or accidental situations, and to foresee the adequate provisions in order to mitigate incidental or accidental possible consequences.

In this context, the safety analyses established in France are carried out accordingly to a deterministic approach, supplemented, when relevant, by probabilistic studies.

All of the justifications and analyzes produced by the operators of nuclear facilities require the definition of assumptions and input data. For instance, these assumptions and data concern the hazards characteristics which are taken into account in the safety demonstration (intensity, duration, etc.) or combinations of hazards that it is reasonable to consider for the demonstration. These characteristics are regularly reassessed in the light of new scientific and technical knowledge.

2.2. The objectives and the implementation of CSAs

The approach of CSAs engaged after the Fukushima-Daiichi accident assumes that very unlikely severe accident situations can still happen by the fact of natural external hazards with an intensity level exceeding those considered until then in the safety demonstration. The main natural hazards considered in CSAs are the earthquake, flooding and climatic phenomena. Extreme situations such that the total loss of electrical supplies or the total loss of cooling sources have also been postulated in CSAs with more degraded assumptions than those previously considered in the safety demonstration (duration of sources loss, numbers of facilities concerned in the same time on a given site, etc.).

The prime objective of CSAs is to assess the response of nuclear facilities in the event of an extreme hazard or an extreme situation (as above mentioned) which would affect the facilities. In France, the analysis has been focused on the identification of potential cliff-edge effects, that is to say, the risk that a small variation of a characteristic related to a hazard or to a degraded situation will lead to a brutal change of the facility behavior, combined most of time with large radiological consequences.

To this effect, the French operators of nuclear facilities have presented analyses of robustness based on an evaluation of safety margins, in terms of resistance of civil engineering structures or equipment, estimated from the design technical specifications (input data related to dimensioning methods or assumptions taken to the seismic spectra used in design studies) or to the construction (equipment anchorages, structural gaps between civil engineering structures) of facilities. This safety margin assessment, mostly based on single margin factors evaluated with an engineer appreciation, have enabled licensees to assess the robustness of their facilities and, where appropriate, to identify points of weakness and the required facility reinforcements.

In 2011, IRSN has estimated [2] that given the uncertainties about the levels of extreme hazards to consider on the one hand, and because of simplified approaches implemented for assessing the behavior of the facilities on the other hand, it is not possible to assess, with a sufficient confidence level, the robustness of facilities facing extreme hazards. In that way, IRSN has recommended that additional studies, based on codified technical rules and methods, be implemented to accurately identify the reinforcements that may be required to ensure the resistance of nuclear facilities against extreme hazards and situations.

However, the important work done by the operators in a very short time to carry out CSAs allowed to identify the systems, structures and components (SSCs) of facilities whose loss or failure may lead to a cliff-edge effect in terms of radiological or toxic consequences. These SSCs are directly involved in the control of fundamental safety functions namely, for the reactors, the reactivity control, the fuel cooling control and the containment of dangerous materials. In addition, these SSCs can be classified in one or the other of the levels of defense-in-depth (prevention of an accident, mitigation of the accident consequences and crisis management) according to their role for safety. From the point of view of IRSN, the identification by the operators of the SSCs whose loss or failure is likely to lead to a cliff-edge effect was globally satisfactory.

Finally, the CSAs determined, for the facilities for which a risk of cliff-edge effect had been identified, a set of material provisions to enable the facility to withstand hazards or situations with intensity levels higher than those considered so far. This set of provisions, completed with organizational measures, constitutes what it has been called the "hardened safety core” [2].

3. The “hardened safety core” concept

The “hardened safety core” (HSC) must ensure ultimate protection of nuclear facilities with the following three objectives:

- prevent a severe accident or limit its progression;
- limit large-scale releases in the event of an accident which it was not possible to control;
- enable the licensee to perform its emergency management duties.

The HSC may be composed of existing SSCs, that might require to be strengthened, or of new SSCs that should be designed and sized to withstand extreme situations.

In a general manner, IRSN considers that choices regarding the definition of the SSCs of HSC and the related technical requirements must guarantee, with a high level of confidence, the ability to ensure their functions in the event of extreme situations.

