IMPLICATIONS OF THE FUKUSHIMA ACCIDENT FOR RESEARCH REACTOR SAFETY

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Abstract

Preliminary findings of Fukushima accident show that there is no evidence of major human errors as in previous accident in the nuclear power industry, namely, Three Mile Island (USA) and Chernobyl (Soviet Union), and that the initiating event, a natural catastrophe of extraordinary magnitude, caused a long term loss of the normal power supply producing the failure of each defence-in depth barriers with the final release of radioactive material to the atmosphere. It is worth noticing that the direct damage caused in Japan by the earthquake and tsunami far exceeded any damage caused by the accident at the nuclear plant. In the light of this event the question whether safety systems of research reactors will cope with this type of scenarios arises. The objective of this works is to present an overview of the current practice commonly used in the safety analysis in research reactors and to assess the capability to mitigate conditions from a beyond-design-basis event like the one occurred at Fukushima power plant.

1. INTRODUCTION

Thirty two years ago the most serious nuclear accident in the American history of the commercial nuclear power industry happened in U.S. Three Mile Island accident was a minor breakdown that was exacerbated by human error because operators were overwhelmed with information, much of it misleading. Although the accident caused no deaths or injuries, however, did lead to tougher safety standards for nuclear plants, particularly in term of improvements in training, quality assurance, engineering, operational surveillance and emergency planning. One of the major lessons learned was repeatedly discussed and is something that confronts the industry and regulator: the need to guard against complacency.

The worst nuclear accident in history happened 25 years ago in Chernobyl has convincingly demonstrated that the cost of ensuring the safety of nuclear facilities is significantly lower than that of dealing with accident consequences. The accident showed the importance of strict compliance with the basic and technical safety principles for nuclear power plants, of continuous safety analysis of operating nuclear power plants and of taking thorough account of the human factor.

While an initial attempt to identify the key lessons from the Fukushima accident are being carried out, some undoubted facts can be mentioned: there is no evidence of major human errors as in the previous accident and the initiating event was of an extraordinary magnitude causing a long term loss of the normal power supply producing the failure of each defence-in depth barriers with the final release of radioactive material to the atmosphere. It is worth to note that the direct damage caused in Japan by the earthquake and tsunami far exceeded any damage caused by the accident at the nuclear plant.

The Fukushima accident created a unique, although unfortunate, frame to seek to learn and improve worldwide nuclear safety, not only for NPPs but for research reactors, as well. It is realistic to think that future safety reviews will require facing such severe scenarios, so it is important to identify the design of engineered safety features that can mitigate undesirable consequences. While a Beyond Design Basis Accident (BDBA) is normally analysed in terms of dose to the public and frequency of occurrence (probabilistic analysis), to consider that the BDBA occurs due to and together with a catastrophic situation adds a severe stress component worse than the dose in itself, giving the analysis a new perspective.

The objective of this works is to present an overview of the current practice commonly used in the safety analysis in research reactors and to assess the capability to mitigate conditions from beyond-design-basis events. An assessment of the evolution and consequences of a combined LOCA event with long term power blackout in research reactors is made, as at present is clearly identify as a BDBA.

To fulfil this analysis a general description of the different types of reactor designs will be presented, from very low to very high power densities, with the focus on one of the basic safety function, "to remove the decay power".

A list of the Engineered Safety Features (ESF) normally considered will be presented as well as a description of their functions.

2. SAFETY APPROACH AND ENGINEERED SAFETY FEATURES IN RR

The basic purpose of reactor safety is to comply with the Safety Objectives, mainly, "to protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defences against radiological hazards maintain the integrity of the multiple barriers to fission product release", following IAEA guidelines 0.

This implies to fulfil the three basic safety functions, also stated by IAEA guidelines, which are:

— Shut down the reactor and maintain it in a safe shutdown state for all operational states or Design Basis Accidents (DBA);
— Provide for adequate removal of heat after shutdown, in particular from the core;
— Confine radioactive material in order to prevent or mitigate its unplanned release to the environment.

