Nuclear Safety
NEA/CSNI/R(2017)12/ADD1
January 2018
www.oecd-nea.org

Developments in Fuel Cycle Facilities after the Fukushima Daiichi Nuclear Power Station Accident

Workshop Proceedings

Aomori City, Japan
15-17 November 2016

Appendix 3

BETTER POLICIES FOR BETTER LIVES

NUCLEAR ENERGY AGENCY

Appendix 3 - Conference papers

Proceedings of the CSNI Workshop on Developments in Fuel Cycle Facilities (FCFs) after the Fukushima Daiichi Nuclear Power Station (NPS) Accident
Aomori City, Japan
15-17 November 2016

Session 1- Feedback of post-Fukushima safety reviews performed for FCFS

Chairpersons: J. Marcano (NRC), O. Nevander (OECD/NEA)

TECHNICAL EVALUATION OF MODIFICATIONS TO A URANIUM HEXAFLUORIDE FACILITY TO PROTECT AGAINST SEISMIC AND TORNADO MISSILE EVENTS

J. Marcano Lozada, P.E.
U.S. Nuclear Regulatory Commission
Mail Stop TWFN-4B16
Washington, DC 20555

jonathan.marcano@nrc.gov

ABSTRACT

On March 11, 2011, the Tohoku–Taiheiyou–Oki earthquake occurred near the east coast of Honshu, Japan. This magnitude 9.0 earthquake and the subsequent tsunami caused significant damage to at least four of the six units of the Fukushima Dai-ichi nuclear power station. As a result, there was a loss of offsite and onsite electrical power systems. Subsequently, the NRC staff performed a systematic evaluation and inspection of selected fuel cycle facilities, in light of the lessons learned from the accident at the Fukushima Dai-ichi Nuclear Power Plant, to confirm that licensees were in compliance with applicable regulatory requirements and license conditions; and to evaluate their readiness to address natural phenomena hazards (NPH) events and other licensing bases events related to NPH. As a result of the inspections, the NRC staff identified a concern at a licensed facility that converts uranium concentrate received from uranium mines to uranium hexafluoride. Specifically, as a result of its inspection activities, the staff found that the facility may not have been adequately protected from an unlikely but credible event, such as an earthquake or tornado. After the inspection, the licensee agreed to fortify its building and processing equipment. This paper provides a summary of the staff technical evaluation to determine that the risk to public and workers for the upgraded licensed facility provides reasonable assurance of adequate protection of worker and public health and safety.

1. Background

In May, 2012, in response to the events at the Fukushima Daiichi site in Japan, the Nuclear Regulatory Commission staff inspected the Honeywell Metropolis Works (MTW) facility, in accordance with NRC

Temporary Instruction (TI) 2600/015 (NRC, 2011). During the inspection, the staff identified concerns related to protection of liquid uranium hexafluoride (UF_6) from a seismic or tornado event and evaluation of Honeywell's UF_6 and hydrogen fluoride (HF) bounding source terms, which Honeywell used as the basis for the MTW facility's Emergency Response Plan (ERP).[1] During the May 2012 inspection, the NRC staff found that the process equipment in Honeywell's Feed Materials Building (FMB) lacked seismic restraints, supports and bracing to ensure equipment integrity during credible seismic events or tornadoes.

The NRC staff documented the results of this inspection in a letter to Honeywell dated August 9, 2012 (NRC, 2012a). In that letter, the staff identified two apparent violations of NRC regulations relating to the ERP. Honeywell, which was in a maintenance shutdown at the time, agreed to keep the MTW facility shut down until it addressed the issues raised by the NRC inspection. On October 15, 2012, the NRC issued a Confirmatory Order (EA-12-157) to Honeywell. The Confirmatory Order formalized Honeywell's commitment to remain shut down until it took adequate corrective actions and those actions had been verified by the NRC (NRC, 2012b).

The Confirmatory Order, in part, identified corrective actions needed to be completed at the MTW facility in order to provide reasonable assurance that public health and safety will be adequately protected. The confirmatory order required Honeywell to provide an evaluation of external events at the MTW facility that clearly defines and provides the safety bases for the following:

- seismic and wind design
- structures, systems, or components relied on to protect workers and the public during both intermediate and high consequence events caused by seismic and wind hazards
- defining intermediate and high consequence events for non-radiological releases caused by these events
- defining "unlikely" and "highly unlikely" as the basis for determining acceptable limits for seismic and wind events[2]

---

1 UF_6 is an NRC-licensed material which reacts with moisture in air to produce HF, a toxic gas.

2 As part of its ISA, Honeywell uses a risk matrix approach to identify accident sequences for which certain combinations of consequences and likelihoods yield an unacceptable risk index. Honeywell defines the permissible likelihood of occurrence of uncontrolled accident sequences, for different consequence categories. Accident sequences with impacts categorized as high consequence must be highly unlikely and accident sequences with impacts categorized as intermediate consequence accident sequences must be unlikely. If these permissible likelihoods are exceeded, Honeywell designates controls to prevent the sequence or to mitigate its consequences. In Honeywell's ISA, these controls

Honeywell was also required to provide, to the NRC, a revised ERP and documentation of the design bases for the MTW facility and equipment as modified by Honeywell to meet the NRC order. Additionally, Honeywell must develop, implement, and have available for NRC inspection, the quality assurance measures that would be applied to the modifications needed for the facility to meet the requirements specified in the Order.

The NRC developed a Technical Evaluation Report (TER) titled, "Technical Evaluation Report for Safety Basis and Corrective Action Plan Leading to Restart," ADAMS Accession
No. ML13190A165, to document the staff's review of Honeywell's evaluation of seismic hazards, structural design and design margins, chemical consequences, integrated safety analysis and controls, and the potential risks from seismic and high wind events. The focus of the staff's evaluation was to determine whether Honeywell's provides an adequate safety basis for the proposed modifications to the MTW facility. The following is a summary of the staff's evaluation.

2. <u>Technical Evaluation</u>

The Confirmatory Order required Honeywell to address the adequacy of the safety basis for seismic and tornado events at the MTW facility. Honeywell used its ISA methodology, as the means to demonstrate, in a risk-informed fashion, that the safety basis for these events is acceptable. In addition to defining and demonstrating the safety basis for these events, Honeywell was also required to provide definitions of the terms "unlikely" (between 10^{-3} and 10^{-4} per event per year) and "highly unlikely" (less than 10^{-4} per event per year) for the purpose of providing limiting risk performance criteria for acceptability of these external events. Honeywell used these definitions to demonstrate that its evaluation of these events meets their risk performance criteria for consequences and accident sequence likelihoods. Honeywell's overall conclusion is that potential high consequence events due to seismic or tornado missile hazards will be highly unlikely, thereby providing an acceptable demonstration of the safety basis for these events. The staff reviewed Honeywell's application of the ISA methodology and the results of the ISA evaluation, and determined that the corrective actions provide an acceptable level of safety for seismic and tornado missile hazards.

Honeywell proposed modifications to the MTW facility to provide fortification of pipes, equipment and components that contain liquid UF_6 (the hazardous material at risk for release). Honeywell also proposed modifications to the building structure. The purpose of these modifications is to prevent and/or mitigate the potential for release of UF_6 due to a seismic or tornado missile event. Honeywell's design basis earthquake (DBE) is the 475-year return period earthquake based on the 2002 USGS seismic hazard maps.

are referred to as plant features and procedures (PFAP).

Honeywell proposes modifications to the structure and piping and equipment restraints to this design basis. For the discussion of the seismic margin assessment, see Sections 7.0 and 8.0 of the TER, the Evaluation Basis Earthquake (EBE) is used. The EBE is the 475-year return period earthquake based on the 2008 USGS seismic hazard maps. Differences between the DBE and EBE are described in Section 7.0 of the TER.

Honeywell's safety basis includes modifications made to meet the design basis as well as additional modifications and controls to the MTW facility which provide additional margin to protect against earthquake ground motions that exceed the DBE and EBE. The staff reviewed the documentation to evaluate the adequacy of these modifications and the controls Honeywell proposed to prevent an UF_6/HF release at the MTW facility or to mitigate the consequences of such a release. The staff systematically evaluated the structural impacts on various systems and components and the possible UF_6/HF releases associated with these expected impacts. The staff also evaluated the assumptions and justifications made by Honeywell in reaching their overall conclusion that potential high consequence events due to a seismic or tornado missile event will be highly unlikely. Key information reviewed by the staff included the seismic impact on the facility and structures; the associated frequency of the seismic event; the structural response of systems and components, including the building structure and the analytical conservatisms and margins associated with the structural analyses; the design of tornado missile barriers, the quantities and locations of liquid UF_6 at risk; various parameters associated with the modeling of releases and determination of impact to the public including the expected reactions and release rates of UF_6/HF; and the consequences of various UF_6/HF release scenarios resulting from a seismic event.

The ISA analysis as performed by Honeywell evaluates hazards and their credibility, determines or estimates the likelihood of events including failures, evaluates the consequences of hazards and their level (high, intermediate, or low), and determines safety controls needed to prevent or mitigate accidents and designates them as plant features and procedures (PFAPs). PFAPs are the safety controls identified and credited by Honeywell in their ISA analysis demonstration of acceptable risk performance. The staff concluded that evaluating seismic and tornado missile events by a method that uses a consequence-likelihood risk matrix is consistent with currently approved practices and is acceptable for the analysis provided by Honeywell to support their safety basis.

Evaluation of Seismic Safety Basis

In evaluating Honeywell's analysis, the staff considered the overall likelihood of the accident sequence to be the product of the frequency of the initiating seismic event; the probability of release of the material at risk (i.e., UF_6 is in the liquid state); the probability of failure of the PFAP that could result in a high consequence release; and the probability that a member of the public could receive an exposure resulting in a high consequence (a probability that takes into account distance to the nearest resident and prevailing wind direction).

Equation 1:

Frequency of initiating seismic event	*	Probability that the material at risk is available for release	*	Probability of failure of the PFAPs resulting in a possible high consequence	*	Probability that public experiences high consequence	$<10^{-4}$/year

To evaluate a range of credible events, the staff considered the possible accident sequences associated with the level of an earthquake in three ranges: (1) the range of seismic loads impacting the facility up to the 475-year return period; (2) the range of seismic loads beyond the 475-year return period, up to the estimated design safety margin limit for the FMB piping supports and vessel restraints, a 1300-year return period earthquake; and (3) the range of seismic loads beyond the 1300-year return period. The staff chose these ranges based on the analysis that Honeywell provided regarding the seismic performance of the Feed Materials Building (FMB) and the restraints for the process equipment and piping. The following is a summary discussion of the staff's evaluation of the accident sequences for each of these three ranges.

For the mitigated accident sequence initiated by a 475-year return period earthquake, no release of UF_6/HF is expected to occur. The staff makes this finding based on Honeywell's evaluations supporting the ability of the proposed MTW facility modifications and PFAP controls to withstand the 475-year seismic event without damage to these safety features, as well as their ability to perform their safety functions and prevent unacceptable consequences.

In addition to demonstrating that the PFAPs provide acceptable prevention or mitigation of the consequences to highly unlikely (1 x 10^{-4}/year), Honeywell also proposed other layers of protection that prevent or mitigate a high consequence event to the public. These layers include seismically activated isolation valves on all tanks that contain liquid UF_6 and modifications to the distillation area of the building to create confinement of possible releases to the first three floors. For the areas with large quantities of liquid UF_6, the isolation valves and confinement provide additional layers of protection against large offsite releases. In Section 9.0 of the TER, Chemical Consequences, the staff discusses the benefits of these additional layers of protection. The staff agrees with Honeywell that these additional protections, although not precisely quantifiable, support a finding that the overall likelihood of a release resulting in high consequences would be at or below the value demonstrated by Honeywell for the ISA analysis of a seismic event with a 475-year return period and conservatively below the performance requirement for highly unlikely.

The staff also considered the contribution of the probability that the material is at risk (i.e. UF_6 in the liquid state) and the probability that a member of the public experiences a high consequence event (taking into account distance to the nearest resident and prevailing wind direction)—factors not directly considered by Honeywell in its analysis. Honeywell's evaluation of the likelihood of a release is based on conservatively assuming that all material at risk has a probability of 1.0 of being available for release when, in fact, certain large amounts are only available for small periods of time (probability <1.0). In addition, Honeywell's evaluation assumes that the probability of meteorological conditions that could result in a high consequence is 1.0 when, in fact, more extreme meteorological conditions that may be needed to result in high consequence events have probabilities up to two orders of magnitude lower. These conservative assumptions provide additional assurance that the likelihood of the accident resulting in high consequences to the public as a result of the 475-year return period event will be reasonably below the performance requirement for highly unlikely.

In addition to the 475-year return period earthquake, the staff evaluated the credible scenarios associated with frequencies of earthquakes beyond the 475-year return period. Honeywell provided information regarding the response of the modified FMB for forces beyond the assumed ground accelerations of the seismic accident sequence. Although Honeywell provided this information to estimate the design margin in the FMB structure for the 475-year earthquake, the staff used this information to make qualitative evaluations of risk for other credible seismic events with ground accelerations greater than those assumed for the 475-year return period. This evaluation is consistent with current staff ISA guidance requiring that all credible events be analyzed as part of the ISA.

Based on the staff's review of the information provided by Honeywell, the median seismic capacity of the FMB is expected to be 2.51 times the EBE, indicating that the structure has the capability to meet structural performance requirements for seismic loads equivalent to an earthquake with a 1700-year return period. In Section 8.0 of the TER, the staff also concluded that the restraints for the UF_6 process equipment and piping are appropriately designed to meet structural performance requirements for seismic loads equivalent to an earthquake with a 1300-year recurrence interval. Because the sequences of equipment and piping failures and the associated extent of damage are complex to characterize, but are closely related to possible releases, the staff has assumed, for the purposes of evaluating risk, that the initiating frequency of an earthquake resulting in significant releases of UF_6 from damaged equipment or piping is the more conservative value of the 1300-year return period.

For return periods greater than 475 years and less than 1300 years, the staff looked at possible accident sequences and qualitatively estimated probabilities of releases based on structural analyses and determinations provided by Honeywell. The 1300-year return period represents Honeywell's assumed design basis for the piping supports and vessel restraints for those components that contain liquid UF_6. For select combinations of pipe and equipment failures the staff also performed consequence estimates as described in Section 9.0 of the TER, Chemical Consequences. For this range of possible sequences, the staff concluded that the likelihood of failures resulting in a high consequence release will increase as the earthquake return period increases. Honeywell has shown, via calculations and evaluations of modifications to tanks, pipes and other components, that the likelihood of failure for these components up to the design basis 1300-year return period is well below that needed to demonstrate acceptable risk performance. The staff reviewed Honeywell's calculations of building response and equipment modifications, as discussed in Section 8.0 of the TER, Design of Structures, Systems, and Components, and concluded that the assumption that the likelihood of failure for up to the 1300-year return period is supportable by a finding that there is acceptable performance of the building and components and that there is reasonable assurance that proposed accident sequences in this range could be demonstrated to be highly unlikely. Honeywell demonstrated that the FMB structure has the capacity to withstand up to the 1700-year return period earthquake without major damage, so the initiating event frequency that could be assumed for a high consequence event may be lower, further supporting the staff determination that the 1300-year event design basis of the components is acceptable. The staff determination is also supported by additional measures of protection or mitigation for the overall likelihood of failure resulting in a major release. Also, similarly conservative assumptions as stated above regarding the availability of material at risk and the meteorology could be credited for this analysis. Although not quantified, these layers of protection and conservative assumptions provide additional assurance that the consequences of events with initiating frequencies up to the 1300-year return period are reasonably below the performance requirement for highly unlikely.

For return periods greater than 1300-years, the staff considered the frequency of the initiating event resulting in a high consequence to be the frequency of the 1300-year return period earthquake ($8x10^{-4}$). The staff considered the probability that the public experiences a high consequence to be the probability of prevailing winds in the direction of the nearest residence. The product of these two probabilities alone is less than 1.07×10^{-4} which nominally meets the likelihood criteria of highly unlikely without consideration of other factors. However, given that the likelihood criterion is only nominally met with the ISA-type demonstration, assuming no additional credit for other factors, the staff further explored a conservative risk evaluation for an individual member of the public. Subsection 5.2.4 of the TER provides further discussion of this evaluation and conclusions regarding adequate protection. The following section provides a summary of the quantitative risk evaluation.

Staff Quantitative Risk Evaluation for Seismic Hazard

The staff conducted an evaluation of risk to individuals from seismic events at the MTW facility with the proposed modifications. This evaluation used results of the seismic structural analyses submitted by Honeywell, but assessed risk and used criteria that are independent of Honeywell's methods and definitions. The purpose was to establish a realistic quantitative basis for an understanding of how risk is limited. The risk to individuals arises from the possibility that seismic structural failures could cause releases of UF_6. The risk results from the staff's evaluation were compared to quantitative risk guidelines. There are no quantitative risk guidelines in NRC guidance directly applicable to this case. Therefore the staff used guidelines from international authorities (ICRP, 1993; ICRP, 1991) that are consistent with NRC qualitative discussions of risk criteria (NRC, 2008; and NRC, 2004). These risk guidelines are expressed as frequencies below which risk of the specified health effect is considered acceptably limited. These guidelines are not risk or safety goals, which are typically described as "insignificant" or "negligible" risk, but higher values for judging acceptability (see NRC 2008, Chap. 4):

1) Risk to public of minor health effects: $< 1x10^{-2}$ /year
2) Risk to public of serious long-lasting health effects: $< 1x10^{-3}$ /year
3) Risk to public of fatality: $< 1x10^{-4}$ /year
4) Risk to workers of fatality: $< 1x10^{-3}$ /year

For worker risk, any large release of UF_6 could result in fatality, if the worker were unprotected and could not escape the plume. However, the frequency of the seismic event is less than 1/1300-years = $7.7x10^{-4}$/year, because Honeywell's safety analyses shows that piping failures will not occur for seismic loads up to this frequency. This frequency value meets the $1x10^{-3}$/year guideline 4 for workers.

The staff also calculated the consequences and bounding frequencies to a member of the public. The individual selected for this evaluation was at the closest offsite residence, and hence at highest risk. If the risk to this individual meets the above guidelines, then the risk to all other individuals also meets it. In evaluating this risk, the staff considered the previously mentioned probability equation for each of these guidelines (see Equation 1 above). The staff identified that, for the MTW facility, the individual at greatest risk is located at the nearest residence, a distance of 1850 feet (ft) in a north northeast (NNE) direction from the FMB. This direction also has the highest wind frequency (wind rose value from the SSW) and so is bounding on other individuals.

The staff evaluated the consequences and likelihood of a range of equipment failure scenarios, such as releases of UF_6 from tanks and piping, for different weather conditions. The likelihood of resulting consequences to the bounding individual offsite for a given scenario was compared to the above-mentioned guidelines. In the likelihood evaluations referred to below, the frequencies of limiting values of the seismic structural analysis were used to evaluate releases of UF_6 from tanks and piping. Specifically, tank failures were associated with the 1/1700-year condition, a frequency of $5.9x10^{-4}$ /yr, and piping failures with the 1/1300-year frequency ($7.7x10^{-4}$/yr). The number and location of tank and piping failures determines the amount of UF_6 that could be released. The frequencies (of $5.9x10^{-4}$ /yr and $7.7x10^{-4}$/yr) used here are upper bounds on the frequency of the tank and piping failures for which chemical consequences were evaluated. Thus, since these frequencies are below the above risk guidelines, the frequencies of actual tank or piping failures would be well below the guidelines. Health consequences resulting from releases depend on weather conditions as well as amount released. Weather frequencies used were based on actual data.

For more frequent weather conditions, unstable and turbulent atmospheric conditions bounded by stability class D, and SSW wind direction, the staff found that the consequences for all possible release scenarios, including the release of the total inventory of liquid UF_6, were below the airborne concentration of HF associated with irreversible or serious health effects and that the likelihood of occurrence was below $5.4x10^{-5}$/year. Thus, for this case, the consequence severity is well below the life threatening airborne concentration and the frequency of occurrence is below the risk guidelines for both the risk of serious health effect ($1x10^{-3}$) and the risk of fatality ($1x10^{-4}$). Therefore, the staff found reasonable assurance that the risk to individual’s offsite, associated with release scenarios for seismic events at or beyond the 1300-year event under normal weather conditions, is limited to an acceptable level.

For less frequent weather conditions characterized by the very stable, hence concentrated, plume, for stability class F, and SSW wind direction, the staff found a range of consequences, depending on the quantity released and the release rate from the building. For example, in the scenario of the loss of liquid UF_6 from the process piping, the consequences were below the irreversible or serious health effects and the likelihood of occurrence about $2x10^{-5}$/year, much less than the 10^{-3}/year guideline. However, for release scenarios involving the loss of larger quantities of liquid UF_6 from additional vessels, including up to the total inventory of liquid UF_6, the consequences were found to cover a wide range of consequences, with some cases above the airborne concentration of HF associated with life-threatening health effects. The likelihood of this range of scenarios was well below $2x10^{-5}$/year. Thus, for this range of scenarios, while the consequence severity ranged from serious health effects to life-threatening, the frequency of occurrence is estimated to be well below the above-mentioned guideline for fatality of $1x10^{-4}$. Therefore, the staff concluded that the risk to individuals offsite, from scenarios when weather conditions are stable, hence producing higher concentrations offsite, is also limited to an acceptable level.

In conclusion, the staff reviewed Honeywell’s demonstration of highly unlikely for the seismic scenarios. Also, the staff independently determined that, with the proposed modifications, the quantitative seismic risk to individuals is acceptably limited and adequate to protect public health, including for seismic events with a return period greater than 1700 years.

Evaluation Tornado-Generated Missile Hazards

Honeywell defined the likelihood of events associated with tornado-generated hazards as follows:

Not unlikely	More than 10^{-4} per event, per year
Unlikely	Between 10^{-4} and 10^{-5} per event, per year
Highly unlikely	Less than 10^{-5} per event, per year

The staff concluded that these definitions are reasonable and consistent with definitions previously used by Honeywell for ISA analysis and approved by the staff for use by other facilities for similar applications and are applicable for evaluation by Honeywell for tornado-related events based on current guidance.

In evaluating the tornado initiated accident sequence, Honeywell assumed that a tornado missile event would be a high consequence event and that the product of the frequency of the initiating tornado missile event and the probability of failure of the PFAPs must be less than 1 x 10^{-5}/year. Equation 2 illustrates this calculation.

Equation 2:

Frequency of initiating tornado missile event	*	Probability of failure of the PFAPs	$<10^{-5}$/year

Honeywell’s ISA analysis assumed an initiating tornado event of 10^{-3}/year and provided two PFAPs: an administrative control to implement safe shutdown procedures and a passive engineered control consisting of armor plate shielding to prevent possible consequences. For the administrative control, Honeywell assumed a likelihood of failure of 10^{-1}/year. For the passive controls, a likelihood of failure 10^{-2}/year was assumed. The overall likelihood for the accident sequence is therefore determined to be 10^{-6}/year, resulting in acceptable risk performance. Section 8.0 of this TER, Design of Structures, Systems, and Components, includes the staff’s review and evaluation of the missile prevention controls, the assumptions used in the analysis of the tornado-generated missile events, and the staff's determination of acceptability.

In conclusion, the staff reviewed Honeywell's demonstration of highly unlikely for tornado missile scenarios and verified that the corrective actions will adequately prevent consequences to the public from the release of hazardous chemicals for the design basis tornado.

3. Technical Evaluation Conclusions

The staff evaluated whether the risk to individuals offsite has been adequately limited. In considering this risk, the staff evaluated Honeywell's assessment of the seismic and tornado hazards; the adequacy of studies that Honeywell conducted to develop proposed seismic corrective actions for the FMB structure, major process equipment, and piping systems; the adequacy of Honeywell's tornado design bases; and Honeywell's assessment of the hazards, consequences, and characterization of risk to individuals offsite. The staff concluded that Honeywell's approach to determining the facility risk levels is consistent with accepted ISA methods and guidance. Further, the staff concluded that the risk levels presented by the facility (as modified), under credible seismic and tornado missile events, are acceptably low and consistent with the risk levels at other operating fuel cycle facilities. The TER also provides the staff's evaluation in each of these areas: Seismic Hazard Assessment; Design of Structures, Systems, and Components; Chemical Consequences; and Other Considerations.

4. References

(ICRP, 1991) International Commission on Radiological Protection, ICRP Publication 60, 1990 Recommendation of the International Commission on Radiological Protection, 1991.

(ICRP, 1993) International Commission on Radiological Protection, ICRP Publication 64, Protection from Potential Exposure: A Conceptual Framework, 1993.

(NRC, 2004) U.S. Nuclear Regulatory Commission, NUREG/BR-0058, -1513, Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission, Revision 4, September 2004.

(NRC, 2008) U.S. Nuclear Regulatory Commission, Risk-Informed Decision-Making for Nuclear Material and Waste Applications, Revision 1, ADAMS Accession No. ML080720238, February 2008.

(NRC, 2011) U.S. Nuclear Regulatory Commission, Temporary Instruction 2600/015, Evaluation of Licensee Strategies for the Prevention and/or Mitigation of Emergencies at Fuel Facilities, ADAMS Accession No. ML111030453, September 30, 2011.

(NRC, 2012a) U.S. Nuclear Regulatory Commission, Letter to Larry Smith, Honeywell, Nuclear Regulatory Commission's Temporary Instruction 2600-015 Inspection Report No. 40-3392/2012-006, ADAMS Accession No. ML12222A163, August 9, 2012.

(NRC, 2012b) U.S. Nuclear Regulatory Commission, Letter to Brett Able, Honeywell, Confirmatory Order, ADAMS Accession No. ML112289A863, October 15, 2012.

APPLICABILITY OF LESSONS LEARNED FROM THE FUKUSHIMA DAI-ICHI ACCIDENT TO FACILITIES OTHER THAN POWER REACTORS IN THE UNITED STATES

Margie Kotzalas
U.S. Nuclear Regulatory Commission
Mail Stop TWFN-4B16
Washington, DC 20555
Margie.Kotzalas@nrc.gov

ABSTRACT

On March 11, 2011, the Tohoku–Taiheiyou–Oki earthquake occurred near the east coast of Honshu, Japan. This magnitude 9.0 earthquake and the subsequent tsunami caused significant damage to at least four of the six units of the Fukushima Dai-ichi nuclear power station. As a result, there was a loss of offsite and onsite electrical power systems. Shortly after the accident, the U.S. Nuclear Regulatory Commission (NRC) performed limited assessments of non-reactor facilities to ensure that no immediate safety concerns existed. After several years and with insights gained from NRC activities related to power reactors and from the results from inspections at fuel cycle facilities, NRC staff more fully evaluated whether additional regulatory actions should be taken for other NRC-licensed materials, devices, and facilities. This paper describes the evaluations performed for radioactive material users and irradiators.

Background

Shortly after the accident at Fukushima Dai-ichi nuclear power station, the NRC performed limited assessments of non-reactor facilities to ensure that no immediate safety concerns existed at those facilities. After several years and with insights gained from NRC activities related to power reactors and from the results from inspections at fuel cycle facilities, NRC staff more fully evaluated whether additional regulatory actions should be taken for other NRC-licensed materials, devices, and facilities. Evaluations were performed for the following: spent fuel storage and transportation systems, radioactive material users, irradiators, low-level waste disposal facilities, uranium recovery facilities and uranium mill tailings, decommissioning reactors, and non-power reactors. This paper discusses the NRC staff's assessments of radioactive material users and irradiators. For both of these areas, NRC staff assessment provides the current regulatory framework and an evaluation of the facilities' capabilities to address or mitigate events such as: flood, seismic, high wind and missiles, snow and ice loads, temperature extremes, fire, and loss of power.

Staff Assessment of Radioactive Material Users

Current Regulatory Framework

The NRC regulates about 2,900 research, medical, industrial, government, and academic materials licensees.

In addition, the NRC has agreements with 37 States, under which the States assume regulatory responsibility for the use of certain radioactive materials. These Agreement States oversee about 18,000 licensees. The quantities that medical and academic licensees possess can range from millicurie quantities of radionuclides to thousands of curies contained in self-shielded irradiators and medical devices. Industrial uses of sealed source devices include a variety of applications and devices that include density gauges, thickness gauges, prompt gamma neutron activation analysis gauges, well logging gauges, moisture density gauges, industrial radiography sources, irradiators, as well as others.

Safety evaluations for sources and devices used in industrial, academic and medical settings are mostly evaluated as part of the sealed source and device registration process. Licensees are authorized to possess and use only those sealed sources and devices specifically approved or registered by NRC or an Agreement State. The NRC or Agreement State evaluates the safety of gauges, radiography source assemblies, exposure devices, source changers, and well- logging sources before authorizing a manufacturer or distributor to distribute the gauges to specific licensees. The safety evaluation is documented in a sealed source and device registration certificate.

Sealed sources are required to satisfy rigorous design and performance criteria. To prevent accidental dispersion from the device into the environment, the licensed material should be a chemical and physical form that is as insoluble and non-dispersible as practical. The material is doubly encapsulated; the capsule should be resistant to extreme changes in temperature, pressure, and vibration; and it is resistant to impact and puncture. The evaluation of the sources and devices include a review of the design, manufacturing, prototype testing, and proposed uses. The sources and devices are designed to survive normal conditions of use and likely accidental conditions.

Self-shielded irradiators (e.g., blood irradiators) incorporate many engineering features to protect individuals from unnecessary radiation exposure. Many other devices (e.g., gamma knife, radiography cameras, gauges, etc.) have some engineering features (shielding, connectors, switches, etc.) that prevent unnecessary exposure. These devices are usually designed for use in an industrial or laboratory environment, i.e., inside a building, protected from the weather, and in many cases without wide variations in temperature and humidity.

Facilities and equipment must be adequate to protect health, minimize danger to life or property, minimize the possibility of contamination, and keep exposure to occupationally exposed workers and the public as low as is reasonably achievable (ALARA). Licensed materials located in an unrestricted area and not in storage must be under the constant surveillance and immediate control of the licensee. Areas where material is used or stored, including below ground bunker storage areas, should (1) be accessible only by authorized persons, and (2) secured or locked when an authorized person is not physically present.

Although most equipment that uses sealed sources is very rugged, accidents occur that can temporarily or permanently damage the equipment. If this occurs while the source is outside its shielded container, there is potential for worker and public exposure. On numerous occasions ice, blowing snow or freezing rain have prevented the retraction of a source into a radiography exposure device. Structural and vehicle fires can damage the outer casing or over pack of a radiography exposure device, but the source generally remains intact and in the shielded position. Several instances have occurred where devices have been washed overboard while on route to a job site or off a deep sea oil drilling platform; divers recovered most, but not all of these devices. In other instances, trucks or tractor trailers have run over moisture density gauges with significant damage to the storage container. In those few instances where the source was dislodged from the source holder, the source was quickly found and re-inserted into the source holder and protective shielding.

In 2006, a nuclear pharmacy reported flood damage from a storm. All radioactive material (sealed sources and radiopharmaceutical material) was present and no removable contamination was present at the pharmacy. A September 2001 tornado damaged multiple buildings at the U.S. Department of Agriculture facility. The buildings contained various forms of carbon-14, tritium, iodine-125, phosphorous-32, and sulfur-35. All fume hoods and storage refrigerators, where the radioactive material was stored, were intact and the tornado did not affect the material. No other instances involving Category 1 or 2 medical or academic sources to external events are documented in the nuclear material event database. No instances in the nuclear material event database document the loss of licensee control of Category 1 industrial, medical or academic sources due to severe weather, earthquake, or flooding.

For more than 30 years, the NRC worked to improve in its emergency preparedness and incident response programs, especially upon reviewing lessons learned after several severe natural disasters. For example, in 2005, Hurricane Katrina struck the Gulf Coast. Hurricane Katrina is described as “the single most catastrophic natural disaster in U.S. history,” with estimated damage exceeding $100 billion. In preparation for and after landfall, the NRC contacted its Category 1 and Category 2 licensees to obtain information on the physical status and the security of facilities and materials in those states potentially affected by the hurricane. Coordination with the Agreement States proved successful in obtaining current information regarding the status of radioactive materials located in those states. The NRC emergency response was coordinated with other Federal agencies. If a licensee had lost control of a radiation source, Federal aerial monitoring system was available to search for and find any missing or misplaced radiation source.

Extreme external events such as earthquakes, tornadoes, hurricanes, flooding, or wildfires occur every year in the United States. Many of these events have the potential to cause the loss of licensee control of radioactive material. One tool that is available to assist the NRC and Agreement States monitor the status of radioactive material is the National Source Tracking System (NSTS). The NSTS is a secure web-based database designed to document the location and status of Category 1 and Category 2 radioactive sources regulated by the NRC and the Agreement States. About 1,300 licensees began reporting their Category 1 and Category 2 source information for inclusion into the NSTS in January 2009. The tracking spans the life cycle of the source from manufacture through shipment receipt, decay, and disposal. NSTS enhances the ability of the NRC and Agreement States to inspect and investigate, communicate information to other government agencies, and verify legitimate ownership and use of nationally tracked sources.

Post-Fukushima Event Evaluation

Although the source integrity of a sealed source after exposure to a natural event is very likely to be retained, loss of control of a source by the licensee is a possibility. The tsunami that struck the northeast coast of Japan destroyed thousands of buildings and created approximately 20 million tons of debris. Five million tons of this debris were swept out to sea (3.5 million tons of debris was deposited along the coast of Japan and another 1.5 million tons became floating debris in the Pacific Ocean).

Since the Fukushima accident, the NRC also has developed a mapping tool to give situational awareness of the NSTS's Category 1 and 2 licensees and sources as distributed across the United States. Tools like this, along with NSTS, are used regularly to monitor and verify source security after natural events like earthquakes, wildfires, tornadoes, hurricanes, and flooding. The greatest concern is the loss of licensee control of radioactive material.

Effect of External Events on Radioactive Material Users

External Event	**Outcome**	**Assessment**
Flood	Challenge to structures and vehicles in which unsealed and sealed sources are used, stored, or transported; potential loss of control	Licensees are required to meet any city, county or state requirements/ regulations regarding building construction. The greatest concern is the loss of licensee control of radioactive material. During the license application process, the location of all unsealed and sealed source materials must be described. For those radioactive materials with activities that exceed those for Category 1 (e.g., irradiators) and Category 2 (e.g., well logging) sources, they must be reported to and tracked in the NSTS. Emergency plans that address natural phenomenon, including an earthquake, a tornado, flooding, or other phenomena, generally are not required by industrial licensees, unless the activity limits exceed those in 10 CFR 30.72, Schedule C. However, the licensee must develop operating and emergency procedures and the operator must demonstrate an understanding of these procedures.
Seismic	Challenge to structures in which unsealed and sealed sources are used or stored. Manufacturer must check medical equipment before patient use.	
High Wind and Missiles	Challenge to structures and vehicles in which unsealed and sealed sources are used, stored, or transported; potential loss of control	
Lightning	Challenge to structures in which unsealed and sealed sources are used, stored, or transported	
Snow and Ice Loads	Challenge to structures and vehicles in which unsealed and sealed sources are used, stored, or transported	
Drought	None	
Temperature Extremes	Failure to retract radiography source because of ice formation.	
External Fire	Challenge to structures and vehicles in which unsealed and sealed sources are used, stored, or transported. Potential damage to shielding material (e.g., lead) upon exposure to extreme heat.	
Loss of Power	Some gauges and devices	

Conclusions

NRC staff concludes that unsealed radioactive materials and sealed sources and devices used in industry, academia, and medicine are appropriately licensed and have sufficient engineering controls to protect the health and safety of workers and members of the public. Worker exposures are kept as low as is reasonably achievable and minimize the danger to life and property. The safety evaluation is documented in a sealed source and device registration certificate conducted before authorizing a manufacturer or distributor to distribute the radioactive sources to specific licensee assures the integrity of the device. Thousands of industrial sources have been exposed to harsh environmental stressors and licensees have developed operational and emergency procedures to deal with unplanned accidents and emergencies. Databases and mapping tools designed to track Category 1 and Category 2 radioactive sources are continually being improved and refined. Finally, specific guidance for licensing radioactive material has been updated and published for public comment. Therefore, no further study or regulatory action is warranted.

Staff Assessment of Irradiators

Current Regulatory Framework

Gamma radiation is routinely used in industry to eliminate disease-causing insects and micro-organisms such as E. coli and salmonella. Food products, food containers, spices, fruits, plants and medical supplies are the products most commonly irradiated. To be effective, radiation exposure upward of hundreds to thousands of gray is required to sterilize these products. The process does not leave a radioactive residue or cause the treated products to become radioactive.

An irradiator is defined as a facility that uses radioactive sealed sources for the irradiation of objects or materials and in which radiation dose rates exceeding 5 gray (500 rads) per hour exist at 1 meter from the sealed radioactive sources, but does not include irradiators in which both the sealed source and the area subject to irradiation are contained within a device (self-shielded) and are not accessible to personnel such as a blood/tissue irradiator. Irradiators generally contain Category 1 sources of cobalt-60 with activities that range from 9 PBq (250 KCi) to 1 EBq (30 MCi).

The NRC and Agreement States license several types of irradiators. Underwater irradiators are devices in which sealed sources always remain underwater and workers do not have direct access to the sources or the area subject to irradiation without entering the pool. Panoramic irradiators include those facilities where the irradiations occur in air and workers potentially could have access to the source while it is in operation. If the source is stored in a pool of water when not in use, the facility is called a panoramic wet-source-storage irradiator. The United States has over 50 wet-source-storage irradiators in use, which are licensed to use between 37 to 520 PBq (1 to 14 MCi) of cobalt-60. If an irradiator's sources are not stored in a pool of water when not in use, it is referred to as a dry-source-storage irradiator. In this case, the sources are stored in a large shielded container. Beam type dry-source storage irradiators emit a narrow beam of radiation which irradiates the target material in air in areas potentially accessible to workers.

In a typical irradiation system, products are loaded into irradiation containers that are conveyed through a maze into a concrete or metal shielded room. Inside the shielded room, the products are exposed to gamma rays coming from a rack of cobalt-60 sources. The concrete or metal shield is designed to limit worker and public exposure to no more than 2 mrem (20 µSv) per hour at 1 meter from the shield.

Post-Fukushima Event Evaluation

The staff's evaluation reviewed the generic facility design and available equipment to determine if engineering controls and barriers are sufficient to protect the health and safety of the public and irradiator workers in the event of severe natural phenomena such as an earthquake or tsunami.

In the Unities States, 55 panoramic, wet-source-storage irradiators and one underwater pool irradiator, each containing a minimum of 37 PBq (1 MCi) of cobalt-60, are licensed for use. Nine of these irradiators are in seismic areas. The NRC defines a seismic area as any location where the probability of a horizontal acceleration in rock of more than 0.3 times the acceleration of gravity in 250 years is greater than 10 percent. Although not required, each of these nine irradiators is equipped with a seismic detector that causes the radiation source holder to become fully shielded automatically in the source storage pool should the seismic detector be actuated. When returned to the storage pool, the radiation reading at the pool surface should be less than 2 mrem per hour.

The average source storage pool holds between 16,700 and 18,100 gallons of water. The water level is typically maintained 3 meters (10 feet) above the top of the source material when the source rack is in the fully shielded position. The pool liner is fabricated using 3/16-inch stainless steel, encased in high density, steel reinforced concrete, generally 40 to 50 cm (15 to 20 inches) thick. The pool is specifically designed without a tie-in to the main shield so that it is free to move independently of the shield in the event of an earthquake. Water chilling systems maintain pool water temperatures between 18 degrees to 27 degrees Celsius (65 degrees to 80 degrees Fahrenheit). The water chilling system is necessary because a single 370 TBq (10,000 Ci) cobalt-60 sealed source has an estimated surface temperature of 130 degrees Celsius (265 degrees Fahrenheit). During normal operations several hundred individual sources are loaded into the source rack. Without chilling capacity, the pool water temperature will begin to elevate and level off after 5 to 7 days at approximately 65 degrees Celsius (150 degree Fahrenheit). The pool water evaporation rate will be slow, such that it would take weeks before the source material is exposed to the air. Water can be manually added to the pool. For example, two wet storage irradiators can be quickly refilled by the fire department through a designed connection point located outside each shield. It is important to note that there is no adverse effect to the source material resulting from the pool being empty. Each source is safety tested to withstand temperatures in excess of 800 degrees Celsius (1,400 degrees Fahrenheit) for 1 hour, which simulates the temperature of many hydrocarbon fires.

In the event the source rack does not return to the shielded position, the radiation shield surrounding the source is extremely robust. The walls of the shield are designed such that the radiation dose rate in areas that are normally occupied during operation of a panoramic irradiation may not exceed 2 millirem per hour at any location 30 cm or more from the wall or ceiling of the room. To achieve this level of protection, the concrete wall and ceiling thickness generally exceeds 165 to180 cm (65 to 70 inches). The design of the walls and roof is required to comply with the structural requirements of the local building code or the American Concrete Institute Standard 318-89; hence, the concrete density, rebar size, the rebar cross spacing and the number of rows is appropriate for use in a seismic area. Assuming a minimum concrete density of 147 pounds/cubic foot, the approximate weight of each radiation shield structure is at least 2,000 tons. An NRC staff review demonstrates that the structural components of irradiators have considerable seismic capacity and will maintain their structural integrity under quite severe ground motion conditions (greater than several g's).

In the event of a significant earthquake, the warehouse roofing and walls may collapse restricting access to the radiation control room and primary building power may be lost for an extended period of time. However, the shield is designed to retain its integrity and to ensure that no radioactive material will be released and that there will not be any off site radiation exposure.

The NRC previously evaluated the potential impact of an earthquake generated tsunami to a pool irradiator built in Honolulu, Hawaii. NRC staff considers this evaluation to be generally bounding for other irradiators. This is because, other than the irradiator located in Hawaii, the licensed irradiator facilities are either located inland or in locations that would not generate tsunamis to the extent possible in Hawaii. For the evaluation of the Hawaii irradiator, at shore, tsunami waves up to 10 m (33 ft) can reach velocities of up to 13 m/s (29 mph). Given the weight of a single sealed source and the shear velocity needed to lift a source out of the storage pool, the wave velocities associated with the largest historical tsunami would be insufficient to remove a source from the bottom of the irradiator pool even if the facility had sustained enough damage that the source rack had been destroyed. In addition, the wave velocity of a wind- generated (i.e., hurricane) storm surge is less than that associated with a tsunami. Therefore, NRC staff concludes that it is unlikely that a tsunami or hurricane will result in a loss of control of radioactive material from an industrial irradiator that would have an adverse effect on public health and safety or the environment.

Effect of External Events on Irradiators

<table>
<tr><th>External Event</th><th>Outcome</th><th>Assessment</th></tr>
<tr><td>Flood</td><td>Challenge to external structures in which irradiator sources are used. No loss of source control.</td><td rowspan="9">Warehouse roofing and walls may be damaged thus restricting access to the irradiator control room and access door until debris is removed.

The concrete/steel reinforced radiation shield, 70 to 74 inches thick, is designed to retain its integrity in the event of an earthquake by designing to the seismic requirements of an appropriate source such as ACI Standard ACI 318-89.

A radiation shield constructed with steel, 12 to 14 inches thick, is designed to retain its integrity in the event of a severe earthquake.

Emergency or abnormal event procedures must address natural phenomenon, including an earthquake, a tornado, flooding, or other phenomena, as appropriate for the facility's geographical location.</td></tr>
<tr><td>Seismic</td><td>Challenge to external structures in which irradiator sources are used. Radiation shield has considerable seismic capacity and will maintain structural integrity under quite severe (several g's) ground</td></tr>
<tr><td>High Wind and Missiles</td><td>Challenge to external structures in which irradiator sources are used. No damage to shield integrity.</td></tr>
<tr><td>Lightning</td><td>Challenge to external structures in which irradiator sources are used.</td></tr>
<tr><td>Snow and Ice Loads</td><td>Challenge to external structures in which irradiator sources are used. No damage to shield integrity.</td></tr>
<tr><td>Drought</td><td>None</td></tr>
<tr><td>Temperature Extremes</td><td>None</td></tr>
<tr><td>External Fire</td><td>Challenge to external structures in which irradiator sources are used. No damage to sealed sources or shield integrity.</td></tr>
<tr><td>Loss of Power</td><td>Source automatically returns to shielded position.
Malfunction of source rack possible, thus requiring mechanical intervention to return sources to shielded position. Control system door interlocks failsafe closed and locked in the event of a power loss. All facility control, monitoring, and security systems would be inoperable once battery backup power is exhausted, estimated to be within 24 hours.</td></tr>
</table>

Conclusions

NRC staff concludes that irradiators are licensed appropriately and have sufficient engineering controls to protect the health and safety of workers and members of the public. Worker exposures are kept as low as is reasonably achievable and minimize the danger to life and property. Based on the available information, it is unlikely that natural phenomena (e.g., tsunamis, earthquakes, hurricanes, or fire) will result in a loss of control of radioactive material from an industrial irradiator that would have an adverse effect on public health and safety or the environment. NRC staff concludes that irradiators are licensed as appropriate to the scope and potential hazard created.

POST-FUKUSHIMA SAFETY REVIEWS PERFORMED AT FUEL CYCLE FACILITIES IN THE UNITED STATES

J. Marcano Lozada, P.E.
U.S. Nuclear Regulatory Commission
Mail Stop TWFN-4B16
Washington, DC 20555

jonathan.marcano@nrc.gov

ABSTRACT

On March 11, 2011, the Tohoku–Taiheiyou–Oki earthquake occurred near the east coast of Honshu, Japan. This magnitude 9.0 earthquake and the subsequent tsunami caused significant damage to at least four of the six units of the Fukushima Dai-ichi nuclear power station. As a result, there was a loss of offsite and onsite electrical power systems. On March 31, 2011, the Nuclear Regulatory Commission (NRC) staff issued Information Notice (IN) 2011-08, "Tohoku-Taiheiyou-Oki Earthquake Effects on Japanese Nuclear Power Plants—for Fuel Cycle Facilities" to inform fuel cycle facilities of the potential challenges associated with preventing or mitigating the effects of natural phenomena events. IN 2011-08 recommended that fuel cycle facilities review the information for applicability to their facilities and consider actions, as appropriate, to ensure that features and preparations necessary to withstand or respond to severe external events from natural phenomena (e.g., earthquakes, tsunamis, floods, tornadoes, and hurricanes) are reasonable and consistent with regulatory requirements. Subsequently, the NRC staff performed a systematic evaluation and inspection of selected fuel cycle facilities, in light of the lessons learned from the accident at the Fukushima Dai-ichi Nuclear Power Plant, to confirm that licensees were in compliance with applicable regulatory requirements and license conditions; and to evaluate their readiness to address natural phenomena hazards (NPH) events and other licensing bases events related to NPH. The staff's assessment considered recommendations published by the NRC to improve the regulatory framework for reactors in the United States and their applicability to fuel cycle facilities to determine whether additional regulatory actions by the NRC were warranted. This assessment included consideration of new seismic hazard information from the U.S. Geological Survey (USGS) for the central and eastern United States, which was the subject of an NRC generic communication to fuel facilities in IN 2010-19, "Updated Probabilistic Seismic Hazard Estimates in Central Eastern United States".

1. Overview of Fuel Cycle Facilities, Academic and Other Institutions

Fuel cycle facilities involved in conversion, enrichment, and fuel fabrication are regulated through a combination of regulatory requirements; licensing; safety oversight (including inspection, assessment of performance, and enforcement); evaluation of operational experience; and regulatory support activities. These facilities turn the uranium that has been removed from ore (as yellowcake) into fuel for nuclear reactors. In this process, the conversion facility converts yellowcake into uranium hexafluoride (UF_6). Next, an enrichment facility heats the solid UF_6 enough to turn it into a gas, which is "enriched," or processed to increase the concentration of the uranium-235 (U_{235}). Then the enriched uranium is manufactured into pellets. These pellets are placed into fuel assemblies and ultimately into nuclear reactors.

Fuel Cycle Process, Facilities, and Associated Hazards

The regulations governing the licensing and operation of fuel cycle facilities are found below. Each type of facility presents different levels of risk to both the worker and offsite from events involving chemical and radiological material.

- 10 CFR Part 20, "Standards for protection against radiation"
- 10 CFR Part 21, "Reporting of defects and noncompliance"
- 10 CFR Part 40, "Domestic licensing of source material"
- 10 CFR Part 70, "Domestic licensing of special nuclear material"
- 10 CFR Part 73, "Physical protection of plants and materials"
- 10 CFR Part 74, "Material control and accounting of special nuclear material"
- 10 CFR Part 76, "Certification of gaseous diffusion plants"

A. Uranium Conversion

Process: After the yellowcake is produced at the mill, the next step is conversion into pure UF_6 gas suitable for use in enrichment operations. During this conversion, impurities are removed and the uranium is combined with fluorine to create the UF_6 gas. The UF_6 is then pressurized and cooled to a liquid. In its liquid state it is drained into 14-ton cylinders where it solidifies after cooling for approximately five days. The UF_6 cylinder, in the solid form, is then shipped to an enrichment plant.

Hazards: As with mining and milling, the primary risks associated with conversion are potential chemical and radiological events. Strong acids and alkalis are used in the conversion process, which involves converting the yellowcake (uranium oxide) powder to very soluble forms, leading to possible inhalation of uranium. In addition, conversion produces extremely corrosive chemicals that could cause fire and explosion hazards. Fire and explosion hazards are also a concern in areas where liquid UF_6 is stored and processed. When liquid UF_6 is released to the atmosphere, it reacts with the moisture in the air to form a dense vapor cloud that contains hydrogen fluoride gas (chemical hazard), a nonradioactive, extremely toxic substance.

Plants: Honeywell International Inc. Metropolis, Illinois.

B. Uranium Enrichment

Process: Enriched uranium is required in commercial light-water reactors to produce a controlled nuclear reaction. Enriching uranium increases the proportion of uranium atoms that can be “split” by fission to release energy (usually in the form of heat) that can be used to produce electricity. Not all uranium atoms are the same. When uranium is mined, it consists of about 99.3 percent uranium-238 (U_{238}), 0.7 percent uranium-235 (U_{235}), and less than 0.01 percent uranium-234 (U_{234}). The fuel for nuclear reactors has to have a higher concentration of U_{235} than exists in natural uranium ore. Normally, the amount of the U_{235} isotope is enriched from 0.7 percent of the uranium mass to about 5 percent.

Hazards: The principal hazards at an enrichment plant are the chemical hazards in handling UF_6. When UF_6 contacts moisture in air, it reacts to form hydrogen fluoride and uranyl fluoride. The chemical hazards of compounds of uranium in soluble form such as UF_6 and uranyl fluoride are much greater than the radiological hazards of those same compounds. In addition, hydrogen fluoride can be very dangerous if inhaled; inhalation is the principal hazard at an enrichment plant. These hazards are controlled by plant design and administrative controls to confine soluble uranium compounds. The radiological hazards are relatively low and containers of natural, enriched, and depleted uranium can be handled without extra shielding. Another hazard for this type of facility is the potential for mishandling the enriched uranium, which could create a criticality accident (inadvertent nuclear chain reaction).

Plants:

- Gaseous Diffusion Uranium Enrichment Facility
  - United States Enrichment Corporation (USEC) Inc. in Paducah, Kentucky (No longer a NRC-licensed facility)

- Gas Centrifuge Uranium Enrichment Facilities
  - American Centrifuge Plant, LLC (USEC) in Piketon, OH (License issued, construction halted)
  - Louisiana Energy Services in Eunice, NM

 – AREVA Enrichment Services Eagle Rock, LLC , Idaho Falls, ID (License issued, construction not started)

- Laser Separation Enrichment Facility
  – GE-Hitachi in Wilmington, NC (License issued, construction not started)

C. Uranium Fuel Fabrication

Process: Fuel fabrication facilities convert enriched UF_6 into fuel for nuclear reactors. Fabrication also can involve mixed oxide (MOX) fuel, which is a combination of uranium and plutonium components. The NRC regulates several different types of nuclear fuel fabrication operations, such as light water reactor low-enriched uranium fuel and light water reactor mixed oxide fuel.

Fuel fabrication for light water power reactors typically begins with receipt of low-enriched uranium (LEU) hexafluoride from an enrichment plant. The UF_6, in solid form in containers, is heated to gaseous form, and the UF_6 gas is chemically processed to form LEU dioxide (UO_2) powder. This powder is then pressed into pellets, sintered into ceramic form, loaded into zircaloy tubes, and constructed into fuel assemblies. Depending on the type of light water reactor, a fuel assembly may contain up to 264 fuel rods and have dimensions of 5 to 9 inches square by about 12 feet long.

