Severe accident mitigation strategy for the generation II PWRs in France – some outcomes of the on-going periodic safety review of the French 1300 MWe PWR series

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Abstract:
The 3rd Periodic Safety Review of the French 1300 MWe PWRs series includes some modifications to increase their robustness in case of a severe accident. Their review is based on both deterministic and probabilistic approaches, keeping in mind that severe accidents frequencies and radiological consequences should be as low as reasonably practicable, severe accidents management strategies should be as safe as possible and the robustness of equipment used for severe accident management should be ensured.
Consequently, the IRSN level 2 probabilistic safety assessment (L2 PSA) studies for the 1300 MWe reactors have been used to re-assess the results of the utility's L2 PSA and rank them to identify the containment failure modes contributing the most to the global risk. This ranking helped the review of plant modifications.
Regarding strategies for accident management, the EDF management of water in the reactor cavity during a severe accident for the 1300 MWe PWRs is presented as well as the IRSN position on this strategy: this is an example where the optimal severe accident management strategy choice is not so easy to define.
Regarding the robustness of equipment used for severe accident management, the interest of a diversification or redundancy of the French emergency filtered containment venting opening is one example among many others.

1 INTRODUCTION

The French electrical utility Électricité de France (EDF) is operating a fleet of 58 standarized pressurized water reactors (PWRs) (3 series of 900, 1300 and 1450 MWe). Periodic Safety Reviews (PSRs) are conducted every 10 years. A PSR is carried out for the whole series considered. Associated studies must be completed early enough so that the modifications to equipment and documents can be deployed on reactors from the start of their ten-yearly outage. The 3rd PSR for the 1300 MWe PWRs (20 reactors) is presently on-going and their 3rd ten-yearly outage is planed from 2015 to 2021. The 3rd ten-yearly outage for the 900 MWe PWRs (34 reactors) is on-going and their 4th PSR should start next year (2014) as their 4th ten-yearly outage is planed from 2019 to 2029.

The severe accidents were not included in the initial design of the Gen II PWR. Nevertheless, all French operating plants include already severe accident management equipment (Passive Autocatalytic Hydrogen Recombiners (PARs), Emergency Filtered Containment Venting System (EFCVS), severe accident instrumentation…) as a result of previous reviews (PSRs, …) [1].
On French Safety Authority demand, EDF has produced a severe accident safety standard including the safety requirements (approach and safety objectives in terms of prevention and mitigation of severe accident, the studies necessary to demonstrate compliance with the objectives, the current practical provisions and their design basis, the requirements applied to equipment).

The severe accident safety standard for the 1300 MWe reactors has been analyzed by IRSN during the 3rd PSR keeping in mind that:

- severe accident frequencies and radiological consequences should be as low as reasonably practicable,
- severe accident management strategies should be as safe as possible (radiological consequences should be as low as reasonably practicable),
- the robustness of equipment used for severe accident management should be ensured. It concerns the behavior of equipment in severe accident conditions (pressure, temperature, radiation, …), their protection against internal and external events and also their reliability (correct needed equipment behavior in severe accident conditions).

In order to review that severe accident frequencies and radiological consequences are as low as reasonably practicable, the IRSN level 2 probabilistic safety assessment (L2 PSA) studies for the French 1300 MWe reactors have been used to re-assess the results of the utility's L2 PSA. Using the obtained containment failure modes frequencies, IRSN ranked them to identify the containment failure modes contributing the most to the global risk. This ranking helped to review the plant modifications proposed by the utility for this PSR and to identify some additional issues. For IRSN, all the risks identified by this work should be either reduced by mitigation measures or better understood through R&D efforts. This work is presented in the §2 of this paper.

As an example of ensuring that severe accident management strategies are as safe as possible, the management of the water in the reactor cavity during a severe accident proposed by EDF for the 1300 MWe reactors is presented in the §3 of this paper as well as the IRSN position on this topic. As a matter of fact, the effective risk associated to a steam explosion[1] after vessel lower head failure with a flooded reactor cavity (for the French PWRs, the reactor cavity can be filled by water after activation of the inner containment spray system) is still a topic of debate.