To this purpose, it must be defined, on the one hand, the required intensity levels to characterize what are extreme hazards and situations (intensity, duration, etc.), on the other hand the methods implemented to justify the resistance and the operability of SSCs of the HSC under extreme conditions. It is on the basis of these data that it will be possible to demonstrate that the main safety functions will be ensured in the event of extreme situations.

For earthquake for instance, ASN has asked the French licensees to justify the SSCs of HSC on the basis of a seismic response spectrum (giving the answer in acceleration of a simple oscillator placed on the ground affected by the earthquake) that meets the following requirements:

- be 50% higher than the seismic spectrum chosen as a reference to the design of new nuclear facilities (determination method of this spectrum is specified in a fundamental safety rule published by the ASN in 2001 [3]);
- be conservative of spectra defined accordingly to a probabilistic manner with a return period of 20 000 years;
- take into account the possible effects due to the facility location including the nature of the soil.

4. The hardened safety core of the High Flux Reactor

4.1.Description of the facility

The High Flux Reactor (RHF) is a research reactor located in the immediate vicinity of Grenoble city (France). First diverged in 1971, this reactor aims at providing, through channels pointing directly towards the reactor core, neutrons source for the purposes of fundamental research mainly devoted to the exploration of matter.

This reactor, operated by the Laue-Langevin Institute (ILL), develops a maximum thermal power of 58.3 MW. The reactor core *(see FIG.1.)*, cooled by heavy water, is composed of one fuel element made of highly-enriched uranium and aluminum alloy.

The reactor is operated by cycles of approximately 50 days. Immediately after the reactor shutdown, the passage in natural convection allows, thanks to the calorific capacity of the reactor pool, to cool the core without need of electrical power supply or external cooling source.

The reactor building is made of a double enclosure *(see FIG.2.)*, one in a 40-cm thick concrete wall (internal enclosure), the other in a 1.1-cm thick metal lining, the annular space between both enclosures being pressurized at 135 mbar.

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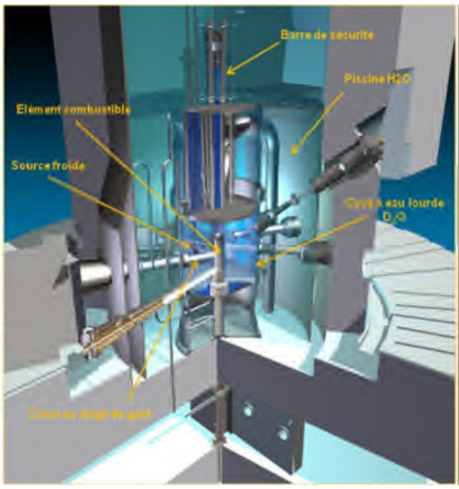

FIG.1. RHF reactor core

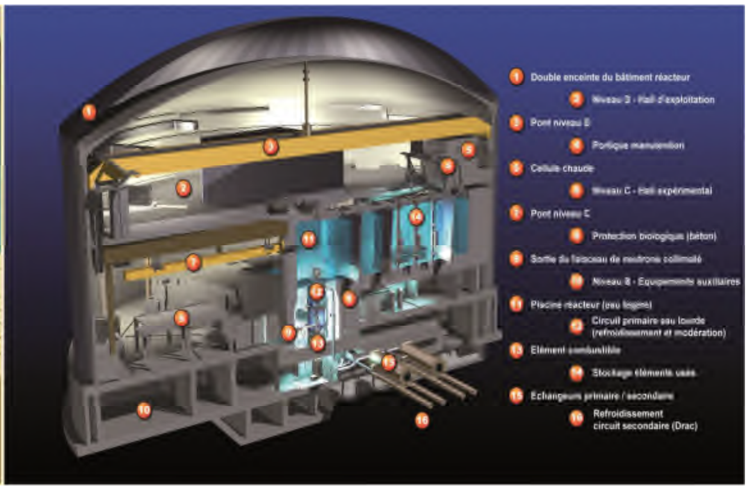

FIG.2. RHF building reactor

4.2.The building and the setup of the RHF hardened safety core

As previously mentioned, IRSN assessed the demonstration carried out by the operators to identify the SSCs whose failure might lead to a cliff-edge effect in case of extreme hazards or extreme situation. For ILL, this identification had been made on the basis of scenarios related to core melt accidents (severe reactivity accident or loss of fuel cooling accident).