To maintain the integrity of the multiple barriers to fission product release and cope with potential human and mechanical failures, several levels of protection are implemented including successive provisions preventing the release of radioactive material to the environment. The classical five levels "defence-in-depth concept" foresees protections against damage to the plant and to the barriers themselves:

— Prevention of deviations from normal operation and of system failures by a sound and conservative design quality assurance, surveillance activities and a general safety culture;
— Control by detection and intervention of such deviations and failures so as to prevent abnormal transients from escalating into accident conditions;
— Control of the consequences of any resulting accident conditions in the unlikely event that the escalation anticipated in the design basis is not arrested by a preceding level;
— Control of severe conditions including prevention accident progression and mitigation of the consequences of a severe accident;
— Mitigation of the radiological consequences of significant releases of radioactive materials.

The safety functions that cope with accidental conditions (Level 3 of defence in depth) are fulfilled by Engineered Safety Features (ESFs) of the plant. These are active or passive features designed to mitigate the consequences of accidents and to keep radiological exposures to the public, the facility staff and the environment within acceptable values.

3. SAFETY EVALUATION – POSTULATED INITIATING EVENTS IN RRS

The term ‘postulated initiating event’ (PIE) refers to an unintended event, including operating errors, equipment failures or external hazards, that directly or indirectly challenges basic safety functions. Any of these events reaching level 3 of the defence-in-depth analysis, i.e., not being arrested by levels 1 and 2, are defined as a DBA.

Events are categorized in accordance with the frequency range of occurrence, typically, events with a frequency of occurrence < 10^{-6} /reactor year are not considered for the design and are considered Beyond Design Bases Accident (BDBA).

Combinations of initial events are usually considered by a probabilistic assessment to determine its likelihood. For example, an analysis of the consequences of a seismic event is included in the design bases, but its combinations with initial events of the other categories are considered BDBA.

Typically, each PIE is analyzed individually assuming “a single failure” on demand of a system or component during the sequence of the event and is usually assigned to one of the following categories, consistent with the type of reactor under study [2]:

— Loss of electrical power supplies;
— Insertion of excess reactivity;
— Loss of flow;
— Loss of coolant;
— Erroneous handling or failure of equipment or components;
— Special internal events;
— External events;
— Human errors.

For the purpose of this work, we will describe only those PIEs of categories (i), (iv) and (vii), as they will be the basis of the BDBA considered for the present analysis:

— Loss of electric power supplies: Low power reactors that can be cooled by natural circulation may not have a standby power supply beyond uninterruptible power systems (batteries) for instrumentation and control. Higher power research reactors may have diesel generators and rely on their functioning for the decay heat removal;
— Loss of coolant: Usually involves the primary coolant boundary rupture, including failure of the piping itself as well as of equipment or failure of beam tubes;
— External events: This category of PIE depends on the site, but may include: aircraft impact; wildfire in surrounding vegetation; industrial activities; military activities due to the presence of a military facility in the vicinity of the site; on-site activities outside the facility; transport accidents and natural disasters such extreme wind; seismic events and local flooding, among others.

4. RESEARCH REACTORS TYPES AND ESF

The safety approach was presented besides the groups of the PIEs that will be considered in the present analysis. The following sections describe the different types of reactors and the ESFs and how the basic safety function of decay heat removal is fulfilled.

4.1. Research reactors types

The description of the different types of research reactors will be based, this time, on the core power, mainly power density, with focus on the power removal issue and its safety features.

On one hand the total power is the figure to measure the radioactive inventory, the energy accumulated in the system and the required amount of water to keep the core under safe condition in case of accident while, on the other hand, the core power density must be considered to identify the need of a forced convection of cooling water even under shutdown.

Some present designs have high power densities, up to five times a PWR power density even when core powers are not so high, for that reason the definition of power for the different types must be taken, only, as an indicative value.

Commonly, high power reactors with high power densities require a pressure vessel and they are identified as tank-in pool reactor types. The difference with an open pool is based, mainly, in the operating pressure having them less than 2 absolute bars on the top of core against some tens of bars for the first type. Both types include a large reactor pool which acts as heat sink when the reactor is under safe shutdown condition, giving a considerable large "coolant inventory to power ratio" in case of accident.

For very low power reactors (i.e. most of TRIGA reactors), natural convection is enough to ensure the adequate cooling of the core in nominal conditions and, consequently, after shut down. In this type of reactors no damage is foreseen due to the low power density, also. They are considered to be out of the scope of this analysis and they will not be commented.