MOX fuel differs from LEU fuel in that the dioxide powder from which the fuel pellets are pressed is a combination of UO_2 and plutonium oxide (PuO_2). Congress directed NRC to regulate the Department of Energy's (DOE's) fabrication of MOX fuel, which uses repurposed plutonium from international nuclear disarmament agreements.

Hazards: Chemical, radiological, and criticality hazards at fuel fabrication facilities are similar to hazards at enrichment plants.

Plants:

- Uranium Fuel Fabrication
  – Global Nuclear Fuel-Americas, LLC in Wilmington, NC
  – Westinghouse Electric Company, LLC in Columbia, SC
  – Nuclear Fuel Services, Inc. in Erwin, TN
  – AREVA NP, Inc. in Richland, WA

  - Babcock & Wilcox Nuclear Operations in Lynchburg, VA

- Mixed Oxide Fuel Fabrication
  - Shaw AREVA MOX Services, LLC in Aiken, SC (Construction Authorization issued, construction ongoing)

D. Uranium Hexafluoride Deconversion

Process: As U_{235} is extracted, converted, and enriched in the uranium recovery, conversion, and enrichment processes for use in fabricating fuel for nuclear reactors, large quantities of depleted uranium hexafluoride (DUF_6), or “tailings,” are produced. These tailings are transferred into 14-ton cylinders which are stored in large yards near the enrichment facilities. A process called "deconversion" is then used to chemically extract the fluoride from the DUF6 stored in the cylinders. This deconversion process produces stable compounds, known as uranium oxides, which are generally suitable for disposal as low-level radioactive waste.

Hazards: Chemical exposure is the dominant hazard at deconversion facilities because uranium chemical compounds and other chemical compounds (such as hydrogen fluoride) are hazardous at low levels of exposure.

Plant: International Isotopes in Hobbs, NM (license issued, construction not started)

E. Academic and Other Institutions

Academic and other institutions use radioactive material in classroom demonstrations, laboratory experiments, and research, and to offer health physics support to other institutional radioactive materials users. These facilities are licensed in accordance with 10 CFR Parts 30, 40, or 70 depending on the type of materials possessed. These programs may vary in size from large, broad-scope programs involving chemical, physical, biological engineering, and biomedical research, to small programs using only gas chromatographs or self-shielded irradiators.

Hazards: These licensees have limited amounts of materials and have demonstrated by NRC-approved evaluation that no member of the public will exceed the thresholds of the regulations in 10 CFR Part 70. That is not to say that accidents cannot occur with these licensees, but because of the limited amount of materials possessed and the primarily sealed nature of the material, the effect to the public and the environment is limited.

2. Post-Fukushima Event Evaluations/Assessment

NRC staff evaluated and inspected selected fuel cycle facilities to confirm that licensees complied with regulatory requirements and license conditions; and to evaluate their readiness under natural phenomena hazards (NPH) events and other licensing bases events related to NPHs. NRC staff's assessment considered the Near-Term Task Force (NTTF) recommendations to determine whether other regulatory actions by the NRC are warranted. The NTTF was created to review the Fukushima events and the possible implications for the safety of U.S. nuclear power plants. On July 12, 2011, the NTTF issued its report, titled "Near-Term Report and Recommendations for Agency Actions Following the Events in Japan" (ADAMS Accession No. ML11186A950), which included 12 recommendations. This assessment included consideration of new seismic hazards information from the U.S. Geological Survey (USGS) for the central and eastern United States which was the subject of an NRC generic communication to fuel facilities in NRC Information Notice 2010-19, "Updated Probabilistic Seismic Hazard Estimates in Central Eastern United States" (ADAMS Accession No. ML102160735).

The NRC staff completed inspections at selected fuel facilities and the results were used to perform a systematic evaluation of the processes and regulations applicable to fuel facilities. Because of the evaluation, NRC staff concludes that the current regulatory approach and requirements of these licensees continues to serve as a basis for reasonable assurance of adequate protection of public health and safety. However, as described in greater detail below, for the Honeywell Metropolis Works Facility, the inspections found potentially significant safety issues and the NRC staff took immediate steps to ensure corrective actions were taken. In addition, NRC staff found generic issues regarding compliance with the current regulatory framework with regards to the treatment of certain natural phenomena events in the facilities' (uranium conversion, enrichment and fuel fabrication) safety assessments. In June 2015, the NRC issued Generic Letter (GL) 2015-01, "Treatment of Natural Phenomena Hazards in Fuel Cycle Facilities" (ADAMS Accession No. ML14328A029). The NRC staff has received responses to the GL from all recipients and is preparing evaluation reports. As part of the closure process, the NRC staff is also independently verifying compliance, as needed, using Temporary Instruction 2600/016, "Inspection of Activities Associated with NRC Generic Letter 2015-01" (ADAMS Accession No. ML15317A506).

A. Uranium Conversion, Enrichment, Fuel Fabrication, and Deconversion Licensees

On March 31, 2011, NRC staff issued Information Notice (IN) 2011-08, "Tohoku–Taiheiyou–Oki Earthquake Effects on Japanese Nuclear Power Plants—for Fuel Cycle Facilities," (ADAMS Accession No. ML110830824) to inform addressees of the potential challenges associated with preventing or mitigating the effects of natural phenomena events. IN 2011-08 recommended that addressees review the information for applicability to their facilities and consider actions, as appropriate, to ensure that features and preparations necessary to withstand or respond to severe external events from natural phenomena (e.g., earthquakes, tsunamis, floods, tornadoes, and hurricanes) are reasonable and consistent with regulatory requirements.

From December 2011 through May 2012, NRC staff conducted inspection activities in accordance with Temporary Instruction (TI) 2600/015, "Evaluation of Licensee Strategies for the Prevention and/or Mitigation of Emergencies at Fuel Facilities" (ADAMS Accession No. ML12286A284). The NRC completed the TI in three phases. In the initial phase, NRC staff reviewed licensing documents, including the safety assessments and EPs. The second phase consisted of NRC inspectors evaluating accident prevention measures and emergency actions through onsite evaluations that focused on credible natural phenomena and loss of utilities that support onsite systems (e.g. electricity and water). The third phase involved assessing whether the strategies and equipment were effective to prevent or mitigate emergencies during selected beyond licensing basis natural events and the extended loss of utilities that support onsite systems. In the review of licensing basis events, the NRC considered the following NPHs: seismic, flooding, and high winds (caused by hurricanes or tornadoes). The NRC also evaluated onsite fires that may result from seismic related equipment failures. Particular attention was given to earthquakes and flooding because of recent events and significant advancements in the state of knowledge of these hazards.

> In addition, during the TI 2600/015 inspections, the staff also considered operating experience on the implementation of mitigation strategies and emergency procedures used by licensees to cope with natural phenomena events. The staff performed interviews of licensee personnel and walk-downs of the facility to assess how the facility coped with the occurrence of previous natural phenomena events (such as floods or hurricanes).

After the implementation of TI 2600/015, the NRC determined that the evaluated facilities had established programs, procedures, and equipment to respond to licensing basis events involving fire, flooding, and loss of utilities. However, based on information obtained from the inspection activities, NRC staff found that the assumptions used by licensees in developing the independent safety analysis and other safety assessments are not clearly described and documented. The NRC primarily attributed this to the lack of available facility design information and significant variations in the level of detail and rigor of implementation in the facility safety assessments with regards to the treatment of natural phenomena events. Therefore, the NRC inspectors were unable to verify that these facilities complied with their licensing basis and regulatory requirements. The NRC inspectors opened unresolved items (URIs) to further assess whether the evaluated licensees complied with license conditions, the requirements of 10 CFR 70.61 and 10 CFR 70.62(c), regarding NPH event accident sequences. A summary of the results of the inspections can be found in the table below. NRC staff has determined that for all the facilities inspected (except the Honeywell Metropolis Works Facility) in consideration of inherent seismic capacity in facility structures, existing safety programs in place, and radiological/chemical source terms, continued operation does not pose an imminent risk to public health and safety.

For the Honeywell Metropolis Works Facility, the NRC determined that the site Emergency Response Plan underestimated the amount of UF_6 and hydrogen fluoride that could potentially be released during credible seismic events or tornadoes. Specifically, the inspectors found that the process equipment in the licensee's Feed Materials Building lacked seismic restraints, supports, and bracing that would ensure process equipment integrity during certain credible seismic events or tornadoes. The NRC issued a confirmatory order that required the licensee to demonstrate its SSCs relied on for safety were adequate for seismic and tornado events. The facility structure and internal components were significantly retrofitted to improve the performance under seismic and tornado events. Additional information is available in NUREG-0090, Volume 36, "Report to Congress on Abnormal Occurrences: Fiscal Year 2013" (ADAMS Accession No. ML14150A073).

Summary of TI 2600/015 results

	Facility	Summary of Issues Identified
Part 76	Paducah (ADAMS Accession No. ML12131A437)	• Tornadoes were not considered a credible event because of the return period chosen for the evaluation basis event. The team determined that a tornado could be considered a credible event for the site if newer data is used to evaluate the probability of occurrence. However, the consequences of a tornado event are bounded by other safety basis events.
Part 70	AREVA (ADAMS Accession No. ML12122A094)	• Unresolved Item (URI) 70-1257/2012-006-01 was opened to further evaluate whether the licensee complied with the requirements of 10 CFR 70.62(c) and 70.61 performance requirements regarding natural phenomena events accident sequences.
	Babcock & Wilcox Nuclear Operations Group (ADAMS Accession No. ML12121A574)	• URI 70-27/2012-006-01 was opened to further evaluate whether the licensee complied with the requirements of 10 CFR 70.62(c) and 70.61 performance requirements regarding natural phenomena events accident sequences.
	Global Nuclear Fuel – Americas (ADAMS Accession No. ML12209A276)	• URI 70-113/2012-006-01 was opened to further evaluate whether the licensee complied with the requirements of 10 CFR 70.62(c) and 70.61 performance requirements regarding natural phenomena events accident sequences.
	Nuclear Fuel Services (ADAMS Accession No. ML12122A186)	• URI 70-143/2012-06-01 was opened to further evaluate whether the licensee complied with Table 2.2 of the license application regarding management measures for items relied on for safety PREP-A and PREP-B. • URI 70-143/2012-006-03 was opened to further evaluate whether the licensee complied with the requirements of 10 CFR 70.62(c) and 70.61 performance requirements regarding natural

	Facility	Summary of Issues Identified
Part 40		phenomena events accident sequences.
	Westinghouse – Columbia Fuels (ADAMS Accession No. ML12122A083)	• URI 70-1151/2011-07-01 was opened to review Westinghouse's response to the failure to ensure that the risk of an earthquake was limited by applying sufficient engineered controls, administrative controls, or both, to the extent needed so that, upon implementation of such controls, the event was highly unlikely. • URI 70-1151/2011-07-02 was opened to review Westinghouse's evaluation regarding whether all nuclear process under an earthquake were subcritical.
	Honeywell (ADAMS Accession No. ML12222A163)	• URI 40-3392/2012-006-01 was opened to evaluate whether the Metropolis Works Facility integrated safety analysis appropriately considered credible high consequence seismic and tornado events and subsequently designated plant features and procedures and management measures to ensure that the accident sequences (public and workers health and safety) remained highly unlikely or the consequences were mitigated to acceptable levels. • Apparent Violation (AV) 40-3392/2012-006-02 was identified for the failure to identify all relevant accident sequences related to credible seismic events and tornadoes that could result in large uranium hexafluoride (UF6) releases for which protective actions may be needed. • AV 40-3392/2012-006-003 was identified for the failure to supply complete and accurate information related to MTW's emergency response plan.

Note: Louisiana Energy Services was not inspected because it is a new facility designed and constructed with more demanding criteria as required by 10 CFR 70.64, "Baseline Design Criteria".

Review of Near-Term Task Force recommendations

NRC staff considered the 12 NTTF recommendations for applicability to fuel cycle facilities and evaluated to determine if any future action is warranted. The following table summarizes the result of the review.

Near-Term Task Force Recommendations and Future Actions

Recommendations		Review Result
1	Establish a logical, systematic, and coherent regulatory framework for adequate protection that appropriately balances defense-in-depth and risk considerations.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.
2	The task force recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection of SSCs.	No Immediate Action. Based on lessons learned from TI 2600/15 inspections staff identified a need to verify compliance regarding treatment of natural phenomena events (see Generic Letter discussion below).
3	The task force recommends, as part of the longer term review, that the NRC evaluate potential enhancements to the capability to prevent or mitigate seismically induced fires and floods.	No Action Needed. Staff assessment noted inherent safety margin in fuel cycle facilities and NRC staff assessment considered new seismic hazards information from USGS which was provided to fuel facilities by NRC Information Notice 2010-19.
4	The task force recommends that the NRC strengthen SBO mitigation capability at all operating and new reactors for design-basis and beyond-design-basis external events.	No Action Needed. Staff assessment considered the inherent safety margin in fuel cycle facilities, including but not limited to, existing safety programs in place and radiological and chemical source terms to determine that SBO mitigation capabilities do not need to be strengthened beyond those that already exist.
5	The task force recommends requiring reliable hardened vent designs in BWR facilities with Mark I and Mark II containments.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees.

Recommendations		Review Result
6	The task force recommends, as part of the longer term review, that the NRC points out insights about hydrogen control and mitigation inside containment or in other buildings as additional information is revealed through further study of the Fukushima Dai-ichi accident.	Not Applicable. There are no credible scenarios that create hydrogen in quantities of concern for these licensees.
7	The task force recommends enhancing spent fuel pool makeup capability and instrumentation for the spent fuel pool.	Not Applicable to fuel cycle facilities.
8	The task force recommends strengthening and integrating onsite emergency response capabilities such as EOPs, SAMGs, and EDMGs.	No Action Needed. Fukushima lessons learned for power reactors showed prolonged loss of AC power and multiunit events could have severe consequences leading to a need to update EP response. No need to upgrade existing EP requirements based on inherent safety margin in fuel cycle facilities, including but not limited to, existing safety programs in place, and considering radiological and chemical source terms associated with fuel cycle facilities.
9	The task force recommends that the NRC require that facility EPs address prolonged SBO and multiunit events.	No Action Needed. See item 8 above.
10	The task force recommends, as part of the longer term review, that the NRC should pursue additional EP topics related to multiunit events and prolonged SBO.	No Action Needed. See item 8 above.
11	The task force recommends, as part of the longer term review, that the NRC should pursue EP topics related to decision-making, radiation monitoring, and public education.	No Action Needed. See item 8 above.

Recommendations		Review Result
12	The task force recommends that the NRC strengthen regulatory oversight of licensee safety performance (i.e., the reactor oversight process) by focusing more attention on defense-in-depth requirements consistent with the recommended defense-in-depth framework.	No Action Needed. No regulatory gaps were identified as a result of the staff's assessment.

Generic Letter: Treatment of Natural Phenomena Hazards in Fuel Cycle Facilities

Because of inspections, NRC staff proposed issuing a generic letter due to the generic applicability of the URIs across the nuclear fuel facility industry. Current NRC regulations require the evaluation of site hazards including natural phenomena events. The purpose of the generic letter was to request information from licensees to verify that compliance is being maintained with regulatory requirements and license conditions regarding the treatment of natural phenomena events. The NRC issued Generic Letter (GL) 2015-01, "Treatment of Natural Phenomena Hazards in Fuel Cycle Facilities" (ADAMS Accession No. ML14328A029). The NRC staff has received responses to the GL from all recipients and is preparing evaluation reports. As part of the closure process, the NRC staff is also independently verifying compliance, as needed, using Temporary Instruction 2600/016, "Inspection of Activities Associated with NRC Generic Letter 2015-01" (ADAMS Accession No. ML15317A506).

B. Academic and Other Institutions (Greater Than Critical Mass)

One of the considerations coming out of the fuel cycle review described above was to evaluate the readiness of greater than critical mass (GTCM) licensees as it applies to the lessons from the Fukushima accident. The readiness of GTCM facilities was evaluated by performing desktop reviews. The recommendations of the NTTF were reviewed for applicability, as well as considerations found during the review of fuel cycle facilities. On the basis of this review, the continued operation of GTCM licensees do not pose an imminent risk to the public health and safety. The current regulatory approach and requirements of these licensees continues to serve as a basis for reasonable assurance of adequate protection of public health and safety. For these facilities, compliance with the current regulatory framework ensures that the consequences of any potential natural phenomena event are low to offsite receptors and workers.

3. Conclusions

Because of the systematic and methodical evaluation of fuel facilities and GTCM licensees in light of the lessons learned from Fukushima, NRC staff concludes that the current regulatory approach and requirements for fuel cycle licensees offers reasonable assurance of adequate protection of public health and safety. For these facilities, compliance with the current regulatory framework ensures that the consequences of any potential natural phenomena event to offsite receptors and workers are within the dose limits stipulated by the regulatory requirements. NRC staff will continue its efforts to resolve concerns with fuel facilities' safety assessments and the supporting documentation with respect to the treatment of NPHs. NRC staff expects to close out Generic Letter 2015-01 activities by mid-year 2017.

Evaluating the Resilience of Nuclear facilities at Sellafield

Andrew B Buchan
Sellafield Ltd, UK

-

Summary

The Sellafield site comprises a wide range of nuclear facilities, including operating facilities associated with the Magnox reprocessing programme, the Thermal Oxide Reprocessing Plant (THORP) and a range of waste treatment plants. The focus of site operations is now moving towards the remediation of legacy facilities enabled by safe, secure and efficient waste management.

The operational life of some of the site facilities currently extends to 2120, requiring the retrieval, treatment, consolidation and safe extended storage of a variety of radioactive materials. Sellafield has utilized its existing safety assessment processes to inform and prioritise studies into beyond design basis events and resilience evaluation required following Fukushima. These studies have been used inform the response to Stress Tests by UK regulators and industry bodies such as ENSREG.

There are significant differences between NPPs, for which the ENSREG *"stress tests"* were originally intended, and the Sellafield site which is instead centred around two reprocessing facilities (Magnox and THORP), with a supporting infrastructure of waste processing and storage facilities, coupled with a legacy of high hazard older facilities. In the case of NPPs, the consequences of a catastrophic failure are promptly realized, leading to significant problems such as fuel failure / meltdown in AGRs and LWRs respectively in the event of complete loss of cooling. At Sellafield, the processes are carried out at comparatively low temperatures and pressures with relatively low rates of change following any loss of cooling. Instead the consequences of catastrophic failure at Sellafield are more directly related to the very large inventories of radioactive materials, including high level liquid wastes and unprocessed fuels, present in specific plants and the condition of the ageing assets, holding legacy wastes.

Previous papers have detailed how a number of analysis techniques have been utilized to facilitate design, operation, resilience evaluation and accident management of facilities supporting the range of operations at Sellafield.

This paper focuses on the higher level interaction issues associated with having a number of high hazard facilities located close to each other on a compact and condensed site. It outlines how both analysis techniques and requirements for engineered solutions have been developed in the following areas.

- Severe Accident Analysis considering serious but very unlikely accidents where off-site consequences are projected to be significant, providing information on their progression, within the facility and also beyond the site boundary.
- Severe Accident Management Strategies which have been developed to deal with such envisaged accident conditions as they develop
- How “domino” effects of severe accident scenarios have been considered in evaluating the effects of a severe accident in one facility upon adjacent nuclear facilities, infrastructure and the wider site.

All of these activities continue to represent real learning from the events at Fukushima.

1 Background

The Sellafield site comprises a wide range of nuclear facilities, including operating facilities associated with the Magnox reprocessing programme, the Thermal Oxide Reprocessing Plant (THORP) and a range of waste treatment plants and interim storage facilities. Interim storage facilities include fuel storage ponds, legacy waste silos, solid intermediate level waste stores and encapsulated waste stores. Whilst reprocessing operations are planned to cease in the next two to four years, the operational life of some of the site facilities currently extends to 2120, requiring the retrieval, treatment, consolidation and safe extended storage of a variety of radioactive materials. Sellafield has utilised its existing safety assessment expertise and experience to inform and prioritise studies into beyond design basis events and resilience evaluation required following Fukushima by UK regulators and bodies such as ENSREG.

There are significant differences between Nuclear Power Plants, for which the European Nuclear Safety Regulators (ENSREG) “stress tests” were originally intended, and the Sellafield site which is instead centred around two reprocessing facilities (Magnox and THORP), with a supporting infrastructure of waste processing and storage facilities, coupled with a legacy of high hazard older facilities. In the former case the consequences of a catastrophic failure are promptly realised, leading to significant problems such as fuel failure / meltdown in Advance Gas Reactors (AGRs) and Light Water Reactors (LWRs) respectively in the event of complete loss of cooling. At Sellafield, the processes are generally carried out at comparatively low temperatures and pressures with relatively low rates of change following any loss of cooling. Instead the consequences of catastrophic failure at Sellafield are more directly related to the very large inventories of radioactive materials, including high level liquid wastes and unprocessed fuels, present in specific plants and to the condition of the ageing assets, which hold legacy wastes.

2 Safety Assessment Approach

The safety assessment of nuclear chemical plants and support facilities at Sellafield has been developed over a number of decades and a range of techniques has been applied to ensure high levels of safety for both the workforce and the general public. Sellafield has been an active participant in best practice benchmarking via peer assists in nuclear operations for a number of years through WANO membership. In addition, Sellafield has been keen to promote good practice in safety assessment of nuclear facilities in both national and international fora[3].

For many years, Sellafield Ltd, formerly British Nuclear Fuels Ltd (BNFL) made use of probabilistic safety assessment (PSA) and comparison with a set of radiological risk criteria, as a principal means of demonstrating the adequacy of safety of new designs of nuclear chemical plant. The relevant radiological risk criteria were defined essentially in such a way that the overall risk from a plant, if the criteria were satisfied, would be at the 'broadly acceptable' level; there was in addition, an overriding obligation to demonstrate that any residual risk was ALARP (as low as reasonably practicable).

Whilst there had always been an element of deterministic assessment of safety, this had not been by use of any formal analysis technique. However, in the late 1990s, the requirement to demonstrate the adequacy of safety by means of a formal and structured deterministic method, in addition to the use of PSA, was addressed. Previous papers highlight the development and evolution of the safety assessment approach[4, 5] and its evolution into severe accident analysis studies[6].

---

[3] CSNI Report, Use and Implementation of Safety Cases by Regulators and Nuclear Site Operators on Fuel Cycle Facilities in NEA Member Countries Jan 2006, CSNI/SEN/FCS(2006)1.

[4] S Drake, Application of Deterministic criteria in addition to Probabilistic Safety Analysis, OECD Fuel Cycle Safety Conference, Paris September 2004.

[5] PW Ball, Developments in Safety Assessment over the Last Thirty Five Years, Journal of the Safety and Reliability Society, Vol 32 No 4

[6] AB Buchan, The Safety Assessment of Long Term Interim Storage at Sellafield International Workshop on IAEA/OECD FCSWG Munich 2013

3 Fukushima and Beyond Design Basis Events

Following the massive earthquake and subsequent tsunami that struck Fukushima Daiichi NPP in Japan on 11 March 2011, Governments, Regulators and the Nuclear Industry established review programmes to examine the robustness of their own facilities, and to provide a focus for the learning emerging from Fukushima. In Europe the EC supported the development of "stress tests" by ENSREG which were deployed by national regulatory bodies. The output from these ENSREG tests has now been publicly reported for UK facilities by the Office for Nuclear Regulation (ONR). In respect of non-NPP facilities e.g. Sellafield, the ONR instructed Site Licence Companies to apply the ENSREG "Stress tests" and to report the resultant findings to provide comprehensive and consistent coverage across UK nuclear installations.
The ENSREG "stress tests" required evaluation of the resilience of a large number of facilities on the Sellafield site and of the integrated site as a whole. Sellafield has an existing programme of safety cases which are subject to regular review and update through the Long Term Periodic Review (LTPR) process, a key component of the site licence arrangements. A safety case is required for all high and medium nuclear hazard facilities on the site; safety cases are also prepared for key utilities and infrastructure systems.
To provide a manageable and proportionate basis for applying the ENSREG "stress tests" across a complex site such as Sellafield with over two hundred active facilities it was necessary to focus on significant plants. Accordingly all safety cases were reviewed for significant potential fault sequences with an off-site consequence threshold of 10 mSv to the critical group. This threshold is an existing accident risk threshold and hence these faults could readily be identified from extant hazard analysis documentation. These safety case documents had been subject to full internal verification and regulatory review, thus providing a substantive foundation for this work.

To support the requirement to analyse beyond design basis events and the subsequent loss of safety critical utilities (as experienced at Fukushima) Sellafield Ltd developed and applied two further processes, Severe Accident Analysis and RESilence Evaluation Process (RESEP), to enable a more developed understanding of both potential fault states and the required responses. These processes use the very developed understanding already in place as a result of a comprehensive program of safety cases underpinned by extensive DBA and PSA studies. The focus of the RESEP was to look at prolonged outage of services; the SAA focus was to look at plant damage states, this included plant process faults within the safety case, and a range of external events.

4 Severe Accident Analysis

Severe Accident Analysis (SAA) was developed to allow a comprehensive evaluation of high consequence scenarios and provide, when used in conjunction with the outputs of the RESEP process, an enhanced understanding of time-related severe emergency management issues for key plants at Sellafield. This understanding has been used as a key input to the development of Severe Accident Management Strategies

(SAMS) for a range of facilities at Sellafield. These give strategic guidance on incident response and management, inform tactical emergency response instructions, inform the procurement or development of equipment specifically required for emergency response and, in addition, aid in developing future emergency exercise programmes.

SAA is hence complementary to existing DBA analysis and PSA techniques and uses a best estimate approach to determine the magnitude and characteristics of accident consequences and considers so-called ‘cliff edge’ effects so as to identify reasonably practicable preventive or mitigating measures for Beyond Design Basis Accidents (BDBA) and to provide a basis for emergency plans and Severe Accident Management Strategies (SAMS).

As there are common fault initiators, consequence escalation pathways and response strategies, plants have been grouped and prioritized, depending on the type, form, magnitude and radiotoxicity of the potential release. The approach to SAA is shown in Figure 1. In developing the approach to SAA, Sellafield Ltd undertook benchmarking visits with EDF (formerly British Energy) who had a well-developed and respected process for reactors; in addition it reviewed severe accident events in the petrochemical process sector such as Deep Water Horizon and Buncefield. Key to the process is the identification of fault states and the key safety functions which have to be maintained to reduce potentially significant offsite doses. An analysis is then carried out of how these key safety functions could be maintained or restored noting any constraints (such as high dose rates, the potential for release of contamination or adverse chemical reaction). These modified responses, along with any identified improvements, are then incorporated into Severe Accident Management Strategies. A typical output is shown below in Table 1.

Table 1: Critical Safety Function Identification

	Shielding	**Cooling**	**Containment**	**Criticality**	**Control**
Safety Case	Bulk shielding is robust	Cooling may be lost to the vessel if the service pipework is disrupted as a result of a H2 ignition	Primary containment may be lost if vessel damaged as a result of a H2 ignition	No issue	It may not be possible to empty vessels using the installed emptying system if the vessel integrity is lost. Large volumes of water may be added to vessel / cell
External Hazard	Bulk shielding is robust to external hazards	The vessel vent system is vulnerable to a seismic event / vehicle impact	Containment is robust to external hazards	No issue	No issue

Red = Potential for Severe Accident consequences, Severe Accident Management Strategies developed
Orange = Significant Consequences could develop but below Severe Accident threshold, some consideration required
Green = No Further consideration required

5 Resilience Evaluation Process (RESEP)

The RESEP process has been developed as a structured and consistent approach to resilience assessment for the Sellafield site that satisfies the requirements of the ENSREG stress tests. Additionally, ENSREG set the background scenarios as being, “the most unfavourable operational states that are permitted under plant technical specifications” with all plants being simultaneously affected and offsite power assumed to be lost for several days, the site isolated from delivery of heavy material for 72 hours and portable lightweight equipment for 24 hours.

RESEP made use of preliminary work undertaken as part of the SAA programme, to develop ‘timelines’ for key plants, indicating their dependence on utilities as they are progressively “stressed”, as well as assessing the robustness of the emergency arrangements/infrastructure to a severe accident scenario. A key focus of the RESEP was to utilise the plant operating staff, rather than the safety case expertise. This was to maximise the potential to identify practical mitigations which could be developed and deployed. The plant RESEP process was designed to be carried out in several stages. Stage one concentrated on the individual plants. Each of the key plants was subject to a plant RESEP which:

- reviewed the robustness of the existing arrangements to prevent/mitigate against the fault. “stressed” these arrangements against a number of beyond Design Basis Events (DBE) such as prolonged loss of utilities (7 day Site Black Out (SBO), seismic and other signification site wide event such as extreme weather)

The output from the plant RESEP identified:

- the critical safety functions (CSF) of the plant
- demand on Site Utilities
- demand on central resource such as Sellafield Fire and Rescue Services
- the extent of plant self-reliance
- any shortfalls in existing arrangements
- future improvement considerations

The data collected from the individual plant reviews was rolled-up to enable the 'key' plant reliance on site utilities i.e. power, water, steam, communications etc. and Sellafield Fire and Rescue Services (SF&RS) to be produced and therefore, consequently, provide the basis for the next stages of the RESEP process:

- stage 2 of the RESEP process concentrated on Site Wide Utilities
- stage 3 of the RESEP process concentrated on the Communications, Emergency Service and Site control functions
- stage 4 of the RESEP process reviewed the preparedness of the current emergency arrangements when challenged with a greater than Design Basis seismic event.

The output from these RESEP stages identified; the “domino” impacts from a loss of utilities, the radiological domino impacts and the command and control “domino” impacts. A key aspect of the RESEP approach was to identify the cliff-edge (see Figure 2) , that point where the Design Basis has been exceeded.

The key characteristics of this approach are as follows:

- Strong representation from plant operators, coupled with a challenging technical team, who have prepared thoroughly
- The process is deterministic (assume that events have happened) and therefore avoids lengthy and complex probabilistic analysis, and debate
- Greater emphasis is placed on events and plants with the potential for off-site radiological consequence, thus channelling effort to where it can bring greatest benefit
- It is not bounded by the time limits of any regulatory guidance and was applied flexibly.
- Each possible line of defence in the plant is progressively challenged and removed. This approach thoroughly tests the boundaries of each plant's resilience.

The output from RESEP has been applied to help further develop and inform Severe Accident Management Strategies (SAMS), by providing timelines which clearly show the timing of degradation in a plant under different challenges. These timelines will also be adopted for use in emergency control centres. RESEP typically leads to a series of considerations (potential actions which can improve resilience and which require further review). The implementation of these potential improvement actions is described in detail elsewhere[7].

6 Meeting Relevant Regulatory Requirements.

The primary regulatory expectations are set out within Safety Assessment Principles, which have been subject to review to reflect learning from the Fukushima Daiichi event[8]. Three forms of analysis can be used to establish a safety case for fault and accident conditions[9] (SAP Fault Analysis FA.1), i.e. design basis analysis (DBA), probabilistic safety analysis (PSA) and severe accident analysis (SAA).

- DBA is focused on the key safety measures for those initiating faults that are most significant in terms of frequency and unmitigated potential consequences (FA.4 and FA.9);
- PSA considers the full range of fault sequences and allows full incorporation of the reliability and failure probability of the safety measures and other features of the design and operations (FA.10 and FA.14); and
- SAA considers significant but highly unlikely accidents where off-site consequences are likely to exceed 100mSv to the critical group and provides information on their progression, both within the facility and also beyond the site boundary (FA.15 and FA.16).

---

[7] A O'Loane Optimising the Resilience of Nuclear Facilities at Sellafield, International Workshop on IAEA/OECD FCSWG Aomori City 2016.

[8] Nuclear Installations Inspectorate, Safety Assessment Principles for Nuclear Facilities (2014 Edition).

[9] IAEA, INSAG 10, Defence in Depth in Nuclear Safety, A Report by the International Nuclear Safety Advisory Group, 1996

Sellafield Ltd is satisfied that it has effectively and efficiently met the required standards set out for the above analysis, noting that its own SAA threshold has been taken as 10mSv to take account for variability in weather conditions which could give rise variability in off site doses. Figure 1 illustrates how the combination of DBA, PSA, SAA, RESEP and SAMS meet the requirements of defence in depth in nuclear safety as highlighted by international best practice such as INSAG 10. Sellafield Ltd has placed considerable emphasis on maximising learning from Fukushima Daiichi. This has included significant effort in sharing experience with other operators, both nationally and internationally such as:

- Support to a UK Fukushima Working group, sponsored by Safety Directors of all UK nuclear licencees
- Involvement in International Atomic Energy Agency (IAEA) and World Association of Nuclear Operators (WANO) conferences

7 Severe Accident Management Strategies

A key aspect of the developing phases of an emergency management timeline is that the actual state of consequences may be quite difficult to predict in detail. As such, whilst protection strategies for the early phase of an incident may be relatively straightforward to characterise and prepare for an identified scenario, such strategies following the incident's onset become increasingly difficult to plan in detail. Therefore SAMS are focused at a high level and are to be used primarily to inform strategic response planning. For a large multi facility reprocessing and waste management complex such as Sellafield the following factors have to be taken into account at both individual facility and site level.

Studies indicate that in a severe accident event, response team stress levels are predicted to be very high, with significant potential for human error in providing an effective response[10]. As a result the SAMS have been subject to review by human factors experts to optimise their effectiveness. A balance has been achieved to ensure that documents are comprehensive but concise, with extensive use made of tables and flow charts. Significant effort has been expended to ensure that the SAMS documents clearly identify the critical safety functions which are affected, that they are consistent in layout and easy to follow e.g. using colour coding. This consistency aids training across a range of plants, and provides a formal, structured and uniform approach for team members. Given the higher degrees of uncertainty inherent in evaluating severe accident development, a range of response options are given. The potential for further fault escalation is highlighted,

[10] J Williams, J Bell, Consolidation of the Error Producing Conditions used in Human Error Assessment and Reduction Technique (HEART), Journal of the Safety and Reliability Society Vol 35, No3.

along with predicted timescales, allowing the efficacy, practicality and potential side effects of response options to be considered well ahead of implementation.

For a diverse and complex nuclear fuel cycle site such as Sellafield, SAMS are key to :

- identifying vulnerable plant areas/systems with a facility
- highlighting the potential for further escalation of the event
- prioritising the response across the site
- dealing with external hazards events at a site as well as facility level

8 Development of Studies to Consider On-Site Domino effects

As part of the further development of Severe Accident Analysis, studies have been carried out to determine the potential effects which may be caused by domino events following or during a Severe Accident Scenario. A domino event is where an initial event sets off a chain of similar or related events in adjacent facilities. These domino effects may be caused by direct radiation, release of contamination, etc.

The work began with two strands of gap analysis/ desktop review. The first strand looked at other Nuclear Licensees and high hazard industries both UK and internationally. It found little research or methodology which can be taken from other domino methodologies which is transferable to a nuclear reprocessing facility such as Sellafield Site. The second strand of gap analysis reviewed existing Sellafield Ltd work streams which characterise or articulate preparedness against domino initiators. Most categories of initiator e.g. seismic, criticality were found to be already included into Sellafield Ltd Emergency Planning by existing Sellafield Ltd work streams & resources. The maturity of those work streams was such that it was sufficient to cross-reference these and then consider which type of hazard had been least developed against domino analysis to date. This led to the conclusion that radiological dose to on-site workers was the area where the Domino study could best improve Emergency Preparedness.

By reviewing the Severe Accident Analysis project and consulting underlying Safety Cases and facility technical experts, 40 faults were identified as being potentially domino significant in their consequences. On-site dose consequence calculations were generated based on the data in current, approved safety assessments and documented in a series of grouped domino effects reports. These consequences were plotted and overlaid onto a site map so that location-specific consequences could be calculated from each fault onto each recipient building. Facilities were identified as domino sensitive recipients if they were buildings which would likely be needed to continue functioning in a post-event scenario, or if they had the potential to result in a high off-site dose consequence if they were to suffer a fault.

This screening identified only 75 facilities as being domino sensitive recipients out of approximately 540 buildings which were accounted for in the underlying consequence assessments. A spreadsheet tool was developed and colour coded (red, amber, green) to indicate the significance of the potential consequences to each recipient building for each fault. In addition a more developed understanding was achieved by overlaying the effects in layers using the proprietary ARC-GIS software system, allowing the development of evacuation and response and recovery scenarios based on location and faults. This has helped indicate potential operability of facilities and services during a severe accident event and further inform the development of the Severe Accident Management Strategies during emergency exercises.

9 Conclusions

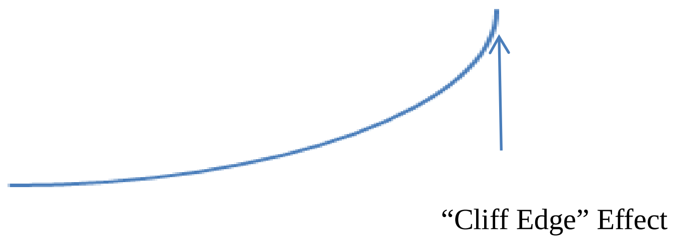
To support the requirement to analyse beyond design basis events and the subsequent loss of safety critical utilities (as experienced at Fukushima), Sellafield Ltd has developed two further processes, Severe Accident Analysis and Resilience Evaluation, to enable a more developed understanding of both potential fault states and the required responses. These processes build on the very developed understanding already in place as a result of a comprehensive programme of safety cases with an associated programme of long term periodic review.

These approaches to resilience analysis and improvement have enabled new learning to be derived, whilst residing effectively within existing IAEA standards and within the existing national regulatory framework. This further demonstrates the effectiveness of both existing standards and regulatory frameworks. The techniques developed since Fukushima have been effectively applied to the range of fuel cycle facilities covering fuel receipt ponds, reprocessing plants, waste treatment facilities and product stores. These reviews have considered the resilience of complex, diverse fuel management facilities, comprising many tens of plants, with different functions. The driving objective was to improve protection to the workforce and the public by maximising the learning from Fukushima. Such improved protection only becomes effective if action is taken. Therefore, the focus was on balancing the depth of analysis with the promptness and practicality of action. As a result, a comprehensive set of Severe Accident Management Strategies has been developed and implemented. These have set the integrated requirements for necessary enhancements in resilience capability. The output of Severe Accident Analysis has been further progressed to derive an understanding of domino events and indicate the potential functionality of facilities and services at a site level during a full range of severe accident events allowing further development of the Severe Accident Management Strategies. The Severe Accident Analysis and resultant Severe Accident Management Strategies have formed key inputs to further enhance of a regular programme of challenging emergency exercises.

Figure 1: Integration of Methods

Equivalent Levels from INSAG 10	Requirement	Sellafield Ltd Process
Level 5	Mitigation of Radiological consequences of significant releases of radioactive materials	Severe Accident Management Strategies
Level 4	Control of Severe Plant Conditions including prevention of accident progression and mitigation of the consequences of severe accidents	Severe Accident Analysis and Resilience Evaluation Process
Level 3	Control of Accidents within the Design Basis	Probabilistic Safety Analysis
Level 2	Control of Abnormal Operation and Detection of failures	Design Basis Analysis
Level 1	Prevention of Abnormal Operation and Failures	Hazard Management Strategy

Figure 2 Attributes for Design Basis and Severe Accident Consideration

	Design Basis		Severe Accident
Consequences			
Uncertainty	Low		High
Engineering	Passive		Mitigative
Time to Respond	Long		Short
Operator Stress	Low		High
Analysis Margins	Fully Conservative		Fully realistic

Results and Consequences of Stress Tests performed for Interim Storage Facilities of Radioactive Material in Germany

Christian Drobniewski

Federal Office for the Safety of Nuclear Waste Management
(Bundesamt für kerntechnische Entsorgungssicherheit BfE)
Christian.Drobniewski@bfe.bund.de

Bundesamt für kerntechnische Entsorgungssicherheit
Willy-Brandt-Straße 5, 38226 Salzgitter, Germany

Abstract

In the light of the accident at the Fukushima Daiichi NPP in March 2011 stress tests were performed for the NPPs in Germany by the RSK (Reactor Safety Commission)[11]. Comparable investigations were performed for all FCFs in Germany. The results of those investigations were presented in two reports by the ESK (Nuclear Waste Management Commission)[12],[13] in 2013 which provide the results for the FCFs for nuclear fuel production and handling, for the storage facilities of spent fuel and high level waste and as well for all FCFs dealing with low and intermediate level waste.

Subsequently the guidelines for the storage facilities for spent fuel and high-level waste respective low and

---

[11]Anlagenspezifische Sicherheitsüberprüfung (RSK-SÜ) deutscher Kernkraftwerke unter Berücksichtigung der der Ereignisse in Fukushima-I (Japan) (2011)

[12] ESK-Stresstest für Anlagen und Einrichtungen der Ver- und Entsorgung in Deutschland
Teil 1: Anlagen der Brennstoffversorgung, Zwischenlager für bestrahlte Brennelemente und Wärme entwickelnde radioaktive Abfälle, Anlagen zur Behandlung bestrahlter Brennelemente (14.03.2013)

[13]ESK-Stresstest für Anlagen und Einrichtungen der Ver-und Entsorgung in Deutschland
Teil 2: Lager für schwach-und mittelradioaktive Abfälle, stationäre Einrichtungen zur Konditionierung schwach-
und mittelradioaktiver Abfälle, Endlager für radioaktive Abfälle (18.10.2013)

intermediate waste were updated in 2013[14],[15] to include the results as well as new developments in the field.

The BfS as the competent authority in Germany for licensing the storage facilities for spent fuel and high level waste in Germany did perform a qualitative examination of extreme events by extrapolation on the design-based accidents and presented[16] this in 2011 at the workshop **Safety Assessment of Fuel Cycle Facilities – Regulatory Approaches and Industry Perspectives** of the WGFCS in Toronto.

In accordance with the waste management concept in Germany, spent fuel is stored in interim storage facilities for a period of up to 40 years until deposition in a geological repository. In twelve on-site interim storages in the vicinity or directly on the sites of the nuclear power plants, spent fuel elements from reactor operation are stored after the necessary period of decay in wet storage basins inside the reactors. Additionally, three central interim storage facilities exist for storage of spent fuel and high-level waste.

This paper focusses in particular on the results and consequences (for regulator, operator, ...) of the above-mentioned investigations on the storage facilities for spent fuel and the resulting implications therein.

---

[14] Leitlinien für die trockene Zwischenlagerung bestrahlter Brennelemente und Wärme entwickelnder radioaktiver Abfälle in Behältern, ESK (10.06.2013)

[15] Leitlinien für die Zwischenlagerung von radioaktiven Abfällen mit vernachlässigbarer Wärmeentwicklung, ESK (10.06.2013)

[16] Dry Interim Storage of Radioactive Material in Germany C. Drobniewski; J.Palmes (2011)

1. Introduction

The goal of the article is to provide an overview over the performed “Stresstests” in Germany and the results and consequences of those investigations.

By the term “storage facility for spent fuel” the article refers to the dry interim storage facilities build for the storage of spent fuel and high level waste in transport and storage casks. Those facilities are located either on-site with the nuclear power plants or as a centralized storage facility on a separate site.

Main safety element of the storage facilities are the storage casks themselves, as they provide the necessary confinement and shielding of the radioactive waste.
The inventory integrity is provided by the structural stability of the steel cask, the special holding structure for the fuel inside and the rod claddings. Leak tightness is by design secured by a double lid closing. Proper shielding of radiation is achieved by the casks and their inbuilt neutron moderator material.
For the licensed radioactive inventory the cask provides thermal stability of the waste, radiation shielding, sub criticality and leak tightness. The storage facilities in Germany are buildings of concrete with wall thicknesses between 0.8 m and 1.2 m. They are sectioned to provide separate rooms for reception, repair, maintenance and storage respectively.

The licensing procedure is defined by the German Atomic Energy Act (AtG[17]) supplemented by the requirements stated in the radiation protection regulations StrSchV[18] and the ESK guidelines[19].

The following section 2 will summarize the performed investigations, for both the ESK documents and the investigation performed by the BfS. Section 3 will focus on the results of the Investigations described in section 2. The final section 4 will summarize the result and concludes the document.

[17] AtG – Gesetz über die friedliche Verwendung der Kernenergie und den Schutz gegen ihre Gefahren (Atomgesetz) – in the present valid version

[18] StrlSchV – Verordnung über den Schutz vor Schäden durch ionisierende Strahlen (Strahlenschutzverordnung) – in the present valid version

[19]Leitlinien für die trockene Zwischenlagerung bestrahlter Brennelemente und Wärme entwickelnder radioaktiver Abfälle in Behältern, ESK (10.06.2013)

2. Stress Test related Investigations in Germany

In Germany stress tests were performed for the NPPs by the Reactor Safety Commission (RSK)[20]. Comparable investigations were performed for all other nuclear installations (eq. FCF's, interim storage facilities) in Germany.
The results of those investigations were presented in two reports by the Nuclear Waste Management Commission (ESK)[21;22] in 2013, which provide the results for the FCFs, for the storage facilities of spent fuel and high level waste and as well for all facilities dealing with low and intermediate level waste.
This article only deals with the investigations concerning the interim storage facilities for spent fuel.

The BfS performed a qualitative examination of extreme events by extrapolation of the design-based accidents[23] in 2011 and provides investigations for all interim storage facilities for spent fuel in Germany.

2.1 Investigation of the ESK

The investigations of the ESK started by assembling the set of relevant impacts and the respective assessment criteria. The set of impacts include earthquake, flood, heavy rainfall, other weather impacts, loss of power supply, internal/external fire, airplane crash and detonation waves. The assessment criteria are divided in several stress levels ("Stresslevel") and safety levels ("Schutzgrade") distinguishing between natural and civilisatory impacts.
The operators of the spent fuel storage facilities were given a questionary to provide documents to the ESK to enable the assessment.

At the time the questionary was supplied there were 12(13) on-site spent storage facilities and 3 centralized storage facilities in operation.

---

[20] Anlagenspezifische Sicherheitsüberprüfung (RSK-SÜ) deutscher Kernkraftwerke unter Berücksichtigung der der Ereignisse in Fukushima-I (Japan) (2011)

[21] ESK-Stresstest für Anlagen und Einrichtungen der Ver- und Entsorgung in Deutschland
Teil 1: Anlagen der Brennstoffversorgung, Zwischenlager für bestrahlte Brennelemente und Wärme entwickelnde radioaktive Abfälle, Anlagen zur Behandlung bestrahlter Brennelemente (14.03.2013)

[22] ESK-Stresstest für Anlagen und Einrichtungen der Ver-und Entsorgung in Deutschland
Teil 2: Lager für schwach-und mittelradioaktiveAbfälle, stationäre Einrichtungen zur Konditionierung schwach-
und mittelradioaktiver Abfälle, Endlager für radioaktive Abfälle (18.10.2013)

[23] Dry Interim Storage of Radioactive Material in Germany C. Drobniewski; J.Palmes (2011)

The main assessment criteria were:

- Do the facilities keep their vital functions at the stress levels / safety levels
- What are the feasible maximum consequences of the stress level impacts
- Are cliff-edge effects feasible?
- On which information basis are the above mentioned questions answered/proven?

For an easier comparison between the BfS investigation and the ESK investigation, only the same subset of impacts and their stress levels are described. The full list is available in the ESK report [11].

The list of described impacts in this article is therefore limited to:

a) Flooding

The main questions for the impact "flooding" included:

- the assessed level of flood height for the license and its basis
- disaster prevention measures relied on or provided
- information on the behaviour of the facility for stress level flooding
- available disaster remediation measures.

The assessment levels are:

- Base level - the design base flood level
- Stress level 1- 1.5 times higher water discharge for river sites and 1 m higher flood level for sea sites
- Stress level 2 - 2 times higher water discharge for river sites and 2 m higher flood level for sea sites
- Stress level 3 – site is not prone to flood due to the enviroment

b) Earthquake

The main questions for the impact "earthquake" included:

- the assessed earthquake for the license and its basis
- information on the behaviour of the facility for stronger earthquakes then the design base earthquake (if available)
- assessed concurrent scenarios and available disaster remediation measures.

The assessment levels are:

- Base level - the design base earthquake
- Stress level - design base earthquake intensity +1

c) Fire

The main questions for the impact "internal fire" included:

- the assessed fire and its basis
- Disaster prevention measures relied on or provided
- information on the behaviour of the facility for stress level fire
- available disaster remediation measures.

The assessment levels are:

- Base level - the design base
- Stress level 1- fire for 1 hour longer then the design base
- Stress level 2 – the facility is not prone to prolonged fire due to the limited amount of burnable material at the site

d) Explosions

The main questions for the impact detonation waves included:

- the assessed detonation wave (if any)
- possible consequences of a stronger detonation wave
- amount of explosive material close by or on site
- if the facility has no assessed explosion wave, which consequences are foreseen if such an explosion were to be assessed (if occurrence cannot be excluded due to site specific circumstances)

The assessment levels are:

- Safety level 1 - the facility keeps its vital functions under detonation wave impact
- Safety level 2- the facility keeps its vital functions under impact of a 20% stronger detonation wave as in safety level 1 (p(t) curve 20% higher).
- Safety level 3 - the facility is not prone to detonation waves due to site specific circumstances

It should be not that the BfS investigation for accidental stress test situations does not cover the (although implicitly covered) airplane crash. This is due to the fact that the accidental crash of a military aircraft is regarded as a beyond design-basis accident scenario and covered as such in the licensing process, while the crash of a large passenger aircraft is regarded as a security issue, which is therefore dealt with separately and does not fit in this extrapolation regime. The ESK however followed the RSK by evaluating the airplane crash scenarios as follows:

The main questions for the impact airplane crash:

- Were the consequences of an airplane crash assessed in the licensing process, and if which?
- Which thermal and mechanical impacts are assessed due to the airplane crash?
- Is the site in a regularly commuted flight zone?

The assessment levels are:

- Safety level 1 - the facility keeps its vital functions under the mechanical and thermal impact of a military aircraft of the type Starfighter
- Safety level 2- the facility keeps its vital functions under the mechanical and thermal impact of a medium size (passenger) plane
- Safety level 3 - - the facility keeps its vital functions under the mechanical and thermal impact of a large size (passenger) plane

2.2 Investigation of the BfS

In [13] a set of extreme scenarios exceeding the design basis accident scenarios were derived and investigated. Those were used to assess safety margins of the storage facilities. Therefore those scenarios were not treated on a quantitative but qualitative level, especially to identify the scaling of accident consequences in dependence of the accident/disaster extent to identify cliff-edge effects if applicable. After a screening of scenarios to consider, the event list included Flooding, Earthquakes and subsequent events, Fire and external Explosions.

a) Flooding

As floods are events based on measured or simulated data, the height of such events can be extrapolated to cover more severe occurrences. The qualitative assessment was done without any set levels of flood height (like the ESK did) as this would need site specific investigations. The goal of this study was however to provide in principle overall applicable results.
Therefore the assessment is done in hypothetical “unlimited” height to identify any cliff-edge effects or severe consequences resulting from the site.

b) Earthquakes

As the design base earthquakes are assessed following the KTA 2201.1 the starting point for any investigation of extreme events is given. As stated for flooding the investigation will not be based on site or facility specific data, but rather in a qualitative way identifying possible consequences of an arbitrary large earthquake exceeding the design base earthquake (which would be different in strength for each site/facility). Extreme earthquakes can in principle lead to 2 possible postulated initiating events (PIE). Firstly the casks can tip which leads to a mechanical impact, and secondly the casks can after tipping be buried under debris of the concrete structures which leads to an additional thermal impact. While the first should be a result of a comparable strong earthquake for all sites mostly, the second is strongly dependent on the construction of the facility and the site specifics, which might even include severe effects like soil liquefaction.

c) Fire

The impact of fire is assessed in the licensing process on the base of possible fire scenarios resulting from the availability of burnable material at the given location. As the storage facilities are not close to any highly burnable surroundings, external fire was not investigated further. The internal fire was investigated in the aspect of thermal impact and feasibility as an accidental condition. Compared to flooding or earthquake which are mainly site dependent, the fire scenarios are highly dependent on the facility design and operation.

d) Explosions

Explosions in a relevant vicinity to a storage facility were considered and as stated for internal fire, the feasibility of occurrence is also investigated.

It should be noted again, that the accidental crash of an airplane was not further investigated as the accidental crash of a military aircraft is covered in the licensing process as a safety related scenario, while the crash of a large (passenger) plane is considered and investigated as a security issue.

3. Results of the Investigations

3.1 Results of the ESK Investigation

In the summary of the ESK report [11] the following conclusions are drawn:

The dry interim storage as it is realized by the storage facilities in Germany is a robust concept which relies mainly on the properties of storage cask. Due to this fact, even for the Stress Test conditions no severe disaster remediation actions will become necessary.