Finally, IRSN has reviewed the requirements defined by EDF for each equipment associated to severe accident management (and the list of equipement). The interest in 1300 MWe EFCVS opening diversification or redundancy is one example and is presented in the §4 of this paper.

2 USE OF IRSN L2 PSA IN THE 3RD PERIODIC SAFETY REVIEW

The IRSN L2 PSA was used to compare the assumptions used in the L2 PSA realised by EDF in the framework of the 3rd PSR of the 1300 MWe reactors. This work allowed a systematic review of all main topics and comparison with the state of the art. Even if the EDF L2 PSA was consistently developed, IRSN considered that the EDF study sometimes underestimated some risks and overestimated some others. A simplified reassessment of the results of the utility's L2 PSA was then performed by IRSN, taking into account lessons learned from this review. This reassessment took into account:

- a more realistic modeling for hydrogen combustion: pressure peaks calculation after combustion, hydrogen combustion between the inner and the outer containment walls[2], impact of late depressurization of the primary circuit, impact of reflooding during in-vessel core degradation, impact of inner spray containment actuation after vessel failure...,
- a more realistic modeling of the fragility curve of the inner containment,

[1] A "steam explosion" is due to a very fast heat transfer between an overheated molten core and liquid water. The large vaporization is so quick that local pressure increase leads to fragment the liquid corium in very small drops. This fragmentation increases again heat exchange surface with water, so the energy transfer, and can lead to an explosion called "steam explosion".

[2] 1300 MWe PWRs series are designed with a double containment (inner and outer containment walls). The space between these two walls, designated as the annulus, is maintained below atmospheric pressure by the Annulus Ventilation System

- the possibility for the reactor cavity to be flooded after the inner spray containment activation and so the steam explosion risk at vessel lower head failure under these configurations,
- the extension of heterogeneous boron dilutions situations to be considered, taking into account heterogeneous dilutions inherent to a severe accident scenario,
- the spent fuel pool accidents (loss of coolant, loss of cooling, …).

Using the frequency of the containment failure modes obtained from this reassessment work, IRSN ranked them by means of a basic risk metric based on the product (accident frequency) x (consequence amplitude). As the objective was to identify the containment failure modes that contribute the most to the global risk without giving too much importance to the figures themselves, the (consequence amplitude) term was choosen to be:
- 1000 for large early releases,
- 100 for the other non acceptable releases,
- 10 for late filtered releases.

The result of this ranking is presented on table 1. For IRSN, all the risks listed in this table should be either reduced by mitigation measures or better understood through R&D efforts.

Table 1: Contribution to global risk of containment failure modes. Simplified ranking.

Containment failure mode	**Contribution to global risk** (frequency x consequence)
I-SGTR (induced steam generator tube rupture)[3]	29.0 %
Reactor containment isolation failure	16.1 %
Reactor containment failure after hydrogen combustion during in-vessel phase (i.e. when the corium is still inside the vessel)	14.4 %
Reactor containment bypasses (heterogeneous dilutions, initial SGTR, interfacing LOCA[4], bypass through equipment hatch access)	10.0 %
Ex-vessel steam explosion	8.7 %
Spent fuel pool accident	6.5 %
Reactor containment failure after hydrogen combustion during ex-vessel phase	4.0 %
Reactor containment failure after direct containment heating[5]	2.7 %
Secondary system rupture inside containment	2.7 %
Secondary containment failure after hydrogen combustion in the annulus	2.0 %
Emergency filtered containment venting	1.7 %
Basemat penetration by the corium	1.3 %
Long term containment overpressurization	1.0 %

This simplified re-assessment of the results of the utility's L2 PSA helped IRSN to review the plant modifications proposed by EDF in the framework of the 3rd PSR of the 1300 MWe reactors and to identify some additional issues to be considered for plant safety