For higher powers, when forced convection is needed to remove the core power, the coolant flow direction, downward or upward, is a main issue. Both directions present advantages and disadvantages and preferences depend, mainly, on the maximum heat flux of the core.

4.1.1. Low power reactors

Normally, for this group of reactors the coolant flowing downwards is the more appropriate direction and provides adequate cooling with the following advantages:

- Reactivity Control enters from above so, the scram action is aided by flow direction;
- From the mechanical point of view, the core pressure drop does not act to unseat the core grid seal as there is no drag force upwards;
- Due to the flow direction and a decay tank, ^{16}N does not reach the pool surface.

The power of this type of reactors is limited, mainly, for the event of a pump stop. After reactor shutdown, a flow reversal occurs during the transition to shutdown cooling mode (natural circulation). This flow reversal together with the low working pressure restricts the power that can be achieved. Typical power densities for this group are lower than 100 kW/l.

A confinement system rather than the integrity of the containment structure is adopted to confine a potential release of radioactive gases and particles. A negative pressure is permanently maintained inside the reactor building by ventilation systems, including exhaust stacks or vents to the external environment, filters and blowers.

4.1.2. Medium power reactors

The core power dealt with within this group, is in general, higher than 10 MW, typically, around 30 – 40 MW and up to 70 MW, with power densities higher than 100 kW/l.

As it was already mentioned the maximum power that could be removed depends, also, on the power density or, more precisely, on the maximum heat flux. This is the reason why reactors of up to 70 MW, like Osiris reactor, can be cooled using this scheme.

Generally speaking, for this type of reactors the forced coolant flow is in the upward direction presenting the following advantages:

- The hydraulic resistance dynamically pressurizes the system;
- There is no flow reversal when the main pump/s stop;
- Shutdown cooling mode, such as natural convection, smoothly establishes after a failure;
- A riser enhances the natural circulation flow under shutdown cooling mode.

The main disadvantage is caused by the drag force of the upward flow, implying that the fuel assemblies must be clamped and the reactivity control elements moves in the opposite direction to the flow in case of reactor trip.

A Confinement system is also commonly adopted for these cases, but some designs have additional features to ensure the negative pressure in case of accident.

4.1.3. High power reactors

For this range of powers and compact cores, i.e., high power densities, the tank-in pool design is commonly adopted.

In general they require forced coolant flow for some hours after shutdown to remain safe. To ensure the coolant flow, batteries or diesel generators are maintained to power the emergency coolant pump operation. After that period the reactor is in a safe condition as long as the fuel remains covered with water.

Unlike the open pool type, the cooling of core is independent of the flow direction taking the corresponding advantages of each direction but, on the other hand and due to pressurization, some additional systems are required to avoid the sudden depressurization of the core cooling systems.

Both, confinement and containment system are adopted in accordance with the radioactive inventory.

4.2. Engineered safety features (ESF)

The need for ESFs is always determined by the analysis of accidents that could occur. It is possible that, for a particular design, the analysis performed for the Safety Analysis Report demonstrates that some ESFs considered “usual” are not needed.

The ESFs are implemented through the design of components, structures and systems of the plant. They include both safety systems and components of safety related systems that perform selected safety functions. Following the IAEA safety Standards [1], a research reactor should have: (i) a Protection system, (ii) a Shutdown system, (iii) a Residual/Emergency Core Cooling system and (iv) a Reactor Containment/Confinement system to fulfil the three basic functions already mentioned.