The documents provided by the operators and local authorities show that the storage facilities provide enough reserves in the design to achieve even the safety in the highest stress levels / safety levels in most cases. One case where the ESK could not decide if the highest level safety was proven was the flooding scenario, as the documents do not provide distinct data for such cases to the extent necessary. Nevertheless the ESK concluded that for flooding a negative effect on the safety appears to be not feasible.

As the KTA 2201.1[24] was updated in 11/2011 the ESK could not provide a complete statement regarding possible impacts thereof on the design base earthquakes for the facilities in the report.

In the meantime the BfS has checked the design base earthquakes for the on-site storage facilities and the centralized storage facilities at Ahaus and Gorleben for compatibility with the updated KTA 2201.1. The results show that the design base earthquakes are still fulfilling the requirements of the updated KTA 2201.1.
For the storage facility in Jülich, the design base earthquake is under investigation in a §6 AtG licensing process.

Further details on the results can be found in the ESK report [11] as it provides detailed results for each facility and for the in this article not explicitly described impacts.

3.2 Results of the BfS Investigations

The results for the identified scenarios as described in the BfS article [13] are given below:

a) Flooding

As the main safety functions are provided by the casks and the leak tightness is provided up to 200m submersion, even extreme floods that exceed the design based flood disaster cannot create any substantial release. As floods are typically no prolonged conditions, corrosive damage to the cask can be excluded. Besides very fast rising flood events (Tsunami etc.) it is also very unlikely that the storage facility itself

[24] KTA 2201.1 Auslegung von Kernkraftwerken gegen seismische Einwirkungen Teil 1: Grundsätze Fassung 2011-11 (Design of Nuclear Power Plants against Seismic Events; Part 1: Principles)

will be damaged and therefore the casks are also safely enclosed at the storage area. If an event occurs that can harm the facility structure, one could argue about consequences comparable to the ones of an extreme Earthquake as described below.

b) Earthquakes

As described, the two possible results of an extreme earthquake are firstly tipping of the casks and secondly burying of the casks under debris of the concrete structures. It is reasonable to assume that the case of tipping casks is the one more likely to happen instead of the collapse of the storage building. Therefore we will take the toppled casks scenario as given when considering the facility collapse.

1) Toppling of the casks in the storage facility:
The mechanical impact of toppling is lesser than the mechanical impact of the accidental crash of a military aircraft. It is therefore conservative to assume that the releases of this case is limited to those resulting from the assessment of this scenario. When extrapolating the releases to the amount of casks possibly affected at the storage facility it leads to no unsustainable consequences.

2) Burying the cask below debris due to collapse of the facility building:
In an event of overwhelmingly strong earthquakes, the possibility of partly or full collapse of the facility building is feasible. In the most extreme case the casks will then be toppled and buried below debris. However, it is still feasible to assume that the mechanical consequences for the casks are of the same order as the ones described under the toppling impact. It is therefore safe to assume that the releases will be in the same dimension. However it is important to focus on the thermal impact due to reduced air cooling of the casks.
Under realistic assumptions, the heat up of the cask will take place in the timescale of weeks. Furthermore the temperatures resulting in this time span are in a range that one can safely assume that the integrity of the casks is provided. Nevertheless, special attention is advised for the working personnel in the recovery / retrieval of the casks, as the neutron moderator material might be lost in the long run (depending on the cask design), imposing special attention to radiation protecting of the working personal.

c) Fire

As seen in the former scenario the thermal condition of the cask is an important issue. While due to an earthquake the thermal impact is on a daily/weekly scale, a fire scenario is of much shorter timescale (shorter duration and higher thermal load).
In the licensing process the fire scenario is typically covered by a generic fire impact scenario with a

duration of 1h. The possible burning times in a storage facility for spent fuel are well below that time span as the amount of burnable material is very limited in the facilities.

Therefore fires with longer durations than 1h are very unlikely (in principle practical eliminated) to happen from natural sources.

Even including man made burnable substances, it is highly unlikely to sustain a fire of relevant extent for a longer period due to the sheer amount of necessary material. In addition the response time of fire-fighters is expected to not exceed hours.

Anyhow, assuming fires with varied duration shows that fire exceeding hours can lead to a cask temperatures inside so that the specified lid seal maximum temperature is exceeded.

In such a case the radiation release would be gradually higher than the releases calculated for the design based accidents, depending on fire duration. However, a temperature endangering the overall integrity of the cask is not feasible with substances viable in an accidental/disaster situation.

d) Explosions

Including explosions of sources closer and/or stronger lead basically to the same impacts as earthquakes. Moreover an explosion has a much shorter thermal impact time, so that the assumptions made in the extreme fire scenario are not applicable. Therefore the thermal/mechanical consequences are more of theoretical value than actual feasible.

An overview of the results are given in Table 1, taken from [13].

Table 1. BfS Results – Overview [13]

Accident/ Disaster	Radiologic consequences compared to limits	Feasible as accident/disaster	Non-radiological impact to surrounding area
Flood	Very limited to none	Unlikely in the extreme extent	Severe flood damage in a large surrounding area
Earthquake 1	Safely below limits	Yes	Depending on the structural integrity, high damage to buildings in the area
Earthquake 2	Safely below limits for a timeframe of several weeks[25]	Yes	Devastating effects on even strong concrete buildings and therefore for most civilian buildings
Fire	Below limits, limits are reachable for long durations (several hours)	No[26]	Only close vicinity will be affected
Explosion	Safely below limits	No[16]	Only close vicinity will be affected

5. Summary and Conclusion

As presented in the article, a wide range of Stress Test conditions (nature based events as well as man-made accidents) were covered by the investigations by the ESK and the BfS.
While the ESK used a fixed set of stress levels/ safety levels to assess and judge the facilities based on the provided documents, the BfS approach was driven by the investigation towards cliff-edge effects and the inquiry of safety margins starting from results proven in the licensing process. In this sense both investigations complement each other. The results show that the interim storage facilities for spent fuel in Germany provide a high level of safety margins.

Following the investigations, the guidelines for storage facilities for spent fuel [11] were updated to include the results depicted and keep the safety assessment for the facilities in line with the scientific knowledge and international standards. In the perpetual revision process the supplementing technical documents (e.g. KTA rules) have also been updated.

---

[25] Depending on the time buried - the release scales.

[26] The amount of burnable/explosive material is not feasible originating from an accidental situation and can therefore be regarded as practically eliminated.

As the ESK states in their report the fulfilment of safety level 3 for the impact of an airplane crash (crash of a large passenger plane) is based on the conducted investigations of the BfS in §6 AtG licensing process for the storage facilities. As mentioned above, in the §6 AtG licensing process, the crash of a large passenger plane is not considered an accidental scenario but as a security issue.

ENUSA-JUZBADO PLANT STRESS TEST APPROACH AND ACTIONS TAKEN

Enrique Escandón Ortiz eoe@enusa.es

Luis Blanco Fernández lbf@enusa.es

Óscar Zurrón Cifuente ozc@enusa.es

ENUSA Industrias Avanzadas, S.A.

Juzbado Fuel Fabrication Plant

Ctra. Salamanca-Ledesma, Km. 26

37115 JUZBADO (Salamanca)

Telephone: +34 923 32 9759

Fax: +34 923 32 1782

ABSTRACT

The Fukushima Daiichi accident posed certain weaknesses and vulnerabilities in plant design, emergency preparedness & response arrangements, and in planning management of an accident beyond design basis. Lessons learned from this accident brought about a systematic review of the safety analysis by the international community, defining and establishing ways of preventing and minimizing the effects of these severe accidents.

Thereby, during 2011 and 2012, ENUSA carried out a deep and systematic review of the design basis and safety analysis of the Juzbado Fuel Fabrication Facility.

The analysis was developed in four steps:

(a) Initial situation: Verification of the design basis, safety systems configuration, procedures associated to postulated accidents and license basis compliance, at the time of the analysis (June 2011).

(b) Robustness analysis: Evaluation of the safety margins for each postulated accident sequence, identification of limit situations or potential weaknesses, and definition of appropriate countermeasures, according to the defence in depth approach.

(c) Beyond design basis situation: Identification, analysis of the consequences and countermeasure definition of "extreme credible external hazards" that could damage structures, systems and important components for the safety (SSCs).

(d) Emergency preparedness and response arrangements: Evaluation of the structures, systems, components and human factors necessary to manage the emergency and mitigate its consequences for each scenario.

All evaluations assume, under a deterministic approach, the sequential loss of the existing defensive lines and system's safety functions.

The situations taken into account, that correspond to more and more degraded conditions, were:

- Earthquakes.
- Floodings.
- Extreme natural external hazards: Hurricanes, snow, heavy rain…
- Sequential loss of systems' safety functions: Loss of Offsite Power (LOOP), Station Black Out (SBO).

Most relevant conclusions were:

- ENUSA Juzbado Plant meet all the requirements of its Design Basis.
- Earthquake is the only credible risk, which can produce a limit situation in combination with the fire caused by the earthquake itself.
- LOOP and SBO do not produce a limit situation because in both cases the Juzbado plant automatically turns into a safe state that can be maintained in time.

According to these conclusions and in order to improve the emergency preparedness and response arrangements, several actions were taken:

- Rearrange H_2 supply pipelining to outside the plant.
- Implementation of a new water supply system for fire extinguishing, capable to operate after an earthquake.
- Improvements in power supply to ensure it and extend its time availability.
- Improvements in emergency procedures.

1. INTRODUCTION

The commissioning of Enusa was on 1985 and its main activity is focused on the first part of the fuel cycle fabrication. Thereby the production scope ranges from pelletizing uranium oxide or uranium oxide with gadolinium to rods loading and final assembly.

Enusa has four manufacturing lines, three devoted to PWR/BWR/VVER fuels and the other devoted to fabrication of Gadolinia rods.

On the other hand, the maximum enrichment authorized is five percent and the licensed capacity is five hundred tonnes of uranium per year.

As a result of the Fukushima Daiichi nuclear power plant accident, Stress Tests have been defined for the European Nuclear Power Plants, which have been focused on analyzing a set of extreme situations. The objective of this analysis is to highlight the strength of the protective measures and to identify appropriate safety improvement plans.

In accordance with this, ENUSA has carried out a deep and systematic review with the following schedule.

In july 2011, ENUSA received a Technical Instruction by the Spanish Regulatory Body, in order to acquire the following commitments:

After having received the technical instruction ENUSA had to give a preliminary report before the 15th august 2011 and a final report with the assessment before the 31st October 2011.

In the case of ENUSA, the assessment only received Spanish Regulatory body on July 2012.

The final report with the comments made by the Regulatory body, was delivered in December 2012 with the commitment of implementing the actions of the final report in the next twelve months and the implemented actions as a result of some complementary comments for the next ten years assessment safety report in June 2015.

2. ENUSA METHODOLOGY

The methodology of the evaluation is based on:

- Evaluation of the response capacity of the facility in case of the occurrence of extreme events.

- Verification of the preventive and mitigative measures, following the standards of "defense in depth" philosophy.

All evaluations assume, under a deterministic approach, the sequential loss of the existing defensive lines and system's safety functions, independently of the likelihood.

So, the following steps are established in the analysis:

(a) Initial situation: Verification of the design basis, safety systems configuration, procedures associated to postulated accidents and license basis compliance, at the time of the analysis.

(b) Robustness analysis: Evaluation of the safety margins for each postulated accident sequence, identification of extreme situations or potential weaknesses, and definition of appropriate countermeasures, according to the defence in depth approach.

(c) Beyond design basis situation: Identification, analysis of the consequences and countermeasure definition of "extreme credible external hazards" that could damage structures, systems and important components for the safety (SSCs).

(d) Emergency preparedness and response arrangements: Evaluation of the structures, systems, components and human factors which is necessary to manage the emergency and mitigate its consequences for each scenario.

The situations considered, that correspond to more and more degraded conditions, were:

- Earthquakes.

- Floodings.

- Extreme natural external hazards: Hurricanes, snow and heavy rain.

In addition, it is considered the sequential loss of the safety functions related with the initial events, in particular the loss of offsite Power that includes Station Black Out.

In the case of Enusa Plant, are out of the scope of the Test, the external aid that could support the Plant during the accidents.

3. RESULTS OF THE STRESS TEST

3.1 Earthquakes.

The facility is located in Juzbado, province of Salamanca, which is situated in the west of Spain. The characterization of the region concludes that the horizontal ground acceleration is equal or minor than 0.07g. In order to increase the safety margins, for the design of the Juzbado Plant, it was considered an earthquake with a peak ground acceleration of 0.15g.

The assessment methodology of the Design Basis Earthquake, as well as the validation of these dates with the time, is based in the following information:

- Geological and seismic region characterization with a return period of 100000 years.
- Historical information about earthquakes with more intensity of the zone.

After the Fukushima accident, Enusa requested an independent evaluation of the Design Basis hypothesis. The new value of peak ground acceleration obtained is 0.10 g, so the value considered in the design of the Juzbado Plant is conservative and the current Design Basis Earthquake is correct.

In case of a severe earthquake, the Manufacturing plant structured and Fuel Bundles handling storage PWR and BWR would remain stable. Although the rest of structures, systems and equipment do not have credibility to maintain their functions, the possible effects over them would be reduced by the maintenance of the Manufacturing plant structured.

Moreover, Enusa has different procedures to respond to an earthquake in the emergency plan and others to check the state of the Plant after the event. However, some necessary equipment to respond to the emergency are located in buildings without earthquake resistant, so there could be no assurance that they will be available in a case of an earthquake.

In parallel with the studies and analysis of the Design Basis hypothesis, there have been made analysis about existing safety margins in the Plant and possible improvements. These analyses have determined the seismic margin of the installation, identifying load combinations and elements that have structural margin minor Also, a new calculation has been made for the critical elements, increasing the earthquake with the same regulation, design criteria and methodology of the License project.

The results obtained are the following:

Table 1

Element	Safety Margin	Corresponding horizontal ground acceleration
Manufacturing plant structured	1.09	0.164g
Fuel Bundles handling storage PWR	1.25	0.190g
Fuel Bundles handling storage BWR	2.09	0.310g
Fire extinguishing pump room	< 1	< 0.150g
Control room	< 1	< 0.150g
Data process room	< 1	< 0.150g

As Table 1 shows, the Fire water pump room, the Control room and the Data process room don’t resist the Design Basis Earthquake. In addition, it have been identified the following “cliff edge” situations during the study:

- Pendulum effect in the Fuel BWR Handling Storage.
- The possibility of apparition of water inside of the Manufacturing building, by leaks of the water supply pipes that conduct water inside of Uranium drums Storage.

According to these results, several actions have been carried ou:

- Implementation of a new water supply system for fire extinguishing, capable of operating after an earthquake.
- Emergency Room Extension with seismic resistance > 0.17 g.
- Pillars reinforcement in the Data process centre.
- Design an implementation of handling clips to avoid Pendulum effect in the Fuel BWR Handling Storage.
- Changing water supply pipes in other areas.
- Installation of an Accelerograph in the Plant.

For the point “beyond design basis situations”, it is assumed an earthquake with a higher acceleration than 0.164g, which involves the partial collapse of the Plant. it is also assumed a critical situation or a fire at the same time of the earthquake.

In both situations the public dose in the limit of the Security Area is lower 5 mSv, which is the established limit in the accidents analysis of the Plant and therefore, these situations are covered by the current systems and procedures.

In accordance with the foregoing, no limit situations and no potential weaknesses have been detected from a radiological point of view, when an earthquake excess the Design Basis and there are other extreme situations at the same time.

From a no radiological point of view, in case that an earthquake excess the Design Basis and a fire happens at the same time, the elements which must keep the safety of the Installation are:

- Fire extinguishing Tanks.
- Fire extinguishing pump.
- Water supply system for fire extinguishing.

Therefore, the new water supply system and the mobile equipment for fire extinguishing must be able to operate after an earthquake of 0.195g.

In addition to the assumptions related with The Stress Tests, a particular study about the effects of a fire inside the Manufacturing building has been made with independence of the initial event.

As a result of the weaknesses identified in this study, the following actions have been carried out:

- Rearrange H_2 supply pipelining outside the plant.
- Replace the H_2 supply pipelining with seismic resistant.
- Place the Components warehouse outside the plant.

3.2 Flooding.

In the Design Basis of the Plant are considered the following situations:

- Maximum water level River return.
- Maximum water level by the collapse of the nearby dam.

In the first situation, it is calculated the maximum level of water for returns period of 500 and 1000 years and taking into account the minimal drainage section. Thus, the maximum level of water in the worst case is 760.10 meters, 30 meters less than Enusa Plant level.

On the other hand, for the second case it is assumed that the dam is to the maximum level and breaks the top third of it. Also, it is assumed that the duration of the accident is equal to 3600 seconds so the water flow increases from 1000 m^3/s to 35000 m^3/s, reaching to 46000 m^3/s in 4500 seconds.

With these hypothesis, the water flow reached in the placed of the Plant is less than 38000 m^3/s required to reach the level of the Installation (790 meters).

The assessment methodology of the Design Basis, as well as the validation of these dates with the time, are based in the following information:

- Return periods of 500 and 1000 years.
- Historical information about events in the zone.

Regarding the Design Basis of a flooding, the Official Documents do not set up any structure, system or special component to achieve and maintain the safety conditions. In the Design Basis of a flooding, the water level does not reach the level of the Factory at any moment.

Therefore, after Fukushima accident Enusa applied for an independent evaluation of the Design Basis hypothesis. The conclusions of these studies reflected the validity of the data and the hypothesis of the Design Basis of a flooding. The studies also concluded that it is not necessary any structure, system or special component to achieve and maintain the safety conditions.

Moreover, it is not necessary any special measures to mitigate the effects of the maximum flooding the Manufacturing plant.

Apart from these studies, there have been made several analysis about existing safety margins in the Plant and possible improvements in the case of a flooding. These analyses represent a possible flooding with consequences of the maximum water level return and the collapse of the two nearby dams. The maximum water level reach is 765.30 meters, which it implies a safety margin of 24.70 meters.

In this case, there have not been identified any “cliff edge” situations and any situation beyond Design Basis, because it represents the occurrence of all the events mentioned.

3.3 Extreme natural external hazards: Hurricanes, snow and heavy rain.

The Manufacturing building has been designed to resist winds with a dynamic pressure equal to 87.5 kg/m^2 as well as an overload by snow equal to 80 kg/m^2. For both parameters, it was taken as reference the Spanish construction code applicable at the time of building the Plant.

In the case of overload by heavy rain, the regulation did not consider this parameter explicitly, so it was considered as reference the value of maximum precipitation in the zone (187 l/h).

As in the other events, Enusa requested an independent evaluation of these design criterions. The conclusions of this evaluation reflected that the definition and values of these parameters have not been changed over time.

According to the extreme natural external hazards, the only structures, systems and necessary components to reach the safety condition are the structure and the closing elements of the Manufacturing building. Therefore, the only requirement of frontal protection to extreme external conditions is maintaining the integrity of structures and the closing elements.

Moreover, in the case of hurricanes there has been identified indirect effects as damages in the chimneys of the ventilation system, but the cut of the ventilation system also causes the blackout of equipment and therefore the powder emissions. Another risk of these events is the possible presence of water in areas where the uranium is not encapsulated, with the consequent criticality risk. However, the drain system, the cover sealing and the individual protection of the units with uranium oxide minimize the likelihood of presence of water inside of these. In summary, these events do not produce leaks greater than the accidents produced in the manufacturing process or auxiliary systems of the Plant.

Therefore, it is not necessary any special measures to mitigate the effects of these events.

In order to maintain safety margins, the Design parameter is compared with the values of the references.

The results obtained are the following:

Table 2

Event	Design parameters	Reference parameter	Safety Margin
Hurricanes/winds	87.5 kg/m^2	75 kg/m^2	1.17
Snow	80 kg/m^2	72 kg/m^2	1.11
Heavy rains	200 l/h	187 l/h	1.07

As Table 2 shows, all the safety margins for these events are higher than one, this verifies the conservatism of the design basis. In addition, according with independent studies there have not been identified “cliff edge” situations and beyond Design Basis situations, so it is not necessary any special measures.

3.4 Loss of Electrical Power supply system.

The power supply system of the Plant is composed of the following elements:

- Normal electrical power supply system: it comes from the external electricity grid and provides services to the manufacturing equipment, ventilation system, special working fluids system and normal lighting system. Furthermore, it provides support to the rest of safety system and the emergency lighting system.

- Emergency electrical power supply system: it is composed by two diesel generators and batteries groups that supply energy to the most important detection and alarm systems.

The power supply system is a necessary system for manufacturing operations and provides support to systems that control the safety of the Installations, but in case of failure, this is not an accident situation.

The lack of electrical power supply would lead in a shutdown of the Plant until achieving a safety conditions, that once achieved, the power supply system is not necessary to maintain the safety condition in the Plant.

Thus, it has been analyzed the impact of the loss of electrical power supply, including station black out.

Loss of external electrical power supply (LOOP)

The lack of the electrical power supply does not suppose any risk, because this situation leads the safe shutdown of the production equipment and the ventilation system, at the same time. More, the valves of the special working fluids system, change hydrogen to nitrogen. Therefore, the failure of the electrical power system does not lead to an accident situation.

On the other hand, Emergency electrical power supply guarantee the functions of the main safety systems of the Plant for a period of not less than eight hours.

Station black out (SBO)

In addition to the diesel generators, there is an emergency electrical power supply system by batteries, that provides power to certain systems as Fire Protection System, Criticality Alarm System, etc. These batteries provide a minimal autonomy of two hours.

Although the loss of electrical power supply in the Plant does not suppose an accident situation, some measures have been taken with the aim of increasing its robustness. These actions are the following:

- Connecting the tanks of both diesel generators to another tank with a capacity of 5000 liters.
- Separating the filling lines of the diesel generators tanks.
- Auxiliary pump with flexible connections for filling the diesel tanks.
- Re-distribution of charges to maximize the capacity of the Diesel Generators.

4. CONCLUSIONS

The main conclusions of this analysis are listed below:

1. Enusa Plant meet all the Design Basis Requirements, no finds have been discovered during the performance of the Stress Test.
2. Only in the case of an earthquake, there have been identified “cliff edge” situations and situations beyond Design Basis.
3. Several improvements related with the different events have been taken to increase the level of Defense in Depth:
    - Implementation of a new water supply system for fire extinguishing, capable of operating after an earthquake.
    - Emergency Room Extension with seismic resistance > 0.17 g.
    - Pillars reinforcement in the Data process centre.
    - Design and implementation of handling clips to avoid Pendulum effect in the Fuel BWR Handling Storage. Change water supply pipes to other areas.

- Installation of an Accelerograph in the Plant.
- Changing water supply pipes in other areas.
- Rearrange H_2 supply pipelining to outside the plant.
- Replacement the H_2 supply pipelining with seismic resistant.
- Place the Components warehouse outside the plant.
- Connecting the tanks of both diesel generators to another tank with a capacity of 5000 liters.
- Separating the filling lines of the diesel generators tanks.
- Auxiliary pump with flexible connections for filling the diesel tanks.
- Re-distribution of charges to maximize the capacity of the Diesel Generators.

Feedback of complementary safety assessments for French fuel cycle facilities and research laboratories and reactors

Session 1

Emeline Cluzel and Michel Guillard, IRSN (Institute for Radiological Protection and Nuclear Safety)

Abstract:

The Complementary Safety Assessments (CSAs), requested in France by the Prime Minister in spring 2011 after the Fukushima-Daiichi accident have been performed by the nuclear licensees in 2011. The facilities have been divided into three categories, depending on their vulnerability to accidents like those at Fukushima and on the importance and the scale of the consequences of any accident affecting them. AREVA nuclear fuel facilities and some CEA research reactors and facilities were part of the top priority facilities. The stresses that have to be considered for the CSAs are natural hazards (earthquake and flooding in particular) higher than what is required in the current design standards and the deterministic loss of power supply and cooling functions. The outcome of the studies carried out by the fuel cycle facility licensees is that these facilities ensure a sufficient safety level. However, the IRSN analysis showed that it is necessary to implement a "*hardened safety core*" (HSC) of robust material and organizational measures aiming, for extreme situations, at:

- preventing a severe accident or limit its progression,
- limiting massive discharges resulting from a non-controlled accident,
- enabling the licensee to fulfil its crisis management duties.

The HSC proposed by the fuel cycle facility licensees have been assessed by IRSN.
Even if complements are necessary, regarding the HSC provisions and associated requirements, IRSN has estimated that the corresponding equipment and measures are about to enhance the ability of the facilities to withstand extreme hazards or supply losses.

Keywords:

Fukushima-Daiichi accident – French Fuel Cycle Facilities – French research reactors and laboratories - Complementary Safety Assessment - Extreme events – Dreaded situations - Key structures, systems and components - Hardened Safety Core

1. Introduction

Following the accident that occurred on the Fukushima Daiichi Nuclear Power Plants (NPPs) on March 11, 2011, numerous actions were undertaken throughout the world to check the robustness of the nuclear facilities and of the existing organizations to face extreme situations that were not taken into account at the design stage of these facilities. In France, the Prime Minister asked the French Nuclear Safety Authority (ASN) to carry out an audit of the safety of all French nuclear facilities, concerning five points: risks of flood, earthquake, loss of power supplies and loss of cooling systems, as well as the operational management of the accidental situations. On May 5, 2011, ASN issued twelve decisions requiring the French nuclear licensees to perform Complementary Safety Assessments (CSAs) of their facilities, based on the specifications attached to the aforementioned decisions and consistent with the decisions for the stress tests requested by the European Council for NPPs.

With the aim of taking into account the first lessons learned from the events that hit the Fukushima Daiichi NPPs, the CSAs reports evaluated the capacity of French nuclear facilities to withstand extreme situations beyond design basis assumptions[27]. ASN fixed the deadlines of delivery of these files according to the safety stakes presented by the nuclear facilities: indeed, three categories ("Batch 1", "Batch 2" and "Batch 3") of facilities were defined according to a decreasing priority depending on their vulnerability to Fukushima type events and on the importance and the scale of the consequences of any accident affecting them. In addition to the French NPPs, AREVA nuclear fuel facilities were part of the top priority facilities, as well as some research reactors and facilities operated by the CEA. The corresponding deadlines for "Batch 1" and "Batch 2" facilities were respectively September 2011 and September 2012. Regarding "Batch 3" facilities, considered as the lowest priority, ASN asked the concerned operators to transmit the CSAs reports on the occasion of any administrative procedure involving a public inquiry or on the occasion of a facility licensing or, at the latest, within the framework of the ten-year safety reassessment report. **Considering the preceding elements, the information presented below primarily concerns the facilities of "Batch 1".** In addition, some elements are also given about "Batch 2" facilities.

French secret nuclear facilities, which are not mentioned in the present article, were subject to a similar approach.

IRSN highlights the following main "key principles", structuring the French CSA approach:

- **All of the nuclear reactors and nuclear facilities are concerned by the CSA approach ("Batch 1", "Batch 2" or "Batch 3" facilities);**

[27] The oldest buildings were generally not designed to withstand a major earthquake. In such cases, the operators have to implement, apart from the CSA approach, a shutdown strategy for the oldest facilities.

- **Remediation means, mitigation means and crisis management means have to enable the operator on each site to be in total autonomy during the first 48 hours after a dreaded situation, whatever the site conditions, without any complementary human or material means brought from outside the site;**

- **Each nuclear site has to be considered as a whole, including all the facilities located on the site (with potential multiple accident situations and aggravating phenomena to deal with simultaneously) and the industrial environment;**

- **"Dreaded situations" have to be identified on each nuclear site, i.e. situations resulting from an extreme natural event or from the deterministic loss of electrical powers and cooling functions which lead to a cliff-edge effect.**

These key principles are detailed in the next sections of the present article.

2. Assessment of the "Batch 1" facilities

2.1. Context

In 2011, the CSAs evaluations included the nuclear power reactors in operation (900, 1,300 and 1,450 MWe PWRs) or under construction (EPR), as well as some nuclear facilities, called facilities of "Batch 1", considered by ASN to be the top priority.

Among these facilities are included the following French fuel cycle facilities (FCFs) operated by AREVA and by the CEA[28]:

AREVA FCFs:

- on the Tricastin site, the uranium conversion, enrichment and treatment facilities,
- on the Romans-sur-Isère site, the uranium fuel manufacturing facility (FBFC plant),

[28] The AREVA and CEA sites housing the French fuel cycle facilities have very varied characteristics, given their geographical position (Channel seaside for La Hague, Paris basin for Saclay, Alps mountain area for Romans-sur-Isère, Rhône river valley for Tricastin and Marcoule, Provence limestone plateau for Cadarache), their different ground and climate specificities, as well as their industrial and natural environment.

- on the La Hague site, the spent fuel reprocessing facilities and facilities under decommissioning or legacy facilities,
- on the Marcoule site, the MOX fuel manufacturing facility (MELOX plant).

CEA FCFs:

- on the Cadarache site, the Jules Horowitz research reactor (RJH), the MASURCA critical model facility and the former MOX fuel manufacturing facility (ATPu facility),
- on the Marcoule site, the PHENIX sodium metal-cooled fast breeder reactor,
- on the Saclay site, the OSIRIS research reactor.

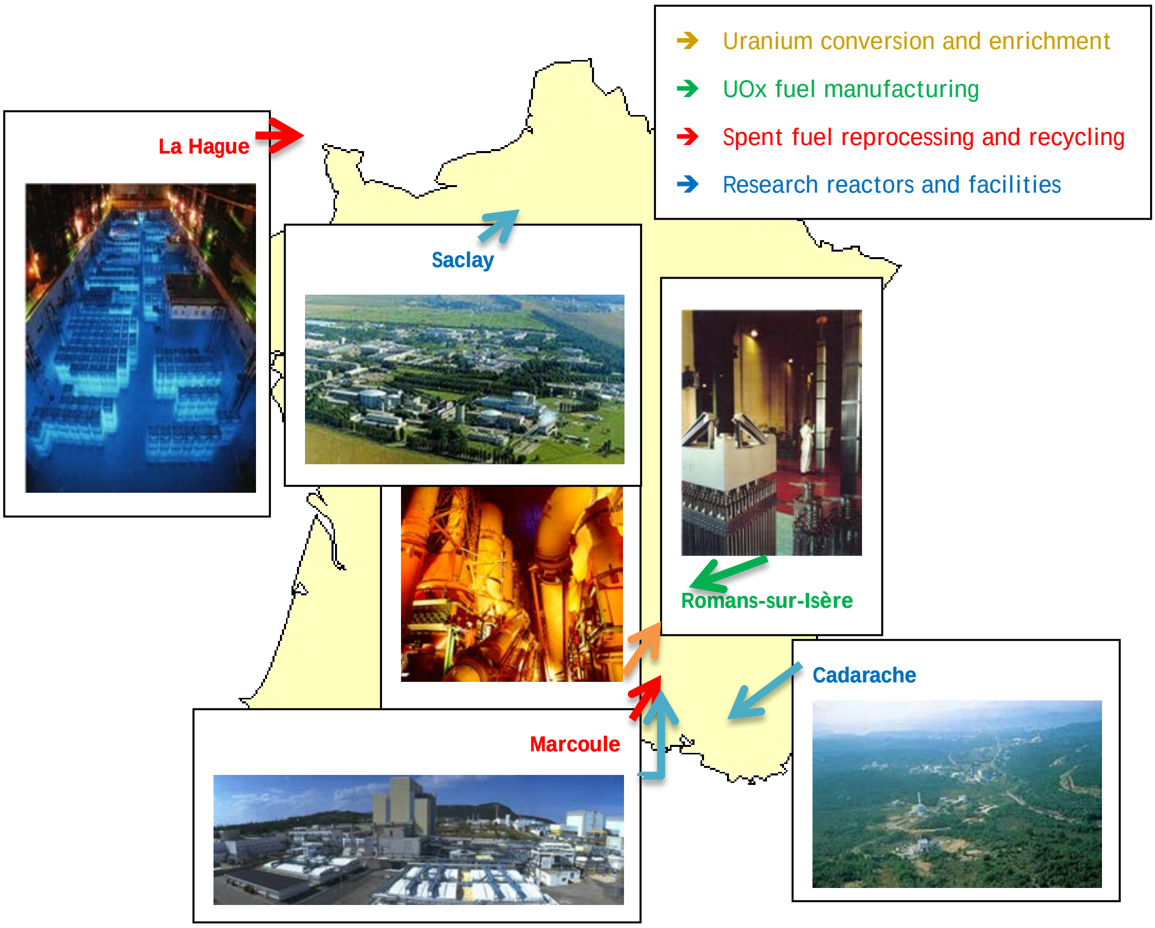


Fig. 1: AREVA and CEA sites

ASN asked the French Nuclear Safety Advisory Committees for reactors, for laboratories and plants to submit their opinion in July 2011 on the methodology proposed by each licensee to evaluate the robustness of the facilities to extreme situations (earthquake, flooding and other extreme natural phenomena) and the consequences of a loss of power supply and of a loss of cooling systems. Additionally, the licensees had to present the organizational and material measures implemented to manage a severe accident that could affect several facilities on a given site simultaneously and the subcontractor monitoring. This methodology was considered satisfactory. The licensees submitted their CSAs reports to ASN on September 15, 2011.

2.2. AREVA and CEA CSAs reports

2.2.1.Extreme situations taken into account

The CSA approach considers earthquake and external flooding as extreme natural hazards, with a higher level than the stress levels taken into account in the current standard for design. In this way, in order to assess the robustness of their facilities, the operators estimated the maximal earthquake and the highest flooding level the facilities are liable to withstand. The CSA reports did not study extreme events concerning other phenomena related to weather (snow, wind, effects induced by a tornado…) nor the impact of aggressions of external origin (due to industrial environment and transportation lines) or of internal origin, liable to be induced by an earthquake or by a flooding (fire, explosion, criticality accident…).

In addition, in the CSA approach, the deterministic loss of cooling and electrical supplies has to be postulated, including their combination, whatever the existing redundancies.

2.2.2.Dreaded situations

Extreme situations have been identified by the operators: these dreaded situations are the transposition of the notion of “severe accident” used for NPPs. They correspond to a degraded state of the facilities which has to be avoided, with liberation of a “potential of hazard”. Given the number of dreaded situations which are possible, a classification was defined by the operators, according to the kinetics of liberation of the “potential of hazard”. The operator analysis was based on the amount of radioactive material liable to be released during accident situations “beyond design”, on the kinetics of the dreaded situation (less than 48 hours in general) and on the level of robustness of the concerned facilities.

The main dreaded situations identified by the operator for the AREVA sites of La Hague, MELOX, Romans and Tricastin as well as for the CEA sites of Cadarache, Marcoule and Saclay are summarised as follows.

For the La Hague AREVA FCFs:

- Loss of cooling of radioactive materials, leading to, due to thermal release:
  - a fuel assembly dewatering in spent fuel pools,
  - a boiling of concentrated fission products solutions,
  - a degradation of concrete pits in the plutonium dioxide container storages,
- Loss of declogging function of the centrifuge settlers used to filter the dissolving solution (release of gaseous ruthenium oxide);
- Loss dilution of radiolysis hydrogen in fine solution tanks and alkali rinsing tanks (air circulation loss), leading to an explosive atmosphere;
- Containment loss of radioactive materials stored in nuclear waste storage silos.

For the other AREVA sites:

- In the MELOX plant, loss or degradation of the dynamic and static containment systems in the facilities processing plutonium dioxide (PuO_2) powder and loss of cooling in the main fuel rod storage (leading to a criticality accident resulting from geometry degradations);
- On the Romans-sur-Isère and Tricastin sites, massive releases of liquid and gaseous uranium hexafluoride (UF_6) and of concentrated or anhydrous hydrogen fluoride (HF) in the environment;
- On the Romans-sur-Isère site, criticality accident in a process facility located at the limit of the site.

For the CEA sites:

- the total loss of electric supply,
- the total loss of cooling systems or failure of heat sink,
- the situations resulting from the conditions of the facilities, site and environment after an external aggression.

2.2.3. Systems, Structures and Components

The licensees have identified the associated Systems, Structures and Components (SSCs) necessary to ensure important safety functions and thus to maintain the facility in a safe state: They are called “key” SSCs. The concerned safety functions are, for example, civil engineering structure stability, containment barrier integrity, radioactive material cooling, radiolysis hydrogen dilution, fire and explosion hazard control…

The “key” SSCs defined by the operators were, either “structural” (civil engineering elements), or “functional” (generally active equipment ensuring specific functions, e.g. detectors, measurement systems, valves, pumps, ventilation systems, backup diesel generators…).

2.3. Results of the IRSN assessment for the CSAs reports

2.3.1.General outcomes

In order to confront the robustness with these exceptional - but nonetheless conceivable - scenarios, the IRSN analysis confirmed that the licensees had to make sure, firstly, of the conformity of their facilities with the safety requirements which are applicable to them. The margins identified in the design are based on the conformity of the "key" SSCs. The conformity review consists in comparing the facility with its design in order to ensure that the changes of the facility and its operation, as a result of the modifications or ageing, comply with applicable regulation and do not compromise the facility safety requirements. A ten year conformity check is performed in the context of the periodic safety reviews but it does not relieve the licensee of its permanent obligation to guarantee the conformity. In this respect, all the licensees have undertaken, in the framework of the CSAs, a review in order to confirm that there is no compliance gaps in the key SSCs contributing to the management of the accident situations considered in the CSA.

Beyond the design basis earthquakes, the licensees presented simplified calculation methods and test results as well as methodologies to evaluate the overall margins for "key" SSCs notably on the basis of currently available data in the field of seismology and on an "expert's opinion". Concerning seismic hazards, IRSN noticed that seismic knowledge is rapidly increasing and has considered that area of improvement can be identified in the explicit integration of uncertainties in the calculation of seismic hazards.

As regards flooding hazards, IRSN identified some measures to be implemented in order to improve the robustness of the facility beyond the design basis.

As regards to the facility robustness, the simplified methods do not allow to consider the values of global margin factors described by licensees as reliable, and IRSN considered necessary that the licensees provide additional checks to support the results.

Furthermore, the IRSN analysis showed the interest to make the safety of nuclear facilities more robust to unlikely risks, significantly higher than those currently included in the initial design of the facilities or included in their periodic safety review. In this sense, significant provisions to improve the safety of French nuclear facilities were proposed. **In particular, IRSN proposed the implementation of a "*hardened safety core*" (HSC), composed of a limited number of "key" SSCs strengthened with high design**

requirements, which aimed at:

- **preventing a severe accident or limiting its progression;**
- **preventing large-scale releases in the event of an uncontrolled accident;**
- **enabling the licensee to perform its emergency management duties.** The emergency organization had to be enhanced to take into account the above mentioned extreme events (see § 2.2.1) for all the facilities of the site.

2.3.2.Specific outcomes for the AREVA and CEA nuclear sites

Even if the behaviour of the AREVA and CEA facilities raises some concerns, in particular with regards to the number of configurations studied and the identification – based mostly on an expert’s opinion – of different levels of hazards leading to the loss of safety functions, the AREVA and CEA licensees proposed structural and organisational improvements of sites, which should, over time, increase the robustness of the facilities and of the emergency management resources. Consequently, IRSN considered satisfactory the AREVA and CEA methodologies, focused on the analysis of dreaded situations leading to significant consequences in the short term, which constituted a first stage in the consideration of the experience feedback from the Fukushima accident.

However, a broader analysis taking into account the recommendations summarised below seemed necessary to IRSN for the support of studies on emergency management that had to be performed mainly in 2012, which constituted a second stage in the Fukushima accident operating feedback.

2.3.3.Crisis management

As regards the crisis management, IRSN verified more particularly the capacity of the last level of the in-depth defence to withstand extreme events, including accidents on several facilities on the same site. To be considered as "strong", the organization and the means of crisis have to remain operational for levels of stresses much higher to those taken into account at the design of the installations. Besides, IRSN assessed that these means must be endowed with a high capacity of adaptation towards situations which would not have been envisaged until now. For example, such an assessment includes the presence of an intervention team equipped with “professional means”, a “highly resistant” crisis management building, measures enabling a reinforcement of the organisation and means of crisis management by external means.

At the stage of these first CSA reports, IRSN had not all the elements to estimate the "robustness" of the organisation and the means of crisis of the operators. During the instruction, IRSN attempted to verify that the operators had clearly identified the axes of improvement as well as the necessary action plans to have in the short-term a “strong” organization and “strong” means of crisis.

Within the specific frame of the CSAs, three themes connected to the organizational and human factors have been developed:

- the organizational and human measures foreseen during accidental situations,
- the organizational processes dedicated to the preservation of the conformity of the facilities relative to their design basis,
- the impact of the recourse to subcontractors on safety and radiation protection.

As regards the human interventions in accidental situations, the CSAs reports transmitted by the licensees did not bring much information.

2.4. ASN position on CSAs reports

Based on the IRSN review, the French Nuclear Safety Advisory Committees met on November 8, 9 and 10, 2011. On November 17, 2011, during a conference jointly organized by ASN and IRSN, IRSN presented its analysis and conclusions to the press and made public its CSAs report assessment. Then, on January 3, 2012, the ASN report on the CSAs was submitted to the Prime Minister and made public. Based on the advice of the Advisory Committees, ASN has considered that the facilities which have been reviewed present a sufficient level of safety requiring no immediate shutdown. At the same time, ASN considered that the continued operation of the facilities required that their robustness to extreme situations shall be improved as rapidly as possible.

ASN prescribed both to AREVA and to CEA, through decisions emitted in June 2012, especially “*the implementation of a “hardened safety core” of strong material and organizational measures aiming, for the extreme situations studied within the framework of the CSA, in:*

- *preventing a serious accident or limit its progress,*
- *limiting the massive discharges in a scenario of accident which would not have been able to be controlled,*
- *enabling the licensee to fulfil its crisis management duties .”*

In addition, depending on the sites, the operators had to consider, in order to define their HSC, a list of extreme events and the aggression effects induced by extreme events (linked to the industrial environment, transportation lines, site internal aggressions, criticality accident…). ASN also asked AREVA and CEA to present “*the applicable requirements to this “hardened safety core”*”.

The ASN position on CSAs reports will be detailed in Session 2.

3. “*Hardened safety core*” reports

The AREVA and CEA licensees transmitted, in the second half-year 2012, “*hardened safety core*” reports, answering to the ASN decisions.

3.1. “Hardened safety cores” identified

AREVA and CEA have defined, for each site, the dreaded situations and considered different possibilities for the HSC based on prevention or mitigation of consequences, depending on the dreaded situation. The HSC can be constituted by new SSCs but also by existing ones, since it is demonstrated that they are able to withstand the considered extreme aggressions. The corresponding types of SSCs which compose the HSC (HSC-SSCs) or interfaced with the HSC (HSC-INT) may be illustrated with the following examples:

- HSC-SSCs may be fixed or mobile equipment, internal or external to buildings. Examples: water supplying systems (tanks, pipes, distribution valve chests, valves, pumps, power supply…), cooling coils, measurement systems, ventilation systems (filters, pipes, ventilators, power supply…), isolation valves, backup diesel generators and diesel oil reserve, lifting devices, nitrogen and air pressurised reserves, air compressors, wall penetrations and connecting boxes (for water or electric supply), seismic valves on water or gaseous networks, seismic detection and cut-off systems, recovery pits of chemical products;
- HSC-INT are mainly civil works of buildings or premises housing HSC-SSCs, support structures of equipment (with their anchor points) potentially missile in premises dedicated to remediation or mitigation actions, neutron screens, last static containment barrier (walls at the building limits, doors, filters, valves…), containers, drip-trays under chemical product tanks.

3.2. IRSN assessment for the “hardened safety cores”

IRSN assessed the elements presented in these reports concerning:
- the proposed HSC of the "Batch 1" facilities;
- the requirements associated with the HSC and the justification retained by the licensees to demonstrate the operational character of its HSC in the conditions which they are brought to operate.

3.2.1.<u>Extreme natural hazards to consider</u>

As regards extreme earthquakes, the licensees proposed standard spectra, which enclose the spectra of earthquakes taken into account at the facility design including a 50% acceleration increase. These spectra aimed at covering extreme seismic scenarios not taken into account in the safety case.

IRSN considered that the levels of hazard proposed by the licensees need to be justified, especially regarding the standard envelope spectra choice and quantitative elements enabling the assessment of the increase in the envelope spectra compared to extreme earthquakes. According to IRSN, complementary justifications, based on supported data and scientific elements, are necessary to consolidate technically the levels of earthquake. In addition, the licensees had to check the impact of the site effects and of the presence of active seismic fault lines in the vicinity of the facilities.

As regards to extreme floods, IRSN considered that the extreme hazard levels proposed by the licensees are broadly satisfactory. Additional justifications seem nevertheless necessary for the AREVA Tricastin site, as regards the resistance of the Donzère-Mondragon channel dikes to an extreme earthquake, a hypothesis held for the definition of the level of the "flood" hazard for this site.

As regards other extreme meteorological phenomena, IRSN considered that the licensees had to complete their consideration of natural phenomena which could lead to dreaded situations, such as tornado, extreme winds, lightning, extreme temperature, hail, snow …

3.2.2.<u>Outcomes for "hardened safety cores" and related safety requirements</u>

Unlike the nuclear power reactors in France which are of similar design, the "Batch 1" facilities of AREVA and CEA have different behaviours in case of extreme hazards, of power supply loss or of cooling system loss. Therefore, an individual analysis was imperative.

At the end of its analysis related to the various facilities of the "Batch 1", IRSN considered that the HSCs proposed by AREVA and CEA have to be punctually completed or strengthened. As examples, IRSN considered that some measures must be proposed to deal with:

- losses of watertightness of the spent fuel storage ponds of the La Hague site after an extreme earthquake, as far as the presented elements do not enable to rule out such losses;
- fires arising in several places of a facility following an extreme situation (earthquake, flood…); such scenarios cannot be totally eliminated;
- discharges of radioactive material in the environment induced by an extreme natural aggression.

Finally, IRSN insisted on the fact that the SSCs of the HSC in the facilities will have to comply with requirements in terms of methods and criteria for design, manufacture, maintenance and operation. According to the IRSN review, these requirements are essential to justify the high reliability of the HSCs which will have to be able to ensure that the facility remains in an acceptable safe condition in case of extreme situations.

Besides, within the framework of the strengthening of the crisis management means, it should be noted that AREVA has planned to build on-site new centres of crisis management and to deploy a “National Task Force” (FINA)[29]. Likewise, CEA has planned to build an on-site new centre of crisis management on the Cadarache site and to deploy a “National Rapid Response Force” (FARN). This strengthening aims at ensuring the implementation of on-site fast intervention means in case of extreme events.

3.3. ASN position on “hardened safety cores”

Based on the IRSN assessment, the Advisory Committee members considered, in April 2013, that even if some complements are necessary, the proposals of "*hardened safety cores*" held by AREVA and CEA for the "Batch 1" facilities will increase the level of long term resistance of these facilities in case of extreme natural aggression or in case of loss of support functions. This improvement was estimated significant, as it involved a better consideration of extreme events with a suitable dimensioning of the “key” SSCs which constitute the “*hardened safety core*”, either at design (for new equipment) or after strengthening (for existing equipment).

ASN prescribed both to AREVA and to CEA, through decisions emitted in January 2015, complementary prescriptions related to the “*Hardened Safety Core*” and to the management of emergency management, applicable to each of the “Batch 1” facilities. In particular, the above mentioned decisions cover the following items:

- The external natural hazards to take into account,
- The definition of a “*hardened safety core*” for each facility and site,

[29]AREVA FINA and CEA FARN: Within the framework of the post-Fukushima experience feedback, AREVA and CEA have decided the principle of a National Task Force designed to face a major accident within their sites or facilities. The FINA or FARN mission is to help any site which would be damaged by supplying it, within 48 hours, with human beings and material means (compressor of air, lighting, pump, generator, etc.). These means are complementary to the means already present on the site, aiming at managing the crisis and limiting the consequences of the accident, in particular in terms of discharges of chemical or radioactive substances in the environment.

- The list of the dreaded situations to deal with,
- The requirements for the design of the new SSCs of the "*hardened safety core*" and for the analysis of the behaviour of the existing SSCs of the "*hardened safety core*",
- The additional studies to carry out, depending on the facilities and sites,
- The management of extreme emergency situations and the human and organisational factors.

The ASN position on HSCs reports will be detailed in Session 2.

4. Assessment of the "Batch 2" facilities

4.1. Context

The CSAs evaluations for the facilities of "Batch 2", considered by ASN to have lower priority, were sent by the licensees In September 2012. They concerned nuclear power reactors decommissioned or under dismantling, and other facilities, in particular the following facilities and nuclear sites operated by CEA:
- nuclear material storage facilities, spent fuel storage facilities and a research laboratory (LECA) located on the Cadarache site,
- a research laboratory (ATALANTE) located on the Marcoule site,
- a research reactor (ORPHEE) located on the Saclay site,
- the common support functions and means on each site of Cadarache and Marcoule.

4.2. CEA CSAs reports

The CSA approach for the "Batch 2" facilities is the same as for the "Batch 1" facilities. The extreme situations considered are earthquakes, flooding and other natural events, with a higher level than the stress levels taken into account for design. The deterministic loss of cooling and electrical supply is postulated, including their combination.

The main dreaded situations identified by CEA for the sites of Cadarache, Marcoule and Saclay are summarised as follows:
- a fire in the LECA research laboratory cells caused by an earthquake, leading to radioactive discharges,
- a degradation of the containment barriers in the ATALANTE research laboratory, leading to a dissemination of radioactive material (aerosols or liquids) in the environment,
- the complete loss of the core cooling systems in the CABRI and ORPHEE research reactors.

The licensees have identified the associated “key” SSCs necessary to ensure important safety functions (civil engineering structure stability, containment barrier integrity, radioactive material cooling, fire and explosion hazard control…). As for the “Batch 1” facilities, the “key” SSCs defined by the operators for the “Batch 2” facilities were, either “structural”, or “functional”. In addition, the licensees had to identify, for the Cadarache and Marcoule sites, the “key” SSCs aiming at maintaining the support functions of the sites if an extreme event occurred, and had to assess their robustness as well as the associated infrastructures.

4.3. Results of the IRSN assessment for the CSAs reports

Given the heterogeneity of the facilities assessed in terms of nature (experimental reactors, experimental laboratories, storage facilities…) and of life phase (operation, shutdown…), IRSN assessment was based on an approach proportional to the safety stakes. The IRSN objective was to identify the facilities for which it was worthwhile going further in the CSA approach.

Regarding the storage facilities, even if the CSAs do not highlight any dreaded situations, IRSN has estimated that the containment is not guaranteed if an extreme event occurs and therefore, that the removal of spent fuels and nuclear waste from these facilities has to be a high priority for CEA. In addition, concerning the RAPSODIE storage facility, IRSN has estimated that the operator has to complete the study of a reaction between sodium and water induced by rains following a major earthquake as well as the corresponding radiological and chemical consequences.

Regarding the LECA research laboratory, the operator has planned the implementation of a seismic detection and cut-off system in order to disconnect the facility power supply, which was estimated satisfactory by IRSN, as it was the main ignition source identified. Likewise, IRSN has estimated satisfactory the robustness of SSCs constituting the first containment barrier in the ATALANTE research laboratory, subject to the verification of conformity for the high activity liquid effluent storage tanks.

Regarding the research reactors CABRI and ORPHEE, the CSAs have been estimated satisfactory by IRSN, subject to the verification, by the operator, of robustness of some SSCs included in the HSC. In addition, IRSN has estimated satisfactory, the implementation in ORPHEE of a seismic detection and cut-off system in order to disconnect the facility power supply.

Regarding the sites of Cadarache and Marcoule, the operators have planned complementary measures aiming at strengthening the existing organisation and means and at justifying the adequacy of the measures defined for the HSC. In addition, IRSN has estimated that the CSAs have to be completed by taking into account the risks induced by site internal transportation and, for the Marcoule site, by completing the study of a fire caused by a major earthquake in the magnesium waste storage pits and in the storages of asphalted drums.

Regarding crisis management, the operators have planned adaptations of their crisis organisation in order to manage the consequences of an extreme event, in considering the site as a whole (size of the site, multiple facilities, industrial environment…) and over time (at least, during 48 hours after an extreme event). IRSN has estimated that the operators have to complete their action plans, especially concerning the availability of the crisis teams and the emergency response teams, the measures and means necessary for the diagnostic and the follow-up of accident situations in any case of extreme event as well as the acquisition and transmission means of weather data and radiological data in the environment.