[3] In severe accident conditions, steam generator tube rupture can be induced by temperature conditions in the reactor coolant system (circulation of superheated steam from the core to the steam generator tubes).
[4] LOCA occuring on outgoing pipes, outside the containment.
[5] Direct containment heating (DCH) is a set of phenomena, starting at the dispersion of corium from the reactor vessel still at elevated pressure, and resulting in the heat up and pressurization of the containment atmosphere. Induced vessel displacement is possible in extreme case.

enhancement. The following subchapters provide information on some plant safety improvements that can be associated to risk reduction in L2 PSA.

2.1 Risk reduction of I-SGTR

From table 1, the risk of containment bypass by induced steam generator tube rupture (accident of core melt without reactor coolant system (RCS) depressurization) is one of the dominant risks for the French 1300 MWe PWRs: this is due to the relatively high station black-out frequency as initiating event. Two modifications are planned by EDF for the 3rd outage of the 1300 MWe PWRs:

- to anticipate the opening of the pressurizer valves in operators procedures,
- to modify the pilot of the pressurizer valves in order to guaranty that the valves stay in open position even if there is not any more power supply; moreover, after this modification, pressurizer valves opening can also be performed with an autonomous electrical supply.

IRSN has considered that these two modifications were relevant and that they should be timely implemented. The complementary safety review performed after the Fukushima accident has then confirmed the importance of this issue.

2.2 Risk reduction of containment isolation failure in case of station black out

The L2 PSA results (from EDF and also IRSN) show the interest to provide an additional electrical supply of some containment isolation valves to improve the reliability of the containment isolation in case of station black out. This modification is planned by EDF for all reactors. Interest was also confirmed during complementary safety review performed after the Fukushima accident.

2.3 Risk reduction of reactor containment failure after hydrogen combustion

The risk of containment failure by hydrogen combustion is not a dominant risk in EDF L2 PSA thanks to the passive autocatalytic hydrogen recombiners (PARs). Nevertheless, based on its own studies and on the review of EDF studies, IRSN considered that if uncertainties were taken into account and if a broader range of situations were considered (especially with regards to water management, i.e. in-vessel water injection, inner containment spray activation,…), this risk was not so residual.
IRSN considered that, for the 1300 MWe PWRs, even with PARs, studies should be continued and safety margins should be evaluated for an extended set of accident scenarios.

2.4 Risk reduction of heterogeneous dilution

The frequency of heterogeneous dilutions is provided by L1 PSAs. The real consequences of these situations are subject to high uncertainties and, in L2 PSAs, they are conservatively associated to large release (both in EDF and in IRSN L2 PSAs).
The utility has concluded that additional analysis was needed on these accidents. IRSN has also emphasized the importance of this issue, and considered that further analysis was needed.

2.5 Risk reduction of containment bypass through equipment hatch access

A severe accident while the containment hatch is open would lead to large release in the environment. Even if the frequency of such accident is low, the risk, for L2 PSA, is significant due to the amplitude of the consequences. The utility has proposed some modifications of the emergency operating procedures in order to have more available time to restore electrical power supplies. The effectiveness of this modification has yet to be rewieved.

2.6 Risk reduction of ex-vessel steam explosion

For the French PWRs, the reactor cavity can be filled by water after the inner containment spray activation. The effective risk associated to a steam explosion after vessel lower head failure is still a topic of debate. This item is detailed in the §3 of this paper in which the EDF management of the water in the reactor cavity during a severe accident on a 1300 MWe reactor is reviewed.

2.7 Risk reduction of spent fuel accidents

The frequency of spent fuel accident is low. Nevertheless, the risk (i.e. (accident frequency) x (consequence amplitude)) associated with these scenarios in L2 PSA is significant due to conservative assumptions related to the amplitude of induced consequences.
IRSN considered that studies should be undertaken in order to get a better understanding of accidents in spent fuel pool. These studies should help to better assess the risks of such accidents but also identify possibilities of mitigation strategies if any.
Precise phenomenological description of accident progression should be first obtained from research programmes.