In that way, ILL accidents of fuel melting (in air or in water) cumulated to a degradation of the building containment function. From these scenarios, ILL has identified the SSCs whose failure may lead to important radiological consequences outside the facility perimeter. IRSN assessment has confirmed that ILL realized an approach matching the specifications of the CSAs, while underlying the effort carried out by ILL to perform an exhaustive search of cliff-edge effects scenarios taking into account all possible initial states of reactor operation.

In 2012, as a following of the CSA of RHF, ILL has proposed to implement a hardened core of material provisions and organizational measures aiming at ensuring the control of the basic safety functions in the event of extreme hazards. On the basis of conclusions of the RHF CSA, the main hazards to be considered are the extreme earthquake and the extreme flooding (induced by dam failures located in the mountains range surrounding Grenoble), as well as the induced effects (explosion of internal or external origin, fire, etc.).

Thus, ILL defined, in a first time, the HSC of the RHF as described below, relying on the levels 3 to 5 of the defense-in-depth principle:

The **"prevention of severe accident" part of the HSC** (level 3 of defense-in-depth):
- the reactor emergency shutdown system (new system called ARS[2]) based on earthquake detection (0,01g),
- the core water supply safeguard systems (existing core water supply systems called CRU and CES, completed by a new groundwater supply system called CEN),
- the emergency fuel lowering system (system called PUC which is an already existing system) for setting the fuel element being handled in a safe position in the fuel storage pool.

The **"mitigation of severe accident" part of the HSC** (level 4 of defense in depth):
- the concrete enclosure building with the related automatic containment isolation devices (system called SIE),
- the containment depressurization seismic circuit (new system called CDS) which will ensure the extraction and the filtration of contaminated air contained in the reactor building in case of severe accident.

The **"emergency management" part of the HSC** (level 5 of defense in depth):
- the bunkered emergency control room (new room called PCS3 located in a new building) which will permit to ensure the control of the aforementioned active systems, the general monitoring of the reactor after accident (temperature and pressure measures, pools water level measures, radioactive releases control, etc.) and the emergency management duties (communication with national and local authorities, communication and information exchange with IRSN, etc.).

During the technical instruction, IRSN has indicated to ILL that the HSC will be useful only if passive SSCs of RHF were not completely deteriorated after an extreme aggression (for

[2] A reactor shutdown system was of course already in place but not designed (I&C in particular) to withstand strong earthquakes. The new ARS system will permit to ensure the reactor shutdown including during the strongest solicitations of an extreme earthquake.

instance the pool walls or the beam tube structures). ILL agreed with this point and decided to modify the content of the HSC of RHF by including passive SSCs necessary to achieve basic safety functions.

From this point on, the HSC of RHF is defined as described in Table I below [4].

TABLE I: SSCs OF THE HARDENED SAFETY CORE OF RHF

« Active » HSC	« Passive » HSC
• Emergency reactor shutdown system (ARS) • Ultimate “drench” circuit (CRU) in association with the emergency water supply circuit (CES) • Underground water circuit (CEN) • Emergency fuel lowering (PUC) • Automatic containment isolation system (SIE) • Containment depressurization seismic circuit (CDS) • PCS3 (means of control and monitoring required for the management of crisis)	• Primary core enclosure and related supporting structures • Fuel handling container • Natural convection flappers • Civil engineering structures and lining of the fuel storage channel and reactor pool • Neutron beam tube nozzles • Concrete reactor enclosure • PCS3 (room and supporting building)

It should be underlined that all new "active" SSCs of HSC satisfy the single failure criterion. Thus, CEN circuit (underground water supply), CDS circuit (which manages the depressurization of reactor building and the accidental release) and all electrical power supplies and monitoring devices required for PCS3 crisis management are redundant.