4.4. ASN position on CSAs reports

Based on the IRSN assessment, the Advisory Committee members considered, in July 2013, that the CSAs reports sent by CEA for its "Batch 2" facilities enable to identify the main elements taking part to the robustness of nuclear facilities and sites regarding extreme situations to be considered according to the ASN specifications and to define priorities in terms of necessary modifications or enhancement.

Based on the advice of the Advisory Committees, ASN prescribed both to CEA, through the decisions emitted in January 2015, complementary prescriptions related to the "*Hardened Safety Core*" and to the management of emergency management, applicable to each of the "Batch 2" facilities and to the Cadarache and Marcoule sites (see § 3.3).

5. Progress on the implementation of the "*hardened safety core*"

5.1. Studies about extreme natural hazards

5.1.1. Seismic event

In its decisions sent in January 2015, ASN specifies the seismic event to take into account for the HSCs, defined by a response spectrum which has to:

- enclose the Safe Shutdown Earthquake (SSE) increased by one and a half times,
- enclose the probabilistic spectra defined with a return period of 20,000 years,
- include the amplification effects caused by the local geology ("site effects"),

- take into account the potentially active seismic fault lines in the vicinity of the site.

The studies sent by AREVA and CEA to define the extreme seismic events for their sites have been assessed by IRSN. The main resulting points were the followings:

- the seismic event defined for the La Hague site meets the ASN requirements,
- the studies sent for the Tricastin and MELOX sites had to be completed to take into account “site effects”,
- the response spectra defined for the Romans-sur-Isère and MELOX sites had to be checked given the probabilistic spectra which may not answer to the ASN requirement about a return period of 20,000 years,
- the studies sent for the Cadarache and Marcoule sites had to be completed to take into account “site effects”.

The investigations of IRSN on the resulting complementary studies are ongoing. Pending the definition of the corresponding earthquake levels, IRSN considered that AREVA had to design the civil works for the HSC buildings to be constructed in the short terms with substantial margins, in order to guarantee their stability under an extreme earthquake. This point is detailed in session 3.

5.1.2.<u>Other events</u>

The tornado phenomena, which were not considered in all the safety case of the French nuclear facilities before the Fukushima-Daiichi accident, are subject of a prescription in the ASN decisions. The licensees have to justify the elements adopted to take into account tornados. The corresponding files are currently assessed by IRSN. In this regard, IRSN had to define a new standard for the evaluation of tornado phenomena, which was inexistent before.

As regards flooding, IRSN considered that the extreme hazard levels proposed by the licensees are broadly satisfactory. However, ASN required additional justifications for the Tricastin site, as regards the resistance of the Donzère-Mondragon channel dikes to an extreme earthquake, a hypothesis held for the definition of the level of "flood" hazard for this site. The corresponding file is currently assessed by IRSN.

For the other extreme meteorological phenomena which could lead to dreaded situations (extreme winds, lightning, extreme temperature, hail, snow …), the evaluations of IRSN are also ongoing.

5.2. *Design of new "hardened safety core" SSCs and analysis of behaviour of existing "hardened safety core" SSCs*

IRSN considers that the HSC SSCs have to comply with the highest quality standards in terms of design, fabrication, maintenance and operation. This point is subject to a prescription in the ASN decisions, requiring to the licensees to justify the design of the new HSC SSCs and the behaviour of the existing HSC SSCs. At the present time, AREVA has sent a file which presents the methodology adopted in order to justify equipment and civil works integrated in the HSCs. Its assessment by IRSN led to consider that AREVA has to complete its approach, both for the design of the new HSC SSCs and the behaviour of the existing HSC SSCs. The revision of this methodology will have to be applied on the AREVA sites to design the new HSC SSCs and to analyse the behaviour of the existing HSC SSCs. This point is detailed in session 3.

5.3. *Strengthening of means of emergency management*

Le licensees have answered to the ASN decisions requiring them to ensure an autonomous functioning of the HSC SSCs during the first 48 hours after a dreaded situation, to implement means associated to the remediation or mitigation scenarii and to commission emergency management premises robust to extreme events. In particular, the licensees have sent information documents about the design of the new crisis management centres for the sites of Cadarache, La Hague, MELOX, Romans-sur-Isère and Tricastin. The assessment of these documents has led IRSN to estimate that the design principles were satisfactory on the whole. However, IRSN has considered that the design provisions have to be completed, notably regarding the dimensioning of these buildings to house the human resources and material as well as for the availability of the functions of these buildings if any dreaded situations occurred. AREVA and CEA have also sent documents for their sites aiming at justifying the adequacy and the availability of the different means associated to the remediation or mitigation of the dreaded situations. Evaluations of IRSN are ongoing.

In addition, AREVA and CEA have sent revisions of their internal emergency plans, aiming at including the management of the dreaded situations in the current crisis organisations. These documents are being assessed by IRSN. Finally, IRSN is also assessing the organisation and modality deployment of the AREVA FINA and of the CEA FARN.

This topic is detailed in session 3.

5.4. Organisational and human factors

AREVA and CEA have sent documents aiming at answering to prescriptions related to organisational and human factors of the ASN decisions sent in July 2012 and January 2015. IRSN has examined the methodology and the organisation deployed in the different sites in order to check their efficiency and robustness if any dreaded situations occurred. IRSN highlights that the analyses and actions carried out by the operators are a step in the right direction but that they need to be completed, especially regarding the measures adopted for the personnel management and the rescue to victims, the justification of the robustness of the site control recovery after an extreme event as well as the validation of this organisation.

This topic is detailed in session 3.

6. Conclusion

Following the accident of the Fukushima-Daiichi NPPs, the French operators performed, at the request of ASN, complementary safety assessments (CSAs). The CSA approach considers natural extreme hazards with a level much higher than the stress levels taken into account at the design stage with the current safety approach. Whatever the existing redundancies, the CSA approach postulates the deterministic loss of cooling and electrical supplies. Extreme situations and scenarios have been identified for each site, as well as the associated “key” SSCs necessary to maintain the facility in a safe state. IRSN has considered necessary to strengthen or implement a limited number of “key” SSCs with high design requirements, called the “hardened safety core” (HSC). In addition, the emergency organization has to be enhanced to take into account extreme events for all the facilities of the site.

In conclusion of its evaluation, IRSN underlined that the HSC measures presented by AREVA and CEA constitute improvements which should strengthen the robustness of the facilities in case of severe accidents. IRSN however considered that the HSCs proposed by AREVA and CEA must be punctually completed or strengthened on certain aspects, in particular to take into account losses of watertightness of the spent fuel storage ponds, as well as fires arising in several places of a facility. As regards extreme hazards, complementary studies are still under assessment by IRSN.

As a conclusion, even if some complements are necessary, regarding the HSC provisions and associated requirements, IRSN has assessed that the corresponding equipment and measures are about to enhance their ability to withstand extreme hazards or supply losses.

As regards emergency preparedness and response, the main outcomes dealt with the possibility to better face the consequences of an extreme event at a whole site scale, and over a long duration. New crisis centres are designed, diagnosis means are reinforced, and the AREVA “National Task Force” (FINA) as the CEA “National Rapid Response Force” (FARN) will be deployed in the coming years. Moreover, in order to ensure the reliability of remediation actions in extreme conditions, the resilience of this organization is being studied through an “organisational and human factors” scope.

===

Session 2- Implementation of post-Fukushima regulatory improvements on FCFS

Chairpersons: A. Buchan (Sellafield), K. Murthy (CNSC)

Current Regulation in Japan and Safety Research in S/NRA/R for Nuclear Fuel Cycle Facilities after the Fukushima Daiichi Nuclear Power Station (1F) Accident

K. Mori, M. Yokotsuka, M. Takanashi and K. Kubota

Regulatory Standard and Research Department, Secretariat of Nuclear Regulation Authority (S/NRA/R), Japan

1-9-9, Roppongi, Minato-ku, Tokyo, Japan 106-8450

Abstract

In responding to the lessons learned from the Fukushima Daiichi Nuclear Power Station (1F) accident, Japanese nuclear regulation was extensively amended such as introducing:

- severe accidents in regulation;
- periodic and comprehensive safety assessment;
- shift to a new regulatory system incorporating the latest knowledge is reflected in existing nuclear facilities;
- the new regulatory requirements based on ‘the graded approach’ concept.

After the above amendment, the new regulatory requirements on countermeasures for severe accidents were developed for spent fuel reprocessing facilities and nuclear fuel fabrication facilities. In this paper, nuclear fuel fabrication facilities include uranium and MOX fuel fabrication facilities, enrichment facilities and conversion facilities. An assessment for the effectiveness of the countermeasures against the severe accidents is required to licensees (spent fuel reprocessing facilities and MOX fuel fabrication facilities only). In addition, strengthening of design basis, which includes clarification of the relationship between the significance of safety function and classification of importance in the seismic design, was made along with the request of strict assessment methods for seismicity and others.

On the other hand, safety research is being performed by S/NRA/R to develop a risk assessment method which licensees are recommended to adopt for periodic and comprehensive safety assessment under new regulation. A major purpose of safety research is to accumulate technical knowledge for the review of the risk assessment. As a part of safety research, procedures of the risk assessment are studied.

This paper describes outlines of current regulation and safety research for spent fuel reprocessing facilities and nuclear fuel fabrication facilities.

1. Introduction[1,2]

Taking into account the lessons learned from the 1F accident, the Act on the Regulation of Nuclear Source Material, Nuclear Fuel Material and Reactors (the Reactor Regulation Act) was amended in June 2012. "The Reactor Regulation Act" requires establishing the new regulatory requirements (NRR) for nuclear fuel cycle facilities. It also requires licensees to conduct periodic and comprehensive safety assessment (PCSA), to submit its results to the regulatory body, and to publish them in order to ensure licensees' continuous safety improvement. PCSA is required to only spent fuel reprocessing facilities (SFRFs) and nuclear fuel fabrication facilities (NFFFs) among nuclear fuel cycle facilities.

In PCSA, the risk assessment is recommended. At present, the risk assessment methods for fuel cycle facilities (FCFs) have not been well established. Therefore, it is necessary to accumulate technical knowledge for the review of the risk assessment. From this situation, safety research on the risk assessment for FCFs is being carried out.

In this paper, outlines of NRR for SFRFs and NFFFs, and relation between PCSA and safety research are described in Chapter 2. Outlines of safety research for SFRFs and NFFFs carried out by S/NRA/R are described in chapter 3.

2. The New Regulatory Requirements and Periodic and Comprehensive Safety Assessment

2.1 Introducing Severe Accident

In "the Reactor Regulation Act", severe accidents are newly introduced into regulation. Severe accidents defined in NRR for SFRFs are listed in Table 1. They are defined as accidents that occur under conditions exceeding design basis.

2.2 Outline of the New Regulatory Requirements

In NRR for FCFs, requirements for the design basis are strengthened and the measures against the severe accidents are required to SFRFs and NFFFs.

2.2.1 The New Regulatory Requirements for Spent Fuel Reprocessing Facilities[2]

In NRR for SFRFs, requirements for the design basis are strengthened as follows:

(i) To clarify the relationship between the significance of safety functions and the classification of importance in the seismic design;

(ii) Regarding natural phenomena;

- To enhance strictly the standards for earthquakes and tsunami;
- To clarify the natural phenomena to be considered in design, e.g. volcanic eruptions; tornadoes, and forest fires, etc.;

Table 1. Severe Accidents of SFRFs in NRR

1) Criticality accident in cells

2) Evaporation to dryness due to the loss of cooling functions

In case where cooling functions are lost, waste liquid etc. boils, resulting in its evaporation to dryness.

3) Explosion of hydrogen generated by radiolysis

In case where hydrogen sweeping functions are lost, hydrogen generated by radiolysis accumulates and explodes.

4) Fire and/or explosion of organic solvents, etc. in cells

5) Fuel damage in spent fuel storage pools

6) Leakage of radioactive materials

(iii) To reinforce comprehensive fire protection measures;

(iv) To clarify the necessary consideration against external man-induced events, internal-missiles, internal leakage of chemicals, etc.;

(v) To enhance the reliability of electric power sources.

Measures against severe accidents are required as follows:

(i) To define severe accidents, and to require countermeasures and assessment of their effectiveness;

(ii) To require countermeasures to suppress the release of radioactive materials and/or radiation outside the site and those against terrorist attacks such as intentional plane crashes.

2.2.2 The New Regulatory Requirements for Nuclear Fuel Fabrication Facilities[2]

In NRR for NFFFs, requirements for the design basis are strengthened as follows:

(i) To clarify the relationship between the significance of safety functions and the classification of importance in the seismic design;

(ii) To enhance strictly the standards for earthquakes and tsunamis for MOX fuel fabrication facilities;

(iii) For uranium fuel fabrication facilities;

- To increase the margin for static seismic force;
- To apply the same requirements to the structures, systems and components (SSCs) important to safety in case of earthquakes and tsunamis as those of MOX fuel fabrication facilities.

Measures against the severe accidents* are required as follows:

*In NFFFs, loss of confinement and criticality accident are defined as the sever accidents in NRR for NFFFs.

(i) To define severe accidents and to require countermeasures and assessment** of their effectiveness;

**MOX fuel fabrication facilities only;

(ii) To require countermeasures for prevention of severe accidents for all type of NFFFs. In addition, to require following measures for MOX fuel fabrication facilities especially;

- Recovering lost functions;
- Restoring facility condition;
- Suppressing a release of radioactive materials/radiation to outside the site (mitigating the consequence);

(iii) To require countermeasures to keep working conditions safe (including the chemical influence due to the leakage of uranium hexafluoride).

2.3 Requirement of Periodic and Comprehensive Safety Assessment

The amended "the Reactor Regulation Act" requires licensees of SFRFs and NFFFs to perform PCSA by themselves at each timing specified by the Ordinance of the NRA. The purpose of PCSA is to ensure that the licensees continuously improve the safety of their facilities.

An internal rule on PCSA recommends that licensees should perform risk assessment for evaluation of severe accidents' possibilities. The internal rule also requires NRA to confirm that evaluation procedures and technical grounds used in a licensee's PCSA are appropriate. Being considered this situation and the maturity of the risk assessment methods, safety research has been carried out for accumulation technical knowledge for the risk assessment for FCFs.

In the next chapter, S/NRA/R's current safety research on the risk assessment for SFRFs and NFFFs is described.

3. Safety Research on Periodic and Comprehensive Safety Assessment for Spent Fuel Reprocessing Facilities and Nuclear Fuel Fabrication Facilities

3.1 Safety Research of the Risk Assessment for Spent Fuel Reprocessing Facilities and Nuclear Fuel Fabrication Facilities

Table 2 summarizes S/NRA/R's safety research concerning the risk assessment for SFRFs and NFFFs. As this table shows, the safety research is categorized into two parts. One is a study of the risk assessment procedures and the other is the accumulation technical knowledge for the severe accidents. Concerning the study of the risk assessment procedures, seismic risk assessment has been carried out since the earthquake is one of natural phenomena which can lead to common cause failures.

In addition, through safety research, S/NRA/R aims to prepare an example of the risk assessment procedure.

Table 2. Outline of S/NRA/R's Safety Research for the Risk Assessment for SFRFs and NFFFs

1. Study of the risk assessment procedure
a) For the internal event b) For the events which results from the earthquake - The accident frequency (Application of the simplified hybrid method* to NFFFs) - The hazard analysis procedure - The risk assessment procedure for the single severe accident - The risk assessment procedure for combination of multiple severe accidents which may occur simultaneously or successively
2. Accumulation technical knowledge for the severe accidents
a) The evaluation method of the chemical influence due to leakage of the uranium hexafluoride b) The evaluation method for the confinement of SSCs against the hydrogen explosion c) Migration behavior of radionuclides under boiling of radioactive solutions (resulting in their evaporation to dryness) ・ Entrainment of mist and generation of gaseous RuO_4 from the boiling of high-level liquid waste ・ Stability of RuO_4 in the vapor of nitric acid and water ・ Removal of radionuclides with the condensation of the vapor in the migration path

*: Simplified hybrid method[3] is the method developed so as to combine the advantage of MA (Seismic Margin Assessment) and that of SPSA (Seismic Probabilistic Safety Assessment).

3.2 Preliminary Basic Flow of the Risk Assessment for Spent Fuel fuel Reprocessing Reprocessing Facilities and Nuclear Fuel Fabrication Facilities

A preliminary basic flow of the risk assessment procedure for SFRFs and NFFFs has been developed. A preliminary one is shown in Figure 1. Table 3 provides simple explanations of the procedure. The specific feature of this flow is that the analysis of multi-severe accidents (which mean the combination of multiple severe accidents that may occur simultaneously and/or subsequently) is required considering the characteristics of SFRFs and NFFFs. In those facilities, various kinds of radioactive or chemical poisonous materials are distributed separately in various physical and chemical forms.

According to item (1), (2), (3), (6), (7), (8) of this procedure in Figure 1 and Table 3, technical knowledge to review the risk assessment procedure in the PCSA have been studied. These include the estimation methods of accident frequency and magnitude of consequence of accidents.

In these evaluations it is allowed to utilize the engineering judgment, semiquantitative or schematic evaluations and the sensitivity analysis instead of the uncertainty analysis. In the next chapter, the analysis of multi-severe accidents is described.

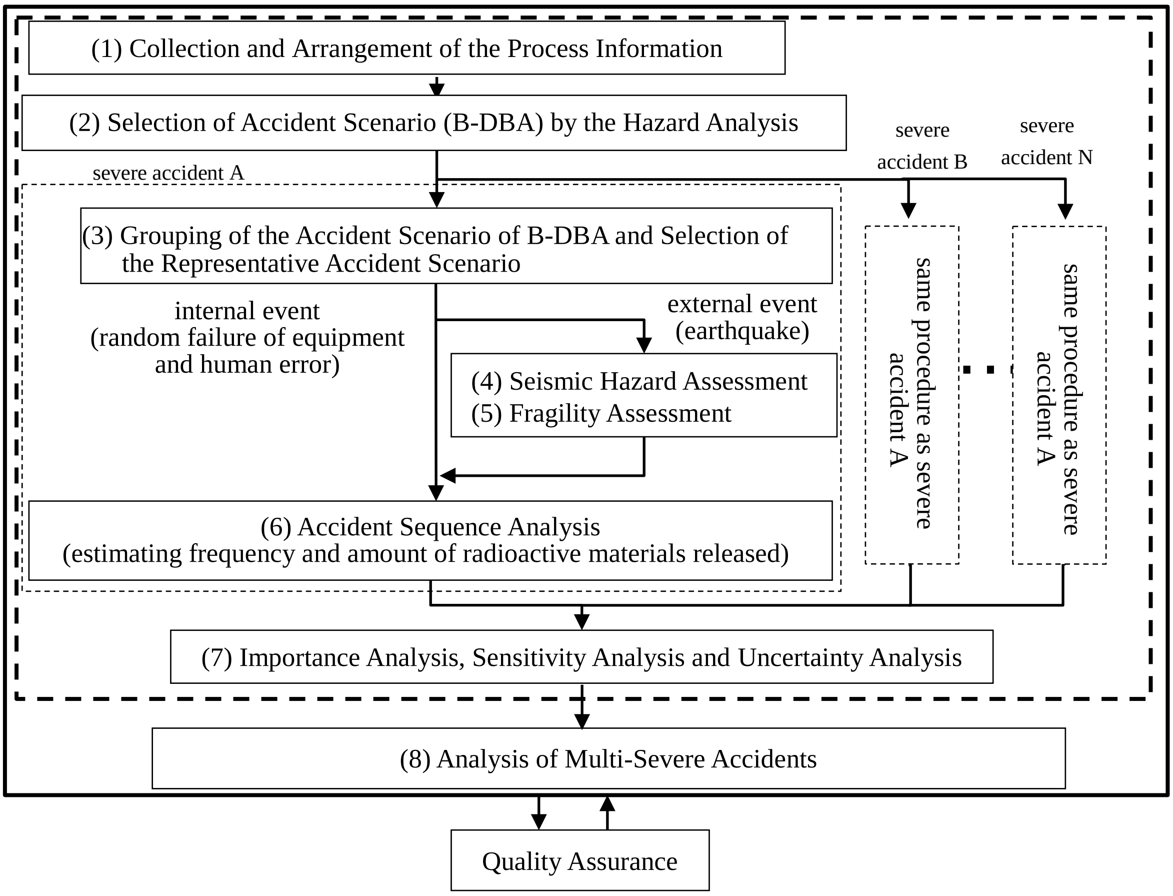


Figure 1. Preliminary Basic Flow of the Risk Assessment in PCSA for SFRFs and NFFFs

Table 3. Outline of the Basic Flow of the Risk Assessment

Item	Outline of the procedure
(1) Collection and Arrangement of the Process Information	Appropriate collection and arrangement of the process information to be used
(2) Selection of Accident Scenario (B-DBA) by the Hazard Analysis	- Identification of initiating events and accident scenarios that could occur in facilities without oversight - Implementation of the hazard analysis considering multiple occurrence of initiating events - Selection of accident scenarios of beyond design bases accidents (B-DBA)
(3) Grouping of the Accident Scenario of B-DBA and Selection of the Representative Accident Scenario	- Grouping of accident scenarios of B-DBA considering common safety measures - Selection of the representative accident scenario for the accident sequence analysis considering the inventory (consequence) and time interval between an initiating event and resultant accident before leading to the severe accident
(4) Seismic Hazard Assessment (5) Fragility Assessment	- Basically the same as those of nuclear power plants
(6) Accident Sequence Analysis	- Identification of the accident sequence for the representative accident scenario with event tree, etc. - Hierarchization of initiation events in case of earthquake - Estimation of the consequence and accident frequency - Selection of the accident sequence that may lead to severe accidents based on results of above procedures
(7) Importance Analysis, Sensitivity Analysis and Uncertainty Analysis	- Conducting importance analysis, sensitivity analysis and uncertainty analysis for the accident sequence selected in (6) - Selection of safety measures and factors influencing risk particularly important for improving the safety
(8) Analysis of Multi-Severe Accidents	- Selection of the representative combination of multiple severe accidents and SSCs based on the evaluation of (1) ~ (7) - Study of the accident sequence considering the mutual interactions of each severe accident
Quality Assurance	- Refer to the quality guidelines on facilities of nuclear power plant

3.3 Study of Multi-Severe Accidents

3.3.1 Background and Purpose of study

In SFRFs and NFFFs, various kinds and forms of radioactive materials and chemicals are processed and distributed in whole facilities. Those materials can, therefore, cause severe accidents simultaneously when safety function is lost by the initiating event such as earthquakes. Although it is important to conduct the risk assessment for multi-severe accidents, a procedure for this assessment has not been practically developed yet. Therefore, a reasonable procedure to conduct the risk assessment for multi-severe accidents has been studied.

3.3.2 Multi-Severe Accidents

Some examples of multi-severe accidents are as follows:

(i) Combination of different severe accidents in the same place;

(ii) Combination of the same severe accidents in different places;

(iii) Combination of different severe accidents in different places;

(iv) Occurrence of severe accidents caused by the other severe accidents.

In this study, “multi-severe accidents” include the accidents which may occur not only simultaneously but also one after another sequentially.

The most important point on the risk assessment for multi-severe accidents is to consider the mutual interactions among severe accidents. The mutual interactions mean that one of severe accidents, when it occurs, may affect other severe accidents phenomena and safety measures for them.

3.3.3 Preliminary Risk Assessment Procedure for Multi-Severe Accidents with Mutual Interactions

The preliminary flow of a risk assessment procedure for multi-severe accidents with mutual interactions is shown in Figure 2. As the first step, the risk assessment is conducted on a unit of structure, e.g. a cell in a reprocessing facility or a process chamber. After above assessment is completed for every unit of structure, the risk assessment is expanded to other units of a building considering mutual interactions among cells, etc. The procedure of the risk assessment for a unit of structure consists of five steps as follows.

Step1. Identifying combination of multi-severe accidents with mutual interactions.

Examples of interactions are as follows.

- Decrease of time interval between an initiating event and a resultant accident
- Increase of source term
- Occurrence of other severe accidents
- Loss of safety functions
- Deterioration of working environment for countermeasures

Step2. Selecting the representative combination of multi-severe accidents.

The identified combinations should be grouped and the representative combination is selected in each group.

Step3. Identifying the place where the representative multi-severe accidents occur.

The place, where the representative multi-severe accidents may occur, should be identified.

Step4. Identifying the accident scenario and the accident sequence.

The scenario and the sequence for single severe accident should be modified considering mutual interactions among severe accidents.

Step5. Conducting sensitivity analysis.

In Step1, the extent of mutual interactions (how mutual interactions affect severe accidents each other) is assumed. In this step, this assumption should be varied to study how the risk assessment results change depending on the extent of mutual interactions.

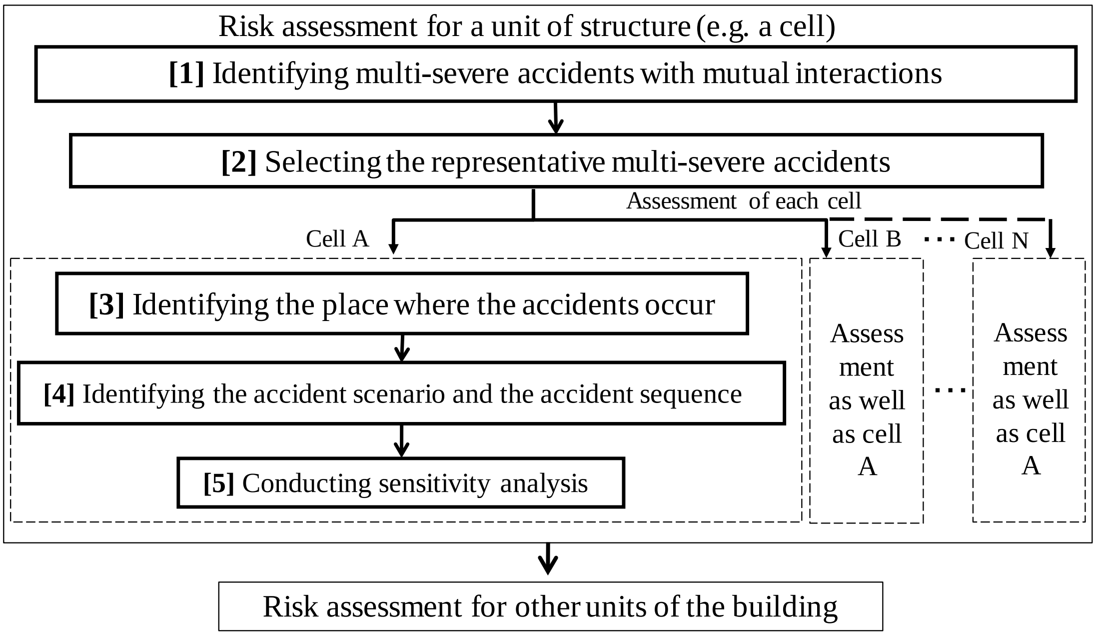


Figure 2. Flow of the Risk Assessment Procedure for Multi-Severe Accidents with Mutual Interactions

4. Summary

In this paper, the outlines of NRR for FCFs after the 1F accident and the safety research conducted by S/NRA/R were described. The major purpose of safety research is to accumulate technical knowledge for the review of the risk assessment in PCSA which is required to licensees in the current regulation. As one of the major results of safety research, the preliminary basic flow of the risk assessment procedure for SFRFs and NFFFs was developed. The specific feature of this flow is that the analysis of multi-severe accidents which mean the combination of multiple severe accidents that may occur simultaneously or subsequently is included considering the characteristics of SFRFs and NFFFs.

The safety research will be conducted continuously to accumulate technical knowledge for the review of risk assessment. Revising the risk assessment procedure based on results of further safety research is a logical next step.

Reference

1 Nuclear Regulation Authority, "Enforcement of the New Regulatory Requirements for Commercial Nuclear Power Reactors", July 8, 2013.

2. Nuclear Regulation Authority, "Outline of the Draft New Regulatory Requirements for Nuclear Fuel Facilities, Research Reactors, and Nuclear Waste Storage/Disposal Facilities".

3. R.P.Kennedy, "Overview of Methods for Seismic PRA and Margin Analysis Including Recent Innovations", Proceedings of the OECD-NEA Workshop on Seismic Risk, Tokyo Japan, 1999.

Implementation of post-Fukushima regulatory improvements on FCFs

Mickael Gandolin, ASN, France

Plan :

1. Context
2. Complementary Safety Assessment as a regulatory requirement
3. Scope of the CSA
4. A graded plan
5. CSA as part of the existing safety improvement process
6. Milestones for the post-Fukushima regulatory programme
7. A consistent approach
8. The post-Fukushima ASN resolutions in details
9. Preliminary feedback

1. Context

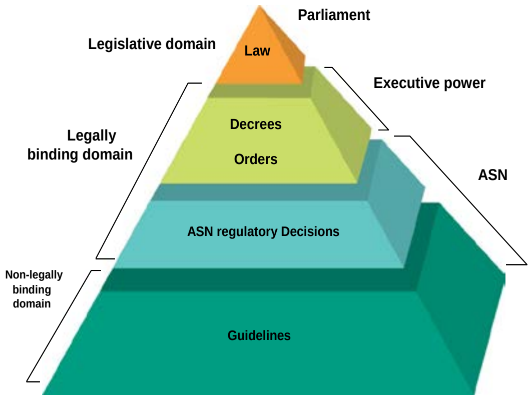
The Fukushima accident, triggered by an earthquake and a tsunami on an exceptional scale, confirmed that despite the precautions taken in the design, construction and operation of the nuclear facilities, an accident is always possible. In this context, and given its knowledge of the 150 French nuclear facilities, through its regulation and oversight, ASN considered in the days following the accident that a complementary assessment of the safety (CSA) of the facilities, with regard to the type of events leading to the Fukushima disaster, should be initiated without delay, even if no immediate emergency measures were necessary.

2. Complementary Safety Assessment as a regulatory requirement

These complementary safety assessments are part of a two-fold approach: on the one hand, performance of a nuclear safety audit on the French civil nuclear facilities in the light of the Fukushima event, which was requested from ASN on 23rd March 2011 by the Prime Minister, pursuant to article 8 of the TSN Act and, on the other, the organization of "stress tests" requested by the European Council at its meeting of 24th and 25th March 2011.
In order to manage the complementary safety assessments, ASN issued twelve resolutions on 5th May requiring the various Licensees of the nuclear facilities to perform these CSA in accordance with precise specifications.

***[ASN resolutions]**: Pursuant to article 4 of the TSN Act, ASN can take regulatory resolutions to point out*

decrees and orders issued concerning nuclear safety or radiation protection, which are submitted to the Government for approval. ASN also issues individual resolutions concerning nuclear activities (for example, commissioning authorization for a basic nuclear installation, authorization to use radioactive material transport packaging, authorization to use radioactive sources, definition of requirements concerning the design, construction, operation or decommissioning of a facility, etc.).
Non-compliance with the ASN resolutions could lead to an offence.



3. Scope of the CSA

The complementary safety assessments concern the robustness of the facilities to extreme situations such as those which led to the Fukushima accident. They complement the permanent safety approach followed.

To ensure consistency between the European and French approaches, the French specifications for the complementary safety assessments were drafted on the basis of the European specifications produced by WENRA (Western European Nuclear Regulators' Association) and approved by ENSREG (European Nuclear Safety REgulators Group) on 25th May 2011. The provisions of the French specifications are consistent with those of the European specifications.

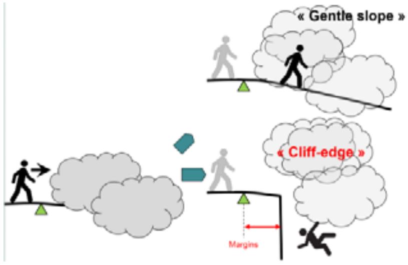
The CSA thus consists of a targeted reassessment of the safety margins of the nuclear facilities in the light of the events which took place in Fukushima, that is extreme natural phenomena (earthquake, flooding and a combination of the two) placing considerable strain on the safety functions of the facilities and leading to a severe accident. The assessment first of all concerns the effects of these natural phenomena; it then looks at the loss of one or more systems important for safety involved in Fukushima (electrical power supplies and cooling systems), regardless of the probability or cause of the loss of these functions; finally, it deals with the organisation and the management of the severe accidents that could result from these events.

Three main aspects are included in this assessment:

- The steps taken in the design of the facility and its conformity with the design requirements applicable to it ;

- The robustness of the facility beyond the level for which it was designed; the licensee in particular identifies the situations leading to a sudden deterioration of the accident sequences ("cliff-edge effects" and presents the measures taken to avoid them) ;
- All possible modifications liable to improve the facility's level of safety.

[Cliff-edge effects]: *High discontinuity in the scenario causing notable and irreversible aggravation of the accident (significant increase in releases, significant decrease in time before undesirable situation is reached, etc.).*



ASN decided to apply the complementary safety assessments to all French nuclear facilities and not simply to the power reactors. Thus, virtually all of the 150 French nuclear facilities will undergo a complementary safety assessment, including for example the EPR reactor currently under construction, or the spent fuel reprocessing plant at La Hague. In this respect, the French specifications have been extended compared to those adopted at the European level by ENSREG.

As of the beginning of the process, the association of stakeholders, particularly HCTISN, asked ASN to place particular emphasis on social, organizational and human factors, especially subcontracting. The Fukushima accident showed that the ability of the Licensees and, as necessary, its subcontractors to organize and work together in the event of a severe accident is a key factor in the management of such a situation. This ability to organize is also a key aspect of accident prevention, facilities maintenance and the quality of their operation. The conditions for the use of subcontracting are also tackled in the French complementary safety assessments.

4. A graded plan

The CSA concern virtually all the 150 basic nuclear installations in France (58 nuclear power generating reactors, EPR reactor under construction, research facilities, and fuel cycle plants).

These facilities have been divided into three categories, depending on their vulnerability to the phenomena which caused the Fukushima accident and on the importance and scale of the consequences of any accident affecting them.

The classification in the tree batches takes in account:

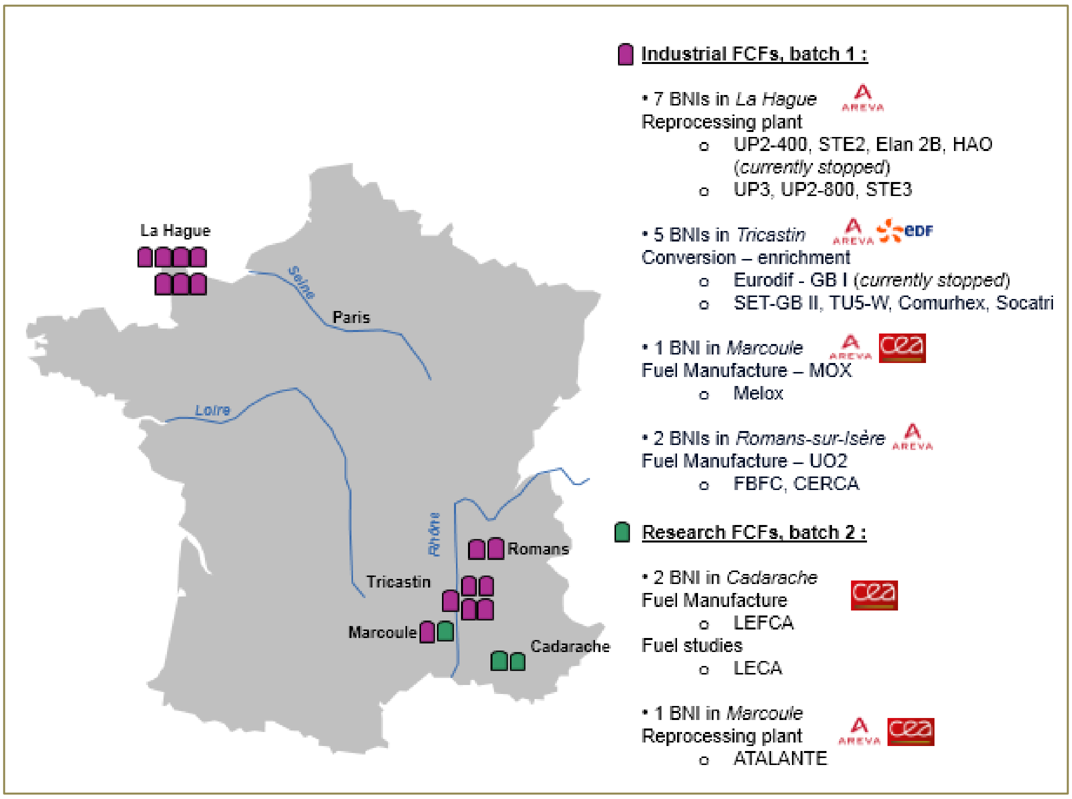
- the type of the facility:
  - nuclear reactors based on the thermal power,
  - fuel cycle facilities based on the annual processing capacity,
  - …
- the amount of radioactive material and hazardous substances,
- the potential off-site releases and the vicinity of the plant,
- the robustness and the independence of the containment barriers.

For the 79 facilities felt to be a priority, first batch, including the 59 power reactors in operation or under construction, the Licensees (AREVA, CEA, EDF, Laue-Langevin Institute) submitted their reports to ASN on 15th September 2011.

For the facilities of lower priority, batch 2, the Licensees are required to submit their reports before 15th September 2012.

Finally, the other facilities, batch 3, will be dealt with through appropriate ASN requests, in particular on the occasion of their next ten-yearly periodic safety review, except for about ten facilities for which decommissioning is nearing completion.

The classification of the French Fuel Cycle Facilities is presented on the map below:



5. CSA as part of the existing safety improvement process

These assessments were carried out in addition to the safety approach performed permanently. The CSA is complementary to existing safety improvement processes:

- periodic safety reviews (PSRs)
- integration of operating experience feedback.

Periodic safety review	*Complementary Safety Assessment*
Required by law: *article L 593-18 of the Code of the Environment*	Required by ASN resolutions
Every 10 years	Once
The technical scope is the overall review of the safety cases. It is presented at the *Title III "Demonstration of nuclear safety" of the Order of 7 February 2012 setting the general rules relative to basic nuclear installations* The main points are: - principle of defense in depth, - a prudent deterministic procedure and shall also include probabilistic analyses of accidents and their consequences, - technical, organizational and human dimensions - 4 safety functions (control of nuclear chain reactions, evacuation of the thermal power produced by the radioactive substances and nuclear reactions, containment of the radioactive substances, protection of persons and the environment against ionising radiation). - list of internal and external hazards to consider.	The technical scope is defined in the ASN resolutions (issued in 2011, 2012 and 2015). Initiating events (beyond design basis) to be taken into account : - Earthquake - Flooding - Other extreme natural events The Consequential loss of safety functions to be assessed using a deterministic approach : - Loss of electrical power, including station blackout (diesels, etc.), - Loss of the ultimate heat sink, - Combination of both The severe accident management issues have to be tackled : - Loss of the core cooling function, - Loss of the spent fuel storage pool cooling function, - Other severe accidents.
The PSR Process is organised in 2 steps: • Conformity with design requirements and Compliance with regulations : - Authorization for creation, Site Licence,	The CSA process includes: • Safety margins taken in the facilities design basis and plant conformity to its design requirements, - assessment of available design

- Environmental & radiological regulatory, - ASN technical resolutions, - environmental permits for the installation, - … • Safety Reassessment: - Reassessment considering the current state of art, - Best practices applicable at national and international levels, in nuclear and non-nuclear activities, - Review of methods used in the safety cases, the accident studies, in the evaluation of natural hazards, … - Improvements and lessons learnt from operating experience, - …	margins, diversity, redundancy, physical separation, … - assessment of the cliff-edge effects beyond which fundamental safety functions would be lost, • Safety margins taken in the facilities design basis and plant conformity to its design requirements, • Possibility of mobile external means and the conditions of their use, • Possibility of mutual help between units from a same site.

6. Milestones for the post-Fukushima regulatory programme

March 11 2011	Fukushima Accident
March 23 2011	Declaration of the French Prime minister asking for national audit to be carried out for all French nuclear facilities, in priority for NPPs
May 5 2011	ASN Commission adopted **12 resolutions** requiring Licensees of French nuclear facilities to conduct a complementary safety assessment (CSA)
June 1st 2011	French Licensees' CSA methodologies sent to ASN
July 6 2011	Advisory expert committee meeting on CSA methodology

July 19 2011	ASN Opinion on Licensees' CSA methodology
Sept. 15 2011	French Licensees' CSA reports sent successfully to ASN (1st batch)
Nov. 8-10 2011	Joint meeting of the French advisory expert committees, based on the French TSO IRSN review of the CSAs
Dec. 2011	ASN's report on CSAs
Jan. 3rd 2012	ASN issued its opinion on French CSAs
June 26 2012	**ASN resolutions**: Complement to CSA + proposals for hardened safety core
June 2012 to April & July 2013	French TSO IRSN reviews complements and proposals (1st & 2nd batches)
April 2013	Joint Meeting of the French advisory expert committees for 1st batch
July 2013	**ASN resolutions** consisted in enforcement notices to implement the transition measures pending the implementation of the *hardened safety core*.
July 2013	Joint Meeting of the French advisory expert committees for 2nd batch
Jan. 8 2015	**ASN resolutions**: Complement to CSA + requirements to define and to design the hardened safety core (1st & 2nd batches)
2016 - 2017	**ASN guidance** to take into account Fukushima accident in the design of the new nuclear facilities (*in progress*)
Until 2018-2020	Long term programme to take in account post-Fukushima feedback actions

7. A consistent approach

Through the following regulatory work, ASN has implemented a common process for all the installations: NPPs, FCFs and RRs, in order to build a consistent approach.

The ASN resolutions and requirements for all the installations have been built from:

- the Licensees' CSA reports which had the same specifications,
- the IRSN assessment findings,
- the French advisory expert committees advices,
- The ASN inspections findings.

ASN has set all the elements which could contribute to increase the robustness of the safety of the facilities. These elements are the basis of the ASN requirements and are of three types.

1/ the requirements that include, first, common provisions to all the installations covered by the CSA. These cover the following topics:

- the enhancements of nuclear standards and guidance,
- the setting of a group of robust materials and organizational provisions for extreme events, , *"hardened safety core"*, to prevent a serious accident or limit its progression, limiting massive releases and allow the Licensees to carry out the expected actions related to emergency preparedness and response ;
- the arrangements to ensure an operative organization and suitable means to manage events which could affect all or part of the facilities on a same site. These include in particular the need for an emergency control room designed to withstand to extreme natural hazards,
- taking into account the social, organizational and human factors in crisis-management operations, including the definition of human actions required to manage an extreme event, the list of the expected skills to manage these situations and whether these skills are likely to be held by contractors.

These generic provisions to all nuclear installations were defined as a result of joint work between the relevant ASN departments in charge of the different types of facilities (NPPs, research reactors, fuel cycle facilities ...).

2 / these common provisions are supplemented, for the installations of a same industrial site (such as Tricastin) by specific requirements related to the site, its environment and its vicinity. These could include to assess potential effects specific to the site generated by earthquake or flooding as dam failures, a severe accident of NPPs nearby… There are specific requirements to all facilities of a same site, as example, a requirement to mitigate the consequences of releases of hydrofluoric acid (HF) or uranium hexafluoride (UF6) for the facilities located on the Tricastin site.

3 / This set of generic and related to the same industrial site requirements is supplemented by the provisions specific to the installation and its operations. These could deal with specific materials to withstand an extreme natural event (earthquake, flooding, tornado …) or complementary assessments of the installation's safety cases in the framework of the CSA, as "*the feared situations*".

8. The post-Fukushima ASN resolutions in details

The first resolutions were taken in May 2011. They defined the scope of the complementary safety assessments in the light of the Fukushima accident. The details are provided above. The deadlines were related to the batches of the installations were fixed in the resolutions. The target dates were September 2011 for most facilities and September 2012 for one of them.

The second resolutions were taken in June 2012. ASN prescribed the Licensees to establish a *hardened safety core* of robust material and organisational measure in order to prevent or mitigate the progress of a major accident, to mitigate large-scale radioactive releases and to enable the Licensees to perform its emergency management duties in the case of an extreme event. The *hardened safety core* have to be designed or qualified with significant margins beyond design basis and, composed of independent and diversified systems, structures and components. The Licensees shall justify the use of undiversified or existing SSCs.
ASN also required the Licensees to assess some cases “feared situations” specific to the installations, interim measures and complementary arrangements to manage emergency situations according to the risks on sites, in particular towards earthquake, flood and chemical risks. Some steps aiming to enhance skills and social and psychological care were also prescribed.

The third resolutions were sent July 2013. They consisted in enforcement notices to implement the interim measures pending the implementation of the hardened safety core.

The fourth resolutions were sent in January 2015. They provide the definition of a “*hardened safety core*” for each facility and site to manage cliff effects. They specify the list of SSCs composing the *hardened safety core* and their qualification requirements:

- New SSCs designed according to industrial standards,
- Existing SSCs verified according to industrial standards, or verified according to methods allowed during PSRs.

The level of external hazards to consider for the hardened safety core have been set. Then, some general and specific requirements were written concerning Systems, Structures and Components design and sizing for the *hardened safety core* as well as additional studies. Finally, ASN required additional arrangements to manage extreme emergency situations and related to organizational and human factors:

- arrangements to ensure the ability of the hardened safety core SSC to work the first 48hrs without any external support and supplies,
- availability in the Emergency Control Room of key parameters related to the safety functions of the facilities (level of water in a pond, T°, …),

- arrangements to provide external support (human resources, additional materials and supplies) to a site affected by an extreme event.

9. Preliminary feedback

The main challenges the ASN had to face**,** were the arbitration between the optimization of time and the ambition of the requested improvements. Indeed ASN expected the Licensees to implement the *hardened safety core* as soon as possible in order to reduce risks on nuclear installations where cliff-effects have been found. The ASN resolutions provide a clear regulatory framework, safety objectives, scope and deadlines to these enhancements programmes. It helps to secure the related resources plan.

However, the definition by the Licensees of these arrangements and the assessment by ASN and IRSN to be sure that they are adequate and fit for purpose take time. It was a big challenge for the Licensees, the French TSO and the regulator. Also the necessary coordination with the department in charge of reactors control was an issue.

Canadian Fuel Cycle Facilities: Licensee response and Canadian nuclear regulatory framework changes as a result of lessons learned from the Fukushima Daiichi Disaster

Kavita Murthy, Julian Amalraj & Jocelyn Truong
Canadian Nuclear Safety Commission, Canada

Abstract

On March 11, 2011, a magnitude 9.0 earthquake followed by a devastating tsunami struck Japan. The combined impact of the earthquake and tsunami caused a severe nuclear accident at the Fukushima Daiichi nuclear power plant. In response, to these events, the Canadian Nuclear Safety Commission (CNSC) required licensees of Nuclear Power Plants and major non-power reactor nuclear facilities, including fuel cycle facilities, to conduct a review of the initial lessons learned from the Fukushima nuclear accident. The Canadian facilities were requested to re-examine safety cases, in particular to investigate:

- *Strengthening their defense-in-depth, with a focus on external hazards such as seismic, flooding, fire and extreme weather events and measures for prevention and mitigation of severe accidents; and*
- *Enhancing their emergency preparedness programs in response to extreme events.*

Licensees were required to report to the CNSC on implementation plans for short-term and long-term measures to address any significant gaps and improvements proposed. Based on the completed reviews of licensee submissions, CNSC staff concluded that the safety cases for fuel cycle facilities are valid; the facilities remain safe and do not pose a significant risk to the health and safety of Canadians or to the environment. Nonetheless, for extreme situations such as those of very low probability, but potentially high impact events, the licensees identified improvements to their installations, equipment and emergency response plans to augment the robustness of their respective facilities and strengthen their emergency response.

In addition to the above, in order to identify opportunities for improvement to the Canadian nuclear regulatory framework, the CNSC also established a Fukushima task force and undertook a comprehensive review of the provisions under existing system. The review concluded that the framework is robust, comprehensive and effectively applied to the whole range of nuclear power plant conditions, including severe accidents. The task forces also made recommendations to further enhance the safety of Canadian nuclear facilities.

Background

The Canadian Nuclear Safety Commission

The Canadian Nuclear Safety Commission (CNSC) is the regulatory body created under the *Nuclear Safety and Control Act* (NSCA) by the Parliament of Canada, *to regulate* the use of nuclear energy and materials to protect the health, safety and security of persons and the environment; *to implement* Canada's international commitments on the peaceful use of nuclear energy; and *to disseminate* objective scientific, technical and regulatory information to the public.

The Canadian nuclear regulatory framework consists of the *Nuclear Safety and Control Act*, (the Act) which was enacted by the Canadian Parliament in the year 2000, the regulations made under the Act, licences and regulatory documents. The Act established the CNSC as the regulatory body for nuclear activities in Canada and among other things, gave the CNSC the power to make regulations and the authority to grant licences. There are 13 regulations made under the Act, several that apply to all activities regulated by the CNSC such as the *Radiation Protection Regulations* and some specific sector-focused regulations such as the *Uranium Mines and Mills Regulations*.

The decision making body with respect to major nuclear facilities in Canada is the independent quasi-judicial administrative tribunal (the Commission), made up of up to seven permanent members. Decisions are made transparently and the Commission provides public and other interested parties opportunities to participate when major decisions are under contemplation. Roughly 800 scientific, technical and professional staff support the Commission. Staff review applications and make recommendations to the Commission and are also responsible for regulatory compliance verification and enforcement.

Regulatory documents serve to clarify the CNSC’s regulatory expectations and generally include two kinds of information: requirements and guidance. When included in the licensing basis, or as a licence condition, requirements are mandatory and must be met by anyone wishing to obtain (or retain) a licence.

Figure 1: The CNSC's regulatory framework is made up of the *Nuclear Safety and Control Act*, the regulations made under the Act, licences accorded by the Commission and regulatory documents.

Fuel Cycle Facilities in Canada

At the present time, Canada has four operating nuclear power plants (NPPs) and one plant in safe shutdown. All of Canada's power reactor fleet are CANDU (Canadian Deuterium-Uranium) reactors. These pressurized heavy water reactors use natural uranium as fuel and heavy water as a coolant and moderator.

Operating commercial fuel cycle facilities (FCFs) in Canada include: Uranium mines and mills and, front-end uranium processing facilities, interim spent fuel storage facilities and one FCF research & development facilities (Chalk River Laboratories). Spent fuel is also stored at the respective reactor sites. There are no enrichment facilities or re-processing facilities in Canada.

This paper focuses on CNSC licensed uranium processing facilities that support front-end nuclear fuel cycle and the three spent fuel waste storage facilities that support the back end nuclear fuel cycle. An overview of CNSC approach to review of lessons learnt from the Fukushima incident, licensee actions taken in response to CNSC review request and regulatory framework actions taken by CNSC in response to the lessons learnt review undertaken specifically related to above mentioned FCFs are detailed in this paper. A brief description of the facility activities are provided below:

Uranium Processing Facilities:

- The Blind River Refinery (BRR) facility is located near Blind River, Ontario, in northern Ontario, Canada. The facility refines uranium concentrates (yellowcake) received from uranium mines worldwide to produce uranium trioxide, an intermediate product of the nuclear fuel cycle.

- The Port Hope Conversion Facility (PHCF) is located in Port Hope, Ontario, situated on the north shore of Lake Ontario approximately 100 kilometers east of the city of Toronto. The facility primarily converts uranium trioxide (UO_3) powder to uranium dioxide (UO_2) and uranium hexafluoride (UF_6). UO_2 is used in the manufacture of CANDU reactor fuel, whereas UF_6 is exported for further processing into fuel for light water reactors.

- Cameco Fuel Manufacturing Facility (CFM) is also situated in Port Hope, Ontario. It receives natural and depleted uranium dioxide powder (UO_2) from PHCF and fabricates ceramic fuel pellets and thereafter manufactures finished fuel bundles for CANDU reactors for use in Canadian nuclear power reactors and research reactors.

- GE Hitachi Nuclear Energy Canada Inc. (GEHC), whose operations are similar to CFM, operates two uranium-processing facilities in Ontario. The Toronto facility processes natural uranium dioxide (UO_2) powder from PHCF into ceramic pellets. The majority of these pellets are shipped to GEH-C's Peterborough facility where they are assembled into CANDU reactor fuel bundles.

Spent fuel waste storage facilities:

- Darlington Waste Management Facility (DWMF) and Pickering Waste Management Facility (PWMF), whose primary purpose is to store spent nuclear fuel in Dry Storage Containers from the Darlington Nuclear Generating Station (NGS) and from the Pickering A and B NGSs respectively.

- Western Waste Management Facility's whose primary purpose is to process and manage low and intermediate level radioactive waste from all of Ontario Power Generation's (OPG) owned nuclear facilities in Ontario and to store spent nuclear fuel in Dry Storage Containers from the Bruce A and B NGSs.