2.8 Risk reduction of direct containment heating

Direct containment heating is associated to vessel rupture at high pressure. The modification proposed by the utility in order to improve reliability of RCS depressurization safety valves (see §2.1 Risk reduction of I-SGTR) will also help to reduce the risk of containment failure after direct containment heating, as it will help to decrease the pressure in the primary circuit.

2.9 Risk reduction of hydrogen combustion in the secondary containment (annulus)

IRSN studies have shown that hydrogen combustion was possible in the secondary containment (called annulus) during the late phase of a severe accident (after vessel rupture and during molten core concrete intercation (MCCI). In the framework of L2 PSA review, EDF has also performed additional analysis on this issue. The EDF results have confirmed that an explosive atmosphere can develop in the annulus during the late phase of MCCI.
This issue will have to be included in the next versions of L2 PSA and, if necessary, will have to be considered in the severe accident management provisions.

2.10 Global risk reduction by mitigating the radiological consequences

The utility has proposed to improve the annulus venting system (strengthening of the electric power and increasing the extraction / filtration capacity rates in case of severe accident) in order to reduce radiological consequences induced by severe accidents.
IRSN considered that the utility should take into account issues related to hydrogen risk in the annulus for the modification design (see above).

3 MANAGEMENT OF THE WATER IN THE REACTOR CAVITY DURING SEVERE ACCIDENT

During a severe accident, the inner containment spray activation is needed to decrease the containment pressure but water can accumulate into the reactor cavity up to the RCS loops. As a matter of fact, on French PWRs, after the inner containment spray activation, the reactor cavity is filled passively with water up to the RCS pipes level within 1.5 to 2.5 hours. In case of vessel lower head failure, the corium can interact with the water with a beneficial effect, its partial cooling, and an a priori negative effect, a violent corium-water interaction called “steam explosion”.

In the framework of the 3rd PSR for the 1300 MWe PWRs, EDF was asked by the French Nuclear Safety Authority to choose the most suitable management of the water in the reactor cavity during a severe accident taking into account the consequences of such steam explosions on the whole 1300 MWe PWRs reactor safety.
EDF evaluated consequently different water management strategies:
- allowing the reactor cavity to be passively flooded up to the maximum level available (RCS pipes level), following inner containment spray activation (FIG. 1),
- flooding voluntarily the reactor cavity up to the maximum level available (RCS pipes level),
- allowing the reactor cavity to be passively flooded up to a relatively low level, corresponding to the reactor sump water level during the accident, thanks to a direct connection between the reactor cavity and the sump (FIG. 2),
- preventing the reactor cavity to be flooded (dry cavity) (FIG. 3).

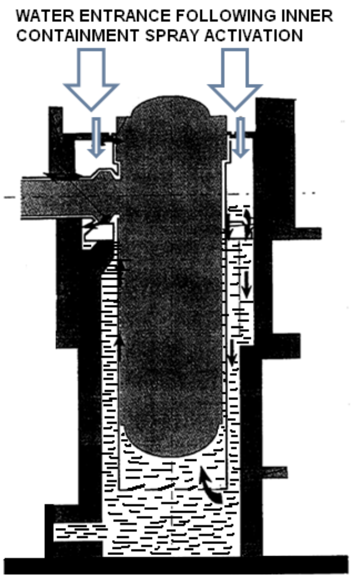

Figure 1: reactor cavity passively flooded up to the maximum level available (schematic view)