Additional systems may be required such as:

- Emergency Make-up Water System (EMWS) which has the function of compensating the loss of water from the reactor in emergency conditions, i.e. during a LOCA;
- Emergency Electrical Power Supply which will be able to supply sufficient power to systems and equipment (all relevant ESFs) to ensure their capability to perform their safety functions when required, including the event of a loss of off-site power

Typically, there are also components or subsystems that perform selected safety functions, such us:

- Reactor Pool Pressure Boundary with the overall safety function of keeping available sufficient quantities of reactor coolant in case of a LOCA;
- Flywheels, pressurized tanks or Pony motors, to ensure a pump coast-down compatible with the required reactor core cooling after reactor shutdown until natural circulation establishes;
- Check or Flap Valves allowing the removal of heat from the core by natural circulation in the long term;
- Components of the facility heating, ventilation, and air conditioning (HVAC) system used to mitigate the consequences of accidents as part of the ESFs of the confinement or containment system;
- Core Cooling System Pressure Boundary: This is considered an ESF in case the pressure boundary design has to be conservative enough to turn its failure in an unlikely event, ensuring that a failure within the DBA scenario will not develop in a larger rupture, namely the Leak-Before-Break and Break Preclusion concept.

As stated previously, the focus of attention is related to the adequate heat removal and control of the energy released in the system to prevent overheating and, in the most severe cases, core melting. Adequate heat removal is required in all modes of reactor plant operation, including normal power generation, decay heat removal during shutdown and in the long-term and accident conditions. The following sections go deeper on the role of these ESF related to decay heat removal.

4.2.1. Residual/Emergency core cooling system (RECCS)

The main function of the RECCS is the removal of the heat from the core once the reactor has been shutdown in the event that the PCS is not running and core cooling by natural circulation is not feasible.

For the case of open pool type the design alternatives are such as to avoid the need of this system. Passive features, such as coolant flow direction, inertia fly-wheels, flap valves and core chimney, are enough to provide an adequate cooling in case of a total loss of power supply.

This is not so for the tank-in pool designs in which a RECCS is provided and is powered by the On-site Emergency Power Supply to cope with black-out events.

4.2.2. Emergency make-up water system (EMWS)

This system has the function of compensating the loss of water from the pool in case of a LOCA in order to maintain the core under water.

Depending on the total power and the maximum heat flux in the reactor this system may be either neglected, specified as a passive system or powered by the On-site Emergency Power Supply.

4.2.3. Reactor pool coolant boundary

This ESF has the overall safety function of keeping available sufficient amounts of reactor coolant. TABLE 1 shows the fraction of power generated in the core vs. time after shutdown, its integrated energy and the equivalent evaporated mass of water/MW of steady state nominal design power.

TABLE 1: INTEGRATED ENERGY AND EQUIVALENT EVAPORATED MASS OF WATER/MW

Time after shutdown	%Steady State Power	Energy (MJ/MW)	Mass of Evaporated Water (kg/MW)
1 s	12.4	0.3	0.12
10 s	7.5	1.1	0.5
100 s	3.7	5.4	2.4
1 h	1.5	75.5	33.5
10 h	0.8	353.8	156.8
1 d	0.6	703.5	311.7
6 d	0.3	2452.1	1086.5
1 m	0.2	6771.3	3000.1

It is essential to achieve the medium term emergency cooling by natural convection while, in the long term, the pool coolant inventory defines the time till the operator action is required to manage the accidental sequence. For example, rough numbers to heat-up and evaporate a water amount of ≈ 50 m^3 (i.e. a pool of 3.5 m ø and 5 m of water column height) for a reactor power of 20 MW, give as a result almost 1 month to evaporate the water above the core and it still remains covered.

For a tank-in pool type reactor, this ESF has an additional function as the EMWS injects water to the PCS from the pool inventory until the removal of the decay heat is compatible with natural circulation.

4.2.4. Provisions for flow and pressure decrease

For the open pool type designs this feature is supplied to provide a coast-down flow compatible with the decay power until natural convection establishes ensuring the adequate cooling of the core.