IRSN has finally concluded that principles and requirements chosen ILL for the definition, the design and the realization of the HSC of RHF were satisfactory. This is likely to meet the expectations of IRSN who considers that the objective of the hardened safety core is to confer on nuclear facilities (and therefore to the RHF in particular) a better robustness to withstand situations not considered up to now in safety demonstrations.

4.3.The building and the setup of the RHF hardened safety core

Since 2012, ILL has placed special emphasis on the strengthening of RHF decided as a result of the accident of Fukushima. The implementation of the HSC must be completed at the end of the first quarter 2016.

In the summer 2015, the state of progress of the implementation of the HSC of the RHF is as described hereafter.

SSCs related to the "prevention" of severe accidents:

- The new equipment of the water underground circuit CEN (*see FIG.3.*) are installed and tested (pumps, suction strainers, piping, power supplies, monitoring and control systems);

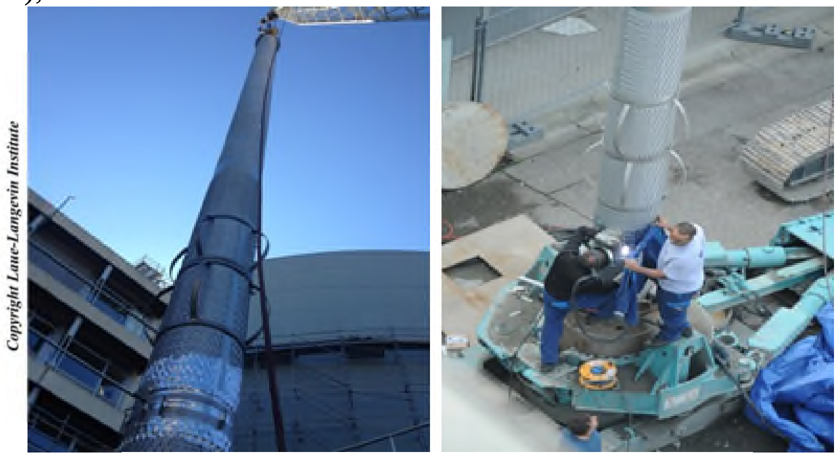

FIG.3. Suction strainers of the new CEN system

SSCs related to the "mitigation" of the consequences of severe accidents:

- The containment depressurization seismic circuit CDS (*see FIG.4.*) is installed and tested (ventilators, HEPA filters, iodine filters, pipes, isolation valves, chimney on the roof of reactor building);