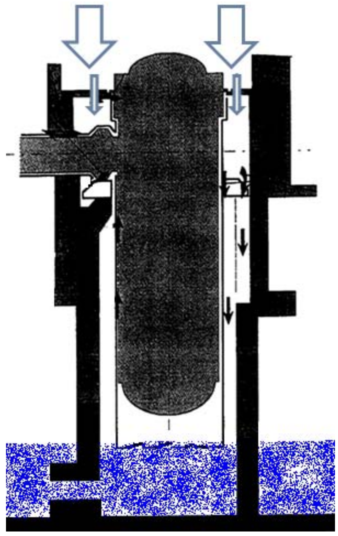

Figure 2: reactor cavity passively flooded up to a relatively low level, thanks to a direct connection between the reactor cavity and the sump
(schematic view)

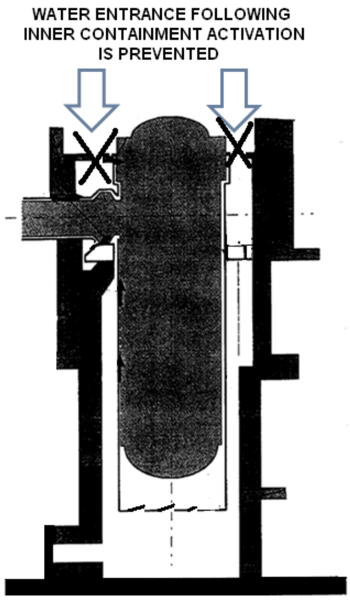

Figure 3: preventing the reactor cavity to be flooded (dry cavity)
(schematic view)

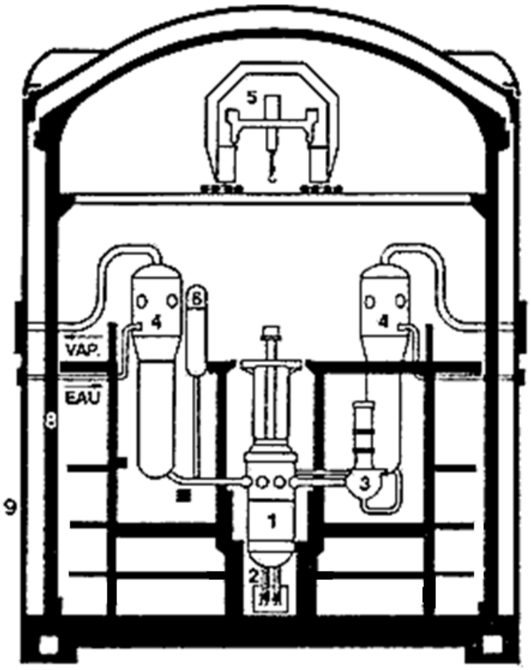

Figure 4: schematic view of a French1300 MWe PWR horizontal floors surrounding the reactor cavity

The French 1300 MWe PWRs have a reactor cavity surrounded by horizontal concrete floors (FIG 4). The extremities of these floors do not touch the inner containment wall and EDF performed calculations to know whether or not, under a steam explosion in the reactor cavity, these floors could impact the inner containment wall and endanger the whole containment integrity. After choosing a "realistic" steam explosion pressure loading in the reactor cavity, EDF performed two-dimensional calculations of the reactor cavity walls and containment internal structures mechanical response for a 1300 MWe reactor. The effect of three-dimensional mechanical calculations for such reactor was investigated using the behavior comparison between two and three-dimensional calculations performed on a 900 MWe PWR. As a conclusion, EDF considered the risk associated to a steam explosion in the reactor cavity to be acceptable and the EDF choice for the 3rd ten-yearly outage of the 1300 MWe reactors was to let the reactor cavity be passively flooded up to the maximum level available (RCS pipes level), following inner containment spray activation.
This choice was also driven by the fact that, for EDF:

- flooding voluntarily the reactor cavity up to the maximum level available would have required significant equipment modifications not feasible in the time frame of the 3rd PSR,
- controlling the reactor cavity to be passively flooded up to a relatively low level would have a beneficial effect by decreasing strongly the steam explosion consequences, but there is no insurance that a large corium pool can be completely cooled in the reactor cavity,
- maintening a dry reactor cavity would eliminate the steam explosion risk after vessel lower head failure ; but there is a need for air circulation in the reactor cavity during normal operation (so difficulties to maintain it dry) and there is also no insurance that a large corium pool can be completely cooled in the reactor cavity by a water injection after vessel lower head failure.