CNSC approach to lessons learnt review for FCFs

In March 2011, the CNSC issued a directive to all major nuclear facilities in Canada to review initial lessons learned from the Fukushima nuclear accident, to re-examine their safety cases and to develop action plans to address any identified gaps. The directive was issued under subsection 12(2) of the *General Nuclear Safety and Control Regulations*, which places an obligation on licensees to respond to a request from the Commission, or a person who is authorized by the Commission, to "conduct a test, analysis, inventory or inspection in respect of the licensed activity or to review or to modify a design, to modify equipment, to modify procedures, or to install a new system or new equipment”

The CNSC convened an internal task force to review the responses from the Canadian NPP operators. To review the lessons learnt for FCFs the CNSC modified the methodology applied to NPPs to take into account the differences in facility designs, activities and the nature of the hazards present at these facilities. CNSC staff approach to the review dealt with key aspects of FCFs which included:

- Strengthening defence-in-depth; and
- Enhancing emergency preparedness and response capabilities
- Improving the regulatory framework
- Communications and public education

Activities related to strengthening defence-in-depth included review of facility’s safety case (design of the facilities, operational reliability, internal and external credible events, facilities’ safety features) and improving safety envelope under site-specific external hazards. Activities under enhancing emergency response included review of facilities’ emergency response plans, including procedures, training and equipment review and update to equipment as well as testing severe accident scenarios. To address the differences in activities, designs and hazard levels, CNSC staff adopted a graded approach and categorized the facilities as shown in Table 1, with the level of risk posed by the facilities decreasing from top to bottom.

Table 1: Major non-power Nuclear Facilities Graded Approach

Facilities where progression of events could lead to severe accidents	
Relevant Facility types	Research reactors Fissile liquid waste in storage
Key considerations	Decay heat per unit of volume/mass Criticality Chemical stability Radiation source terms
Facilities with significant source term (radiological or chemical)	
Relevant Facilities types	Nuclear material / Isotope processing Uranium mines and mills Uranium processing Waste management
Key considerations	Radiation and chemical source terms Distance from public and environment Existing design basis
Other facilities	
Relevant Facilities	Class I particle accelerators greater than 50 MeV
Key considerations	Radiation source terms Risk of releases to the environment

CNSC staff also considered for each major facility the measures in place to prevent or mitigate the progression of an accident and to ensure the protection of the public and the environment.

Canadian FCF licensee improvements

All major nuclear facility licensees responded to the CNSC request to review lessons learnt from the Fukushima incident within the stipulated timeframe. They assessed their respective nuclear facility safety case and operations reliability as well consequence for a range of hazards, including: seismic, flooding, tornadoes, extreme weather and transportation hazards. The licensees submitted to the CNSC the results of their reviews and their plans to enhance the safety of the facilities where warranted.

To address the CNSC's request, uranium processing facilities either retained a third party expert to conduct a thorough review of their operations and/or conducted an in-house review of its safety analysis documentation. A summary of the review results categorized by facility are provided in Table 2 below:

Table 2: Uranium Processing Facilities

Fukushima review related Improvements and Enhancements	
Cameco – Blind River Refinery	To assess critical spare parts.
	To develop a procedure for the periodic review of preventive maintenance tasks on critical safety systems.
	To model the release of radioactive material to the environment in the event of a fire and revise emergency response plans accordingly.
	To model the release of uranium in the event of a tornado and revise emergency response plans accordingly.
	To assess additional controls for localized flooding (i.e., lateral expansion of perimeter ditch).
	To review flood modeling analysis discrepancy between the Eldorado studies and the potential predicted by Brookfield Power – Mississagi Power.
	Emergency response plans were reviewed and found adequate.
Cameco – Port Hope Conversion Facility	To review inspection program for safety significant structural components.
	To complete implementation of systematic approach to training (SAT) program for all high risk positions.
	To assess consequences of activating emergency stop switches at the UF_6 plant without further operator intervention and /or with full loss of service.
	To revise operator's manuals with respect to emergency shutdown procedures.
	To assess critical spare parts through the operational reliability program.

Fukushima review related Improvements and Enhancements	
	To model accidental releases of HF, UO_2 and UF_6 from a beyond design basis transportation accident, earthquake or other events for PHCF and revise emergency response plans accordingly.
	To review and revise the emergency response plans to include alternate access routes to the facility.
	To test the revised emergency response plan.
Cameco Fuel Manufacturing	To conduct a review of the preventive maintenance tasks.
	To align the critical spare parts project with a corporate project to identify and evaluate critical spare parts
	To review and revise safety report to include hydrogen fire/explosion
	To update emergency response plans to include all buildings within the facility perimeter.
	To review and update pre-incident plan with Port Hope fire department.
GE – Hitachi Toronto & Peterborough – Fuel Fabrication Facilities	To conduct a review of the sintering furnaces current configuration against the NFPA 86 standard (GEHC used a third party)
	To review and update the Peterborough facilities safety analysis report to ensure usage of consistent methodologies and outputs between the two GEHC facilities
	To increase firefighter training frequency from every two years to annually.
	To submit updated emergency response plans to emergency response agencies in Toronto and Peterborough.

All five uranium processing licensee reviews concluded that individual facilities were safe with respect to the public, workers and the environment and that the facilities have the capacity to mitigate both natural and man-made risks. Furthermore, the operating licences for all three Cameco facilities were renewed in 2012 by the Commission following public hearings. A detailed safety assessment and review of their existing programs was an integral part of the review.

The used fuel Waste Management facility’s operator OPG also re-examined its safety case, defense-in-depth concepts and emergency preparedness procedures. Initiating hazard conditions such as power supply blackout, fire, seismic and flood events, both from a design basis event perspective and from a beyond design basis event for the waste management facilities were also reviewed. As a result of this review, OPG determined that there were no significant issues requiring immediate corrective or compensatory measures. A summary of the improvements is listed in Table 3.

Table 3: OPG Spent Fuel Storage Facilities

Fukushima review related Improvements and Enhancements
Create a procedure for the post-event inspection of the waste management facilities and equipment.
Create a procedure for the manual pump-out and sampling of the water sampling stations.
Create a procedure for a fire watch, following a post-event fire system impairment.
Request that Bruce Power update their seismic event procedure to include information notifications to the WWMF.
Perform a review of CO2 system availability in the event of a loss of power for an extended period of time.
Perform a flood hazard assessment at the WWMF site, to determine the potential for flooding under extreme rainfall conditions.
Examine the strength of the dry storage container welding bay walls in case of an extreme event at the PWMF and WWMF.
Assess the seismic robustness of above ground storage structures, to determine if there is any adverse impact on the release of radioactive material due to extreme seismic events, and the consequential impact, if any, on the dose rate to the public.
Revise internal programs and procedures to improve the post-event response.
Review of the need for additional contracts for external emergency services.
Purchase additional emergency equipment.

In line with CNSC expectations that safety reports be reviewed and updated on a periodic basis to reflect the current state of the facilities, licensees in some cases updated their safety reports and emergency response programs documents to capture the most recent analyses and to reflect design changes to their facilities. As a result of the reviews, licensees self-identified certain improvements and enhancements to their facilities, equipment and emergency response plans to strengthen their safety case and emergency preparedness. CNSC staff independently reviewed these reports and accepted the relevant conclusions. The reviews confirmed that there is no FCF facility where an extreme external event could cause a consequential severe nuclear accident. In most cases an extreme external event would not result in any offsite consequences. The implementation of the licensee self-identified improvements were tracked through regular compliance activities, desktop reviews and compliance inspections the results of which were reported to the CNSC Commission and the public through annual performance reports.

CNSC regulatory framework improvements

The CNSC's Integrated Action Plan is a document that describes specific actions to be implemented by staff, licensees and affected federal and provincial stakeholders to strengthen the defence-in-depth of Canadian nuclear power plants (NPPs) and major nuclear facilities (Class I nuclear facilities and uranium mines and mills), enhance emergency preparedness, as well as improve regulatory oversight and crisis communication capabilities. The Integrated Action Plan encompasses all public and stakeholders' recommendations and comments received during public consultations, as well as the outcomes from two independent reviews: one by the International Atomic Energy Agency (IAEA) Integrated Regulatory Review Service (IRRS) follow-up mission, and the second by an external advisory committee (EAC) established by the President of the CNSC.

To implement the actions identified in the area of improving regulatory framework processes the CNSC proposes to amend the *Class I Nuclear Facilities Regulations,* the *Uranium Mines and Mills Regulations* and the *Radiation Protection Regulations*.

The proposals are briefly summarized below:

- update and clarification of requirements for radiation protection during an emergency with current international standards and practices in order to ensure that doses to persons participating in emergency control are optimized and appropriate for the type of action being undertaken during the emergency response.
- ensure that human performance and fitness for duty are addressed by Class I nuclear facility licensees to support workers in conducting their daily tasks and to be prepared to effectively respond to nuclear emergencies , in recognition of the fact that recovery efforts following a severe accident depend to a large extent on the capabilities of workers to carry out tasks
- ensure that nuclear power plant licensees undertake regular reviews against modern codes and standards to identify safety improvements to their facilities to ensure their continued safe operation
- modernize the Class I Nuclear Facilities Regulations and the Uranium Mines and Mills Regulations to reflect the nuclear industry's best practice of placing paramount focus on safety through the implementation of a management system, in recognition of the fact that an established management system that integrates safety culture in to normal operations is better positioned to deal with extreme events.

Following the normal process under Canadian law for amending federal regulations, the proposed *Regulations Amending Certain Regulations Made Under the Nuclear Safety and Control Act* were published in the *Canada Gazette, Part I* on October 8, 2016 for a 30-day public consultation period. The formal federally mandated 30-day consultation period was preceded by extensive consultations by CNSC staff, with workshops and discussion papers on the proposed amendments. Discussion papers are a means to solicit early feedback on CNSC policies and approaches and play an integral part in the development of new regulations or regulatory documents. Two discussion were published in 2013: DIS-13-01, Proposals to Amend the *Radiation Protection Regulations*, and DIS-13-02, Proposed Amendments to Regulations Made Under the *Nuclear Safety and Control Act*. Notice of the consultations was also posted on the Government of Canada's Consulting with Canadians Web site. The 30-day consultation period allows the public to review and comment on the specific language proposed by CNSC staff. The Commission will make a final decision on the proposed amendments in a public meeting after taking into consideration the comments from the public.

CNSC staff also identified improvement initiatives for its regulatory documents (REGDOCs). Several new REGDOCs were published in specific areas of accident management, emergency preparedness (sets requirements and guidance related to the development of emergency measures for licensees), safety analysis and probabilistic assessment, design of nuclear facilities, environmental protection, nuclear emergency preparedness and response and public information. Specific regulatory documents that were added to the FCF licences and licence conditions handbook include:

- REGDOC-2.10.1, *Nuclear Emergency Preparedness and Response*, published in 2014 which also consolidates existing information from G-225, *Emergency Planning at Class I Nuclear Facilities and Uranium Mines and Mills*, and RD-353, *Testing the Implementation of Emergency Measures*.

- REGDOC 2.2.2, *Personnel Training*; published in 2014 that ensures human and organizational performance lessons learned from Fukushima are incorporated into the CNSC regulatory framework for FCFs, such as procedural development, training and decision making under adverse conditions, minimum staff complement, and human and organizational aspects of emergency preparedness and management.

- Regulatory Document RD/GD-99.3, *Public Information and Disclosure*, published in April 2012, which was developed to enhance licensee communications to the public concerning nuclear activities.

Finally, CNSC also established a standardized licence condition requirement for a safety analysis programs for FCFs and initiated development of REGDOC-2.4.5, *Safety Analysis for Class1B facilities* to clarify its requirements and guidance in terms of what is expected in a 'Safety Analysis Report', and what is expected to be included in a licensees 'Safety Analysis Program'.

Conclusion

The CNSC developed a CNSC staff Action Plan in response to the severe nuclear accident at Fukushima Daiichi nuclear power plant including a request to licensees under subsection 12(2) of the *General Nuclear*

Safety and Control Regulations and a review of Canada's regulatory framework to ensure it meets CNSC mandate. As a result of the review:

- Licensees identified improvements to their installations, equipment and emergency response plans to augment the robustness of their respective facilities and strengthen emergency response.

- CNSC Fukushima Task Force recommendations resulted in several improvements to the CNSC regulatory framework, including modifications to *Class 1 Regulations* and the *Radiation Protection Regulations* and development of several regulatory documents and standardized licence conditions for better clarity of requirements and associated guidance.

Overall, based on licensee assessments and CNSC staff's own independent review of the actions taken, CNSC concluded that the Canadian FCFs are safe and do not pose a significant risk to the health and safety of Canadians or to the environment and licensees have demonstrated a strong commitment to nuclear safety. CNSC staff also concluded that Canada's FCFs have adequate mitigation measures in place to reduce the impact to the public and environment as much as reasonably possible from an extreme natural disaster.

References

1. CMD 13-M34*, Status update on the CNSC Integrated Action Plan: Lessons Learned from the Fukushima Accident*, CNSC, 2013 August.

2. CMD 12-M56, *Status update on the CNSC Action Plan: Lessons Learned from the Fukushima Accident*, CNSC, 2012 October.

3. CMD 12-M40.A, *Update on the CNSC Action Plan: Lessons Learned from the Fukushima Nuclear Accident*, CNSC, 2012 August.

4. CMD 12-M23.B, *CNSC Action Plan: Lessons Learned from the Fukushima Nuclear Accident*, CNSC, 2012 May.

5. Terms of Reference, CNSC Task Force Review of Japan Nuclear Event: Implications for Canadian Nuclear Power Plants, CNSC, 2011 April.

6. *CNSC Staff Action Plan on the CNSC Fukushima Task Force Recommendations*, CNSC, 2011 December, INFO-0828.

7. Report of the External Advisory Committee Examining the Response of the Canadian Nuclear Safety Commission to the 2011 Japanese Nuclear Event, 2012 April.

8. *The Official Report of the Fukushima Nuclear Accident Independent Investigation Commission - Executive Summary*, The National Diet of Japan, 2012.

9. *Fukushima Nuclear Emergency EOC After Action Report*, CNSC, 2012 January.

10. *Fukushima Nuclear Emergency EOC Improvement Plan*, CNSC, 2012 March.

The New IAEA Safety Report on Safety Reassessment of Nuclear Fuel Cycle Facilities in Light of the Feedback from the Accident at the Fukushima Daiichi Nuclear Power Plant

Ramon Gater, IAEA

In 2015, an IAEA Safety Report on *Safety Reassessment for Nuclear Fuel Cycle Facilities in Light of the Accident at the Fukushima Daiichi Nuclear Power Plant* was approved for publication. This publication will provide guidance on performing safety reassessments, in accordance with a graded approach for nuclear fuel cycle facilities of all types, in the light of the feedback from the accident. Although it primarily focuses on operating nuclear fuel cycle facilities, this publication can also be applied to facilities that are in design and construction phases. It is not intended to replace any of the requirements or guidance provided by the relevant IAEA safety standards. However, the publication is intended for use in conjunction with IAEA safety standards concerning safety analysis, evaluation of seismic and external hazards, and emergency preparedness and response for nuclear fuel cycle facilities.

This presentation will provide an overview of the publication and its technical contents, as well as feedback from its use by the IAEA Member States. Additionally, recent developments in the concepts of “severe accident” and “design extension conditions” will be presented along with discussions on use of a graded approach in application of the IAEA safety standards in accordance with the potential hazards of the facilities.

==

Session 3 - Implementation of post-Fukushima technical and operational improvements on FCFS
Chairpersons: V. Lhomme (IRSN), Y. Ueda (S/NRA/R)

Optimising the Resilience of Nuclear facilities at Sellafield

Anita O'Loane
Sellafield Ltd, UK

Summary

Sellafield Ltd. has implemented a suite of work to enhance the site response to a beyond design event or severe accident.

Of the 200 plants that contain nuclear inventory, several have common safety functions that must be maintained in order to prevent a significant release to the public. These plants have been designed over several decades to achieve various functions, and with a varying level of technology depending on the time of design. Therefore a range of responses, equipment and deployment strategies have been developed and adapted to manage the response to beyond design basis or severe accident scenarios across the range of Sellafield plants.

In order to determine an appropriate response, technical underpinning was carried out to understand the challenge. Adoption of existing designs from within the nuclear sector and learning taken from external companies enabled the delivery of efficient solutions. It was recognised that a standardised solution could not be applied in all cases. Commercial Off The Shelf (COTS) equipment was used where possible and existing solutions /equipment were reused, if deemed appropriate, to meet the required function.

The combination of events that could be faced cannot be fully predicted in advance. Emergency responders have been provided with a dedicated toolbox of equipment from which a flexible response can be developed. These are supported with appropriate training in equipment deployment and operation, and maintenance regimes that ensure availability of equipment on demand.

1. Purpose

This paper presents an overview of the mitigation and accident response improvements carried out at Sellafield which enhance the site response to a beyond design basis event or severe accident.

This report uses specific examples to demonstrate the Sellafield Ltd. approach to delivering these improvements.

2. Introduction

Following the events at Fukushima, and in response to regulatory requirements based on the European Nuclear Safety Regulators (ENSREG) stress tests, a Resilience Evaluation Process (RESEP) was developed and implemented across the Sellafield site. This process provided insight into the consequences following a beyond design basis event or severe accident at Sellafield[30].

There are around 650 buildings on the Sellafield site, 200 of which contain nuclear inventory. Sellafield Ltd. initiated internal workshops with the plant owners, operators and systems engineers to identify the areas where improvements could be made. The objective of these workshops was to reduce the potential for a >10mSv release event occurring and enhance the site response capability to a beyond design basis event.

The suite of work carried out ensures that the site is able to achieve the following:

1st Resilience Goal: - for plants to be self-reliant for the first 24 hours in a loss of site utilities of Site Black Out .
2nd Resilience Goal: - for the site to be self-sufficient for 3 days (with an aspiration to achieve 7 days) in a Loss Of Off-site Power event.
3rd Resilience Goal: - for the site to be self-sufficient for 3 days in the event of a radiological challenge.

This suite of work to identify potential capability gaps resulted in:

- 138 Recommendations – generated from ONR, WANO, Sellafield Ltd and ENSREG stress test findings
- 353 Guidance on Functional Requirements (GoFRs) and Prompt Improvement Justifications (PIJs) – recommendations generated through the RESEP Process

30 The Application of Resilience Learning to UK Spent Fuel Management Facilities, P Hallington, Sellafield Ltd, IAEA Vienna, 19-22 March 2012, IAEA-CN-209-025

As work progressed to understand the requirements of these recommendations, GoFRs and PIJs, it was possible for SL to more accurately characterise the requirements through the use of less conservative technical parameters. This front end work informed by the Severe Accident Analysis[31] and RESEP studies has allowed a full underpinning of technical assumptions. This enabled the programme to create a fit for purpose response with regard to equipment requirements and prioritisation of resource deployment in an extreme event.

Simple fixes are employed which require minimal operator intervention. Due to the diversity of operations across the Sellafield nuclear facilities it has not been appropriate to implement a “one size fits all” response. This has resulted in a multi-faceted approach, requiring both a range of specific solutions and development of a flexible generic capability. This is a range of equipment which enhances the ability to respond to a severe event, it provides responders with a “toolbox” of equipment from which a flexible accident response can be deployed. This enables response to the wide range of events which could occur at varying levels of event severity on a complex nuclear fuel cycle facility.

In order to demonstrate that the GoFRs and PIJs generated have been satisfactorily addressed, engineering and subject matter expert judgement is used to verify the solution, ensuring that it meets its intended function. This enables the programme to formally close out these issues with either transfer of ownership to the client plant owner or technical justification to support the removal of scope.

This paper will focus on the following areas:

- Issues with application of a standardised solution to prevent/mitigate a release from Sellafield nuclear facilities
- The use of Commercial Off The Shelf (COTS) equipment over bespoke design where possible
- Requirements Management

A concentrated effort has been made to focus on the key areas summarised above. As a result of this pragmatic and proportionate approach the overall programme cost was reduced by two thirds of the original estimate. The programme saving has been reallocated within the business, addressing other high hazard areas in order to achieve an overall site risk reduction.

31 The Safety Assessment of Long Term Interim Storage at Sellafield, A B Buchan, Sellafield Ltd, UK, OECD /NEA/ International Workshop on Safety of Long Term Interim Storage Facilities, Munich 2013

Significant progress has been made in implementation of the programme, making the site more resilient against beyond design basis or severe accident scenarios.

3. Optimisation of Supporting Functions

Utility Supply

The stress tests and workshops identified that there was a reliance on the Emergency response and also Site Utilities.

In order to increase the site response to a beyond design basis loss of power, additional Mobile Diesel Alternators were purchased. A suite of work identified the facility power requirements in an extended loss of power scenario. It was recognised that the higher the power output rating from a Mobile Diesel Alternator, the more flexible and versatile it was to a beyond design basis event.

Standard Mobile Diesel Alternator sizes were introduced across site. This reduces the risk of error in deployment and also increases Mobile Diesel Alternator flexibility. In addition to a more efficient deployment, standardising the units reduces the maintenance burden for the equipment.

In addition to the standardisation and increase in the utilities Mobile Diesel Alternator fleet, the programme also prepositioned dedicated Mobile Diesel Alternator sets to the areas that are known to require additional power supplies in an extreme event. This helps to reduce the reliance on the utilities team to deploy power generating sets during an event.

As with several other pieces of dedicated equipment, these MDAs have been designed and located using the Sellafield site flood model to predict likely flood levels and flood risk areas. This minimises the effect of flooding risk on Mobil Diesel Alternator supply as far as possible response in a flooding event.

The supporting power arrangements can readily be extended for longer periods using existing on site stocks of consumables. Longevity will be dictated by available stocks of fuel, Sellafield has access to fuel to support emergency arrangements for several weeks.

The Mobile Diesel Alternators have been trailer mounted and are stored in multiple and diverse locations around the site and are expected to be deployable following a seismic event, using Sellafield Ltd.’s existing transport fleet which have the ability to use diverse route options to get to point of use at all key buildings.

The trailers also ensure that the equipment sits above the expected flood level based on the Sellafield flood model.

Offsite Monitoring

RIMNET (Radioactive Incident Monitoring NETwork) is the national radiation monitoring emergency response system set up by the UK Government to help deal with the consequences of overseas nuclear accidents. It is also used to support the national response to domestic nuclear accidents and to collate routine (pre-event baseline) environmental monitoring data. Sellafield's offsite monitoring capability has been enhanced with RIMNET Gamma Monitoring units which are simple to operate and can be rapidly deployed. It provides real time monitoring of gamma radiation, uses GPS to transmit its location, can be secured in a public location and is a low power design (battery life >100 hours). The units can also be used for reassurance purposes where there is no increase in gamma dose rate from normal background levels.

The fleet of district monitoring vehicles was expanded and include a combination of installed and handheld instrumentation designed to measure, record and transfer radiological and meteorological data from the vehicle to a tactical control centre in real time. They have the capacity to measure particulate in air, volatile radioactivity, ground deposition and are capable of isotopic characterisation. Air within the vehicle is monitored for crew protection. The vehicles are deployable to a wide range of locations, making them particularly useful for tracking plume centre lines, identifying areas of maximum radioactivity (hotspots) and monitoring at population centres. Operating instructions, training and personal protective equipment are used to minimise the risk of radiological exposure of operators.

Fuel Management

A fuel management strategy has been developed to support the transfer of fuel for continued operation of standalone equipment in an emergency.

This strategy considers the site fuel demand, location of fuel stores and provides recommendations on the location of mobile and temporary bowser storage to support emergency operations. The strategy provides guidance on how fuel can be transferred during a resilience loss of power event, the transfer equipment and the man resource required to maintain supply in an emergency. It also considers the normal maintenance of emergency fuel stocks, ensuring that fuel degradation will not affect equipment performance.

It is recognised that fuel stocks and demands may change across site as Sellafield is constantly reducing both conventional and radiological risks. Therefore agreement is in place at a site level to maintain a minimum stock of fuel at all times which can support a minimum 14 days of emergency operations. There will be a periodic review to ensure the strategy remains relevant.

4. Issues with application of a standardised solution to prevent/mitigate a release from Sellafield nuclear facilities

Due to the varied nature and design of plants across the site, a standardised solution cannot be applied to maintain the same Critical Safety Functions on different plants.

Several plants across the Sellafield site generate hydrogen as a result of radiolysis. Following a Loss of power or beyond design basis event, this has the potential to build up to dangerous concentrations in some plants, ignite and cause a loss of primary containment. The example below considers two of these plants.

Although the fault is the same, the plants have a very different design. Therefore it was not possible to apply a common solution. Mitigation of the fault is achieved through two different methods.

Plant 1

A passive venting system was designed and installed on Plant 1. If all other forced extract ventilation systems are lost, the passive ventilation stacks will allow hydrogen arising within the facility storage compartments to be released safely without the need for Operator Intervention.

The passive ventilation system is based on the density differential created between hydrogen and air, driving a gas exchange between two areas. The hydrogen generated by the waste will create a region of lower density in the compartment ullage compared to atmospheric air in the operations area. The hydrogen rises and leaves the ullage via a ‘vent’. This release of ullage hydrogen causes the higher density air from the operations area to be drawn into the ullage via a separate vent. The rate of exchange of air between the ullage and the operations area is a function of two factors:

1. The density difference i.e. hydrogen concentration
2. The resistance in the system i.e. the vent diameter and length

A full scale test rig was developed which released hydrogen from beneath water to fully simulate the condition found on Plant 1. The trials were conducted using the plant’s most onerous chronic hydrogen release rate and minimised the environmental factors that could in reality aid the dispersal of hydrogen i.e.

temperature and air flow. The vent diameter was sized to ensure a safe hydrogen concentration in the ullage of less than 4%.

Passive ventilation is provided by penetrations between the operations floor and the compartment ullage space. In order to protect workers from dose uptake and also prevent the penetration being blocked by operator action, a shielding assembly is located over the penetration (see figure 4). The shielding assemblies are tolerant to a 0.25g seismic event. A third penetration/passive vent provides diversity of location and therefore a level of fault tolerance for potential impacts from falling objects in a seismic event.

An accumulation of hydrogen in the building structure is not deemed credible. This is due to:

- the large volume of the overbuilding
- structure leak paths,
- provision of numerous openings in the structure including vents in the overbuilding roof

The benefits of using a passive venting system are:

- provides management of chronic hydrogen against external hazards
- requires no external services
- requires no operator intervention

Plant 2

In order to prevent hydrogen deflagration(s) within Plant 2, purge air must be maintained and there must be a clear ventilation route from the vessels through the plant off gas system.

Electrostatic Precipitators are used in the Plant 2 ventilation system to knock out vapour and particulate which then drain into a seal pot. In order to maintain a clear ventilation route these seal pots are emptied using steam ejectors.

Following an extended loss of services or beyond design basis event a steam supply is used to empty the seal pots and maintain the vent route. In order to maintain the hydrogen purge a supply of compressed air must also be available to the plant.

Therefore work was carried out to enhance the resilience of the steam and compressed air supplies. Due to the Plant 2 design it was not be possible to retrofit a passive venting system as developed for Plant 1.

Two units were procured, one to generate high pressure steam and the other to produce compressed air.

The units consisted of a series of Commercially available, Off The Shelf (COTS) items including power generation (see figure 5) with a sufficient volume of fuel to meet the Sellafield Ltd. Resilience Goals. Both units are housed within standard ISO freight containers and can be moved if required elsewhere on site. This enables Plant 2 to maintain the Critical Safety Function without intervention from beyond the plant team for 24 hours. These arrangements can readily be extended for longer periods using existing on site stocks of consumables.

The units were designed to start up on demand, ensure that operator intervention is minimised and that the level of skill required to operate the equipment is low. On loss of mains power supply both the compressor and steam boiler units automatically switch to a diesel fuelled generator power supply. The compressor has been sized to supply the normal plant compressed air load as this minimises operator intervention to direct supplies within the plant to the area that needs it.

Although plant 1 and plant 2 are challenged by the same risk, the risk mitigation can only be achieved by taking two different approaches. This is a result of very different plant designs to achieve different operating functions.

Several examples exist across site where individual solutions are required to address the same risk. Where common solutions have been established, work has been required to ensure that the interfacing components are compatible. This is a consequence of evolving plant designs and standards over several decades.

5. The use of Commercial Off The Shelf (COTS) equipment over bespoke design where possible

It was recognised at the beginning of the programme that while the technical underpinning to support development of efficient solutions was being carried out there was a period of time where the site would be vulnerable to the risks identified. Therefore engineering judgement, based on consequence and risk, was used to purchase items which would mitigate the interim risk. High level functional requirements were established, COTS equipment was purchased to meet these requirements and embedding/training implemented.

Where necessary, these items were reassessed against the enhanced technical understanding to ensure the required function could be achieved. Interim measures have been justified as permanent solutions where appropriate.

The equipment must be available when needed. COTS equipment has been used where practicable as it is readily available and potential spares are more widely distributed and simpler maintenance compared to a bespoke design. It could also be delivered in a quicker time frame due to reduction in product development compared to a bespoke system. Close working relationships were developed with suppliers and any functionality that drove a bespoke solution was challenged. When key functionality could not be achieved via COTS, stakeholders were consulted and advised of the impacts to cost / schedule.

The combination of events which may be faced cannot be predicted in advance. A concentrated effort was applied to marrying the equipment provided, engineered or otherwise, with emergency arrangements to manage the risk as opposed to attempting to engineer out all conceivable risks in advance. Therefore a suite of flexible generic equipment was identified and provided, enhancing Sellafield Ltd. ability to respond to a severe event. The equipment supports the critical safety functions:

- Cooling
- Containment
- Control
- Chemistry
- Ventilation
- Shielding

This equipment provides responders with a “toolbox” of equipment from which a flexible accident response can be developed. It was recognised that the toolbox needs to provide sufficient depth (equipment) to facilitate an effective response during an emergency. Where possible equipment was selected that required minimal skill, intervention and effort to deploy. This approach was carefully considered against a set of bounding cases in order to demonstrate the ability to respond reliably to the event range which could occur.

The programme is managing the deployment, care, maintenance and drilling/exercising of this equipment in order to maintain the capability.

Pumping Capability

The Severe Accident Management Strategies workshops identified a number of requirements relating to the supply of cooling water, emergency pumping capability, water demand, water stocks and fire water demand.

It was identified that these requirements could be satisfied through the supply of flexible, transportable pumping equipment. Therefore a range of pumps and pumping arrangements were examined by Sellafield Ltd. looking across the industry and others to obtain Learning from Experience on similar applications. This work concluded that a COTS arrangement was available which satisfied the functional requirements necessary to meet the site water pumping demands following a beyond design basis event.

The Hytrans pump set was selected due to:

- ease of deployment;
- compatibility with Sellafield water piping supply;
- optimised suction side layout, minimising risk of pump failure;
- ability to move large volumes of water quickly from various sources around Sellafield site.

An additional benefit from use of the Hytrans equipment is that it is nationally recognised by emergency responders as a standard item and can be operated by a large number of national responders. This includes local county fire and rescue services. As a result a number of site personnel and likely responders are already trained in the use of this equipment. Fifty of these systems are already in service within UK Fire and Rescue and are in regular use.

This COTS equipment has been used for similar scenarios across the UK and as such is recognised as being a ‘tried and tested’ solution for resilience scenarios such as flooding.

Diesel driven pumping units have also been procured which can provide movement of water at a more local plant level. They are lightweight, portable and are capable of providing high flowrates and high pressure water. Several of these pumps have been dedicated to plants to maintain their critical safety functions and units are also available as part of the flexible generic toolbox equipment. The pumps are electric push button start, therefore they are charged in between use to make sure they are available on demand. They can be operated and deployed by a single person in an emergency.

Metal Fire Fighting

The overall site response to a metal fire, or conditions that could result in a metal fire, were improved through the provision and implementation of off the shelf metal firefighting products. A review was undertaken to identify potential types of metal fires on the site along with suitable firefighting media both COTS and in development. The metal firefighting media and equipment was chosen to meet the Sellafield requirements across the range of scenarios identified. For example, in order to deploy specific dry powder for alkali metal fires, long length lances were attached to the system to enable application within a store compartment. It is well understood that the response time to unstable conditions is crucial in minimising the release from and propagation of a metal fire. Therefore additional equipment was installed on specific plants and also made readily available on the Sellafield Fire and Rescue tenders. This equipment is compatible with existing firefighting trolleys on site and with the vehicle transporting mechanism on existing Fire Appliances.

Aerial release abatement - wetting

In order to mitigate the risk of aerial release, water misting sprayers and water supplies are available local to plant. These provide suppression of aerial particulate and wetting of dry material, to agglomerate particles and significantly reduce the potential for dispersion. The units have a large throw distance and a range of throw angles to enable flexible operating. They are commercial units, typically used for dust suppression on large scale construction sites.

In addition to this, new fire tenders have been purchased and are available for use on site. The tenders can be deployed remotely and are capable of foam spraying, thus can also be used to stop material from drying out.

Where water release has the potential to deposit solids during movement, tarpaulins and water spraying/dilution using the small size flexible generic pumps can be used to prevent drying and aerial mobilisation of particulate. Fixative sprays discussed below immobilise areas of contamination where sludge has reached (or almost reached) dryness, preventing any secondary aerial release.

Decontamination – strippable coatings

There are a range of scenarios that give rise to loose contamination in the open environment, therefore a number of strippable coating products were considered as part of a Response to an event on the Sellafield site. The requirement was to prevent an event getting worse in terms of inventory release or geographical area affected.

Historically these types of decontamination products have been used in nuclear facilities such as glove boxes, cells and fume hoods to remove loose or mildly adhered contamination. Work was carried out to measure their performance on materials typically found in the open environment such as concrete, brick, tarmac, cladding and foliage. This allowed the programme to identify the conditions under which adequate spraying can be achieved to further inform a specification for equipment requirements.

There is potential for the equipment motive force to cause re-suspension when in close proximity to loose contamination, analogous to a bow wave. Therefore commercially available, diesel powered airless positive displacement systems with spray nozzles were selected and trialled for fixative deployment. A removable shroud minimises fixative dispersal by wind / air disturbances. The system also has a water flush tank to wash lines after use to prevent fouling of hoses and spray nozzles when spraying ceases.

Selection of fixative materials were based on:

- The decontamination factor,
- Speed of application,
- Complexity of equipment required to deploy,
- Degree of coverage
- Time to cure/dry;
- And avoided products that could over time enter external drains or enable contamination to become mobile again.

The fixative materials have a shelf life, therefore to avoid large and repeated disposal costs of a "Resilience only" stock of products; the material is available from stores for 'normal' use. Product(s) are rotated within stores whilst holding sufficient inventory to support the Sellafield requirement.

6. Requirements Management

Resilience has been achieved primarily through addressing key vulnerabilities. The process used to define vulnerable areas of site had the potential to include some "business as usual" activities rather than identifying the key site needs and requirements for a beyond design basis event. This presented a risk to the delivery of the programme mission:- "*To enable Sellafield site to be resilient to a defined set of Beyond Design Basis events, as far as is reasonably practicable, by improving the capability to respond to a Beyond Design Basis event and by increasing the levels of protection to the Public and workforce*"

This vulnerability was addressed in two ways. A Sentencing Board reviewed the plant outputs – Guidance on Functional Requirements or GoFRs and Prompt Improvement Justifications or PIJs. The board would sanction funds and resource to complete GoFRs/PIJs according to overall programme priority.

The delivery team then established a set of more detailed functional requirements, challenging any requirements that were not necessary to meet Resilience Goals.

Underpinning work was completed to remove unnecessary requirements and unnecessary conservatisms. Technical assurance for this work was carried out through a Technical Committee which consisted of Technical, Pre-Operations, Engineering, Safety, Emergency Management and Key Stakeholder functions. This ensured that essential work to achieve the programme mission was delivered and that scope which did not represent a key need was removed from the programme. Non-essential scope that was removed from the Resilience programme was passed on to the plant owner for further consideration as "business as usual".

> Underpinning technical work allowed the programme to fully understand the consequence and specific requirement for response. The programme used realistic technical assumptions rather than conservative values used for safety case assessments. This allowed the programme to focus resource and funding on providing the most effective response to a beyond design basis event.

This approach, which challenges the use of normal Safety Case values, is in line with the philosophy of other nuclear operators.[3]

> A refined understanding of the event gives responders better information on which to base their decisions. This information has been translated into simple guidance documents to assist in deployment of response capabilities. The guidance is intended to aid responders and reduce human factors concerns such as stress under duress. It enables response teams to determine where their concentrated effort should lie. These arrangements are fully integrated into overall programme of emergency exercises.

7. Conclusion

A number of techniques have been employed on the programme to optimise the delivery of a resilient response to a beyond design basis event. This results in a more refined understanding of the site risk and consequence.

Through robust technical challenge and the use of realistic technical assumptions rather than conservative safety case assessment values, the integrated programme teams were able to challenge the initial programme cost estimate and also any unnecessary/deflective work.

---

[3] Safety Considerations for a Wet Interim Spent Fuel Store at Conceptual Design Stage, M Astoux, EDF CIDEN, France OECD /NEA/ International Workshop on Safety of Long Term Interim Storage Facilities, Munich 2013

The flexible generic or “toolbox” approach provides strength in depth and enables the required capability to be achieved using COTS equipment

This optimisation has led to the delivery of “fit for purpose” solutions.

Work to implement the learning from Fukushima has been carried out and is now considered to be Business as Usual.

"Post Fukushima Improvements: How does the French TSO (IRSN) tackle human and organizational factors in stress tests set up by French Fuel Cycle Facilities?"

Session 3

Beauquier Sophie, Menuet Lise (IRSN, Radiation protection and Nuclear Safety Institute, France)

Summary : Five years after the Fukushima Dai-ichi accident, there is more evidence now on the way the human factors and organizational structures and the management contributed to or hindered the resolution of the crisis [7].This paper is focused on the main improvements, as regards human and organizational dimensions, implemented or planned by the French Fuel Cycle Facilities (FCFs) licensees, in the context of the safety assessments (so-called "stress tests") required by the French nuclear safety authority (ASN) after this accident.

Lessons learned ABOUt humand and organizational factors (HOF)

Examples of nuclear accidents like that of Fukushima Dai-ichi offer a rich case for studying how crisis management functions under such dire circumstances. These findings can be applicable to a far broader range of crisis management, and can provide insight in emergency preparedness measures. Learning about how humans and organizations interacted with the event can point to complementary lessons on how to mitigate and manage crises if or when they do occur.

Human and organizational factors were keys in determining the way the Fukushima Dai-ichi accident unfolded. These factors have been integrated in the analysis made in different countries or by international organisations such as NEA [11] and IAEA [6]. In IRSN an analysis have been fully dedicated to human and organisational factors in extreme situations [7]. In this paper, we will focus on following lessons :

- what measures should be taken to improve the feasibility of the human actions required by extreme situation[32] management following a major emergency (earthquake, flood, etc.);
- the necessary skills for dealing with such situations;
- the training and preparation measures that would help teach these skills and provide practice improving the capacity of people to deal with such situations ;
- the psychological support measures for intervening staff that should be put into place before and during crisis management situations.

Assessment of the feasibility of human actions and minimal staff required in extreme situations

The level of the earthquake and tsunami experienced in March 2011 was not anticipated and the complexity of subsequent events was not considered. As a consequence of this basic assumption, there were insufficient infrastructure, protocols, procedures, equipment and training in place to respond to such a situation. As a result, workers were forced to improvise and adapt to the evolving situation under severe time pressure, with recurrent earthquake aftershocks, the risks and fear of additional tsunami waves, sometimes in complete darkness, in areas scattered with debris and objects (potentially contaminated water, high radiation dose rates and a range of other physical safety hazards). The workers' ability to work effectively was also affected by the need to wear cumbersome full body protective clothing and full face masks and, in some instances, self-breathing apparatuses [6].

From the moment the tsunami knocked out power to their control room, the operators were plunged into a

[32] In this instance, IRSN considers an extreme situation to be one involving: real and possible serious consequences, significant time pressure, a high degree of uncertainty as to the origins of the difficulties we face as well as the consequences of any actions taken by personnel involved in managing the situation, a loss of organisational structure and a loss of leadership for the personnel involved, rapid and disjointed events that personnel have to deal with.

situation of extreme, almost complete uncertainty. They did not know at first what had caused the loss of power, and when they did learn it was almost unbelievable. The procedures of the plant had not been prepared for a total station blackout: the manuals were no more applicable; the operators were left to interpret the chaos themselves. Without instruments, without functioning controls and without standardized guidance, the shift team was forced to predict the reactor state according to a limited amount of information. The operators scoured the emergency procedures and manuals with flashlights, hoping for guidance. After the loss of power, however, the operators had access to none of the indicators they needed. Based on the last set of parameters and on their limited experience with emergency shutdowns, they had no way of understanding the situation of the reactor or knowing what, if any, effect their actions had on it [7].

In terms of reactor No. 1's isolation condenser (IC), no intervention manual was available. Repairs to this important mechanism were not able to be carried out quickly enough to fix the loss of continuous current, leading to the failure of this crucial remediation action. Furthermore, in the instruction manual for serious accidents, entire sections of the diagram showing other important remediation operations were missing.

Moreover, while some equipment was available at the Fukushima Dai-ichi NPP, in many cases the quantity or capacity was not enough to address the situation at hand. For example, the station had the minimum number of charcoal masks required, but a sufficient number was not available for all plant workers. Not only did the station fail to have the equipment ready available, but TEPCO was also not prepared to immediately secure and implement a number of the necessary equipment and materials.

It now seems crucial that nuclear facility licensees, as part of the human actions required to manage post-emergency extreme situations, anticipate the intervention conditions that may be faced by intervening personnel (radiological ambience, heat, lack of electricity**,** difficulties accessing certain equipment and facilities**, unavailability of** essential data and information about the status of the reactors and facilities**,** lack of emergency documentation describing what actions should be taken in such a situation**,...).**

Another lesson learned is that the decisions made in the first days after an accident and their implementation depended on the available personnel and equipment on site. When external resources and reinforcements arrived on site (with skills and equipment suited to the situation), the most urgent decisions and actions had already been made by the local staff who had to identify effective but non-predefined solutions for resolving unpredicted problems that arose. The first support that came from the outside was a helicopter, and that did not arrive until five days after the tsunami. It meant that those on site, including subcontractors, were the essential resources for managing the most critical phase of the accident and carrying out the on-site interventions required to manage site safety and secure the facilities. Their knowledge of the site and the difficulties involved in receiving outside put them on the front line. These factors highlight the fact that, when managing an extreme situation**,** plans to send in technical reinforcement teams and materials were not enough to guarantee the organisational capacity required for adapting to an unpredictable context [3]. Indeed, **a staffing plan strategy on how to respond to a multi units, high stress and long duration event did not exist at the Fukushima Dai-ichi NPP. As a result, operators worked for several days with minimal rest, and personnel carried out their functions continuously for a number of weeks.** It now seems important that nuclear facility operators define which actions should be carried out in an extreme situation. Such a list of actions should be based on skills and knowledge that may be available on site without the possibility of immediate reinforcement. As we will see further on, some licensees have integrated this requirement into their strategy by setting a minimum number of personnel to be present on site, including during non-working hours. This staff is meant to be housed in premises built to resist earthquakes and must be present at all time to ensure safe operation and adequate emergency response capability.

Necessary skills for managing extreme situations

Post-emergency extreme situation management requires human intervention within a limited timeframe, leading to the question of what resources are available to analyse the situation, launch proper actions, think up and implement appropriate technical solutions. This is the case for diagnostic actions that should be carried out after an earthquake or for manual actions that are necessary to put facilities into a safe mode.

Subcontractors' skills

Particular attention should be paid to the skills required to properly carry out these interventions. These skills should be immediately "operational", which means that staff should have an appropriate knowledge of the facilities, of the actions that must be taken, and of their associated risks.

On this topic, the operational experience of the Fukushima accident shows that the subcontractors had to be mobilised for specific operations (notably to drive fire trucks). But their contracts didn't foresee the possibility for them to get involved in such unexpected actions. This absence of any anticipation slowed down the implementation of some priority actions (e.g. injecting water into the reactor). Thus, all of the skills available on site after an accident, including subcontractors' resources must be considered when the licensees make their staffing plans.

Non technical skills

When people are faced with unexpected events, they have to be creative to anticipate possible scenarios and corresponding barriers or imagine real-time solutions. Coping with the unexpected often means finding a solution to a problem that has no straightforward solutions. Operators rely on their knowledge, improvisation and imagination skills to create a totally new and original solution.

The Fukushima accident illustrates the importance of these "non-technical"[33] skills in a crisis situation, such as leadership skills (ability to make the team adhere to a common goal and to insure its coherence, etc.), stress management skills, the ability to anticipate and adapt (i.e. the ability to fulfil an unforeseen role), creativity, etc.

Faced with the loss of electricity, the Superintendent of Fukushima site first tasked two of his emergency response centre (ERC) teams to deal with it : the electricity team to assess the damages and prescribe a solution, and the restoration team to work specifically on getting indicators back on line. Finding that batteries to power the indicator lamps were not available on site, the recovery team was able to find four 6-volt batteries from sub-contractors and two 12-volt batteries from buses parked on the site. This realization reflects a kind of improvised solution.

Another possibility not defined in the accident management measures was the idea of using fire engines as a pumping mechanism for injecting water, which the superintendent considered an important option "*based on his memory of indoor pipes soundness in the buildings at the Kashiwazaki-Kariwa NPS after the Chuetsu-oki Earthquake*". The idea of using fire truck came up early, but its implementation was delayed while attempting the other possibilities and understanding the procedures. In one hand restoring lost capacities is not necessarily faster than the development of new possibility. In another hand there is a reluctance to make the decision to test a new possibility in a crisis context. An internal coordination was necessary to decide on the implementation of new solution. In this context, the cross function team was very useful to set up new solutions. The recovery team was particularly helpful in implementing innovative solutions. They took on the responsibility of poorly-defined emergency situations that required several fields of expertise. This team was able to use the available resources in a creative and efficient manner [7].

[33] Here, "*technical*" skills are skills that allow someone to accomplish "tasks" and non-technical skills are those involved with managing interactions, as well as communication, cognitive, organisational, and coordination skills, etc.

In high-pressure environments, professional effectiveness is generally explained by technical expertise for it allows forming rapid and low-cost relevant hypotheses regarding situations. But in critical situations – which need rapid answers but fall out of the field of immediate expertise (new or atypical situations) – technical expertise is not sufficient to drive the decision making process and behaviours [2]. Thus, an area for progress in managing these situations is promoting the improvement of “non-technical skills”

Training and preparation measures

Anticipate stress effects

Unexpected also means stress, and when addressing the effects of stress, they include a loss of willingness to communicate and a loss of communication abilities, a loss of team perspective, the occurrence of aggressiveness against colleagues and team members…

Unexpected events particularly stress leaders, who may be challenged to make faster decisions, under higher levels of uncertainty, with less time to think and consult team members, while the impact of even minor decisions may be huge. Front line managers may lose their leadership very quickly if they lose control of their emotions, or demonstrate they are losing their vision of what to do.

In Fukushima, the Site Superintendent explained that after the earthquake, and then after the tsunami, everybody was in a stunning state so that he asked them to calm down and not stress or panic, and to select sedately which equipment to check [3].

Some of the stress felt by the Fukushima staff arose from the fact that they did not initially have any information about their families. Once they got this information, they were more able to focus on the actions that needed to be taken and the decisions to be made. In this context, it therefore seems important that nuclear facility operators also decide on measures to relieve the staff involved in an accident situation of any of the stress and psychological pressure created by a lack of information about their family members’ safety.

Develop organizational resilience

The lack of preparedness resulted in personnel being exposed to extremely challenging working conditions in trying to respond to the events at the Fukushima Daiichi NPP. This seriously affected their ability to be effective in the mitigation of the accident. The feedback from the accident demonstrates the difficulties experienced due to the plant employees suddenly being confronted with an extreme situation for which they were only slightly prepared and trained. The operators on the field were experienced in running the reactors, although in a way that was very dependent on specific procedures and remote indicators and controls. They were trained for certain types of emergencies, and types that fell outside of their realm did not enter into their training or expectations. The operators were also hampered by something that deepened the uncertainty: lack of knowledge about practice with the emergency systems. Confusion about the functioning of the Isolation Condenser (IC) at all levels and the differences across the reactors hindered the operators' ability to effectively address the response, even with a robust approach to uncertainty. While their experiential knowledge and feel for the reactors proved useful to some extent, the amount of training and practice they had for the IC does not seem to have been enough to give them a sense for that technology in the same way.

These aspects of the operational feed-back of Fukushima crisis show us that organizations need to be prepared for the unexpected, i.e. to be "prepared to be unprepared". This preparation and training do not only mean to prepare technically and organizationally for all identified scenarios. It also means to develop generic competencies and resources within the organization that help the personnel to quickly and flexibly adapt to new situations, to improvise and develop new solutions for unknown problems. In other words, to be resilient in unexpected situations. This kind of human and organizational resilience capabilities needs to be developed under normal operation and well in advance. In this context, the training programs used in high-risk industries are now more focused on teaching participants how to acquire and develop a skill rather than how to execute pre-defined or required tasks; simulating more realistic and complex situations based on existing experiences.

The lessons from the analysis of Fukushima accident shows how difficult it is for an organisation to cope with an accident in critical material and organisational conditions. It leads us to consider the capacity of an organisation to "become resilient[34]" in order to return as quickly as possible to a status that guarantees that, at least, the situation will not get any worse, even in emergency conditions and under strong societal pressure [3].

In this context, it would be valuable to identify factors that might help licensees to become resilient more quickly after an accident occurs, in order to avoid the continuous chain of events that followed the tsunami at Fukushima on the 11th of March 2011. Research works that are being conducted on organisational resilience highlight the fact that in such situations, the organisation's ability : to readjust, to redefine goals, roles, and responsibilities, to create a collective understanding of the situation, to develop shared skills that lead to alternative solutions; are a cornerstone of organizational resilience.

In the case of Fukushima, staff members tried to find ways to reduce their exposure as much as possible and keep working. Workers in the field, whether maintaining the fire engines for water injection or laying cables in an attempt to restore electricity, found ways to negotiate their exposure to risk, mainly by taking turns and limiting their time in the field [7]. In the control room where the workers were stationed, they made incremental adjustments to their working habits: sitting or crouching on the floor, or moving over to the side of the room with the least radiation.

When the shift supervisor Izawa received the order to decide which of the operators should take a dangerous job in the field (everyone was aware of the dangers of going into the reactor building), he started

[34] A system's resilience is defined as its ability to adjust its functioning prior to, during, or following changes or disturbances and thereby sustain required operations under both expected and unexpected conditions [5].

by excluding the young operators. Izawa offered himself, but the other senior operators refused to let him go. Three teams of two senior operators each were eventually chosen by a discussion among the operators. The names of the senior and more experienced operators were written on the white board in order of age. They then began to decide on teams of pairs to do the required tasks.

The adaptability of the workers has been preserved through a decentralization mode of organization. Giving those closest to the system the ability to make decisions and act immediately allows them to quickly reach a consensus and effective decisions. It also seems that these local decision-making processes tend to maintain and reinforce values (like excluding younger operators from dangerous missions) and social structures (by refusing to let the shift supervisor leave the control room for risky work in the field) ([7], [10]).

Human and organizational requirements for FCF frencH sites

Twelve days after the accident at the Fukushima Dai-Ichi plant, the French Prime Minister asked nuclear facility operators to carry out supplementary safety assessments. These assessments involved evaluating the response of these facilities to a wide range of extreme situations: earthquakes, floods, loss of electricity supply or coolant source, managing serious accidents that have a lasting effect on all or part of the site's facilities, and crisis organisation.

The French Nuclear Safety Authority then issued several recommendations for licencees, including some that covered the social, organisational, and human dimensions of crisis management. These requirements include mainly :

- the identification of the human actions required in an extreme situation to quickly provide useful information and data to management and to the control centre for them to be able to take appropriate decisions and launch adequate actions for the resolution of the crisis. This list should

take into account: priorization actions to be taken (based on their importance, their duration, etc.), the intervention times necessary to carry out the required actions, the operational documentation that operators need to guide them in carrying out these actions, the intervention conditions (ambient radiation and heat, darkness, difficulty accessing certain locations and equipment, etc.).

- the definition of the resources and skills required to ensure adequate emergency response, including local resources and external support (such as subcontractors, emergency services, national force);
- the development of a training program for people involved in the resolution of the crisis in order to strengthen the individual and collective capabilities to deal with stressful situations, thus preparing organizations to become more resilient to extreme situations.