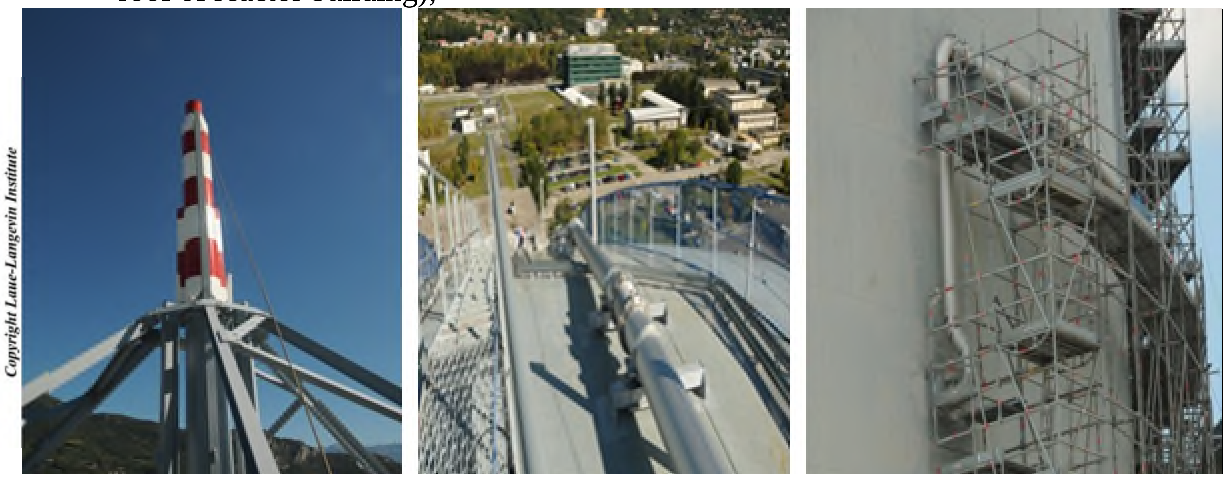

FIG.4. New CDS circuit: parts located outside the reactor building

SSCs related to "emergency management":

- The PCS3 building (*see FIG.5.*) has been built between 2012 and 2013 and its equipment, as well as the cable connections between the new building and the BR, was installed between 2013 and 2014 (*see FIG.6.*). Since the beginning of 2015, all relevant information for crisis management are available at the command post of PCS3 (water level in the reactor pool and fuel storage, radioactivity measures in the reactor building, wind speed and direction, communication devices, etc.). Moreover, equipment required for the human habitability of PCS3 in situation of accident is already installed (air conditioning systems, HEPA and iodine filters, NBC[3] filtration system, etc.). Nevertheless, the effective control of the active systems of the HSC (CEN, CDS, etc.) from the PCS3 still requires an authorization from ASN after a review by IRSN.

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[3] Nuclear, Biological, Chemical.

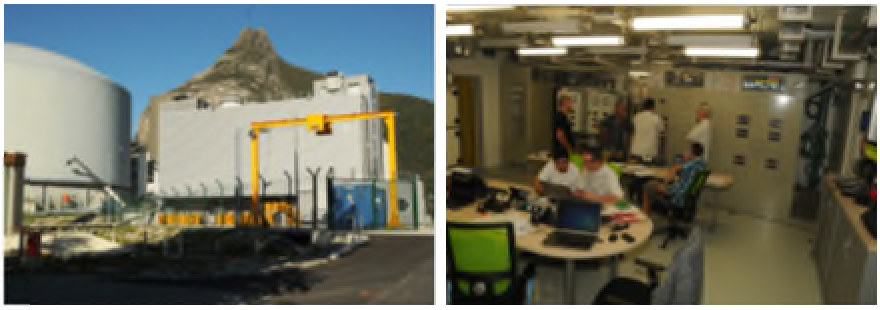

FIG.5. External view of new PCS3 (building) and emergency control

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FIG.6. Communication tools and air conditioning systems of PCS3

5. Conclusion

As a result of the accident of Fukushima-Daiichi, Complementary Safety Assessments (CSAs) have been carried out by licensees of French nuclear facilities to evaluate the behavior of these facilities with regard to extreme hazards and situations, mainly targeted on the earthquake, flooding, climatic phenomena and the total loss of power supplies and sources of fuel cooling. The evaluation by IRSN of these CSAs has led to the emergence of the concept of “hardened safety core (HSC)” whose objective is to increase the resistance of nuclear installations to extreme hazards of extreme situations whose characteristics are superior to those considered up to now in safety demonstrations.

The Laue-Langevin Institute (ILL) has fully developed the concept of HSC for the High flux reactor (RHF) and many technical exchanges with IRSN have been done since 2011 playing an important part in the definition and the implementation of the HSC at the facility.

IRSN considers finally that the reinforcements of existing SSCs made by ILL and the new SSCs put in place on the RHF are likely to significantly improve the robustness of the installation in the event of extreme natural hazards.

6. References

[1] ASN, Decisions requiring that French Nuclear Facility Licensees conduct a Complementary Safety Assessment of their Basic Nuclear Installations in the light of the accident that occurred on the Fukushima Daiichi Nuclear Power Plant (2011).

[2] IRSN, Report No. 679 (November 2011).

[3] ASN, Fundamental Safety Rule No. 2001-01 concerning basic nuclear facility.

[4] IRSN, Report No. 2013-00005 (March 2013).