IRSN considers that some conditions must be fulfilled to obtain a sufficiently strong steam explosion in a reactor cavity leading to a loss of containment tightness: for example a large and constraint quantity of water in the reactor cavity and a fragmentation of the jet corium falling in the water after vessel lower head failure. The risk analysis has also to take into account the resistance of the internal structures and equipment in the containment, and the beneficial role of water in the reactor cavity for cooling the corium inside the vessel.

Firstly, the IRSN methodology to review the water management in the reactor cavity strategy proposed by EDF was based on a set of two and three-dimensional calculations of mechanical loading on a 1300 MWe reactor cavity walls obtained with the MC3D code (thermo-hydraulic multiphase flow code used at IRSN to model fuel-coolant interactions).
Some reactor cavity loadings have been choosen by IRSN for configurations allowing energetic steam explosion in an “academic” configuration (central vessel lower head breach for instance). Using these reactor cavity loadings, IRSN performed then three-dimensional mechanical calculations of the 1300 MWe reactor containment internal structures response.

Due to the multiple configurations that can exist in a real accident (for instance lateral breach) and to the remaining uncertainties in the field of steam explosion modelisation (corium fragmentation modelling…), IRSN cannot not guarantee that the loading he has used in reactor cavity mechanical calculations can be considered as conservative even if more severe than the EDF loading.

The structure response calculations have shown that the extremities of the horizontal concrete floors surrounding the reactor cavity can move or not up to touch the inner containment wall depending on the 1300 MWe PWR type. They also showed a strong mechanical degradation of the reactor cavity and of the surrounding concrete structures.

These results were then complemented by considerations on the plant management possibilities in various situations depending on the status of the internal structures and equipment in the containment after the steam explosion.

For IRSN, this analysis:

- raises questions about potential consequences, for example on heavy equipments (vessel, SGs) and containment penetrations tightness,
- shows a potential for equipment dysfunctions (if not destruction): what is the status of water injection pipes? how to inject water? where is going the injected water if the cavity walls are damaged? could the debris created by the steam explosion lead to sump clogging and consequently to the loss of the inner containment spray under water recirculation mode?

All these issues show that the possibility of an intense steam explosion in the reactor cavity threats the efficiency of accident management in addition to a possible containment tightness damage.

Moreover, IRSN also considered that the beneficial impact of a severe accident management strategy based on the reactor cavity flooding (voluntarily or not) on the in-vessel corium cooling is limited:

- both a water injection into the vessel (if the core is melting, it is probably because there is no water injection into RCS) and a fully flooded reactor cavity (it needs fews hours) are required to avoid a vessel failure,
- even if these two previous conditions are fulfilled, the vessel lower head failure may be inevitable with a high decay heat level and a large corium melt relocated in the vessel lower head, i.e. for accidents leading to a fast core melting.

Consequently, in order to reduce as low as reasonably practicable the radiological consequences of a severe accident, IRSN is favourable to severe accident management strategies preventing strong energetic phenomena, such as a steam explosion (at vessel lower head failure). In particular, the reactor cavity should be kept dry until the vessel lower head failure. It eliminates a risk of large early radioactive release and increases the chances to keep functionnal the structure, system and components (SSCs) needed for severe accident management after vessel failure (e.g. water injection means) and the chances for corium cooling outside the vessel, in particular due to its spreading on a larger area.

4 OPENING OF THE FRENCH EMERGENCY FILTERED CONTAINMENT VENTING SYSTEM

After the TMI-2 accident, EDF has installed on all PWRs an emergency filtered containment venting system (EFCVS) based on a sand bed filter concept in order to ensure a voluntary filtered containment venting during a severe accident, if necessary. All French EDF PWRs presently in operation are consequently fitted out with sand bed filters with prefiltration (metallic filter) in the containment building at the entrance of the system (FIG. 5).