Organizational MEASURES PUT INTO PLACE by French FCFs since Fukushima ACCIDENT

As part of the ECSs, the relevant nuclear facility licensees have identified the essential human actions that must be taken in facilities that are likely to be impacted by an extreme hazard.

In this context, they gathered internal experienced people in operational fields (such as safety, criticality, radioprotection,…) so that they could approve the feasibility of the required human actions based on intervention times and conditions. These experts drew on their knowledge of the activities required in post-emergency extreme situations in order to identify any inconsistencies or incompatibilities in the action plan.

Licensees also relied on crisis drills and scenarios to assess the feasibility of the required human actions in as realistic situations as possible (victims with fake injuries, use of smoke machines to reduce visibility, simulated media pressure or inaccessible roads...) and to ensure that the right actions were taken given the intervention conditions that are likely to occur.

These exercises also aimed to simulate potential stress factors (unavailability of the personnel decontamination facility, a facility manager who is not able to reach the crisis centre, delayed arrival of some key crisis managers, loss of information technology or telecommunications hardware, etc.).

In most cases, the exercise scenario is not communicated to the participants beforehand in order to create situations where they have to face unforeseen or unanticipated problems and assess their ability to carry out non-predefined actions in a harsh context.

In addition, licensees identified the skills that would be mobilised in an extreme situation. These skills are for the most part those that are regularly present and mobilised as part of daily operations in the persons of the facility manager, the safety engineer, the security engineer, the local first responder team, the radioprotection team.

At the end of their initiative, licensees did not identify any necessary supplementary skills to carry out the required actions in post-emergency extreme situations. Their analysis did not lead them to develop the "non-technical" skills they deemed necessary in these situations, even though the leadership style and non-technical skills play an important role in managing the unexpected.

In such a context, IRSN considers that nuclear facility licensees should further continue their reflections until they have identified the "non-technical" skills required for managing extreme situations. Supplementary preparedness measures (training, preparation) that would allow the staff on site to mobilise these skills in case of extreme emergency situation should be dealt with.

Among the two major French FCF licensees, one has defined a minimum qualified staff number required for managing extreme situations. To build their case, the licensees invoke on one hand that no staff shortages were noted during the crisis exercises or during the management of real crisis situations. On the other hand, if they are not enough staff on site to carry out the required actions, standby personnel outside of the affected site would be called in. However, this strategy does not take access road accessibility into account.

As for the specific case of necessary skills that might be provided by subcontractors, the licensees mentioned that besides maintenance services led by site support services (maintenance for electromechanical hardware, radioprotection equipment, remote alarm equipment, etc.), they would not call on any service providers in case of a crisis.

The outcome of the Fukushima accident, however, highlights the importance of a site's ability to rely on all available on-site resources while pending external reinforcements, including service provider companies. Such conditions are to *be dealt* with in special agreements. One main reason is they may be present on site when an accident occurs and may have special skills and enough knowledge of the affected facility to contribute to actions that need to be taken after an extreme event. On this topic, one licensee mentioned that the emergency instructions and procedures covering “diagnostic actions” (facility status diagnostic from a civil engineering, radiation, and contamination points of view) that should be taken in a crisis situation are distributed to service providers working in the facility. Tests of how these emergency instructions used by service provider personnel are also planned.

In terms of preparation and training measures, it should be noted that French nuclear facility operators created crisis management trainings several years ago. These trainings present the necessary information for understanding the crisis management organisational structure and how the crisis cell works (understanding roles, emergency instruction sheets, materials).

One licensee included in its training program one session dedicated to understanding individual and collective behaviour in crisis situations (effects of panic, cognitive biases, how teams work together, positive behaviours). Another part of the training is dedicated to an exercise that takes place at night and that aims to confront trainees with a stressful situation: participants are asked to take on a different role than the one they would normally fill in a crisis organisational structure. This learning method seems quite useful in helping people to prepare for crisis situations, which often entail disruptions to the normal distribution of pre-defined roles.

For another licensee, crisis management training discussed improvisation and cognitive biases, which influence decision-making in a crisis situation, stressing the importance of making decisions together to limit these biases' interference. In this context, this operator developed visual management tools to help with specific diagnostic/forecasting tasks that make sharing information and decision-making easier within crisis operational center.

At last it should be noted that one nuclear facility operator plans on creating a unit whose mission would be to make sure that team members would be able to contact their families during a crisis. It is thus really important to make sure that crisis team members have information about their families and reciprocally.

Conclusion

Many different measures have been put into place by nuclear facility operators following the French Nuclear Safety Authority's recommendations with regard to the social, human, and organisational aspects of crises. These measures should be able to strengthen involved personnel's ability to manage post-emergency extreme situations. Nevertheless, it still seems necessary that the relevant licensees:

- take the likely intervention conditions (**ambient radiation and heat, availability of information to pass on, accessibility of different locations or equipment**, possibility of a complication such as a fire, etc.) more into account as they identify and evaluate the feasibility of required human actions in extreme situations;
- identify the "non-technical" skills that are necessary to manage such situations and define any supplementary training measures that are needed to gain these skills;
- implement measures to guarantee the ability of on-site resources to cope with an unforeseen situation and to carry out emergency actions in the first hours after an extreme event, before external reinforcements arrive on site.

When the survival of critical infrastructure such as nuclear facilities is threatened, effective solutions must be found even though significant resources have been destroyed by the accident. This raises the vision of a future in which nuclear safety oversight authorities require operators to demonstrate their ability to implement an effective engineering thinking strategy in an emergency situation and, more generally, to demonstrate their capacity (skills, expertise, methods, etc.) to ensure a rapid transition into resilience [3].

A kind of organizational resilience was mobilised in managing the crisis that followed in the wake of Hurricane Katrina. A study [1] shows that the planned crisis organisational structure was neither efficient nor adapted to bringing the system back to a nominal level of functioning or managing the resulting disruption and destruction. A self-organising structure was then put into place on the initiative of local actors who developed cooperative structures that were totally independent from the planned structures, which were weakened and rendered inoperable by the scale of the damage. In the hours that followed the destruction of communication systems, cooperative structures were improvised on the initiative of local actors. Teams of volunteers spontaneously showed up to reconnect lines of communication and offered their help in providing access to network technology. These teams' intervention allowed connections to be restored between crisis centres and the outside world, sometimes without official organisations knowing what was going on and trying to shut down these actions because they were taking place outside of their control. Such spontaneous interventions are an example of self-organising structures and emerging processes that cannot be anticipated but that can be understood as resilience factors.

The crisis management process after the ERIKA oil tanker sank off the coast of France in 1999 also illustrates the virtues of adaptation, self-organisation, and improvisation that lead to the emergence of parallel local networks that proved to be highly effective in managing the effects of this crisis in a chaotic environment [9].

These adaptation and self-organisation mechanisms were also used by the Fukushima plant personnel, especially in terms of identifying decisions that needed to be made and actions that needed to be carried out in an extreme situation that, for some of them, presented a high risk of exposure to ionising radiation.

In conclusion, it seems that all the research works conducted on organisational resilience that promote innovation, adaptation, and self-organisation should be better integrated into licencees' future reflections on this topic. Researches should focus on the measures to put into place in order to promote "swift resilience" [3] at the organisational and team levels and the possibility of reconfiguring structures to handle unforeseen situations in unfamiliar conditions and in an environment that is particularly hostile to intervening personnel. With organisational resilience as its guiding principle, crisis management will pursue new pathways that for the moment remain largely unexplored.

Beyond human and organisational measures to provide or reinforce in extreme situations management, IRSN also considers a major lesson learnt from the Fukushima accident : « In spite of the fact that TEPCO and the regulators were aware of the risk from such natural disasters, neither had taken steps to put preventive measures in place. It was this lack of preparation that led to the severity of this accident" [12].

Bibliography

1. Comfort L. K., Haase T. (2006), "Communication, Coherence, and Collective Action: The Impact of Hurricane Katrina on Communications Infrastructure", Public Works Management and Policy, Vol 10 n°4, April 2006.
2. Fornette M-P & al. (2014), "How to improve training programs for the management of complex and unforeseen situations ?", HFES Europe Annual Conference.
3. Guarnieri F. et al. (2015), *L'accident de Fukushima Dai Ichi- Le récit du directeur de la centrale. Volume 1 L'anéantissement*. Presses des Mines, collection Libres opinions

4. Guarnieri F. ; Travadel S. (2014) : « L'ingénierie de l'urgence, l'oubliée du retour d'expérience de Fukushima-Dai ichi » Congrès de Maîtrise des Risques et Sûreté de Fonctionnement - IMDR, Octobre, Dijon, France.
5. Hollnagel E., Leveson N., Woods D. D., (2005), *Resilience Engineering. Concepts and Precepts*, Aldershot, UK: Ashgate.
6. IAEA (2014), "Report on Human and Organizational Factors in Nuclear Safety in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant", Vienna.
7. Report IRSN PSN-SRDS/SFOHREX-2015/01 "A Human and Organizational Factors Perspective on the Fukushima Nuclear Accident"
8. Pariès J. (2016), "What the nuclear sector could learn from resilience engineering", IAEA International Conference on Human and Organizational aspects of assuring nuclear safety, Vienna, 22-26 February.
9. Wybo J-L. et al. (2002), « L'Erika : exemple d'une gestion de crise », *Revue de la gendarmerie nationale*, Direction générale de la gendarmerie nationale.
10. Gisquet E. (2016), "Tragic choices at Fukushima" EGOS, Athens, May 2016.
11. AEN Report n°7284 (2016) : "Five years after the Fukushima Daiichi Accident : Nuclear Safety Improvements and Lessons Learnt", OECD- AEN.
12. Official report of the Fukushima Nuclear Accident Independent Investigation Commission (2012), National Diet of Japan

“Feedback of Fukushima Accident for Fuel Cycle Facilities Emergency Organization”

Nicolas SENDECKI (IRSN, France)

Introduction

The Fukushima accident, triggered by an earthquake and tsunami of exceptional magnitude, confirmed that, despite the precautions taken in the design, construction and operation of nuclear facilities, an accident is always possible. In the days following this event, the French safety authorities (ASN) considered that Stress Tests should be initiated without delay for the Fuel Cycle Facilities sites (FCFs). The operators submitted their Stress Tests reports to ASN on 15th September 2011. On 17th November 2011, during a conference jointly organized by ASN and IRSN, IRSN presented its analysis and conclusions to the press and made public its Stress Test report.

To confront these exceptional - but nonetheless conceivable - event, IRSN recommends adopting an additional safety requirement level, entitled “hardened safety core”, which would guarantee that the vital basic functions of nuclear facilities are sustained over several days, thus enabling off-site resources to intervene.

Regarding discussions between nuclear operators, ASN and IRSN, solutions adopted to enhance FCFs emergency management in case of extreme natural hazards are :

- a new emergency management center more robust than those already built on the FCFs to withstand extreme events and to host emergency team members and perform their tasks under safe radiological and chemical conditions on site ;
- a national task force deployed on FCFs site in order to assist the local Emergency Organization and to manage the damaging effects from extreme natural hazard with complementary human and material resources.

- an emergency organization upgraded to take into account extreme events for the Fuel Cycle Facilities sites (FCFs) and to be sure that the following emergency actions will be guaranteed in these extreme conditions :
    - Perform diagnosis of the facilities and the site after an extreme natural hazard ;
    - Help people, injured or not on site ;
    - Alert and mobilize extra emergency teams ;
    - Alert the French authorities and the local population ;
    - Ensure effective flow of information between the emergency teams ;
    - Collect environmental data necessary for the emergency management ;
    - Protect emergency mobile materials permanently available on the site ;
    - Ensure the protection of emergency team members during their missions ;
    - Ensure the interface between neighboring operators and Emergency Organization.

The possibility of an accident as evidenced by the Fukushima disaster have made it necessary for French government to reconsider how its responds to nuclear and radiological emergency. French government decided to write a National Response Plan published on February 2014 that explains how to handle a nuclear or radiological emergency and is a decision-making guide for emergencies.

1. Fukushima Accident feedback to be taken into account for the new Emergency Organization

The Fukushima accident feedback has shown that from extreme natural hazards, a nuclear site could encounter several radioactive and chemical releases over several days from different installations. It might also be confronted with different damaging effects from extreme natural hazards such as fires, explosions, spraying fuel, collapsed structures or debris that could make some buildings inaccessible,... All these events will complicate emergency management on the Fuel Cycle Facilities sites (FCFs) but emergency actions should nevertheless be guarantees in these extreme conditions. Nuclear operators, ASN and IRSN decided that the following actions should be taken into account after extreme natural hazards on the Fuel Cycle Facilities sites (FCFs).

1.1. Perform diagnosis of the facilities and the site after an extreme natural hazard

After an extreme natural hazard, the immediate diagnosis of the facilities and the site is essential. The aim of this action is to characterize the condition of nuclear installations, the degradation of non-nuclear buildings, the conditions of internal and external access near the site and the radiological and toxic conditions on the site. Moreover, it is necessary to establish human losses as a result of the extreme natural hazard : people present on site before the event, those present at the meeting point, the number of deaths. To perform these actions on the Fuel Cycle Facilities sites (FCFs), site patrol and cameras with video material being transferred to the emergency management building have been chosen.

1.2. Help people, injured or not on site

After an extreme natural hazard, the operator may have to assist a large number of people who have been injured. This has to be done in a largely inaccessible and dangerous environment. On Fuel Cycle Facilities sites (FCFs) safety team have the responsibility to search for people who need medical attention. Injured people will be escorted and assisted to the meeting points where they will receive medical attention. An airlock system to ensure a radiological control of the victims will also be in place at the meeting point.

1.3. Alert and mobilize extra emergency teams

The safety team on duty will be able to make independent decisions regarding the safety on site for the first 48 hours. However, in case of extreme natural hazard, people on site will try also to contact emergency team members outside the site. Nuclear operator must dedicate, inside the local or national emergency organization, a person which will be tasked to identify the emergency team members available and the means to get them to the site in a potentially inaccessible and dangerous environment.

1.4. Alert the French authorities and the population

The French authorities and the local population need to be alerted as quickly as possible when there has been an extreme natural hazard on Fuel Cycle Facilities sites (FCFs). The most effective method of communication could be either:

- at least two satellite phones which have been provided to the Fuel Cycle Facilities sites (FCFs) of which at least one is used exclusively to alert the national authorities and the national emergency organization ;

- a new emergency siren, responsible for warning people located in the immediate vicinity, will be located on the roof of the emergency management center building. This new siren is more robust than siren on site because new siren is guaranteed to work in the event of an extreme natural hazard.

1.5. Ensure effective flow of information between the emergency teams

Local and national emergency teams must have quick access to the technical data from nuclear facilities in order to evaluate the technical data and make safety decisions. Monitoring FCFs technical data on site is only available locally. Nuclear operators plan to collect technical data from nuclear facilities by sending patrols to the facilities. Technical data from nuclear installations must be sent also automatically to the emergency management center to limit human intervention as much as possible. These communication means must be withstand extreme natural hazards.

1.6. Collect environmental data necessary for the emergency management

It is essential that local and national emergency teams have, during the emergency management, the environmental data from the site (meteorological data: wind speed, wind direction, temperatures,…). Meteorological measurement mobile means, easily deployed on site after an extreme hazard have been provided on FCF site. Moreover, emergency teams need to know radiological and chemical conditions for all premises necessary for the emergency management, in particular those in which personnel should stay. Radiological and chemical measurements outside and inside building in order to follow contamination on the CFC sites have been provided.

1.7. Protect emergency mobile materials permanently available on the site

Emergency mobile materials and their reinforcement (extinguishing fire, pumps and distribution water, filtration systems, mobile electric generators…) will remain stored in the shed of the emergency center building or in containers protected from the natural extreme hazards. The emergency mobile materials will therefore be continuously available following an extreme natural event during at least 48 hours. Other means will be provided by the emergency national task force if necessary beyond two days.

1.8. Ensure the protection of emergency team members during their missions

Nuclear operators decide to improve the protection of their emergency team members in order to work safely after extreme natural hazards. These protections are tight uniform, masks,...

Radiation protection equipments are also strengthened with new measuring equipment, mobile dosimeter which has been provided on site.

All these materials are available after an extreme natural hazard onsite for all members of the emergency team in order to perform their mission in total autonomy during 48 hours. Beyond two days, the supply of extra materials will be provided by the emergency national task force.

1.9. Ensure the interface between neighboring operators and Emergency Organization

Nuclear operators have indicated to have strengthened coordination with neighboring operators. Chemical and radioactive risks of neighboring facilities must be known between operators. Operator agreements allow to be quickly alerted to any hazard from a nearby operator and coordinate emergency management.

2. New emergency management center

The main adjustment made on FCFs sites by nuclear operators to manage the consequences of several long duration accidents from many installations due to extreme natural hazard are the construction of new emergency management centers more robust than those already built on the FCFs to withstand extreme events. Nuclear operators, ASN and IRSN decided that every CFC sites have a new emergency management centers with the followings requirements.

2.1.Emergency management center accessibility

The emergency management center must remain accessible to emergency team members even during extreme natural hazards. It will be equipped with at least two entrances: main and emergency entrance geographically separated to avoid losing the accessibility of the emergency management center and protected against the extreme natural hazards. For example, nuclear operators must take into account the flooding risk even for a non-flood zone with heightening the building and doors on the CFC sites.

2.2. Emergency management center habitability

The emergency management centers must allow the emergency team members to perform their tasks under safe radiological and chemical conditions on site. The habitability of emergency management centers must be guaranteed that the ventilation system has a constant and reliable power supply. In addition, the emergency team members must live inside the emergency management center independently, as a minimum for 48 hours.

Ventilation system

The emergency management center requires a higher pressure than the outside in order to protect the emergency team members from radiological and/or chemical releases. A filtration system is designed to protect against radiological or chemicals releases from the nuclear facilities on site and outside the site. In the case of non-filterable pollutants, it must be possible to switch the ventilation from an active state to a static state for a short period. Two air inlets must be created in order to avoid losing ventilation due to any pipe blockages. All equipment must withstand extreme natural hazards.

For example, on the CFC sites, if the emergency management center is located near a nuclear power plant, the risk of iodine release from power plant unit must take into account and iodine traps must be provided within the ventilation system. In addition, the nuclear operator will design these filters for a long duration. If the emergency management center is located near chemical facilities with high chemical risks, suitable chemical traps must be provided.

Power supply

The emergency management center must be electrically independent and self-sufficient during the first 48 hours with a sufficient margin to ensure full ventilation operation. In the case of power supply loss, the power back up must be immediate to maintain ventilation and thus allow safe working conditions for the safety emergency team inside emergency building center. Power supply must be totally independent.

Emergency management center must have at least two electric generators (fixed or mobile) resistant to extreme natural hazards. Each of them must have the power capacity to supply the ventilation system and all the equipment necessary for the full operation and accommodation of the emergency management center. Electric generators must be able to operate in any case without maintenance for a significantly period more than 48 hours. If nuclear operators use electric mobile generators, they must be dedicated and pre-positioned to be able to start immediately to supply the emergency management center. Ventilation must be connected directly to the electric generator without using an electrical board.

In addition, the nuclear operator must provide a fuel capacity powering the generator during a time sufficient to allow recovery by the external emergency team. The equipment associated with power generator (tank, pipes, etc.) must be designed for extreme natural hazards.

Decontamination means

The emergency management center must have a decontamination area with water supplies and able to receive the potentially contaminated water. This equipment must be designed to extreme natural events and human capacity potentially present in the emergency building center (emergency team members and stakeholders).

2.3. Emergency management center Operability

The emergency management center is able to host all the emergency team members and the stakeholders. The emergency team members can work in good conditions in case of site isolation. Moreover, the emergency organization has sufficient means of communication and the intervention resources are sufficient and appropriate to accidental situations envisaged on the CFC sites.

Capacity

The emergency management center will be used during extreme natural events. The emergency management center must be designed to host all the emergency organization (internal and external emergency team). Storage area for emergency materials will be added to the emergency building center. Also, if the nuclear site is shared with different operators on site, the emergency team members of all operators can be accommodated in the building. Emergency management centers capacity for CFC sites is from 50 to 60 persons.

Independence of the emergency team

Emergency team members must have sufficient resources to ensure the independent operation of emergency management center for at least 48 hours before to receive support from the external national task force.

Emergency management center includes rest room beds (12 beds), sanitary water supply (showers and tank capacity of $5m^3$), food and water for 48h00,…

Facilities and environment monitoring

Emergency organization should have as soon as possible reliable information from the nuclear facilities to the emergency management center. Technical data from nuclear installations must be also sent automatically to the emergency management centers to limit human intervention as much as possible. These communication means must be withstand extreme natural hazards.

Emergency teams need to know radiological and chemical conditions inside the emergency management center. Radiological and chemical measurements outside and inside emergency management center in order to follow contamination on the CFC sites have been provided.

Communication means

The means of communication of the emergency management center are:

- a new emergency siren, responsible for warning people located in the immediate vicinity, will be located on the roof of the emergency management center building.
- satellite phones used to alert the national authorities and the national emergency organization ;

These communication means must be withstand extreme natural hazards.

3. National Task Force

The deployment of a national task force will help the local Emergency Organization on the FCFs site to manage the damaging effects from extreme natural hazard with complementary human and material resources. National task force was established to provide human and material resources within 48 hours (24 hours for CEA), in order to help nuclear operator that have to brave extreme natural hazards.

National Task Force Organization

Fuel Cycle Facilities sites (FCFs) that brave extreme natural hazards can request national task force from the emergency national organization. National emergency organization in Paris decides National task force mobilization on site after analyzing sites conditions. National task force has many intervention units: a back office unit, a logistics unit, an engineering unit and an intervention unit. These units will be on site within 48 hours (24 hours for CEA). An Emergency unit for delivering first aid (fire extinguishers, rescue victims, protection, reinforcement mobile emergency means,…) can be deployed, and be on site within 12 hours, if necessary.

The members that make up the National task force are from the other sites which are not impacted by the extreme natural hazards. Depending on their skills, National Task Force members will be allocated to one of the units mentioned previously. All the National task force members are volunteers.

National Task Force Missions

Following actions have been identified to be taken into account to help Fuel Cycle Facilities sites (FCFs) that have to brave extreme natural hazards:

- supply additional consumables and equipments including individual protection equipment and fuel ;

- set up a logistics platform to manage stakeholder inputs and outputs on the site ;

- characterize the environmental situation ;

- site decontamination ;

- deploy a mobile and autonomous network for audio and data transmissions ;

- recover the local containment ;

- clear the site access ;

- rescue and protect the sites ;

- develop and share new technologies for the material and logistical resources.

The material and logistical resources used by national task force are from:

- Means mobilized on group sites, other than the emergency site;
- Means mobilized from partners such as EDF ;
- Means mobilized from external providers (fuels, lifting and handling, generator ...).

4. National Response Plan from French government

The possibility of an accident as evidenced by the Fukushima disaster have made it necessary for French government to reconsider how its responds to nuclear and radiological emergency. The national response plan from French government provides reference information on how to prepare for a nuclear or radiological emergency and make the appropriate decisions in the event of an emergency. It covers the emergency phase (including preparation for the post-accident phase), the period in which the public must be protected and assisted and the accident must be handled so that the situation is brought back under control.

Each emergency management stage in the plan is based on constant communication with each type of public and attention to the public demand for transparent information. Its objectives are to ensure that :

- the general public is protected, particularly from exposure to radioactivity;
- injured persons or people who are exposed to radioactivity receive assistance;
- economic and social continuity is not disrupted ;
- the measures required to manage the post-accident phase and restore society and its economic and social activities to normal are proactively implemented ;
- European and international relations are coordinated.

5. Conclusion

The review conducted by nuclear operators, ASN and IRSN for the emergency organization in case of extreme natural hazards revealed three main evolutions :

1. Emergency organization has to be enhanced to take into account extreme events for all the facilities of the site. Fuel Cycle Facilities operators have already reinforced their emergency organization but emergency actions should nevertheless be guarantees in these extreme conditions.

2. New emergency management center must be able to host all the local emergency organization and the habitability of emergency management centers must be guaranteed that the ventilation system has a constant and reliable power supply. The emergency team members must live inside the emergency management center independently, as a minimum for 48 hours. The means of communication and the intervention resources are sufficient and appropriate to accidental situations envisaged on the site. Fuel Cycle Facilities operators have forecasted commissioning the new emergency management center in 2017.

3. The deployment of a national task force will help the local Emergency Organization on the FCFs site to manage the damaging effects from extreme natural hazard with complementary human and material resources. Fuel Cycle Facilities operators would like to have National task force operational from beginning of 2017.

Human Performance Under Extreme Conditions With Respect to a Resilient Organization

Wolfgang Preischl (GRS mbH)

Abstract

The major findings of a task initiated by NEA/CSNI Working Group on Human and Organizational Factors after Fukushima accident in order to share experiences and to further develop knowledge of human and organizational performance under extreme conditions will be presented.

Synopsis

After the Fukushima Dai-ichi accident a number of initiatives have been undertaken internationally to learn from the accident and to implement lessons learned to improve nuclear safety. The accident has shown in particular the challenges in supporting reliable human performance under extreme conditions. Acknowledging that further work is needed to be better prepared for the HOF (Human and Organizational Factors) challenges of the extreme conditions that may be present in severe accidents, the NEA's Working Group on Human and Organizational Factors (WGHOF, one of the working groups for the Committee on the Safety of Nuclear Installations, CSNI) initiated a task with the objectives to:

- share experiences and knowledge of human and organizational performance under extreme conditions,
- identify specific currently applied HOF principles in nuclear and other high risk industries and compare them with the available knowledge,
- provide a basis for improvements and necessary research taking into account HOF issues in the design and use of measures, and

- Make recommendations with the aim to achieve the best level of human and organizational performance as possible under extreme conditions.

In order to move those issues forward WGHOF hosted together with the Swiss Federal Nuclear Safety Inspectorate ENSI a workshop entitled “Human Performance under Extreme Conditions with respect to a Resilient Organization”. The workshop took place in Brugg, Switzerland in February 2014. The workshop was conducted with participation of a number of invited key speakers from academic research and a range of industries, including nuclear. Experts came from nuclear authorities, research centers, technical support organizations, training simulator centers, utilities and from non-nuclear field (aircraft accident investigation, firefighting, military, design of resilient organizations).

Assuming a beyond design basis accident the quality of precautions in the following three HOF areas is contributing to the possibility to prevent or mitigate severe consequences:

- **Human capabilities** (e.g. availability of competencies, individual and collective stress handling).
- Provision of necessary **infrastructure** taking into account human factors engineering (HFE) aspects of the technical systems, work aids and tools, procedures, needed information and physical environment.
- **Organizational** aspects such as defined roles and responsibilities, cooperation and coordination, communication, task and work flow, organizational culture. Multiple organizational levels and entities should be considered to include the broad system of actors coping with an extreme event (e.g. government and regulatory bodies, the licensee organization, and the operating/plant (site) organization).

The major findings of that task will be presented. With respect to the three areas of human capabilities, organization and infrastructure good practices as well as knowledge gaps and needs for further research are described.

The traditional approach to such accidents is to seek improvements in reliability that should prevent recurrence and provide staff with measures (procedures and equipment) that can be applied. The difficulty with this approach is that the increased complexity can lead to unanticipated situations that render the pre-planned responses inapplicable and ineffective. One of the fundamental conclusions from the workshop is that in addition to reliability, the focus should be on increasing resilience through improving flexibility.

Post-CSA improvements at La Hague reprocessing site

Emeline Cluzel and Michel Guillard (IRSN, France)

ABSTRACT: The La Hague site is composed of legacy facilities, facilities under dismantling and reprocessing plants in operation. They are all concerned by Fukushima-Daiichi accident feedback. Eight dreaded situations have been identified for La Hague site and two potential dreaded situations are currently under examination and discussion. The French TSO, IRSN, assessed the definition and identification of dreaded situations and hardened safety core (HSC). This assessment led to complete the safety case. After recalling the reprocessing process, the presentation will detail each of the situations and the corresponding HSC means. The presentation will focus on the loss of cooling of spent fuel pools and show how an ultimate external water supply strengthens defence-in-depth preventing the dewatering of spent fuel assemblies. New mobile equipment has been provisioned and is meant to be deployed by emergency crisis teams. Multi-facility aspects and aggravating factors (fire, explosion…) are taken into account. Implementation of HSC systems and components sometimes requests material modifications of the plant. For existing HSC systems, structures and components, a demonstration of their robustness is necessary. A global update of the post-Fukushima improvements realized so far and of future plans will be part of the presentation. The presentation will include legacy waste facilities. Those facilities do not always meet current safety standards and by extension do not present any safety margins. Priority for such facilities is to retrieve radioactive material and put it in proper packages. But this process being a long-term project, dreaded situations generated by legacy waste facilities are also to be considered.

Outline:

1. PRESENTATION OF THE FACILITIES AND PROCESS
2. COMPLEMENTARY SAFETY ASSESSMENT
3. DREADED SITUATIONS AND HARDENED SAFETY CORE (HSC)
4. PROGRESS ON THE IMPLEMENTATION OF THE HARDENED SAFETY CORE (HSC)
5. CONCLUSION

1. PRESENTATION OF THE FACILITY AND PROCESS

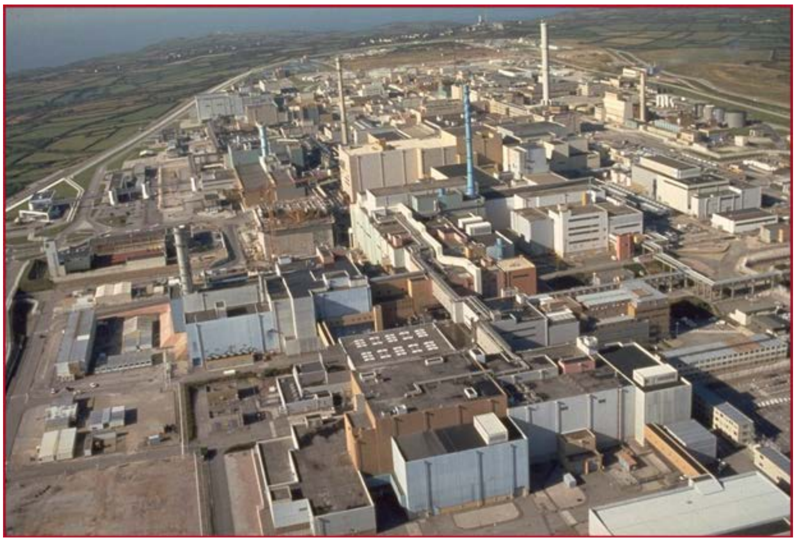


Figure 1: View of La Hague site

La Hague site is composed of two operating reprocessing plants (UP3-A and UP2-800) and several units in decommissioning. Figure 2 below shows the different steps of reprocessing from the unloading of spent fuel to the vitrification of fission products and the storage of resulting canisters.

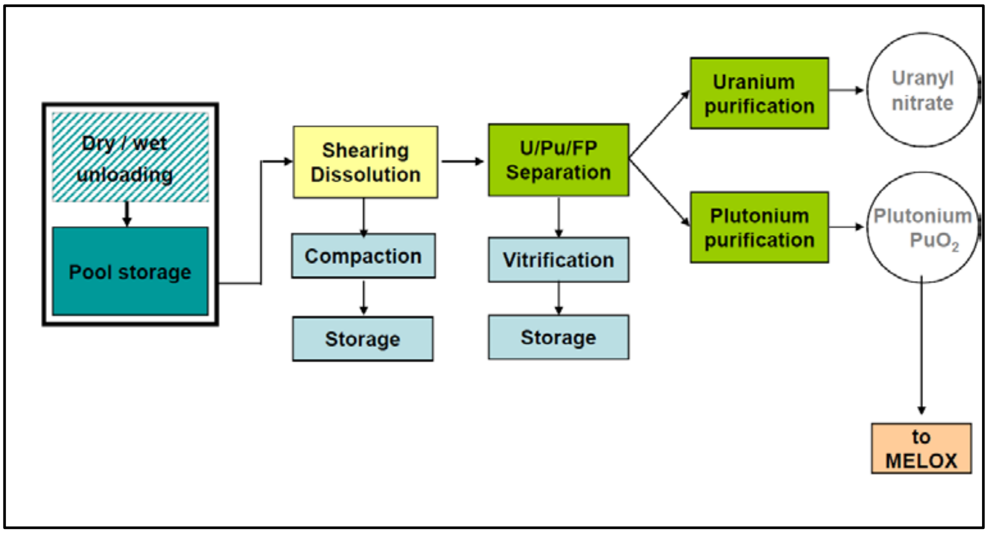


Figure 2: La Hague process in a nutshell

Spent fuel is unloaded from the transport cask either through a dry or a wet unloading unit and is transferred to one of the four storage pools. After a few years in the pool, it is sent to one of the two reprocessing lines. First, the spent fuel assembly is sheared for fuel dissolution. Insoluble pieces (hulls and end-fittings) are separated from dissolved fuel and sent to the compaction unit to make compacted waste canisters. Dissolved material is sent to separation units.

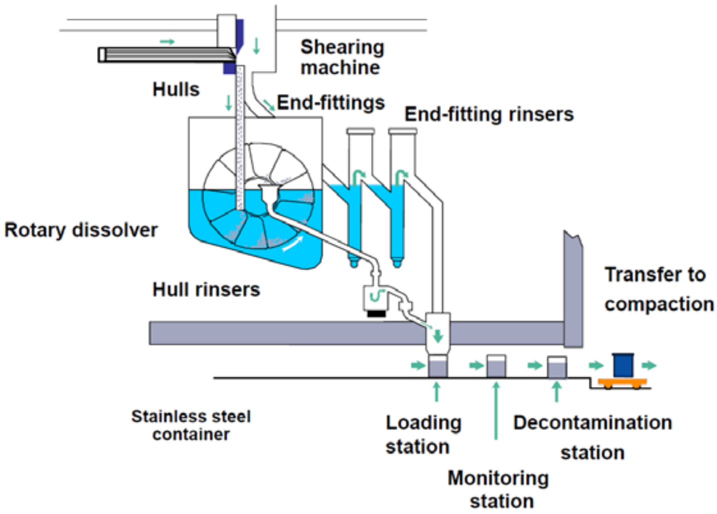


Figure 3: shearing and dissolving process

Separation is based on the liquid extraction PUREX process. First, fission products are separated from uranium and plutonium. Then plutonium is separated from uranium and they are purified so as to be re-used to make new fuel pellets. Uranium, under the form of uranyl nitrate, is sent to Tricastin site and plutonium under the form plutonium dioxide is sent to MELOX MOX manufacturing plant.

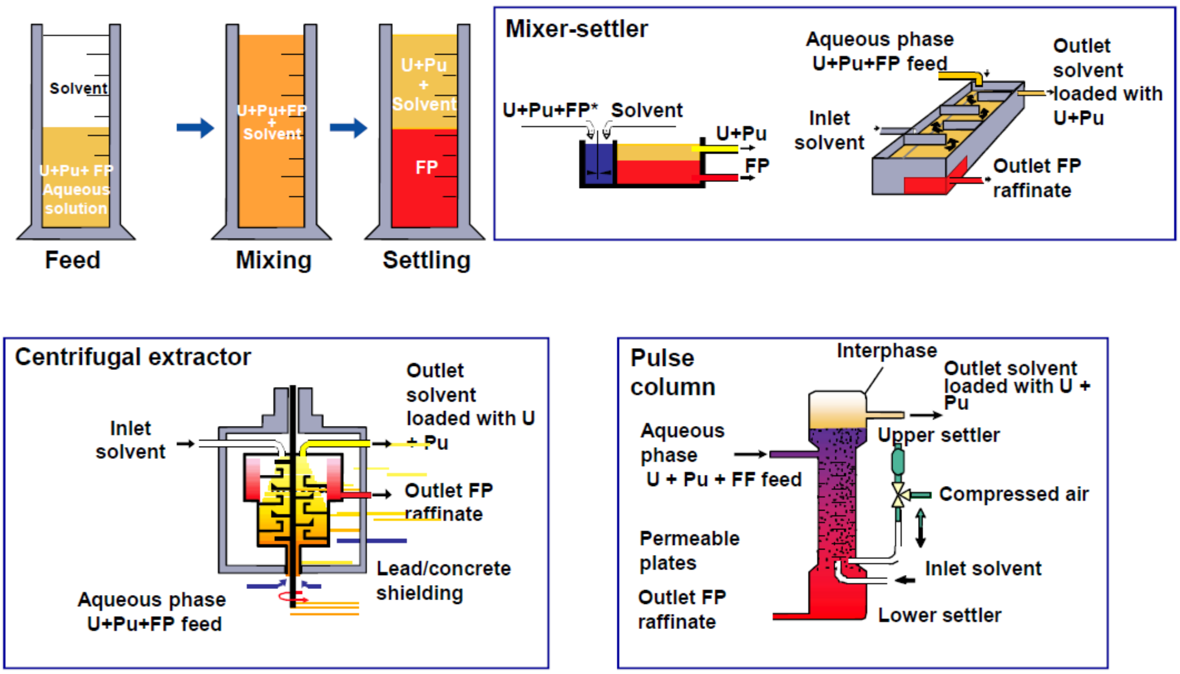


Figure 4: Separation process and devices

Fission products solutions are concentrated and stored in tanks equipped with cooling systems before being sent to vitrification. Fission products and minor actinides are incorporated in a glass matrix directly into a canister. Such canisters are stored in air-cooled pits.

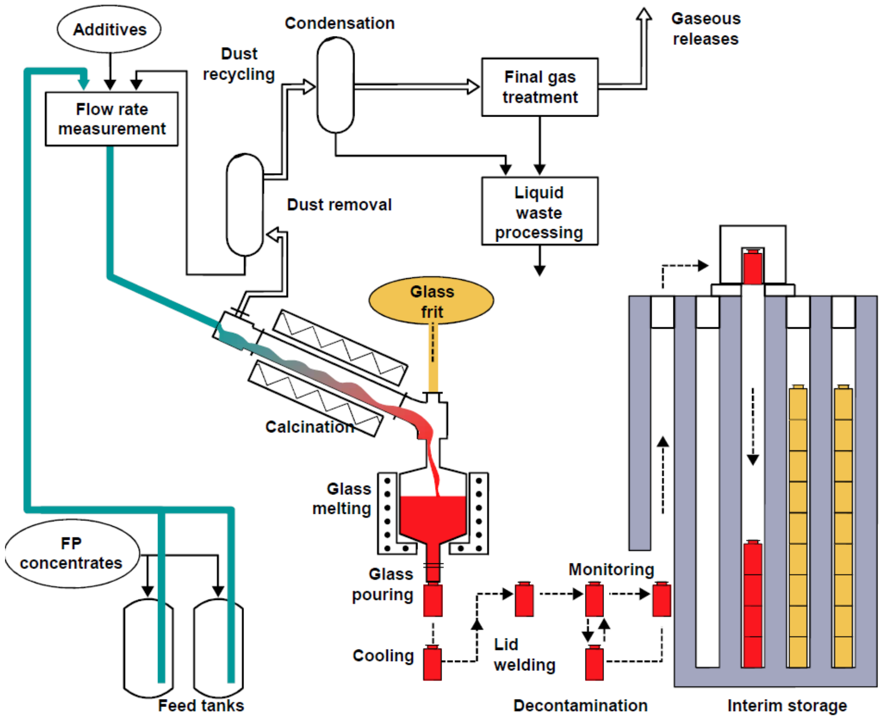


Figure 5: vitrification process

2. COMPLEMENTARY SAFETY ASSESSMENT (CSA)

As explained in article 1 untitled "*Feedback of complementary safety assessments for French FCFs*", the La Hague site, operated by AREVA NC, was considered as a top-priority site regarding Fukushima-Daiichi accident feedback. The licensee performed a complementary safety assessment in 2011. He assessed the robustness of the facilities to extreme natural events and evaluated safety margins. He studied the potential consequences of a total loss of electrical powers (SBO) and of cooling functions (all back-ups and ultimate back-ups lost). He identified cliff-edge effects (see article by ASN for a definition), key structures, systems and components (SSCs) necessary to maintain facilities in a safe state and ensure fundamental safety functions. He also identified potential improvements to safety. He assessed his emergency organization and evaluated his capacity to deal with an accident affecting several facilities at the same time.

La Hague site is situated at the top of a high cliff in Normandy and no major water stream passes by the site. The site is thus not vulnerable to flooding and there is an important safety margin with respect to heavy rain.

Figure 6: La Hague localization

The Cotentin region where La Hague site is situated is not a very seismic region of France. Most of the facilities of La Hague site were designed with respect to a rule called DSN 79 which leads to a design seismic spectrum higher than the actual design earthquake given by current rules for seismic design of nuclear installations. Therefore, the operating plants have margins with respect to earthquake phenomena. On the contrary, facilities under decommissioning or legacy facilities do not meet current safety standards, in particular with respect to earthquake and by extension do not present any safety margins.

The CSA was reviewed and assessed by the TSO IRSN and by expert committees. The first conclusion was that the reprocessing plants could continue to operate; the second conclusion was that nevertheless improvements to safety could be realized. It was also recalled that all facilities had to perform a periodic safety review. These conclusions were written as prescriptions in ASN decisions.

In response to the regulator (ASN) decision of June 2012, released after IRSN and expert committee CSA reviews, the licensee identified the dreaded situations that is to say situations resulting from an extreme natural event or from the deterministic loss of electrical powers and cooling functions which lead to a cliff-edge effect. He also defined a HSC of SSCs meant to prevent severe accident or limit its progression, limit massive radioactive releases and enable the operator to perform emergency actions.

3. DREADED SITUATIONS AND HARDENED SAFETY CORE (HSC)

The licensee identified dreaded situations, depending on the delay before cliff-edge effect (calculated with realistic or standard values depending on the case) and the potential of radioactive or releases. Here is the list:

a. Loss of the cooling function of the spent fuel pools ;

b. Loss of the cooling function of the fission products tanks ;

c. Loss of the cooling function of the condensers of the evaporators ;

d. Loss of the unclogging function of the centrifuge settlers ;

e. Loss of dilution of radiolysis hydrogen in fines solution tanks and alkali rinsing tanks ;

f. Loss of the cooling function in PuO_2 storage pits ;

g. Loss of containment function in legacy waste wet silos.

Each dreaded situation is detailed below and the corresponding HSC meant to prevent the situation and its consequences is presented. HSC is made both of new and of previously-existing SSCs. In addition to HSC SSCs, AREVA defined interfacing SSCs, which are necessary to the HSC. For example, most often, buildings are part of interfacing SSCs. The requirements on the interfacing SSCs are the same as for the HSC SSCs. IRSN considered that distinguishing interfacing SSCs from HSC SSCs is acceptable as long as the requirements for both are equivalent. IRSN also considered that additional and more precise demonstrations of robustness were necessary to be able to categorize existing SSCs as HSC SSCs.

3.1 Loss of the cooling function of the spent fuel pools (dreaded situation a)

The loss of cooling functions of the spent fuel pools leads to the ebullition of water. The first cliff-edge effect corresponds to the loss of the radiation protection provided by water when its level has decreased significantly (2 m), which prevents any intervention by the pool. The second cliff-edge effect is the dewatering of spent fuel. The delays before cliff-edge effects are quite long, compared to other dreaded situations, nevertheless it is considered a dreaded situations due to the severity of potential consequences. The remediation scenario proposed by AREVA NC consists in bringing water from an external robust source to the pools so as to compensate evaporation and prevent spent fuel uncovering. The external water supply is the storm water tank located at the West side of the site. This tank is connected to the Moulinets dam located down the cliff by new HSC underground pipes. The local emergency crew is in charge of deploying the equipment for water remediation. Mobile pipes, of fire hose type, and ready-to-use pumps are stored in trucks and ready to be deployed and connected between the West storm-water tank and a filtration and distribution mobile unit. Other hoses are then connected from this unit to the pools.

Interfacing SSCs	Pool buildings Liners, concrete walls, bearing pads
HSC existing SSCs	Moulinets dam West storm-water tank Water-level measurement devices
HSC new SSCs	Pumps Generators and associated wiring Pipes Trucks Filtration and distribution unit Local connections and distribution Oil reserve Additional water-level measurement devices

Table 1: HSC and interfacing SSCs for dreaded situation (a) loss of cooling of spent fuel pools

Since the conformity of pools of La Hague site is not yet verified and since complete demonstration of the behavior of the pools and its potential aggressors in case of an extreme earthquake is not yet available and could be difficult to demonstrate with a good level of confidence, for more robustness of the demonstration, IRSN asked AREVA NC to consider at the same time as the loss of the cooling function, the draining of the pool due to a breach. For the most recent pools, a pump is installed below the pool to re-inject leaking water. In case of the oldest pool, there is no retention below the pool so water leaking from the breach is lost and must be compensated by an external supply. AREVA has shown that the delays involved and the number and flow rate of the pumps could cover this kind of situation.

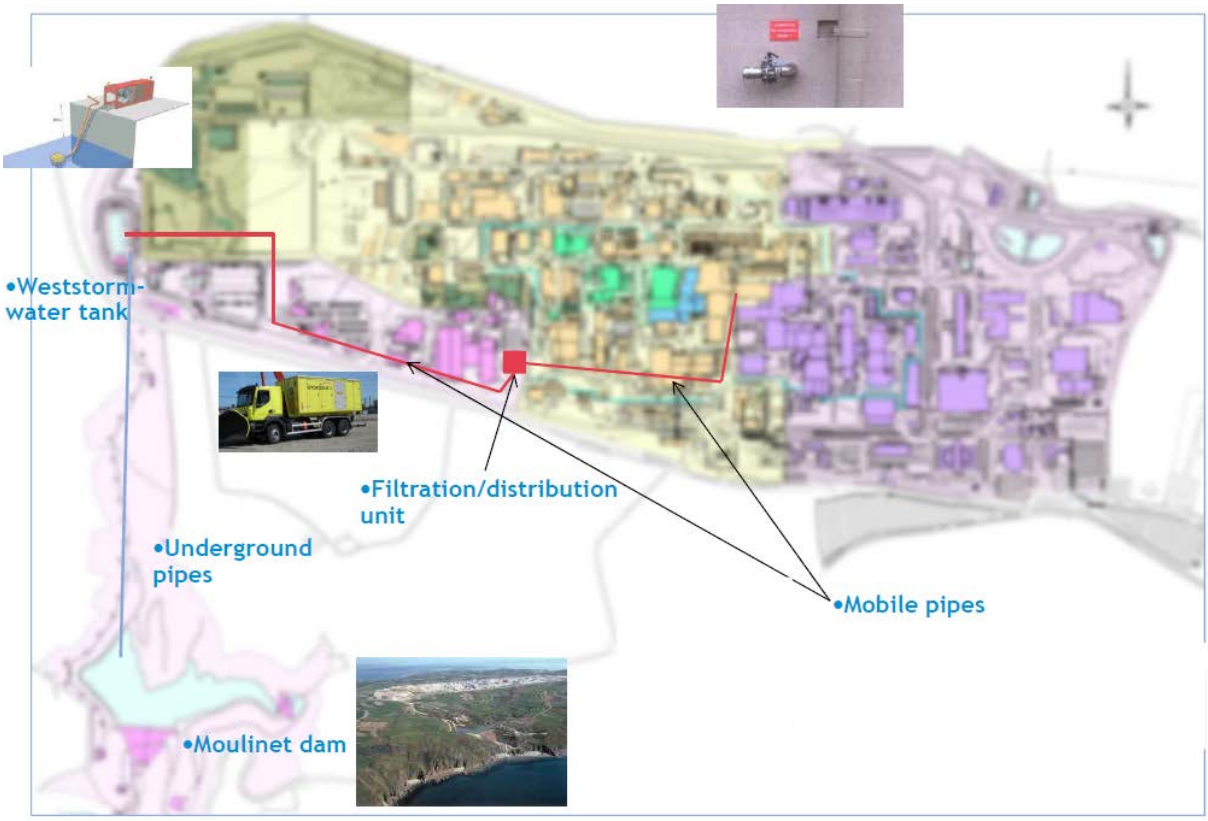


Figure 7: water remediation scheme of dreaded situations 1, 2 and 3

3.2 Loss of the cooling function of the fission products tanks and loss of cooling function of the condensers of the evaporators (dreaded situations b and c)

The remediation scenarios for these two dreaded situations, which could lead to fission products releases, consist in supplying water, in an open loop configuration, to the cooling coils of the fission products tanks and to internal cooling circuits of the condensers of the fission products evaporators. Contrary to the previous case, water can be recycled. AREVA NC is building a dedicated underground tank, robust to extreme earthquake, for water recycling. Basically, the remediation strategy is similar for dreaded situations (a), (b) and (c). The licensee has verified that the number of pumps, the length of fire hoses and flow rate are sufficient to cope with these situations at the same time. IRSN asked AREVA NC to take into account some margins for the flow rate calculation.

3.3 Loss of the unclogging function of the centrifuge settler (dreaded situation d)

The centrifuge settler is used for clarification in workshops dedicated to shearing and dissolution to separate fines (insoluble). This dreaded situation of the loss of the unclogging function of the centrifuge settler is the one with the shortest delay before cliff-edge effect. In case of loss of the unclogging function, it is necessary to rotate manually the cake of fines agglomerated on the bowl of the centrifuge settler to prevent overheat and ruthenium discharges. The tools used by the operators to realize this operation are part of the HSC. This remediation solution was defined by AREVA NC before the CSA.

3.4 Loss of dilution of radiolysis hydrogen in fines solution tanks and alkali rinsing tanks (dreaded situation e)

The dreaded situation is the accumulation of hydrogen from radiolysis up to the limit of inflammability leading to a risk of explosion. All tanks which could lead to accumulation of radiolysis hydrogen up to the inflammability limit in a delay shorter than 120 hours are considered by AREVA NC. Initially, this criteria was fixed at 48 hours but was extended after IRSN review and in response to an ASN prescription. Such tanks are present in several workshops. The remediation strategy consists in supplying dilution air to those tanks through new nozzles close to the high-activity cells. Air is supplied by compressed air cylinders in a first time and then by electrical compressors.

SSCs in interface	- ***Civil engineering units*** - ***Potentially damaging equipment in the facilities dedicated to remediation actions*** - ***Tanks and their anchorages***
HSC existing SSCs	- **Compressed air injection pipes**
HSC new SSCs	- **Fix compressed air cylinders** - **Electric compressors, generators and associated wiring** - **Mobiles pipes** - **Air expansion and distribution system** - **Rotameter** - **Oil reserve** - **Nozzles** - **Wall penetrations**

Table 2: HSC and interfacing SSCs for dreaded situation (e) loss of dilution of radiolysis hydrogen

3.5 Loss of the cooling function in PuO2 storage pits (dreaded situation f)

This dreaded situation is the alteration of concrete by overheat, subsequent to the loss of cooling of PuO_2 storage pits. The remediation strategy consists in supplying power to the extraction ventilators with mobile diesel groups.

SSC in interface	- ***Civil engineering (structures), in particular storage pits***
HSC existing devices	- **Intake and exhaust air networks** - **Exhaust air fans**
HSC new devices	- **Generator and associated power cables** - **Starting devices** - **Switching devices** - **Measuring system for the exhaust air temperature** - **Oil reserve** - **Wall penetrations**

Table 3: HSC and interfacing SSCs for dreaded situation (f) loss of cooling in Pu02 storage pits

3.6 Loss of containment function in legacy waste wet silos (dreaded situation g)

This dreaded situation is the loss of containment of liquid radioactive legacy waste storages which do not meet current safety standards, in particular seismic standard, and leakage to the water table. The kinetics is slow and the remediation strategy consists in pumping the underground water. IRSN considers that this dreaded situation confirms the necessity of retrieving this legacy waste as soon as possible.

3.7 Additional dreaded situations

IRSN identified three additional dreaded situations which are the following:

h. Fire in the glove boxes in the plutonium purification units ;

i. Fire in solvent cells in separation and purification units ;

j. Fire in legacy magnesium waste silos.

Fire in dry plutonium cells and in solvent cells was already identified as reference scenario in La Hague site emergency plan. IRSN considers that the existing design fire prevention measures could be in default after an extreme earthquake. IRSN also considers that a fire in a legacy waste silo could be triggered by an extreme earthquake and that current tests are not very representative of the silo configuration. Consequently, they cannot be used to rule out the occurrence of such a fire. Since the potential consequences of a legacy magnesium waste silo fire could be severe, this scenario is considered by IRSN as a dreaded situation.

Situations (h) and (i) are still under investigation. IRSN has released a recommendation to ASN for AREVA NC to install an automatic shutdown system on seismic detection so as to diminish substantially the probability of ignition of a fire since the major ignition sources are electrical sources.