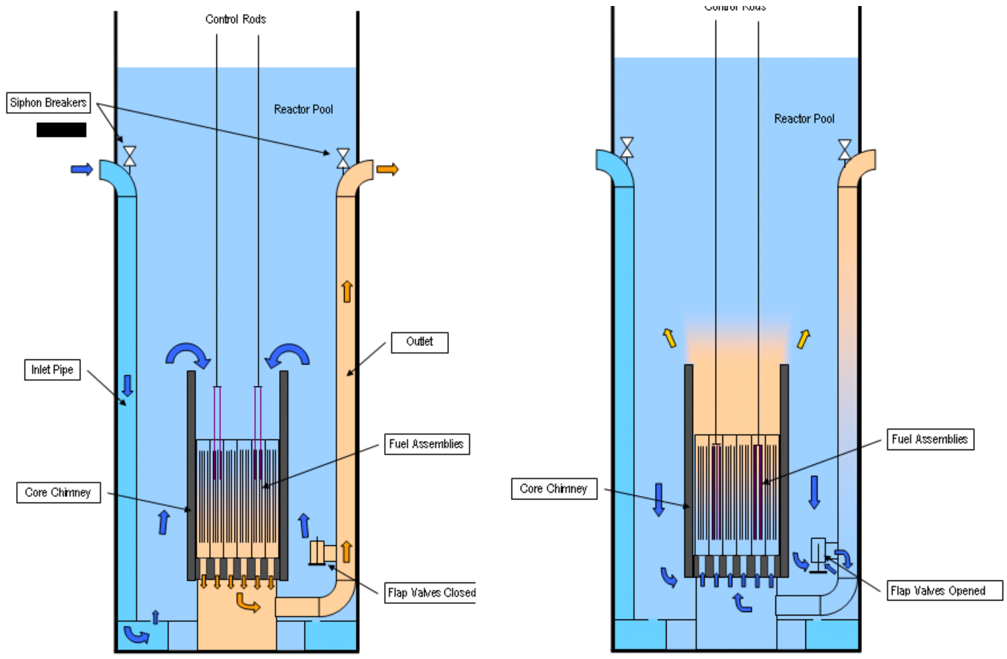
Depending on flow direction this component allows either a delay for flow reversal, (for downward flow), or postpones the natural circulation regime, for the upward flow option.

For the pressurised tank-in pool design the flow coast-down is not enough as a slow pressure decrease is also required and most demanding. For this type of reactors this feature, flow and pressure decrease, provides the cooling condition until the RECCS starts running.

4.2.5. Valves for natural circulation

The function of these valves is to connect the PCS line/s to the reactor pool and they have the safety function of allowing natural circulation to remove the decay heat of the core only after the PCS pump coast-down is complete. In other words, these valves are designed to deal with a black-out scenario.

In some designs flap valves play the role of siphon breakers, avoiding the total drainage of the pool in case of a LOCA. Other designs adopt redundant flap valves at two different heights of the PCS pipes in order to cope with the combination of LOFA + Black-out events.
Figure 1 shows a typical open-pool type reactor with downward flow under nominal cooling conditions and natural circulation after a black-out event occurs.



Forced downward flow   Natural Circulation

Fig. 1. Core cooling modes.

5. KEY ISSUES OBSERVED AT FUKUSHIMA

Clearly, research reactors do not have the same heat removal concerns as NPPs. They have a much lower heat load, much lower operating temperature and much shorter operating cycle.

However, in the light of recent event some questions arise regarding if safety systems of RRs will function after a severe earthquake and if combined initiating events can happen.

Following there is a list of the key issues observed at Fukushima and how RR could manage these issues but previously there is a comparison in Table 2 of RRs and a BWR similar to Fukushima plant in terms of total power and power density and maximum heat flux. As it was already mentioned, the total power is related to the coolant demand to ensure core cooling in the long term while the power density/heat flux/thermodynamic conditions are related to the cooling demand in the short term.

TABLE 2: COMPARISON OF RRS AND A TYPICAL BWR

	Low Power	Medium Power	High Power	NPP
Facility	*RA-3*	*OPAL*	*FRM-2*	*BWR*
Power (MW_{th})	10	20	20	3600
q'''_{ave} (kW/l)	60	250	1100	60
q''_{max} (W/cm^2)	100	210	440	110
P_{in} (bara)	≈ 2	4	20	70

5.1. Emergency power supply

Fukushima issue: The loss of offsite power due to the earthquake and onsite AC power due to the tsunami, resulted in a complete station blackout which, in turn, led to fuel overheating and damage. A single external event (the tsunami) disabled all diesel generators at the station simultaneously causing what a priori was considered independent PIEs.

Regarding research reactors, in general, low and medium power reactors have a large ratio of water inventory to power so; they do not need electricity to overcome a failure in the electrical power supply. The cooling of the reactor is ensured by a coast-down flow compatible with the decay power until natural convection establishes. However, high power reactors (tank-in pool designs) require the actuation of the RECCS for some period after shutdown before the core can be cooled by natural circulation.