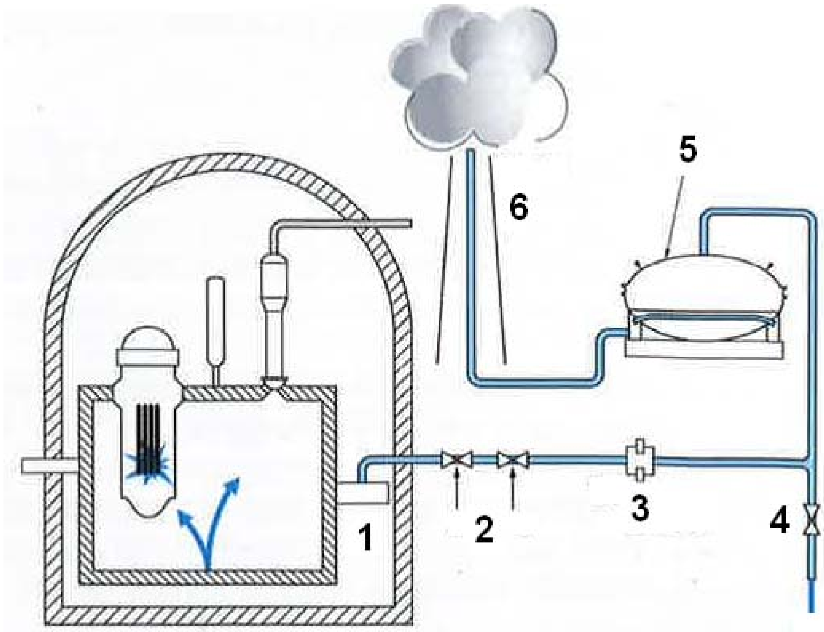

Figure 5: schematic view of the French1300 MWe PWR EFCVS
1 = Metallic prefilter ;
2 = Two isolation manual valves in series, operated locally behind a radiation shield
3 = Pressure letdown orifice ; 4 = Filtered dry air supply during normal operation
5 = Sand bed filter ; 6 = Plant stack, with small vent stack

The opening of the EFCVS is manual by means of two isolation manual valves in series, operated locally behind a radiation shield. IRSN noticed that, in other countries, the opening of EFCVs, is either redundant or diversified (manual or motorized valves, rupture disks …).
Consequently, in the framework of the 3rd PSR for 1300 MWe reactors, EDF was asked, by the french advisory group, to perform a feasibility study on the redondancy or diversification of the French EFCVS opening.

It can also be reminded here that, in the framework of long term operation, EDF should improve the efficiency of the present EFCVS to reduce the severe accident radiological consequences [2].

5 CONCLUSIONS

The 3rd Periodic Safety Review of the French 1300 MWe PWRs series includes the reactors upgrades to increase their robustness in the case of a severe accident. The analysis of these upgrades is based on both deterministic and probabilistic approaches.

The utility L2 PSA review by IRSN has confirmed the benefits of some modifications proposed by EDF (additional electrical supply of the containment isolation valves, new electrical command of RCS depressurization safety valves…). It helps also identifying some risks that can still be reduced by mitigation measures or better understood through R&D efforts.

The optimization of severe accident management strategies remains an area were progress is expected as shown by the management of the water in the reactor cavity. This is an example where the optimal strategy is not so easy to define. IRSN has concluded that solutions able to avoid possibility of energetic phenomena must be prefered because it eliminates a risk of an early containment failure (large early radioactive release) or additional equipment damage that can threat the long term accident management.

For on-going and future activities, the robustness of equipement to be used in severe accident conditions has a major interest. It concerns the behavior of equipement in severe accident conditions (pressure, temperature, radiation, …), their protection against internal and external events and also their reliability. The interest of a diversification or redundancy of the French ECVS opening is one example among many others.

6 REFERENCES

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