Scenario (j) is that of a fire in silo 115 or silo 130 which are storages of legacy waste containing magnesium (from the uranium-graphite-gas reactor period). The dreaded situation considered is a fire in the silo initiated by an extreme earthquake. This situation was identified by IRSN and listed by ASN in its January 2015 decision as a dreaded situation. Those silos are not designed to withstand an earthquake so the remediation scenario is not easy to define and has to take into account several possible configurations depending on the state of the silo, its containment and its equipment. Fire detection is based on human detection and the robust mean to extinguish a fire proposed by AREVA NC is water. Water would be pumped in the West storm-water tank as for situations (a), (b) and (c). There is a dedicated pump for this situation.

3.8 Requirements for the HSC

As we see in the details of each dreaded situations, except for what is concerning legacy waste, AREVA NC strategy is based on prevention. There are very few mitigation means. This strategy of AREVA NC is due to the limited confidence in potential remediation means for some of the dreaded situations.

Remediation scenarios must be manageable in case of aggravating factors such as a fire, explosion or chemical releases. Those aggravating factors are taken into account for routes, delays and means definition and design. They are also taken into account for human interventions.

Some HSC equipment can appear for several situations. It was checked that there was enough pieces of equipment to deal with all dreaded situations initiated at the same time. In addition to this equipment, some crisis equipment is part of the hardened safety core. Such equipment is stored in a new robust building which is presented in another paper by IRSN.

HSC SSCs must be highly reliable. All HSC equipment is, by regulation, considered as equipment important to the protection and safety, which means that there are constraining requirements for its design, fabrication, qualification (for new SSCs) and maintenance (for new and existing SSCs). Equipment must be functional under extreme temperatures and since it is part of the HSC it must be robust to extreme events, in particular extreme earthquake. For La Hague site, the hardened safety core seismic spectrum (see article 1) is not much different from the 1979 design spectrum.

Extreme wind and tornadoes stress definition by AREVA are under evaluation by IRSN. There is currently no referenced rule to take into account tornado phenomena for nuclear facilities in France.

The licensee carried out many studies of the behavior of the existing HSC SSCs under the hardened safety core earthquake. Such demonstrations are being evaluated by IRSN. For most SSCs, there is a good confidence. Only oldest facilities are concerned and some components which were designed with little margin or outdated rules.

4. PROGRESS ON THE IMPLEMENTATION OF HSC

ASN 2015 decision imposes that the hardened safety core is operational by December 31st 2016, except for the bunker crisis center. All crisis equipment (trucks…) is already in place, drills are being realized and the building where they are stored is nearly finished. The water recycling tank is also nearly finished.

Reinforcements of the Moulinets dam have been realized and the underground pipes between the dam and the West storm water tank are in place.

AREVA NC has proposed a very pro-active agenda. It is dealing with several building works on the site at the same time.

AREVA NC has realized modifications of its facilities to install HSC SSCs. Some of these modifications were realized under internal authorizations, some were realized under ASN authorizations. For example, IRSN assessed a demand of AREVA to realize new nozzles, which would be used to connect remediation circuits. IRSN recommended that AREVA NC realizes conformity and aging examinations of the existing circuits. Today, what is left to verify is the robustness and conformity of some existing HSC SSCs. For some SSCs located in high-activity cells, conformity is difficult to verify.

5. CONCLUSION

After Fukushima-Daiichi accident, AREVA NC has performed a CSA taking into account all the facilities of the site. It has identified dreaded situations which are equivalent to severe accidents for a power reactor. This was a complex task since the workshops are numerous, very different and handle different substances. IRSN has verified the relevance of the choice of dreaded situation and corresponding HSC SSCs. We can conclude that defence-in-depth has been strengthened by a new layer of dispositions. The HSC, which will be operational by December 31st 2016 on the La Hague site, is meant to be very robust that is why it is important to complete robustness and conformity demonstrations. At this stage engineer judgment cannot be sufficient. It is also important now to prepare emergency teams to use HSC systems and components and verify the feasibility of remediation scenarios with drills taking into account multi-facilities aspects and aggravating factors. IRSN is involved in the assessment of those aspects.

Oversight of emergency planning zones around nuclear fuel cycle facilities and nuclear power plants in Sweden

Angelica Öhrn, Swedish Radiation Safety Authority

The Swedish Radiation Safety Authority has been assigned by the Swedish government to perform an oversight of the exisiting emergency planning zones located around nuclear fuel facilities and nuclear power plants in Sweden. The assignment also includes a change of these zones if necessary. The presentation focus on how the work will be carried out and how lessons learnt from the Fukushima accident will be used when performing the work.

Improvements following the complementary safety assessments for the French fuel cycle facilities and research laboratories and reactors located in the sites of Cadarache, Marcoule, Romans-sur-Isère, Tricastin and Saclay

Session 3

Emeline Cluzel and Michel Guillard, IRSN (Institute for Radiological Protection and Nuclear Safety)

Abstract:
Apart from the reprocessing facilities at La Hague site, subject to another article, some improvements had also to be carried out for the other French fuel cycle facilities, in order to implement, in most of the cases, a "hardened safety core" (HSC) of robust material and organizational measures, as a result of the assessment by IRSN, the French TSO, of the complementary safety assessments (CSAs) performed after the Fukushima-Daiichi accident in 2011. The IRSN conclusions about the dreaded situations and HSC led the operators to complete their safety case. Most of these facilities are operated by AREVA (sites of Marcoule, Romans-sur-Isère and Tricastin) or CEA (sites of Cadarache, Marcoule and Saclay). Due to the processes implemented, some sites are characterised by the predominance of chemical hazards. The other sites are characterised by the predominance of radioactive hazards, with a great diversity of radioactive materials. Consequently, the systems, structures and components (SSCs) included in the HSC may be very different from a site to another. For the existing HSC SSCs, a demonstration of their robustness is necessary, which can be a great deal at stakes for the licensees. These specificities have been assessed by IRSN.

Keywords:
Fukushima-Daiichi accident – French Fuel Cycle Facilities – French research reactors and laboratories - Complementary Safety Assessment - Extreme events - Dreaded situations - Key structures, systems and components - Hardened Safety Core

7. Introduction

Following the accident that occurred on the Fukushima Daiichi Nuclear Power Plants (NPPs) on March 11, 2011, the French Nuclear Safety Authority (ASN) issued decisions requiring the French nuclear licensees to perform Complementary Safety Assessments (CSAs) of their facilities, based on the specifications attached to the aforementioned decisions and consistent with the decisions for the stress tests requested by the European Council. The CSA reports evaluated the capacity of the French nuclear facilities to withstand extreme situations beyond design assumptions, concerning five points: risks of flood, earthquake, loss of power supplies and loss of cooling systems, as well as the operational management of the accidental situations. The licensees identified the dreaded situations, *i.e.* situations resulting from an extreme natural event or from the deterministic loss of electrical powers and cooling functions which lead to a cliff-edge effect.

ASN fixed the deadlines of delivery of these files according to the safety stakes presented by the nuclear facilities: three categories ("Batch 1", "Batch 2" and "Batch 3") of facilities were defined according to a decreasing priority depending on their vulnerability to Fukushima type events and on the importance and the scale of the consequences of any accident affecting them. The top priority facilities ("Batch 1") included the French NPPs, the AREVA nuclear fuel facilities and some of the research reactors and facilities operated by CEA. The other CEA facilities were either at a lower priority ("Batch 2") or considered at the lowest priority ("Batch 3")[35]. The corresponding deadlines for the "Batch 1" and "Batch 2" facilities were respectively September 2011 and September 2012.

The feedback of post-CSA improvements at La Hague reprocessing site is subject to another article in the framework of the present session. The other French fuel cycle facilities operated by AREVA are located in Tricastin site (uranium conversion, enrichment and treatment facilities), in Romans-sur-Isère site (uranium fuel manufacturing facilities) and in Marcoule site (MOX fuel assemblies manufacturing MELOX plant). The facilities operated by CEA are located in Cadarache, Marcoule and Saclay sites (research reactors and laboratories).

The Tricastin and Romans-sur-Isère sites are characterized by the predominance of chemical hazards, due to the processes implemented. Consequently, the dreaded situations for these sites are mainly resulting from toxic and radioactive material: discharge of liquid uranium hexafluoride (UF_6) and discharge of hydrogen fluoride (HF). Concerning MELOX plant, the dreaded situations are the geometrical loss of rod layers in the main rod storage (STE), caused by the heat release of the plutonium-based material and the discharge of plutonium in the

[35] "Batch 3" facilities: ASN asked the concerned operators to transmit the CSAs reports on the occasion of any administrative procedure involving a public inquiry or on the occasion of a facility licensing or, at the latest, within the framework of the ten-year safety reassessment report.

environment, caused by a loss of the plutonium dioxide powder containment after an extreme earthquake.

The three CEA sites are characterised by the diversity of nuclear facility types (reactors, laboratories, nuclear material storage facilities, spent fuel storage facilities, waste treatment facilities…). The corresponding dreaded situations are varied: for example, fire or degradation of the containment barriers in research laboratories, total loss of the core cooling in research reactors…

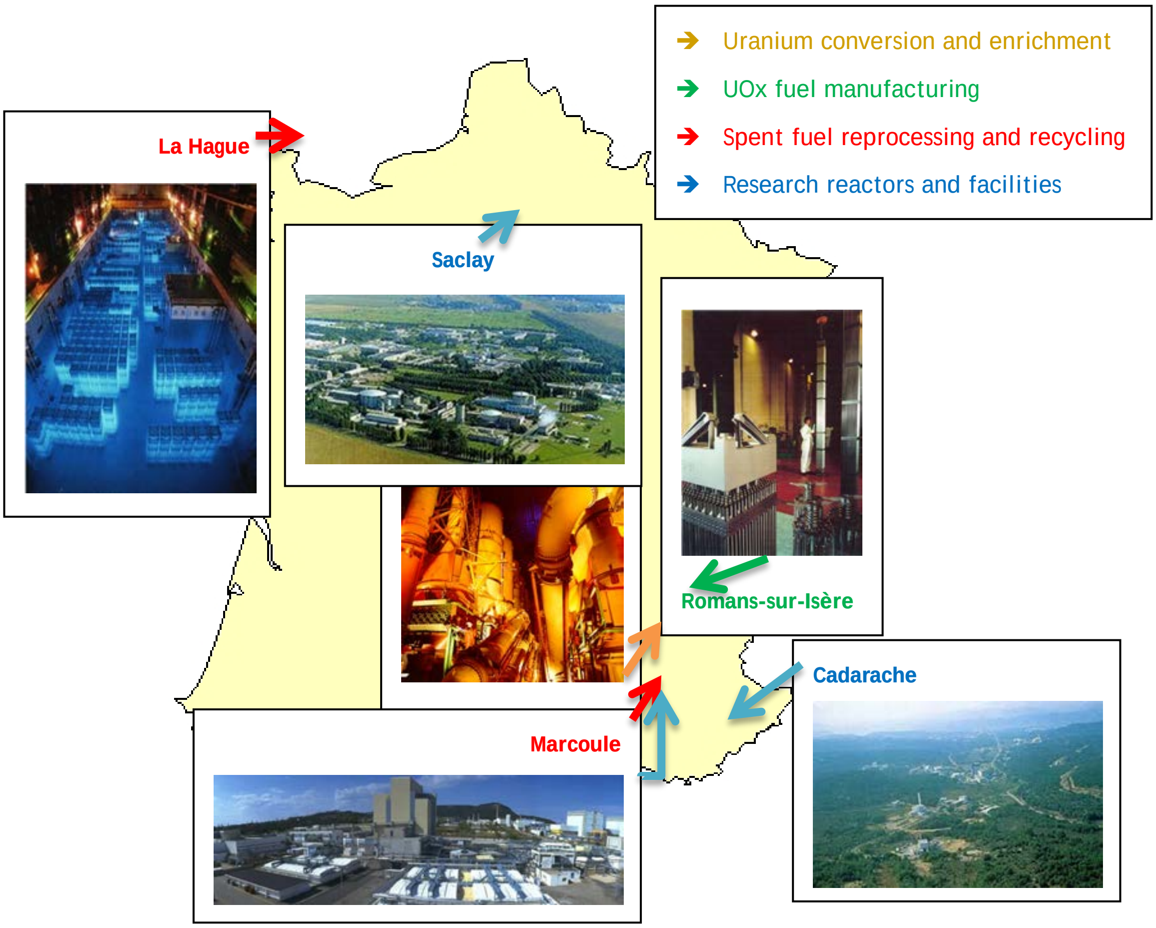


Fig. 1: AREVA and CEA sites

Given the safety stakes and the future of the facilities, the present paper focusses on the aforementioned AREVA sites and on several research reactors and facilities of the Cadarache and Marcoule CEA sites.

8. Site specificities and facility processes

8.1. Site specificities

The aforementioned AREVA and CEA sites housing the French fuel cycle facilities have very varied characteristics, given their geographical position (Paris basin for Saclay, Alps mountain area for Romans-sur-Isère, Rhône river valley for Tricastin and Marcoule, Provence limestone plateau for Cadarache), their different ground (rock, sediments) and climate specificities (oceanic, mountain, submediterranean), the seismic activity in the surroundings, as well as their industrial and natural environment.

The Tricastin and Marcoule sites are at a distance from each other of 20 kilometres along the Rhône river Valley. The AREVA Tricastin site is located at only 2 kilometres from the four NPPs operated by EDF in this site and the MELOX site is at the limit of the CEA Marcoule site. The site of Romans-sur-Isère is in an urban environment, whereas the sites of Cadarache, Marcoule and Tricastin are in a rural environment.

8.2. Facility processes

8.2.1.AREVA sites of Tricastin and Romans-sur-Isère

Uranium conversion and enrichment facilities are located at the Tricastin site as well as facilities dedicated to reprocessed uranium conversion and depleted uranium conversion. The main features of these facilities are the following:

- Uranium conversion is carried out at the COMURHEX facility. It involves a fluorine production unit by electrolysis of anhydrous hydrogen fluoride (HF) and a conversion unit (production of uranium hexafluoride from uranium tetrafluoride by fluorination). Its capacity is about 15,000 tons U per year.
- Uranium enrichment in 235 isotope was carried out, up to 2012, with the two main processes implemented for the uranium isotopic separation, using gaseous UF_6: the gaseous diffusion in the EURODIF plant and the gaseous centrifugation in the Georges Besse II plant (GB II plant). At the present time, only the GB II plant produces enriched uranium. Its capacity is 7.5 million of “separation work unit” (SWU per year), which corresponds to 1,850 tons of enriched uranium.

- Reprocessed uranium (uranyl nitrate) is converted at the TU5 facility and depleted uranium (UF_6) is converted at the W facility. In both cases, the final product is triuranium octoxide (U_3O_8). The reprocessed uranium is converted by a wet process using hydrogen peroxide and the depleted uranium is converted by a dry process using steam water and hydrogen. With the latter, the main by-product is HF.

Given the characteristics of raw materials, products and processes implemented, the main risks are of chemical origins, especially due to the amount of UF_6 and HF stored in the facilities. HF is highly toxic, highly corrosive and highly reactive. In addition, UF_6 is very toxic (radiological and chemical toxicities of uranium) and unstable in presence of water or moisture. In case of a leak, UF_6 hydrolysis is an exothermal reaction which produces a lot of HF ($UF_6 + 2H_2O \rightarrow UO_2F_2 + 4HF$), with formation of a dense white cloud.
Uranium fuel is manufactured in the Romans-sur-Isère facilities. The starting product is UF_6 enriched in 235 isotope, coming from the Tricastin site. The process of uranium fuel manufacturing includes:

- the conversion of UF_6 to uranium dioxide (UO_2) with a step of hydrolysis ($UF_6 + H_2O \rightarrow UO_2F_2 + 4HF$) followed by a step of pyrohydrolysis ($UO_2F_2 + H_2 \rightarrow UO_2 + 2HF$);
- the pellet manufacturing, with a step of UO_2 powder preparation, a step of pre-compacting and granulation, a step of pelletizing under pressure, a step of sintering and a step of grinding;
- the rod assembly manufacturing, with a step of cladding of pellets into zirconium rods, a step of assembly of rods by layer in a frame and a step of assembly check followed by its storage.

The plant capacity is 1,800 tons per year. After the conversion step, HF is recovered from the gaseous effluents with water (scrubbers), giving HF solutions around 50% weight, recycled in the industry. As for the Tricastin site activities, chemical hazards are predominant, because of the huge amounts of UF_6 and HF implemented during the processes. In addition, nuclear hazards depend on the processes carried out (spread hazards, external exposure and criticality).

8.2.2.AREVA site of MELOX at Marcoule

The French MOX fuel (mixed uranium and plutonium oxide) is manufactured at the MELOX plant, located in the site of Marcoule. The process implemented in the MELOX facilities, similar to uranium fuel fabrication from the pressing of powder, includes the following steps:

- reception and storage of depleted uranium dioxide and plutonium dioxide,
- blending of uranium and plutonium dioxides,

- pressing of the powder MOX blend into pellets, sintering and grinding of MOX pellets,
- cladding of MOX pellets into zirconium rods,
- assembly of rods by layer in a frame, MOX assembly check and assembly storage.

The capacity of the facility is 195 tons of MOX fuel per year. Given the characteristics of plutonium (high radiotoxicity, gamma and neutron radiations, low critical mass, thermal power, radiolysis by irradiation of hydrogenated substances), the main risks are of nuclear origins: external and internal exposure, radioactive material dissemination, criticality, heat generation and radiolysis. All along the pelletizing process (blending, pressing, sintering, grinding), safety mainly relies on containment (three static barriers, use of glove boxes…) and mass control. All along the rod assembly process (cladding, assembly), safety mainly relies on radiation protection and geometry control (especially at the rod storages).

8.2.3. *<u>CEA sites of Cadarache and Marcoule</u>*

The following facilities operated by CEA were part of the "Batch 1":

In the Cadarache site,

- the Jules Horowitz reactor (RJH), a new research reactor which will be commissioned in the coming years,
- the MASURCA critical model facility,
- the old MOX fuel manufacturing facility (ATPu facility), shutdown definitively several years ago and under clean-up operations;

In the Marcoule site, the PHENIX sodium metal-cooled fast breeder reactor, which has been shut down and which is still storing around 1,000 tons of liquid sodium.

In addition, the main facilities included into the "Batch 2" were:

- the LECA research laboratory and some storage facilities (nuclear material, spent fuel, nuclear waste), located at Cadarache;
- the ATALANTE laboratory, an R&D and expertise nuclear laboratory in which a wide range of activities linked with nuclear fuel cycle are implemented,
- the common support functions and means on each site of Cadarache and Marcoule, necessary for the remediation or mitigation of any dreaded situations which could occurred on these sites.

9. Dreaded situations and "hardened safety core"

The general lessons learnt from the feedback of CSAs for the French FCFs are subject to an article in session 1.

It is important to recall that, in order to deal with the dreaded situations, the operators have designed on-site new centres of crisis management, housing diagnosis means of the facilities and site conditions (meteorological and radiological conditions), as well as human and material means necessary for the remediation or the mitigation of the dreaded situations. These buildings are designed to be robust to any extreme natural event and are part of the "hardened safety core" (HSC).

9.1. AREVA sites of Romans-sur-Isère and Tricastin

Fig. 2: Romans-sur-Isère site

Fig. 3: Tricastin site

9.1.1.Dreaded situations

The common dreaded situations which may result from extreme natural hazards at Romans-sur-Isère and Tricastin sites have chemical origins:

- A massive release of liquid and gaseous UF_6 during the heating of UF_6 containers (at Tricastin: vaporisation of UF_6 for sampling; at Romans-sur-Isère: vaporisation of UF_6 before its conversion to UO_2);
- A massive release of HF in the environment (at Tricastin: storage of concentrated or anhydrous HF; at Romans-sur-Isère: storage of concentrated HF).

In addition, another dreaded situation is taken into account for a facility at Romans-sur-Isère site: a criticality accident, given the low level of radiation attenuation brought by the facility walls (metal structure) and the short distance to the limit of the site (several meters).

Furthermore, the AREVA site of Tricastin is very close to the EDF site of Tricastin (2 kilometres), which involves four 900 MWe PWRs. This particular industrial environment leads to take into account the consequences of the dreaded situations on each site, which could affect the other site and constitute aggravating effects for the remediation or mitigation actions (e.*g.*: reactor meltdown at the EDF site, massive discharges of UF_6 or HF at the AREVA site).

9.1.2. “Hardened safety core”

The “hardened safety core” (HSC) defined by the licensees for the remediation and the mitigation of the dreaded situations “release of liquid and gaseous UF_6” and “release of HF” are similar at the Tricastin and at the Romans-sur-Isère sites. The following main SSCs are included in the HSC.

For the dreaded situation “release of liquid and gaseous UF_6”, the operator proposed:

- complementary means of detection of UF_6 leak: surveillance cameras (robust to extreme natural events),
- the existing first static containment barrier of UF_6 (robust to an extreme earthquake): UF_6 containers, flexible pipes and isolation valves of UF_6 containers, isolation valves of UF_6 crystallisers, seismic detection and cut-off system (for closing of isolation valves, power and fluid supply cut-off);
- the existing third static containment barrier (building walls, floors, ceilings and roofs robust to an extreme earthquake);
- at Romans-sur-Isère, the existing treatment systems of UF_6 releases (air extraction and scrubbers) inside the concerned premises;
- at Tricastin, a new system (“water wall”) around the “UF_6 emission” building of the W facility (Tricastin site), aimed at reducing the UF_6 and HF spreading in the atmosphere.

IRSN estimated that the existing treatment systems of UF_6 releases at Romans-sur-Isère should be difficult to operate after an extreme earthquake. In addition, IRSN estimated that the leaktightness of the UF_6 second static containment barrier should be improved as well, in order to reduce massive releases of UF_6 in the premises. Consequently, IRSN required the following complementary SSCs for the HSC:

- the second static containment barrier (UF_6 container ovens robust to an extreme earthquake);
- at Romans-sur-Isère, a mobile and autonomous device for the extraction of UF_6 from the premises and for its treatment.

For the dreaded situation “release of HF”, the operator proposed:

- complementary means of detection of HF leak: surveillance cameras (robust to extreme natural events),

- the existing drip-trays under HF tanks and the backup collection pit, including a new liquid detection at the bottom of the backup collection pit and a new fixed piping for HF recovery (SSCs robust to an extreme earthquake)
- a new mobile pumping system for HF recovery from the backup collection pit,
- a new metal decking above the backup collection pit, system aimed at reducing the surface exchange of HF solution with air in the collection pit (reduction of the rate of evaporation of HF).

These measures were estimated appropriate by IRSN.

Regarding the dreaded situation "criticality accident" at Romans-sur-Isère site, the operator proposed:

- to transfer radioactive materials from a part of the concerned building to a new process building, designed to withstand the different extreme natural events,
- the strengthening of the existing building against extreme hazards and of criticality control means to an extreme earthquake (location of nuclear material in equipment, keeping geometry in nuclear material storages…).

These measures were estimated acceptable by IRSN, pending their achievement in the coming years.

9.2. AREVA site of MELOX

Fig. 4: MELOX site

9.2.1. Dreaded situations

The dreaded situations which may result from extreme natural hazards at MELOX plant are due to the characteristics of plutonium:

- Loss or deterioration, after an extreme earthquake, of the high depressure air exhaust system (HD exhaust system) in premises where plutonium dioxide powder is implemented (“powder premises”), leading to plutonium releases in the environment. After such an earthquake, a degradation of the glove box containment is postulated. In addition, the occurrence of a fire in some “powder premises” is considered as an aggravating factor;
- Loss of the cooling system in the main rod storage (STE storage), leading to a heating of the storage which would cause simultaneously geometrical modifications of the rod layers and degradations of neutron screens between the rod layers, combined effects liable to lead to a criticality accident.

9.2.2.“Hardened safety core”

The HSC defined by the operator for the dreaded situation “loss of HD exhaust system” relies on the restart, after repair if needed, of one of the three fans of the HD exhaust system and the maintenance of the availability of the filtration last level. The HSC involved the main existing following SSCs:

- HD exhaust system, redundant seismic valves on the liquid fluids and argon / hydrogen networks and seismic segregation logs of the HD ventilation network;
- Seismic detection and cut-off system;
- Electric power supply of HD fans;
- Third static containment barrier (walls of premises at the building limits, doors, filters, valves…).

IRSN estimated that the availability of the HD exhaust system after an extreme earthquake is not demonstrated and that the maintenance of Pu containment is a top priority. In addition, regarding the numerous sources of fire, the high fire loads in the “powder premises” and the potential degradation of the first static barrier after an extreme earthquake, IRSN has estimated that several simultaneous fires could occur in the “powder” premises. Consequently, the operator had to study complementary static containment means and to implement:

- fire detection means and firefighting means operational after an extreme earthquake, enabling to face several simultaneous fires in the process premises;
- a reinforcement of means enabling to limit the propagation of a fire outside the limit of the second containment barrier.

The HSC defined by the operator for the dreaded situation “loss of cooling in STE storage” relies on the recovery of the rod storage cooling, by providing on the cooling water recyclers with water (pumped directly from the Rhône River) and electric power supply in less than

48 hours after the loss of cooling[36]. The HSC involved the main following SSCs:

- STE storage cooling water recyclers, ventilation ducts between the recyclers and the storage and piping system supplying recyclers with water;
- Wall penetrations and connecting boxes, mobile piping and pump aimed at supplying one recycler with water, water tank and settling system;
- Backup diesel generator.

In its assessment, IRSN underlined that a criticality accident in the STE storage with a degradation of its geometry would necessarily result from the presence of a moderator (e.g. use of water to fight a fire), that is *a priori* possible in the rod storages. Moreover, the criteria taken into account to define the intervention pathways were not specified by the operator. Consequently, regarding a criticality accident in the process premises, considered as an aggravating factor, the operator had:

- to implement a procedure about firefighting means (use of water or hydrogenated products) for the storages;
- to take into account a criticality accident to define the intervention pathways.

In addition, IRSN estimated that the operator had to complete the demonstration of robustness of some parts of civil engineering structures to an extreme earthquake (see § 4.1 hereafter).

9.3. CEA sites of Cadarache and Marcoule

Fig. 5: Marcoule site

Fig. 6: Cadarache site

9.3.1. Dreaded situations

[36] As explained in the presentation "Feedback of complementary safety assessments for French fuel cycle facilities" (Session 1), AREVA has decided the implementation of a "National Task Force" (FINA) designed to face a major accident within their sites or facilities. The FINA mission is to help any site which would be damaged by supplying it, within 48 hours, with human beings and material means. These means are complementary to the means already present on the site, aiming at managing the crisis and limiting the consequences of the accident, in particular in terms of discharges of chemical or radioactive substances in the environment.

The dreaded situations which may result from extreme natural hazards at Cadarache and at Marcoule sites are caused by the nuclear or chemical materials implemented in some of the research laboratories and reactors. ASN specifies the following extreme situations:

- total loss of electric supply,
- total loss of cooling systems or failure of heat sink,
- external aggressions selected,
- situations resulting from the state of the facilities, site and environment after an external aggression.

More specifically, the main dreaded situations are as follows:

- on the Cadarache site,
  - complete cooling loss of the RJH reactor, leading to a core meltdown or spent fuel storage meltdown,
  - complete loss of the core cooling system or fire due to a sodium-water reaction in the CABRI research reactor,
  - fire in the LECA research laboratory cells caused by an earthquake,
- on the Marcoule site,
  - fire due to a sodium-water reaction in the PHENIX former sodium metal-cooled fast breeder reactor,
  - degradation of the containment barriers in the ATALANTE research laboratory caused by an earthquake.

9.3.2.“Hardened safety core”

Some research facilities and reactors did not require any HSC. This was estimated acceptable by IRSN. The HSC defined by CEA concerns, in particular, the research reactors RJH and PHENIX, given the potential consequences if a dreaded situation occurred. The operator strategy relies on the restoration of the cooling function (RJH) before meltdown and on the remediation or mitigation of a sodium fire (PHENIX). The main SSCs included in the HSC are as follows:

- For RJH,
  - existing SSCs for dreaded situation prevention: primary coolant pump, natural convection valves of the primary circuit, piping and valves of the complementary water supply in reactor pool;

  - existing SSCs for mitigation: means of radiological activity measurement, pressure sensor, shut-off valves on ventilation ducts and deflation valves with the associated systems in the reactor building,
  - new SSCs for supervision and crisis management: remote report of valve positions, reactor pool water temperature and level measurements, means of diagnosis of the reactor conditions, mobile intervention means and backup power supply;
- For PHENIX, complementary means of sodium leak detection, water detection, gamma-rays measurement, video surveillance cameras, diagnosis of the reactor conditions, backup power supply and sodium firefighting.

Because of the absence of cliff-edge effects, CEA has not defined any HSC for the other facilities in the Cadarache and Marcoule sites. However, in order to reduce fire hazards after an earthquake in the LECA laboratory, the operator has planned to implement an automatic seismic detection and power supply cut-off system. In addition, given the high temperature processes implemented in this laboratory, IRSN has required means enabling detection and intervention as soon as possible if any fire occurred after an earthquake.

10. Progress on the implementation of the "hardened safety core"

10.1. Extreme natural hazards and dreaded situations

IRSN recalls that, in its decisions sent in January 2015, ASN specifies the seismic event to take into account for the HSCs, defined by a response spectrum which has to:

- *enclose the Safe Shutdown Earthquake (SSE) increased by one and a half times,*
- *enclose the probabilistic spectra defined with a return period of 20.000 years,*
- *include the amplification effects caused by the local geology ("site effects"),*
- *take into account the potentially active seismic fault lines in the vicinity of the site.*

The licensees have answered to the aforementioned prescription of the ASN decisions. The studies sent by AREVA and CEA to define the extreme seismic events for their sites have been assessed by IRSN. Given its conclusions (cf. article "Feedback of CSAs for French FCFs" in session 1), ASN required the operators of Romans-sur-Isère, MELOX, Cadarache and Marcoule sites to perform complementary studies in order, in particular, to take into account "site effects". Likewise, ASN required the operators of Romans-sur-Isère and MELOX sites to check that their probabilistic spectra meet the ASN requirement with a return period of 20,000 years. In addition, AREVA had to estimate the “site effects” on the Tricastin site, by using seismic recording stations, the installation of which was already scheduled.

The current situation is the following:

- The aforementioned complements have been sent, except for the Tricastin site, the operator of which is waiting for sufficient data to be able to estimate the “site effects” on the site;
- The complements sent for Cadarache are satisfactory (rocky soil for setting up HSC SSCs);
- The complementary study sent for Romans-sur-Isère site has been assessed by IRSN, which estimated that the enclosure of the probabilistic spectra defined with a return period of 20,000 years by the response spectrum proposed for an extreme earthquake is still not demonstrated;
- The IRSN assessment of the complements sent for MELOX and Marcoule are ongoing.

Pending the definition of the response spectra corresponding to the HSC extreme earthquake for the sites of Marcoule, MELOX and Tricastin, IRSN has considered that the licensees have to design the new HSC SSCs (buildings and equipment) to be constructed in the short terms with substantial margins (e.g. ASN required for the Tricastin site of taking into account the potential “site “effects” by adding an increase of 30% to the response spectra).

Regarding other extreme natural aggressions, the assessment of IRSN is still ongoing. In particular, the IRSN position about tornado phenomena, including the associated effects (extreme wind, pressure/depressure effects, projectiles), should be finalised in the coming months.

10.2. Design of the new “hardened safety core” SSCs and behaviour of the existing ones

IRSN recalls that the HSC SSCs have to comply with the highest quality standards in terms of design, fabrication, maintenance and operation. **In particular, IRSN considers that the new HSC SSCs have to be designed in considering the actions of extreme aggressions as normal actions (to withstand permanently) and not as accident actions (to withstand in exceptional circumstances).**

This point is subject to a prescription in the ASN decisions sent in January 2015, requiring to the licensees to justify the design of the new HSC SSCs and the behaviour of the existing HSC SSCs. AREVA has sent a general methodology aimed at justifying equipment and civil works integrated in the HSCs (new or existing SSCs). IRSN has estimated that some complements are necessary for this methodology. In particular, regarding the design of the new HSC SSCs, IRSN required to apply rules and criteria relevant to normal conditions, as extreme situations have to be considered like “normal situations”. Likewise, regarding the behaviour of the existing HSC SSCs, IRSN required complementary justifications in order to demonstrate the behaviour control of civil engineering structures when applying the “graduated approach” proposed by AREVA: “basic analysis”, based on the design data, followed, if necessary, by an “interim analysis”, based on the original calculations or on an updated calculation, and then, if necessary, by an “in-depth analysis”, based on specific non-linear calculations.

The revision of this methodology will have to be applied on the AREVA sites to check the design of the new HSC SSCs and the behaviour of the existing HSC SSCs.

Currently, CEA has not sent such a methodology, relevant for the design or the analysis of behaviour of all its HSC SSCs.

10.2.1. Design of the new "hardened safety core"

On each site, the main part of the new HSC SSCs is the new centre of crisis management. These crisis centres include command premises housing crisis teams, premises housing intervention teams and premises storing mobile HSC SSCs (remediation or mitigation equipment). The corresponding buildings and equipment are designed to withstand the effects of extreme natural aggressions, in considering a requirement of functionality. In addition, the command premises are designed to ensure the permanent presence of the crisis teams during at least the first 48 hours following an extreme event, whatever the site conditions (radioactive or chemical release). For this purpose, the crisis centres have their own power supply (backup diesel generator with its oil reserve) and the command premises are operated in constant slight overpressure (ventilation network including convenient sets of filters and spare parts). In addition, the command premises include food and drink reserves, water reserves (showers, toilets) and resting rooms. They also have remote information of the site and facility conditions. Lastly, the crisis centres have external means of measurement for radiological, chemical and meteorological conditions.

ASN has required the remediation and mitigation means to be operational by the end of 2016. The commissioning deadlines of the new crisis centres are between the end of 2016 and 2018, depending on the sites.

10.2.2. Behaviour of the existing "hardened safety core"

The existing HSC SSCs involve buildings or parts of buildings and equipment. Civil engineering structures have to be robust to the different natural extreme aggressions, taking into account interactions between building blocks in case of an extreme earthquake. For example, regarding the MELOX process building, IRSN estimated that the operator had to complete the justification of some particular civil engineering elements (linking galleries, exhaust stack, powder premise ceiling) which could damage the containment function of the building. This complementary analysis from the operator is ongoing.

Regarding buildings with a metallic structure and metal sidings, the justification of their behaviour in case of extreme wind or tornado turns out to be a great concern.

The existing buildings have generally been designed with reference to the Safe Shutdown earthquake (SSE) and the margins taken into account at design for the civil works have been estimated by the operators. In addition, the buildings must not constitute an aggressor of HSC

SSCs. Consequently, in most cases, the operators have carried out reinforcements of civil works and then checked their robustness with reference to the extreme earthquake defined for HSC. Some justifications of the reinforcements are being evaluated by IRSN but most of them are still to be produced by the operators.
As a result, the implementation of a shutdown strategy for the oldest facilities in the short term has been proposed by the Tricastin site operator and the design of a new process and storage building has been planned by the Romans-sur-Isère site operator.

The approach of the operators to justify the existing HSC equipment is similar. In most cases, some strengthening had to be realised (anchorage points, support structures…), taking into account equipment in interface with HSC SSCs and potential aggression of other equipment. Most of justifications are still to be produced by the operators.

Furthermore, the use of new mobile means for remediation or mitigation generally needs adaptations in the facilities, such as the addition of connecting boxes for backup power supply, for water or other fluid supply, the creation of backup cooling loops and of recovery devices for toxic products (HF, UF_6). These adaptations also require being part of the HSC SSCs and designed accordingly, as new equipment.

10.3. Consideration of industrial environment

Industrial environment has to be considered by the operators, in order to meet prescriptions of the ASN decisions.

As it is mentioned above (see § 2.1), the AREVA Tricastin site is located at only 2 kilometres from four 900 MWe PWRs operated by EDF and the MELOX site is at the limit of the CEA Marcoule site. In addition, the Tricastin site is at a distance of 20 kilometres from the MELOX and CEA Marcoule sites. This industrial environment implies, for the operator of each site, to take into account the consequences of the dreaded situations on the neighbouring site, as potential aggravating effects for its own remediation or mitigations actions.

Regarding the Tricastin sites, the operator of AREVA Tricastin site has studied the potential impact of a reactor meltdown at the EDF site and has decided, in addition to the use of self-contained breathing apparatus for intervention teams, to add iodine filters to the ventilation system of the new crisis management building.

In addition, given the kinetics of this kind of accident, the operator is currently studying the way of management of the crisis teams and intervention teams (renewal, reduction of workforces after safe shutdown state in facilities…). Similarly, EDF is currently studying how to manage the workforces on his site if discharges of UF_6 or HF at the AREVA site are impacting the EDF site. Moreover, industrial environment of the Tricastin sites, which involve several facilities with chemical hazards, is also considered in the potential consequences of extreme natural hazards which may have an impact on all this area (for nuclear facilities and chemical facilities).

Furthermore, the Romans-sur-Isère site has an urban environment, including chemical facilities which have not been designed to withstand extreme natural aggressions and could cause discharges of toxic products, in particular ammoniac, as well as combustion products containing hydrocarbons, hydrogen chloride, hydrogen cyanide or carbon monoxide. The operator has estimated that the site should not be impacted by significant toxic effects. In any case, the operator has planned to supply the intervention teams with self-contained breathing apparatus. The IRSN assessment is ongoing.

Regarding the MELOX and CEA Marcoule sites, the potential impact of the dreaded situations on the other part of the site is being studied. In addition, IRSN considers that the impact on the Marcoule sites of a reactor meltdown at the EDF site of Tricastin has to be taken into account, especially for the new crisis management buildings.

Lastly, IRSN notes that most of the operator analyses of impact of on-site aggravating phenomena (fire, explosion, transport of hazardous materials…) are under assessment by IRSN.

10.4. Definition of the "hardened safety core"

In ASN decisions sent in January 2015, requirements concern the definition of the "hardened safety core". The operators had to send a list of HSC SSCs and of SSCs in interface with HSC SSCs (HSC-INT), i.e. SSCs whose functioning or integrity is necessary to the HSC functions, including their functional requirements. The new SSCs and the existing ones had to be clearly identified. The safety requirements had to be the same for HSC SSCs and for SSCs in interface. The operator had also to identify the potential aggressors of the HSC SSCs. The new SSCs have to be designed, constructed and operated in order to fulfil their functions during the time necessary to reach and maintain safe conditions for the facilities concerned. In addition, the operators had to:

- Define the SSCs operating conditions for extreme temperatures and take into account significant margins (at least 5° C) for water loops and equipment using oil or other liquid fluids;
- Design SSCs power distributions as independent and reliable as possible.

The corresponding lists of HSC SSCs and HSC-INT, including the safety requirements, have been sent by the operators. As regards the AREVA facilities, the lists define the HCS SSCs, HSC-INT and SSCs potential aggressor of HSC SSCs. They involve the concerned buildings, the dreaded situations and extreme aggression considered, as well as the HSC function to fulfil and the corresponding functional requirement. Their assessment by IRSN led to consider that they are, as a whole, satisfactory.

The assessment of the CEA lists is ongoing.

11. Conclusion

Following the accident of the Fukushima-Daiichi NPPs, the French operators performed, at the request of ASN, complementary safety assessments (CSAs). The CSA approach considers natural extreme events with a level much higher than the stress levels taken into account at the design stage with the current safety approach. Whatever the existing redundancies, the CSA approach postulates the deterministic loss of cooling and electrical supplies. Extreme situations and dreaded situations have been identified for each site, as well as the associated "hardened safety core" (HSC), a limited number of "key" SSCs with high design requirements, necessary to maintain the facility in an acceptable safe condition.

The French fuel cycle facilities other than the La Hague reprocessing facilities are located at Tricastin site (uranium conversion, enrichment and treatment facilities), at Romans-sur-Isère site (uranium fuel manufacture facilities) and at Marcoule site (MOX fuel assembly manufacture at MELOX plant). The predominance of chemical hazards on Tricastin and Romans-sur-Isère sites has led to identify discharges of liquid uranium hexafluoride (UF_6) and of hydrogen fluoride (HF) as dreaded situations. Concerning MELOX plant, the dreaded situations (criticality accident, plutonium discharge in the environment) are caused by the heat release of the radioactive material and the potential loss of its containment. Concerning the facilities operated by CEA, the main dreaded situations are due to the research reactors (core meltdown, sodium fire). The operators of these sites have planned the implementation of the corresponding remediation and mitigation means and integrated them in the HSC, which were estimated by IRSN, as a whole, satisfactory.

This article focusses on some site specificities which had to be taken into account for CSAs: the vicinity between AREVA FCFs and NPPs at Tricastin site and an urban environment at Romans-sur-Isère site. New fixed and mobile equipment and measures have been provisioned and are meant to be deployed by emergency crisis teams. Multi-facility aspects and aggravating factors (fire, explosion…) are taken into account. Implementation of HSC SSCs sometimes requests material modifications of the existing facility. For existing HSC SSCs, a demonstration of their robustness is necessary. Most of these specificities have been assessed or are currently being evaluated by IRSN.

As a conclusion, even if some complements are necessary, regarding the HSC provisions and associated requirements, IRSN has assessed that the corresponding equipment and measures are about to enhance their ability to withstand extreme hazards or supply losses.

Session 4 – Post-Fukushima Studies and R&D on Accident Scenarios and source terms for FCFs

Chairpersons: K. Tonoike (JAEA), K. Mori (S/NRA/R)

1. (5) Behavior of Volatilized Ru in the Presence of H2O, HNO3 and NOx Gases through Leak Path in a Reprocessing Plant
   Yuki Shibata. Japan Nuclear Fuel Limited, Japan
2. (6) Experiment on airborne release fraction in hydrogen explosion accident at Reprocessing plant
   Takahiro Ishio. Japan Nuclear Fuel Limited, Japan
3. (20) Development of Standard Procedure for Consequence Analysis of Criticality Accident in Fuel Cycle Facilities
   Yuichi Yamane, Hitoshi Abe. Japan Atomic Energy Agency, Japan
4. (22) Experimental Evaluation of Release and Transport Behavior of Gaseous Ruthenium under Boiling Accident in Reprocessing Plant
   Naoki Yoshida, Shinsuke Tashiro, Yuki Amano, Kazuo Yoshida, Yuichi Yamane, Hitoshi Abe; Japan Atomic Energy Agency, Japan

Behavior of Volatilized Ru in the Presence of H_2O, HNO_3 and NOx Gases through Leak Path in a Reprocessing Plant

Yuki Shibata, Takashi Kodama, Kazushi Ogaki
Japan Nuclear Fuel Ltd., Obuchi, Rokkasho-mura, Aomori 039-3212, Japan

Abstract : In cooling malfunction accident of a high level liquid waste (HLLW) tank, the behavior of Ru attracts much attention due to the potential risk, since Ru could be oxidized to a volatile chemical form in the boiling and drying out of the HLLW, and might be discharged to neighboring cells and finally to the environment.
Many experiments show that when the vapor from boiling or drying HLLW is sent to a condenser at around room temperature, most of the volatilized Ru can be recovered from the condensed liquid (for example Philippe et. al.). To understand the behavior of Ru at higher temperatures, we carried out experiments in which 60 mL simulated HLLW was heated to dryness, and the resulting vapor was sent to a 9.6 L stainless steel box whose temperature was maintained above 110°C, and we found that the Ru leak pass factor (LPF) through the box was 1/1000 at 110°C, and increased with the increasing box temperature. These results suggest that the dissolution of RuO_4 into condensed liquid might be a key in the RuO_4 transfer process.

We have hence carried out experiments in which RuO_4 gas is mixed with H_2O vapor, HNO_3 vapor and NOx, and the gas mixture is sent to a flask whose temperature is controlled at various levels. The results show that the existence of NOx in the gas and the residence time would be most significant factors to decrease the Ru-LPF through the flask.

1. Introduction

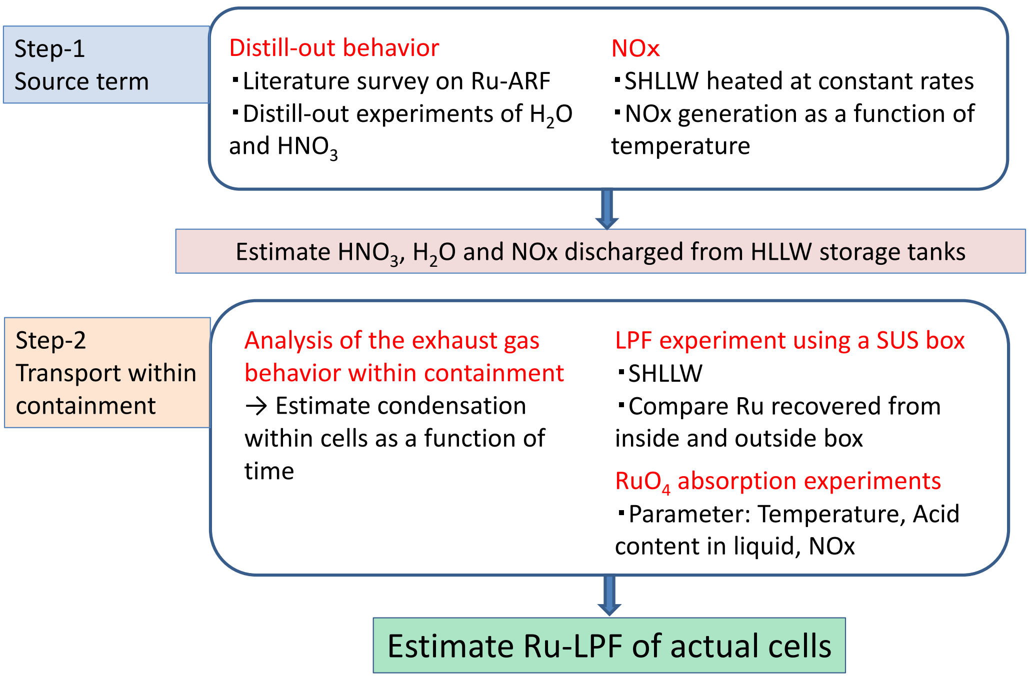
The malfunction of the cooling system of high level liquid waste (HLLW) tanks may result in boiling and drying out of the HLLW, while Ru could be oxidized to a volatile chemical form and might be discharged to neighboring cells and finally to the environment. This does not mean, however, that all of the volatilized Ru is discharged to the environment, but a part of the volatilized Ru may be absorbed into condensed liquid or decomposed to RuO_2 and deposited onto the wall/floor along the pathway. It is therefore of particular significance to evaluate the leak path factor (LPF) of the volatilized Ru along the pathway [1]. Here, LPF is defined as the ratio of the amount of Ru flowing out of a containment to that flowing in. The purpose of this paper is to introduce the outline of the research investigations carried out to estimate the values of Ru-LPF.

The present research is being carried out under the scope shown in Fig.1. The purpose of Step-1 is to estimate the flow rate and composition of the gas discharged from HLLW storage tanks. Major components are HNO_3, H_2O and nitrogen oxides (NOx). For this purpose we have carried out laboratory experiments using a simulated HLLW (SHLLW) which are heated in a glass flask and evaporated to dryness. The vapor from the flask is condensed in a glass condenser and collected for chemical analyses. Since NOx generated during dryness is considerably absorbed into the condensates, we have carried out another experiments in which a small amount of SHLLW is heated at temperatures increasing linearly with time in the nitrogen atmosphere and NOx contents in the off-gas stream is analyzed. The behavior of Ru is summarized based on the literature.

The purpose of Step-2 is to understand the transport behavior of the volatilized Ru within containments and to estimate the values of Ru-LPF. The gas discharged from storage tanks would enter neighboring cells and come in contact with the cold cell wall. We have estimated the condensation of the gas along the pathway by calculating the heat transfer between the gas and the cell wall and the vapor-liquid equilibrium. Hence, we can estimate the environmental conditions of the containments where the volatilized Ru is transferred.

Figure 1. **Outline of the present investigation**

ARF : Airborne Release Fraction, LPF : Leak Path Factor



We have carried out two kinds of laboratory experiments to get data on the Ru transport within containments. One is box experiments to get insight into the phenomena in containments [2].

The experimental apparatus consists of a small glass tank in which 60mL SHLLW is heated to dryness, and a 9.6 L stainless steel box which accepts the vapor from the small tank. The other experiments are now being carried out to get fundamental data on RuO_4 dissolution into nitric acid solutions at conditions covering a wide range of the temperature and acid concentration of the solutions and the NOx content in the gas phase.

2. Source term from HLLW storage tanks

2-1 Ru volatilization in HLLW storage tanks

Igarashi *et al.* [3] carried out calcination experiments, in which SHLLW samples of 2 mL were heated stepwise to 135°C, 200°C, 300°C, 400°C and 500°C with each temperature kept for one hour and measured the amounts of volatilized Ru, showing that the Ru in SHLLW was volatilized at temperatures between 135°C and 300°C.

Philippe *et al.* [4] carried out experiments using a real HLLW and showed that, during boiling and drying up to 160°C, the airborne release fraction (ARF) [1] of Ru was 12 %, whereas those of nonvolatile ^{137}Cs and α-emitters were both about 1×10^{-3} %. The ARF is defined here as the ratio of the amount of a component flowing out of the vessel to the total amount of the component initially contained in the vessel. Ten thousand times larger ARF of Ru than those of nonvolatile elements is explained by its volatility. They also showed that the Ru volatilization began from the solution temperature of 119 °C and the solution acidity of about 6M and continued up to 160°C.

Tashiro *et al.* [5] carried out distill-out experiments using a SHLLW at a heating rate expected for HLLW with 5W/L decay heat, and reported that the Ru volatilization began from 120°C and terminated by 300°C and that the graph of the Ru volatilization rate plotted against the sample temperature showed two peaks at 140°C and 240°C, with Ru-ARF of 8.8%.

These results suggest that the volatilization of Ru occurs at temperatures between 119°C and 300°C, though the Ru-ARF seems varied in value depending on experimental conditions such as heating rate.

2-2 Estimation of the composition and flow rate of vapor discharged from storage tanks

To get data on the composition and flow rate of the vapor discharged from storage tanks as a function of time, we have carried out distill-out experiments using an apparatus described in detail in the previous paper. A 60-mL sample of SHLLW-J08 with the composition shown in Table 1 was heated in a glass reactor with a diameter of 24 mm and a height of 600 mm from the room temperature to 300 °C in about six hours and the vapor from the reactor was sent to a condenser cooled with water of 15°C. Condensate samples were collected at every 10-15 minutes for chemical analyses.

The results are shown in Figure 2. Boiling began in an hour when the sample temperature reached 104 °C, and the temperature reached 300 °C five hours later. The amount of distilled water increased linearly with time from start to finish, whereas that of distilled HNO3 increased slowly just after boiling began and became to increase sharply from three hours later, when the sample in the reactor was concentrated by 1.6 times by volume. Total amount of HNO_3 distilled was 167 mmol, by 39 % larger than 120 mmol that is the initial amount of HNO_3 in the SHLLW. This is because some of the nitrate salts in the SHLLW are considered to release HNO_3 during distill out (for example, Fe, Zr [6]) and the NOx evolved during distill out was apparently absorbed into condensed liquid at the condenser.

Figures 3(a) and 3(b) show the normalized amounts of distill-out liquid and nitric acid as functions of temperature. The vertical axis of Figure 3(a) is the relative volume of distilled liquid to the initial volume, and that of Figure 3(b) is the molar amount of distilled HNO_3 divided by the total amount estimated by extrapolation. These results cover a wide range of duration time of distill out experiments from 1h40m to 21h. The distill-out liquid volume and nitric acid of these 4 runs can be approximated, respectively, by a single curve, suggesting that the distill-out behaviors of H_2O and nitric acid are controlled primarily by the temperature. Hence, we have assumed that the distill-out behavior in a real plant can be estimated from the temperature of HLLW using experimental results as shown in Figure 2.

Though high concentration of NOx is generated near the drying and calcination phase, exact measurement is difficult because the NOx is significantly absorbed into condensed liquid at the condenser. We therefore measured the NOx by a different method, in which the thermogravimeter/differential thermal analyzer (DG/DTA) was used to measure the NOx discharged from a small amount of SHLLW heated at temperatures increasing linearly with time in the nitrogen atmosphere.