It is assumed this is the major key issue and new regulation will seek to verify and assess the capability of the plant to mitigate conditions from BDBA and take appropriate actions if vulnerabilities are identified. As an example, in the next section a combined initial event of total black-out with LOCA event is analyzed for a medium power reactor.

5.2. Fresh Water Supply

Fukushima issue: The unavailability of a large amount of fresh water for the cooling system after the earthquake caused an unprecedented emergency response, injecting sea water into the core.

For research reactors, as stated previously, considering the large water inventory and the low rate of evaporation, in case of an emergency, the unavailability of a large quantity of fresh water can not be considered as a mayor issue. Some designs include stored onsite water by the EMWS which can provide enough pressure from gravity to supply water to the reactor core in the event that all pumps are lost.

5.3. Hydrogen Generation

Fukushima issue: The loss of power supply caused a deficient fuel cooling leading to the fuel overheating, enabling rapid oxidation of the zirconium cladding and generating large amounts of hydrogen, which is extremely flammable and led to the explosion/destruction of the reactor buildings.

Research reactors fuel has aluminium clad and the hydrogen production due to steam oxidation of aluminium is minimal so, a severe damaging hydrogen explosion such as the one occurred at the Japanese NPP is not a believable scenario. However, some developments of new fuel with UMo are considering the use of zirconium cladding for plate type fuel.

5.4. Spent fuel pools

Fukushima issue: One of the major issues in the Fukushima accident involved the spent fuel pools. Some radioactivity releases from the Fukushima plant might be caused from the spent fuel pools. Lack of the cooling (due to loss of power supply) combined with the elevated location (damaged from hydrogen explosions) and earthquake-induced water leakage have aggravated the accident.

In research reactor, the stored energy and radioactive inventory is orders of magnitude lower than a NPP. Additionally, the dispersed fuel used in research reactor has a significantly different behaviour in term of fission product retention.

Some research reactors have the spent fuel pools of stainless-steel lined and built into the concrete structure seismically qualified.

In order to evaluate the consequence of a possible BDBA in this system, it would be important to identify the safety factor precluding structural damage and the sources of emergency water that can be provided, after a severe earthquake and loss of electricity.

5.5. Containment failure

Fukushima issue: Due to the station blackout, the containment was vented to prevent containment over-pressurization. Some vented gases leaked into the reactor building, which had no ventilation (again due to the station blackout).

In most of the research reactors the confinement is not achieved by the integrity of the structure. The building boundaries, access doors with sealing airlocks, pipe penetrations and electric cable penetrations etc., can be assumed "airtight", but an inwards leakage rate is accepted at the nominal negative pressure. The air is ventilated and conditioned by a single system. To ensure that the safety function of this ventilation system remains operational in case of accident, backup power systems are designed to remain running after an event. Confinement systems can be isolated, but usually are not capable of supporting positive internal pressure without leaks.

Only in case of a long-term loss of electrical power combined with an event with radioactive release in the facility, the safety function of the confinement could be threatened.

6. CONCLUSIONS

Fukushima accident has opened a new discussion regarding the safety features of research reactors and how this kind of accidents can be managed.

Several designs were considered emphasizing the basic safety function of “decay heat removal”, considering a wide range of core powers, from some MW up to some tens of MW.

A list of ESF has been analysed and considering the different design characteristics it is concluded that a BDBA like the one occurred at Fukushima can be managed properly by means of passive systems and components and that there is no need of any emergency power supply, particularly, for the case of open pool type reactors. For the designs of the tank-in pool type farther analysis are required to minimize or avoid the need of emergency power sources.

Last but not least, some findings arise involving nuclear designers, operating organizations and regulatory authorities, for example, the continuous update of natural hazards database, the defence-in-depth, physical separation, diversity and redundancy concepts applied to extreme external events and periodic review and/or upgrade of safety analysis concerning these events, just to mention a few.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Research Reactors, IAEA Safety Standards Series NS-R-4, IAEA, Vienna (2005).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Research Reactors, IAEA Safety Reports Series 55, IAEA, Vienna (2008).