Table 1. **Compositions of SHLLW**

Nitrates	KJ (mol/L)	H15 (mol/L)	J08 (mol/L)	Nitrates	KJ (mol/L)	H15 (mol/L)	J08 (mol/L)
(HNO_3)	2	2	2	$Fe(NO_3)_3$	0.025	0.024	0.025
$CsNO_3$	0.069	0.065	0.068	$Ni(NO_3)_2$	0.003	0.003	0.003
$Sr(NO_3)_2$	0.034	0.032	0.033	$RbNO_3$	-	0.015	0.015
$Nd(NO_3)_3$	0.095	0.092	0.096	$Y(NO_3)_3$	-	0.018	0.018
$Gd(NO_3)_3$	0.09	0.087	0.09	$Rh(NO_3)_2$	-	0.014	0.014
$Ce(NO_3)_3$	0.099	0.096	0.1	TeO_2	-	0.016	0.016
$RuNO(NO_3)_3$	0.091	0.092	0.091	$Ba(NO_3)_2$	-	0.04	0.04

$Pd(NO_3)_2$	0.043	0.045	0.045	$La(NO_3)_3$	-	0.03	0.03
$ZrO(NO_3)_2$	0.157	0.173	0.18	$Pr(NO_3)_3$	-	0.027	0.027
$Mn(NO_3)_2$	-	-	0.051	$Sm(NO_3)_3$	-	0.017	0.017
Mo*	0.077	0.113	0.118				

*Powdered metal of Mo was dissolved in 4mol/L nitric acid

Figure 2. **Results of distill-out experiment using 60 mL SHLLW-J08**

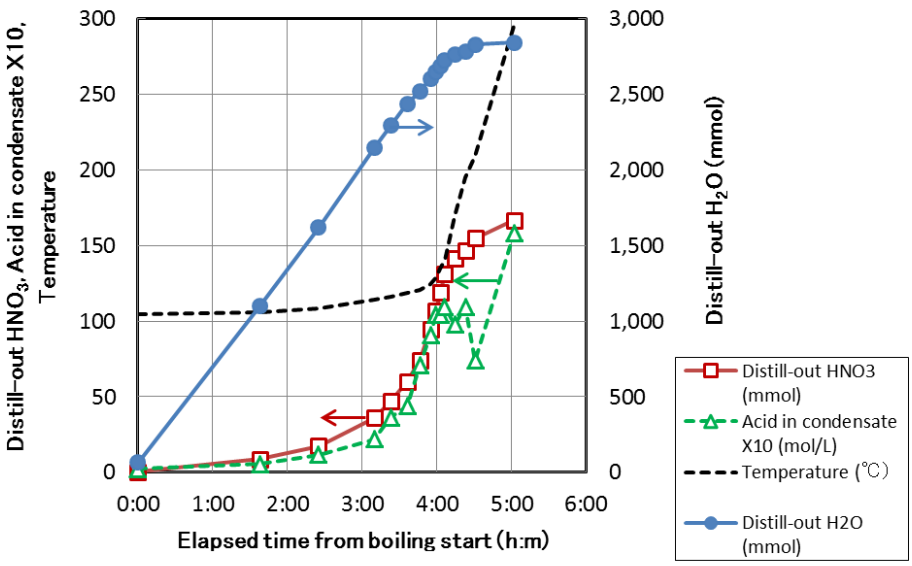
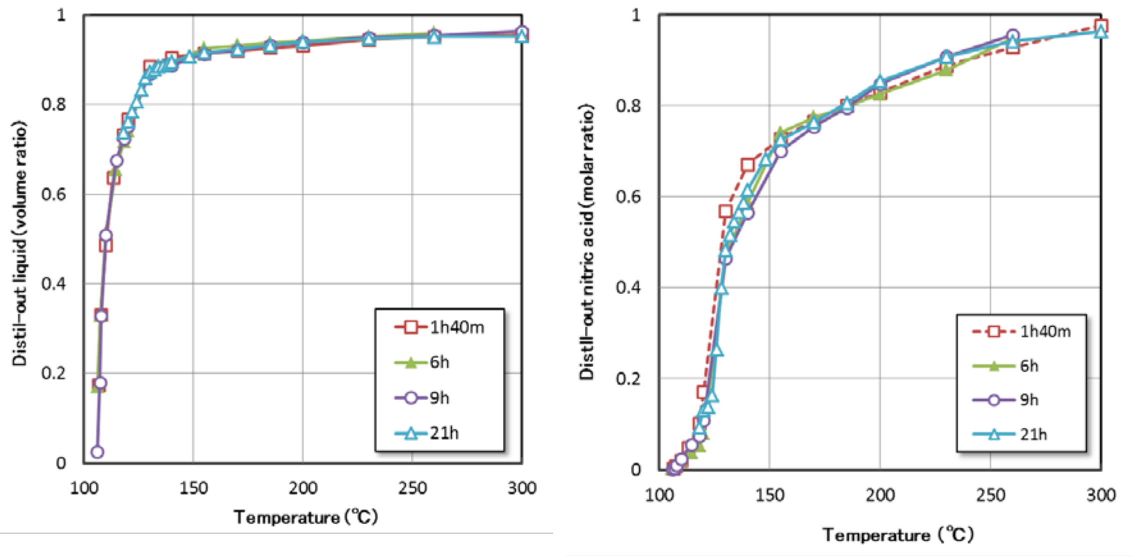


Figure 3. **Distill-out liquid volume and nitric acid as functions of temperature**

Parameter : Duration of distill-out experiment



(a) H_2O (b) HNO_3

Figure 4 shows the effect of heating rate on the NO_2 generation rate from SHLLW-KJ. The vertical axis denotes the amount of NO_2 generated from $1m^3$ SHLLW between a temperature step of 5°C. This figure shows that slower heating rate results in the shift of NO_2 generation curve towards lower temperature. We adopt the rate of 0.3°C /min hereafter, since this is close to the real condition.

Figure 5 shows the amounts of NO_2 generated from SHLLW-H15 and SHLLW-KJ heated at 0.3°C /min. NOx generation of SHLLW-H15 is represented primarily by one large peak at 280°C with a small shoulder at 230°C, whereas that of SHLLW-KJ is represented by two or three peaks and the highest peak temperature is lower than that of SHLLW-H15 by about 20°C. This difference would be owing to the addition of Ba, Y, and Lanthanide nitrates to SHLLW-KJ [7]. The result of SHLLW-H15 is used hereafter since the composition of SHLLW-H15 is considered close to that of real HLLW.

Figure 4. **Effect of heating rate on NO_2** generation from SHLLW-KJ13

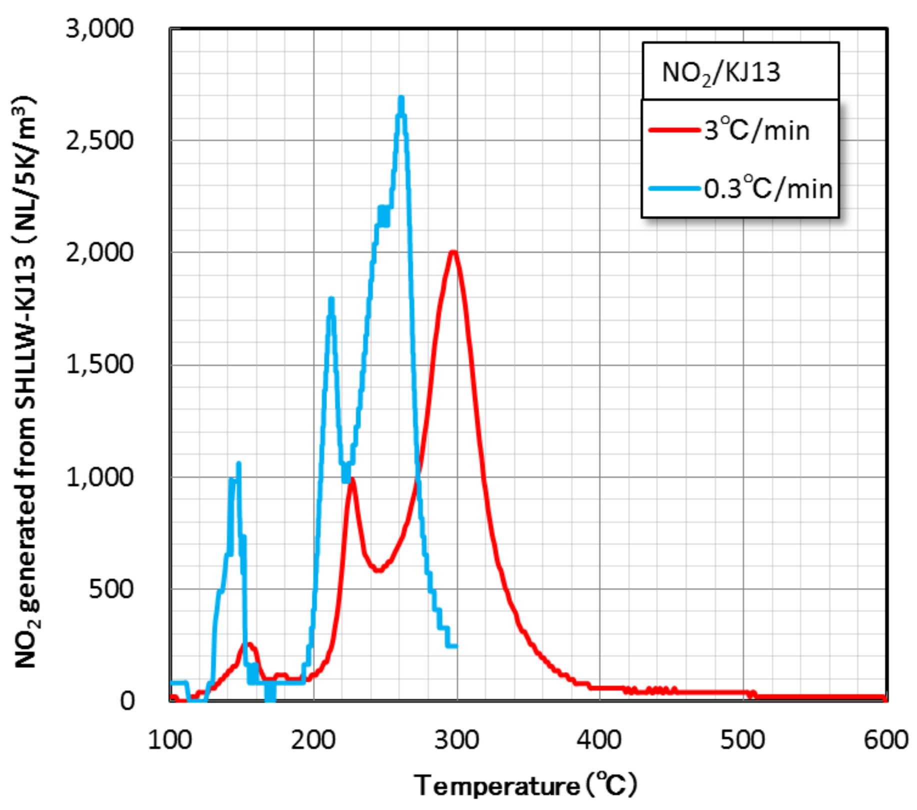
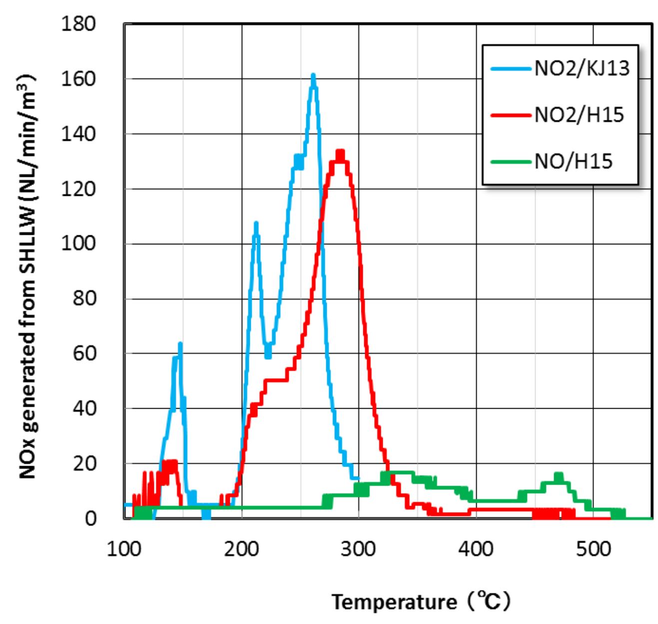


Figure 5. **NOx generated from 1m³-SHLLW heated at 0.3℃/min**



The amount of NO generated simultaneously with the NO_2 is shown in Figure 5, indicating that NOx species generated above 330°C is mostly NO, probably attributable to the thermal decomposition of Cs and Sr nitrates. Though not measured, oxygen is considered to generate during distill out, as suggested by the stoichiometric equation of metal nitrate thermal decomposition. Hereafter we assume that the amount of oxygen generation is 20% of NOx generation.

We carried out many NOx generation experiments using SHLLWs with different compositions [7] and showed that, for example, nitrate groups in Zr and Fe nitrates are all liberated as nitric acid and does not contribute to NOx generation, whereas nitrate groups in alkaline and alkaline earth metal nitrates and lanthanide metal nitrates are all liberated as NOx. The nitrate group in Ru nitrate is liberated as NOx by a quarter or a third and the residual as nitric acid. We also showed that some of metal oxides formed during distill and dry-out can catalyze the decomposition of nitric acid to NOx.

In addition to these NOx sources, we should consider the formation of nitrous acid resulting from the radiolysis of HLLW. With the G-value of 0.22/100eV measured using the SHLLW-KJ and the γ-ray source [7] and the decay heat of 10 W/L, we estimated the nitrous acid formation in the HLLW of $360m^3$ to be 7.12 Nm^3/h.

Hence, we have all data necessary to estimate the distill-out behavior of the HLLW. The estimated results of the flow rate of each component discharged from the storage tanks are shown in Figures 6 and 7. The volume of HLLW and its internal decay heat are assumed to be 360 m^3 and 10 W/L, respectively. The time origin of Figure 6 is the time when the HLLW begins boiling. The vapor discharged from the tanks is composed mostly of water until 40 h, since water is more easily vaporized than nitric acid. At this time, the nitric acid concentration of the HLLW reaches 7.6 M and apparent nitric acid evaporation begins. At 47 h, the temperature arrives at 120°C, when about 80 vol% of the HLLW is already evaporated, and the Ru volatilization should begin according to the literature (Section 2-1). The period of time while the volatile Ru is present in the discharged gas is shown by a yellow colored band. The composition of the discharged gas is shown in Figure 7 versus the HLLW temperature. The NOx content exceeds 1 % when the temperature arrives at 130°C, and increases sharply with time. At temperatures above 300°C, the discharged gas contains only NOx+O_2 with no condensible gas. The most important point shown in these figures is that the volatile Ru is always present in the atmosphere consisting of H_2O, HNO_3, NOx and O_2.

Figure 6. **Estimated gas flow rates at the exit of storage tanks containing HLLW of 360 m³ with the decay heat of 10 W/L**

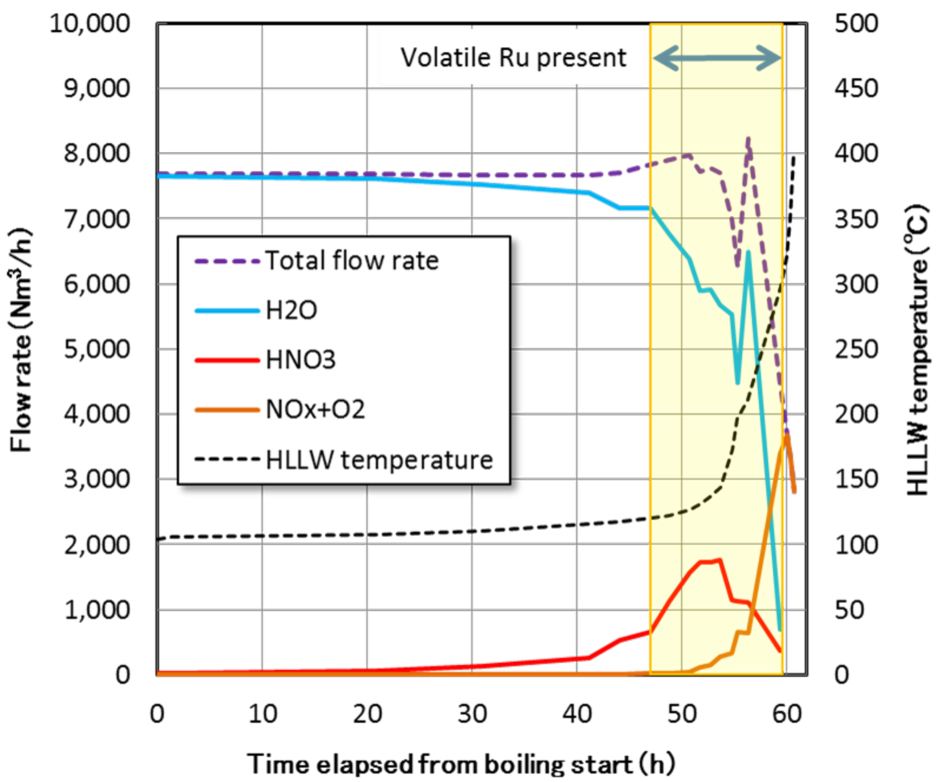
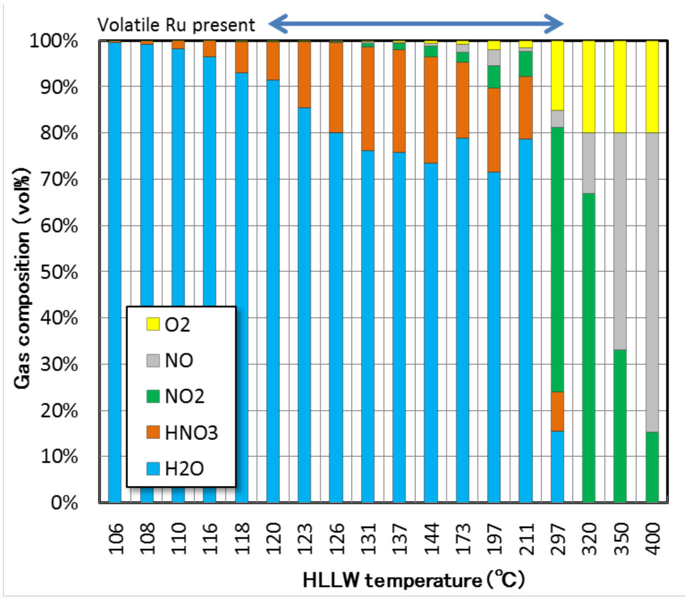


Figure 7. **Estimated gas composition at the exit of the storage tanks**



3. Transport within containments

3-1 Condensation in containments

Many experiments show that when the vapor from boiling or drying HLLW is sent to a condenser at around room temperature, most of the volatilized Ru can be recovered from the condensed liquid [2, 4, 5, 6]. However, the behavior of the volatilized Ru at higher temperatures is left unknown. Abe et al. [6] reported that RuO_4 in water vapor was decomposed to black particulate matter at 150°C, but hardly be decomposed in HNO_3 and H_2O vapor. The behavior of the volatile Ru seems thus complicated.

In the boiling accident, the vapor discharged from HLLW storage tanks is assumed to be led to neighboring cells, where it is mixed with the ambient air and comes to contact with the cold cell walls, and a significant amount of the vapor would be condensed. Along with the condensation, the volatilized Ru may undergo chemical reactions with nitrogen oxides and dissolution into the condensed liquid. Since these reactions depend on the atmospheric conditions along the pathway, we should know the temperature and acid concentration of condensates and the composition of the gas. The followings are assumed in calculation of heat transfer between the gas and the cell wall and resulting condensation.

a) The gas discharged from storage tanks enters directly into an imaginary cell (we call Cell-1) of 3,000 m^3 volume and 1,900 m^2 surface area bounded by thick concrete walls, with drainage holes on the floor. The effluent gas of Cell-1 then enters Cell-2 of the same size as Cell-1, without drainage holes on the floor. No condensed liquid is accumulated on the floor of Cell-1 and some liquid is accumulated on the floor of Cell-2.

b) The amount of HLLW is 360 m^3 with a decay heat of 10 W/L.

c) The initial temperatures of the cell and the concrete are 30°C.

d) The time change of the temperature distribution within the cell wall is approximated by the theoretical solution for the semi-infinite medium whose surface is maintained at a constant temperature T_s at time *t* = *0*, and throughout which the temperature is initially T_a; namely,

$$\frac{T - T_a}{T_s - T_a} = erf\left(\frac{x}{\sqrt{4at}}\right), \quad (1)$$

where a is the thermal diffusivity of the wall material and *x* is the depth from the surface. The total amount of heat Q_i absorbed by the wall between a time interval from t_i to t_{i+1} is given by

$$Q_i = 2A\lambda(T_s - T_a)\frac{\sqrt{t_{i+1}} - \sqrt{t_i}}{\sqrt{\pi a}}, \quad (2)$$

where *A* is the surface area of the wall and λ is the thermal conductivity of the wall material [8]. Though the surface temperature changes with time as shown later, we assumed that cooling heat flux could be calculated by Equation (2) with T_s changing with time.

To check the validity of assumption d), we carried out numerical simulation and found that the effects of the thickness of the wall and T_s changing with time are not significant in calculation of the heat flux.

e) No ventilation air is introduced.

f) Though the NOx is partly dissolved into condensates and generates nitrous acid in the liquid, the partial pressures of components are calculated under the assumption of no dissolution of the NOx.

g) The temperature and nitric acid concentration of condensate are calculated based on the gas-liquid equilibrium. The vapor pressures of H_2O and HNO_3 of nitric acid solutions in [9] are used.

h) The total pressure of the gas phase is assumed to be 101.3 kPa.

i) The gas superficial velocity in the cell is of the order of cm/s, and heat transfer between the gas and the condensate on the cell wall is considered to be controlled by natural convection. The values of heat transfer coefficient used are 1.3-4.4 $W/m^2/K$.

Summary of condensation calculation in Cells-1 and 2 is shown in Table 2. The total amount of vapor discharged from the storage tanks is 367 ton and the amount of condensate in Cell-1 is 79 ton with the nitric acid concentration of 5.2 M and that in Cell-2 is 69 ton with 3.2 M. The latter is equal to the amount and nitric acid concentration of the liquid pool accumulating on the floor of Cell-2.

Figure 8 shows the time changes in temperature of storage tanks, gas, condensate and wall surface, as well as the time change in nitric acid concentration of condensate. The x-axis is the time elapsed from boiling start. At times later than 50 h the temperature of condensate decreases slowly. This is because a sharp increase in the molar fraction of NOx+O_2 in the gas results in a decrease in the molar fraction of H_2O+HNO_3 and in a decrease in the condensation temperature. The condensation temperature shown in this figure is a calculated value derived from the condition that the sum total of the partial pressures of components in the gas should be 101.3 kPa. For the condensation to occur actually, the gas should be cooled to the calculated condensation temperature. Therefore, if the surface temperature is higher than the calculated value, the wall cannot cool the gas and no condensation would occur.

Table 2. **Condensation in Cell-1 and Cell-2**

			Storage tanks	Cell−1	Cell−2
Amount discharged	H_2O	ton	307	250	193
	HNO_3	ton	60	39	27
	Sum	ton	367	289	220
Condensate	H_2O	ton		57	57
	HNO_3	ton		21	12
	Sum	ton		79	69
	Acid	mol/L		5.2	3.2

The temperature designated as 'cell wall' in Figure 8 is the temperature of the wall surface estimated by numerical simulation. Starting from the temperature distribution within the concrete wall at 52.74 h calculated by Equation (2), the temperature distribution was calculated numerically under the boundary conditions that the wall surface is dry and heated by the gas as

$$\text{at } x = 0\text{、} t > 0\text{、} h_G(T_G - T_s) = -\lambda \frac{\partial T}{\partial x}\Big|_{x=0} \qquad (3)$$

where h_G is the heat transfer coefficient between the gas and the wall surface, T_G is the gas temperature and T_s is the surface temperature. The thickness of the wall was assumed 2 m and the boundary at the other side was assumed adiabatic. Figure 8 shows that the wall temperature is lower than the condensation temperature until 57 h, beyond which the wall temperature is higher than the condensation temperature, meaning that no condensation occurs after 57 h. At time period colored tint brown, the gas consists of only NOx and O_2 and no condensation occurs.

A similar calculation was conducted for Cell-2. It was assumed that Cell-2 had no drainage on the floor where a pool of the condensate would appear. Calculation showed that the temperature of the pool liquid ranged between 100°C and 104°C and the acid concentration between 0 M and 3.0 M.

Figure 9 shows the transition of the nitric acid concentration and the temperature of the condensates at Cell-1 and Cell-2 and that of the pool liquids at Cell-2. The nitric acid concentrations and temperatures of condensates while volatile Ru is present scatter within a narrow range between 12 M and 14 M and between 116°C and 118°C, while those of pool liquids at Cell-2 scatter within a range between 1.8 M and 3 M and between 102°C and 104°C. The target of this investigation is to clarify the behavior of volatile Ru in these conditions.

3-2 Box experiments

We have carried out box experiments to get insight into the phenomena within a compartment [2]. The experimental apparatus consists of a small tank in which 60 mL of SHLLW is heated to dryness in about 6 hours, and a 9.6 L stainless steel box which mimics the neighboring cell accepting the vapor from the small tank, and a condenser where the vapor coming out from the box is cooled to collect the condensate. The residence time of the vapor in the SUS box is about 30 min while boiling continues.

Figure 8. **Condensation in Cell-1**

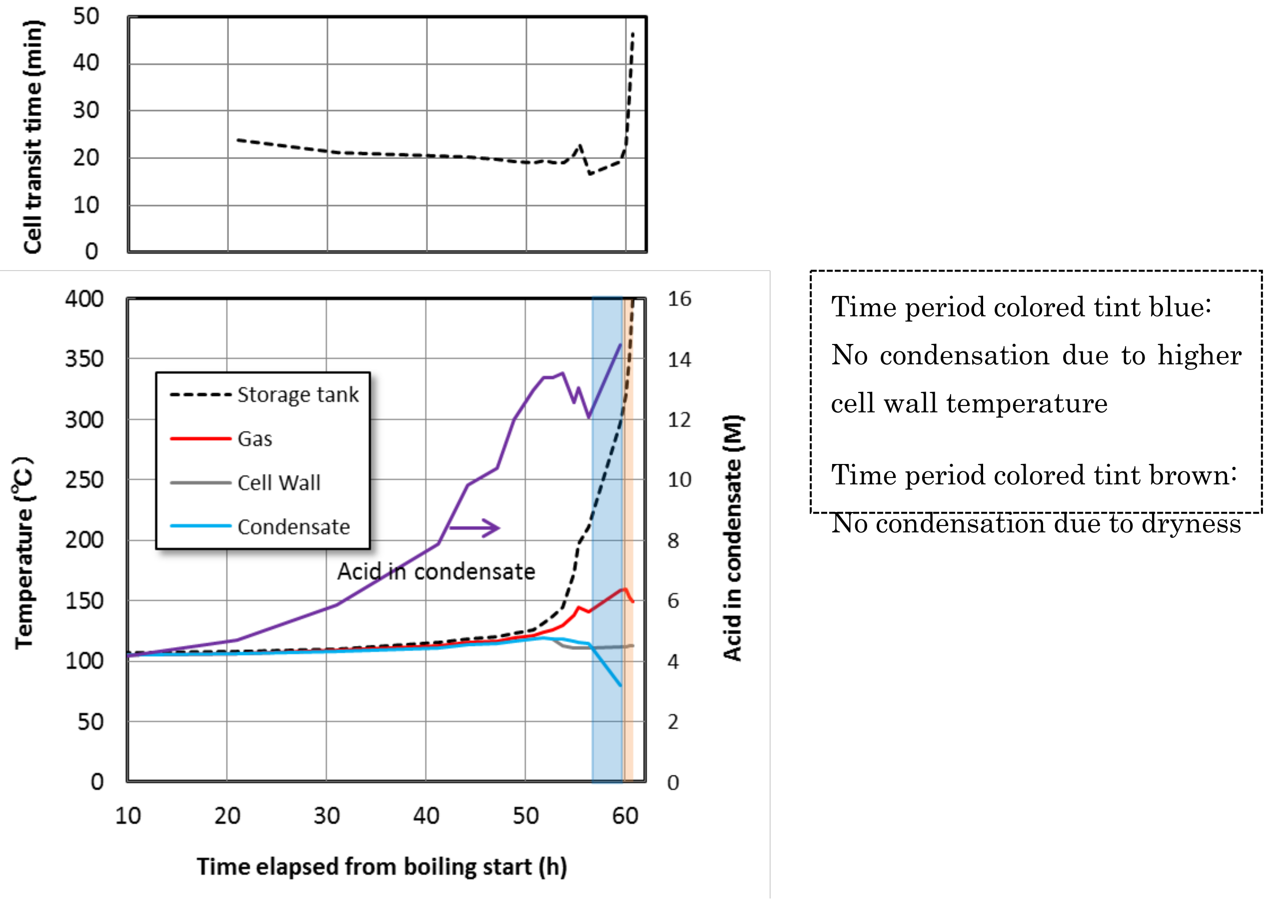


Figure 9. **Temperature and nitric acid concentration of the condensates at Cell-1 and Cell-2**

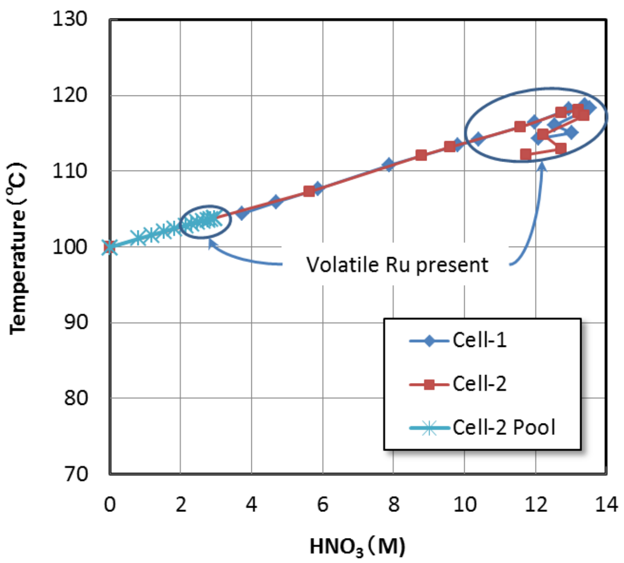
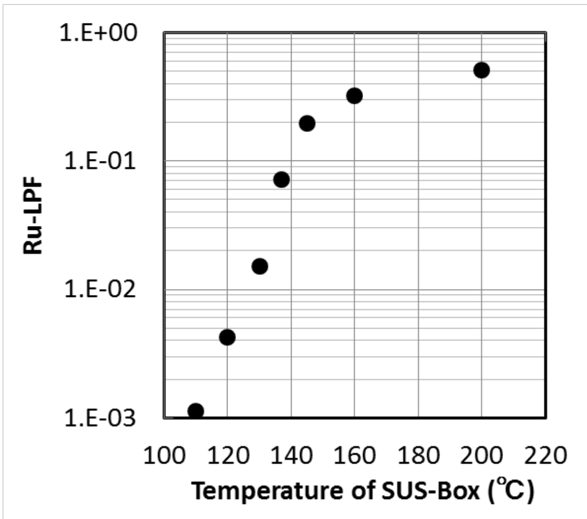


Figure 10 shows the Ru-LPF of the SUS box plotted against the box temperature. The Ru-LPF is defined as the ratio of Ru recovered from the condensates to the total amount of Ru recovered from both the condensate and the inside of the box. This figure shows that the Ru-LPF is 0.1% when tha box temperature is maintained below 110°C.

An important point shown in this figure is that the value of LPF increases with the box temperature. This suggests that Ru trapping in the box is controlled not by the thermal decomposition of RuO_4 but by the dissolution into the condensed liquid.

Figure 10. **Ru-LPF of SUS-Box plotted against the box temperature**



3-3 RuO_4 absorption experiments

Two different types of experiments are being carried out to get quantitative data to understand the RuO_4 behavior. In type I experiments, gas containing RuO_4 and NOx is sent to a flask of 0.6-L inner volume and comes in contact with nitric acid solution. The flask temperature is controlled at a designated value. NOx gas is supplied from a gas cylinder. The vapor coming out of the flask is sent to a condenser and the condensate is collected for chemical analyses. NOx content in the vapor could not be measured due to interference of RuO_4 against the NOx sensor. The residence time of the gas ranged between 16 s and 144s.

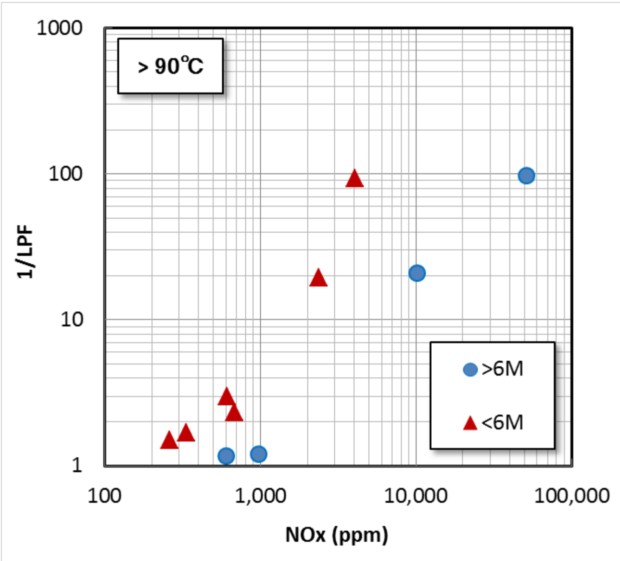
Figure 11-left shows the correlation between Ru-LPF and the NOx concentration at temperatures higher than 90°C. It is apparent that the values of 1/LPF, which equal to the decontamination factor (DF), increase with higher NOx concentrations and are larger at lower nitric acid concentrations of the absorption liquid. At temperatures lower than 60°C, no apparent dependence of the LPF on NOx concentration was observed (Figure 11-right).

In type II experiments, super-heated nitric acid vapor is sent to a flask where it is mixed with RuO_4 and NOx, a part of the vapor condensates as a result of contact with the cool wall of the flask and RuO_4 is partly dissolved into the condensate. The flask was set in a box, whose temperature was controlled by circulating heated air of designated temperatures lower than condensation temperatures. Lower box temperatures result in more condensation.

Figure 12 shows the correlation between the values of Ru-LPF and the residence time of the gas. The temperatures and nitric acid concentrations of the condensates ranged from 52°C to 116°C and from 0 M to 10.5 M, respectively. It is evident that the value of 1/LPF=DF increases with increasing residence time. It is found, in addition, that DF values of square symbols corresponding to NOx > 56,000 ppm are larger than those of circles symbols corresponding to

NOx < 11,000 ppm by one order of magnitude. Data of both the symbols are taken at temperatures higher than 103°C and at nitric acid concentrations higher than 8.9 M. Evident correlation between 1/LPF and NOx is shown in Figure 13. Triangle symbols in Figures11 and 12 stand for data at temperatures of 52°C and 55°C, showing that the temperature is one of important factors controlling the value of Ru-LPF. These experiments are now under way, and more detailed results will be reported in the near future.

Figure 11. **Correlation between Ru-LPF and NOx under non-condensation conditions**



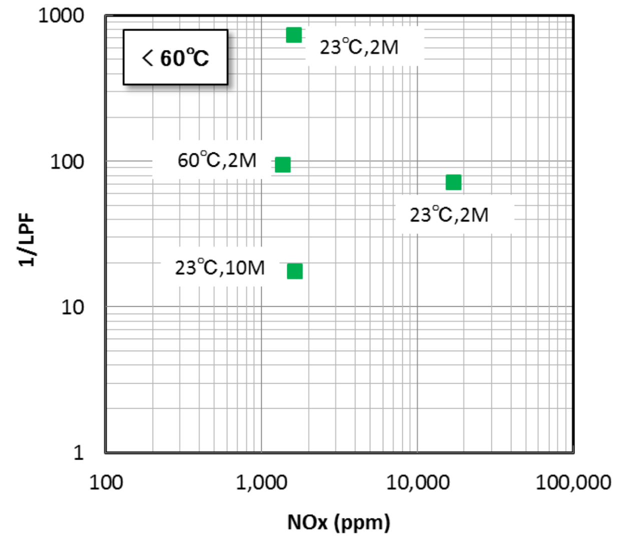


Figure 12. **Correlation between Ru-LPF and residence time under condensation conditions**

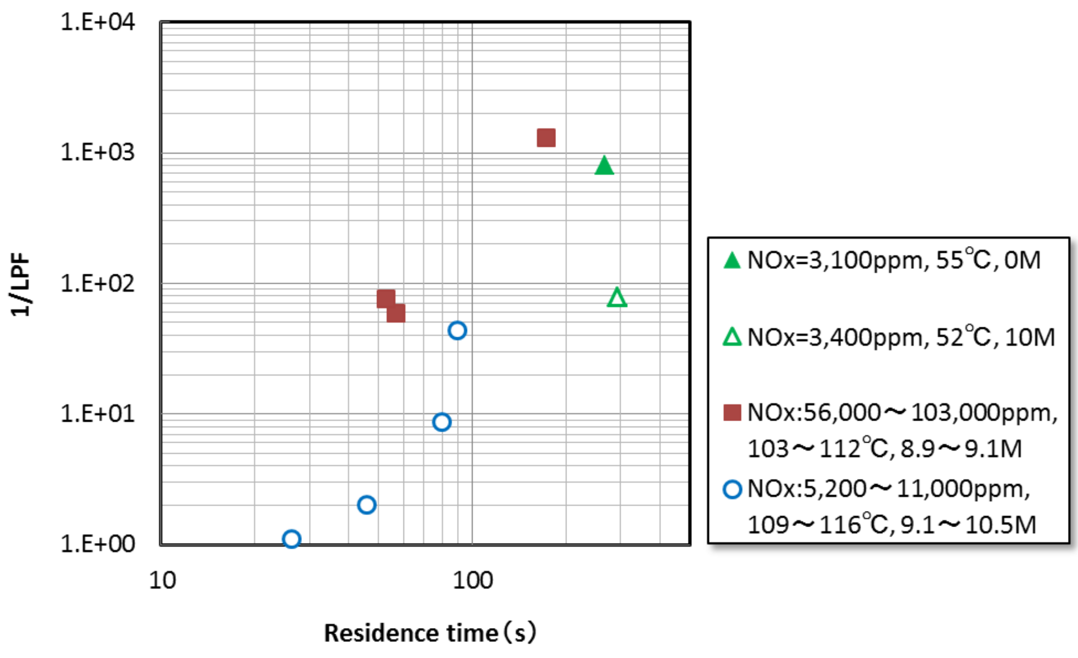
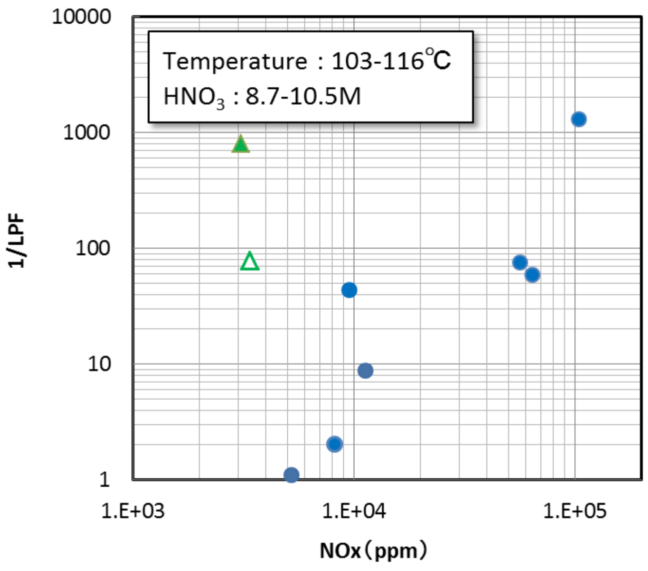


Figure 13. **Correlation between Ru-LPF and NOx under condensation conditions**



4. Conclusion

To predict the behavior of volatile Ru in cells/containments, we estimated

1) changes in flow rate and composition of the gas discharged from HLLW storage tanks based on laboratory experiments, and
2) temperatures and nitric acid concentrations of condensates on the cell walls, assuming that the gas discharged from the storage tanks is transferred to a neighboring Cell-1 and subsequently to Cell-2, both with a volume of 3,000 m^3, with heat transfer between the gas and the cell walls and gas-liquid equilibrium taken in account,
and we carried out volatile Ru absorption experiments in which temperatures, nitric acid concentrations of condensates and NOx contents in the gas were varied.

[Literature]
[1] US Nuclear Regulatory Commission. Nuclear fuel cycle facility accident analysis handbook. 1988; NUREG/CR-6410.
[2] Kodama T, Shibata Y, Takahashi N, Matsuoka S, Kumagai M, Hayashi S, Suzuki K. Experiments on the leak path factor of ruthenium volatilized from high level liquid waste tanks in a reprocessing plant in case of the boiling and drying. J Nucl Sci Technol. Vol52 Iss.4, 2015.
[3] Igarashi H, Kato K, Takahashi T. Volatilization Behavior of semivolatile elements in vitrification of high level liquid waste. PNC TN8410 91-274. 1991.
[4] Phillipe M, Mercier J. P, Gue J. P. Behavior of ruthenium in the case of shutdown of the cooling system of HLLW storage tanks. Proc. The 21st DOE/NRC Nuclear Air Cleaning Conference, 1990 Aug; San Diego, USA; Session13: 831-843.
[5] Tashiro S, Uchiyama G, Amano Y, Abe H, Yamane Y, Yoshida K, Release of radioactive materials from simulated high-level liquid waste at boiling accident in reprocessing plant. Nuclear Technol, 2015, Vol190, 207-213.
[6] Management group on the study of the behavior of released radioactive materials in a reprocessing plant, Study of the behavior of released radioactive materials in a reprocessing plant, February 2014 [in Japanese]; http://iss.ndl.go.jp/books/R100000002-I025394315-00.
[7] Shibata et al. NOx Production in Concentration and Drying of Simulated High Level Radioactive Liquid Waste. 2016. (in preparation).
[8] Crank J. Mathematics of diffusion, Second edition. 1975. Clarendon Press. Oxford.
[9] Perry R H and Green D W, editors. Perry's Chemical Engineers' Handbook - 7th ed, McGraw Hills, 1997.

Experiment on airborne release fraction in hydrogen explosion accident at Reprocessing plant

Takahiro ISHIO[1, 2], Takashi KODAMA[1, 3], Kazushi OGAKI[1, 4]
Takashi KOBAYASHI[5] and Takahiro CHIKAZAWA[5]

1: Japan Nuclear Fuel Limited
4-108, Okitsuke, Obuchi, Rokkasho-mura, Kamikita-gun, Aomori 039-3212, JAPAN
TEL: +81-175-71-2382, FAX: +81-175-71-2101
E-mail: [2.] takahiro.ishio@jnfl.co.jp [3.] takashi.kodama@jnfl.co.jp
[4.] kazushi.oogaki@jnfl.co.jp

5: Mitsubishi Materials Corporation
1002-14, Mukohyama, Naka-shi, Ibaraki 311-0102, JAPAN
TEL: +81-29-295-5069, FAX: +81-29-295-7286

(Abstract)
As one of the severe accidents of the reprocessing plant, there is hydrogen explosion due to radiolysis. Previous reports about hydrogen in nuclear fuel cycle facilities are mainly researched about G value (generated yield) and the damage of the apparatus by the explosion. However, there is little study about airborne release fraction (ARF) of radioactive element in hydrogen explosion. And more, previous reports didn't describe details of procedure. Therefore, we planned experiments on ARF in hydrogen explosion, "(1) small scale test", "(2) annular vessel test".

In "(1) small scale test", we experimented pressure release test, using test equipment based on previous report "NUREG/CR-3093" referred to data in "Accident Analysis Handbook"(NUREG/CR-6410). First, test equipment receiving $Ce(NO_3)_3$ solution were pressurized by compressed air till predetermined pressure. Next, after predetermined amount of time, the release valve was opened. In this experiment, we studied the factor affecting ARF, by changing parameters, such as "volume of solution", "pressure of gas phase", "depth of solution", "surface area" and "concentration of $Ce(NO_3)_3$". As a result, ARF in the small scale test was about 10 times smaller than previous report "NUREG-CR/3093". Furthermore, we confirmed the change of ARF by the parameters.

In "(2) annular vessel test", we experimented hydrogen explosion test, using test equipment simulated gas phase part of the annular vessel receiving concentrated Pu solution in the reprocessing plant. We studied the factor affecting ARF, by changing parameters, such as "concentration of H_2 in gas phase" and "ignition position". As a result, ARF in the annular vessel test was ca. $1\times10^{-6}\sim10^{-10}$ in concentration of H_2 ranging from 15 to 30 vol% and this is far smaller than previous report "NUREG-CR/3093". This is a substantial result for evaluating accurately environment assessment in hydrogen explosion.

In conclusion, ARF in experiments is far smaller than result of previous report "NUREG/CR-3093". Test equipment is simulated the vessel in the reprocessing plant, so we get the useful data for evaluating environment assessment in hydrogen explosion. In the future, we continue the research for applying these data to another type vessel.

We hope that the experiment on ARF in hydrogen explosion accident is carried out in other institutions.

1. Introduction

H_2 gas generation reaction occurs due to radiolysis in some of nuclear fuel cycle equipment. In normal operation, H_2 concentration in equipment is keeping lower than lower limit of explosion by scavenging air. However, if scavenging function is broken by the abnormal accident, H_2 concentration increases, and a hydrogen explosion might occur. The hazard evaluation of the hydrogen explosion accident is decided by radiological release course and aerosol quantity released to the facilities outside. Fig. 1-1 shows application image of "Five-Factors Formula" which is method to evaluate a source term easily.

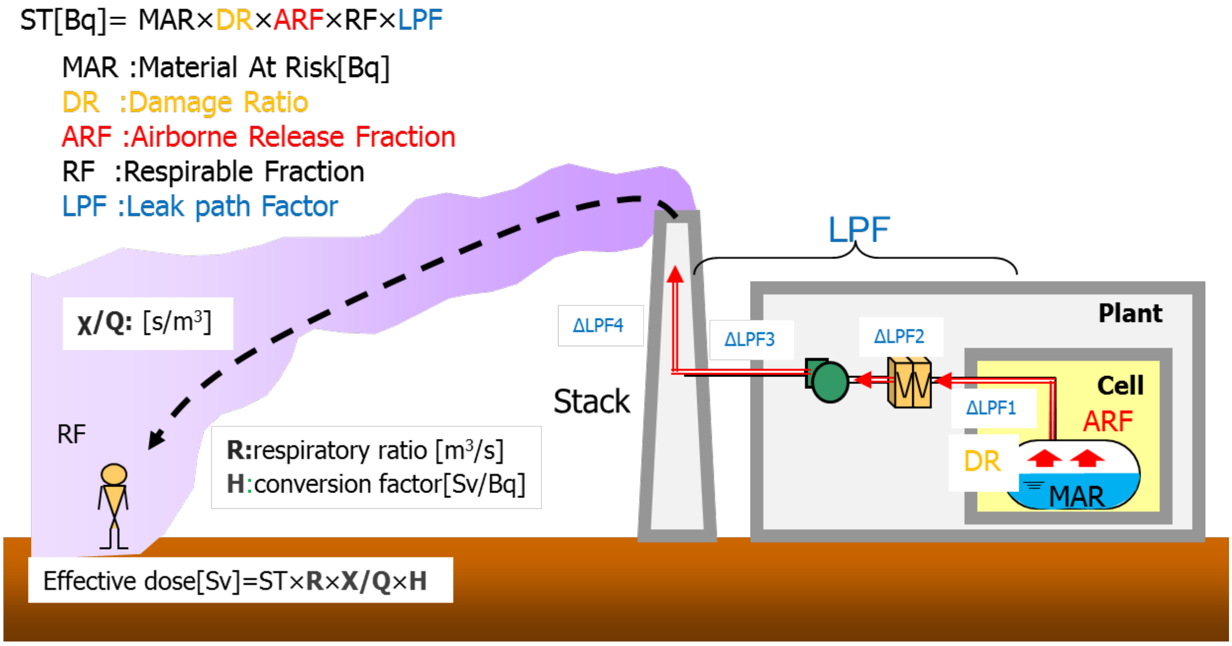


Fig. 1-1 Hazard evaluation model

Now, the ARF using in safety assessment of the reprocessing plant refers to NUREG report by United States Department of Energy (DOE) [1] [2]. However, a judgment is difficult whether we can apply a test result of the NUREG report to the own reprocessing plant, because this report doesn't describe detail of test condition.

We experimented to investigate of reproducibility using test equipment based on NUREG report [2]. And, we investigated of parameter to affect the ARF. Furthermore, we also experimented hydrogen explosion test, using test equipment simulated gas phase part of the annular vessel receiving concentrated Pu solution in the reprocessing plant. From these test results, we consider the release mechanism from solution phase to gas phase, and are intended to acquire a parameter for safety assessment that accorded with the actual equipment in the reprocessing plant.

2. Experiment

2.1 Small scale test

2.1.1 Experimental and procedures

In small scale test, we investigate about parameter to affect the ARF using test equipment based on NUREG report [2]. Fig. 2-1 is schematically shown the apparatus for small scale test. Fig. 2-2 is shown the photo of apparatus. We made 2 kinds of test vessels. Table 2-1 is shown about main dimension of test vessel parts. We loaded Cerium nitrate (simulated Plutonium nitrate) solution in test vessel prior to each start of test. We kept gas phase level in all of test using solution level adjustment block.

The test procedures are summarized as follows.

(1) Cerium nitrate solution was loaded into test vessel.
(2) Valve 1 was closed.
(3) Air feeding pipe were connected to apparatus for small scale test.
(4) We closed test chamber, filter was attached with test chamber.
(5) Air was filled into vessel by opening a valve 2 until a target pressure is attained, then the valve 2 was closed automatically.
(6) Valve 1 was opened automatically the predetermined time (holding time) later.
(7) Scavenging air was feed into test chamber by opening valve 3. The mist which floated adsorbed it on the chamber wall or a filter.
(8) Valve 3 was closed.
(9) Test chamber was washed. We collected washing solution.
(10) The concentration of Ce in washing solution was measured by ICP-MS.
(11) ARF was evaluated with formula (A).

ARF (-) = (Amount of released Ce) / (Amount of initial loaded Ce)…(A)

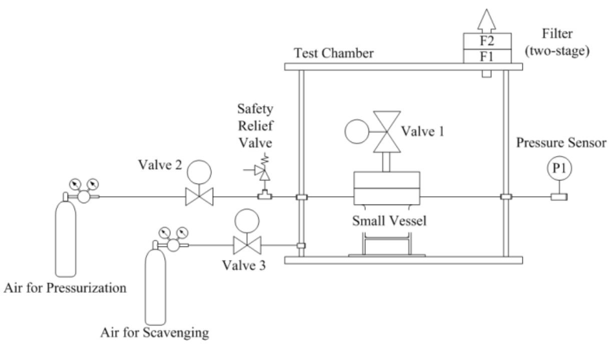


Fig. 2-1 Apparatus for small scale test

Fig. 2-2 Photograph of apparatus for small scale test

Table 2-1 Main dimensions of the test vessel part

Vessel	Vessel No.1	Vessel No.2
Diameter	11.3cm	6.7cm
Surface area	100cm^2	35cm^2
Depth	8.0cm	14.5cm
Inner Volume	800cm^3	508cm^3

2.1.2 Test conditions and parameters

Table. 2-2 was shown parameter of small scale test. And, Table. 2-3 was shown test conditions of small scale test. The parameters in this test were pressure, concentration of Ce, pressure holding time, volume of solution (depth of solution) and surface of solution.

Table 2-2 Parameters of small scale test

<table>
<tr><th rowspan="2">Run</th><th rowspan="2">small vessel</th><th colspan="2">Height*
[cm]</th><th rowspan="2">pressure
[MPa]</th><th rowspan="2">pressure holding time
[s]</th><th rowspan="2">Ce concentration of solution
[gCe/L]</th></tr>
<tr><th>Solution phase</th><th>Gas phase</th></tr>
<tr><td>1</td><td rowspan="10">Vessel No.1</td><td rowspan="3">3.5</td><td rowspan="10">4.5</td><td>3.45</td><td rowspan="8">1</td><td rowspan="6">140</td></tr>
<tr><td>2</td><td>1.72</td></tr>
<tr><td>3</td><td>0.34</td></tr>
<tr><td>4</td><td rowspan="3">1</td><td>3.45</td></tr>
<tr><td>5</td><td>1.72</td></tr>
<tr><td>6</td><td>0.34</td></tr>
<tr><td>7</td><td rowspan="4">3.5</td><td rowspan="4">3.45</td><td>280</td></tr>
<tr><td>8</td><td>14</td></tr>
<tr><td>9</td><td>3600</td><td rowspan="2">140</td></tr>
<tr><td>10</td><td>600</td></tr>
<tr><td>11</td><td rowspan="3">Vessel No.2</td><td>10</td><td>4.5</td><td rowspan="3">3.45</td><td rowspan="3">1</td><td rowspan="3">140</td></tr>
<tr><td>12</td><td>7.0</td><td>4.5</td></tr>
<tr><td>13</td><td>3.5</td><td>4.5</td></tr>
</table>

*height of solution phase and gas phase can be changed by adjustment block

Table 2-3 Test conditions of small scale test

type of gas		air
Max supply flow rate	[L/min]	20
Pressurization time	[s]	60
Pressure holding time	[s]	as described in Table. 2-2
Scavenging air volume	[L]	250

2.2 Annular vessel test

2.2.1 Experimental and procedures

In annular vessel test, we confirmed the phenomenon of the real hydrogen explosion. Fig. 2-3 and Fig. 2-4 are schematically shown apparatus for annular vessel test. Fig. 2-5 is shown the photo of apparatus. And, Table 2-4 is shown main dimensions of annular vessel. This annular vessel was modified upper part of actual annular vessel in reprocessing plant. 43L Cerium nitrate solution were loaded into vessel. Twelve pressure sensors were installed each 30 degrees in a vessel. The test procedures are summarized as follows.

(1) Cerium nitrate solution was loaded into test vessel.
(2) Air feeding pipe were connected to apparatus for annular vessel test.
(3) H_2 + Air mixture gas were feed into annular vessel by opening a valve B and valve C.
(4) When hydrogen concentration into annular vessel reached target value checked by hydrogen concentration meter, H_2 + Air mixture gas were stopped by closing valve B and valve C.
(5) We ignited hydrogen at the same time to opening off gas line valve A.
(6) Valve A was closed.
(7) After the explosion, scavenging air was feed into annular vessel. Remaining H_2 gas was scavenged.

(8) We washed collecting tank, pipe and filter. And we collected washing solution.
(9) The concentration of Ce in washing solution was measured by ICP-MS.
(10) ARF was evaluated with formula (A).
ARF (-) = (Amount of released Ce) / (Amount of initial loaded Ce)…(A)

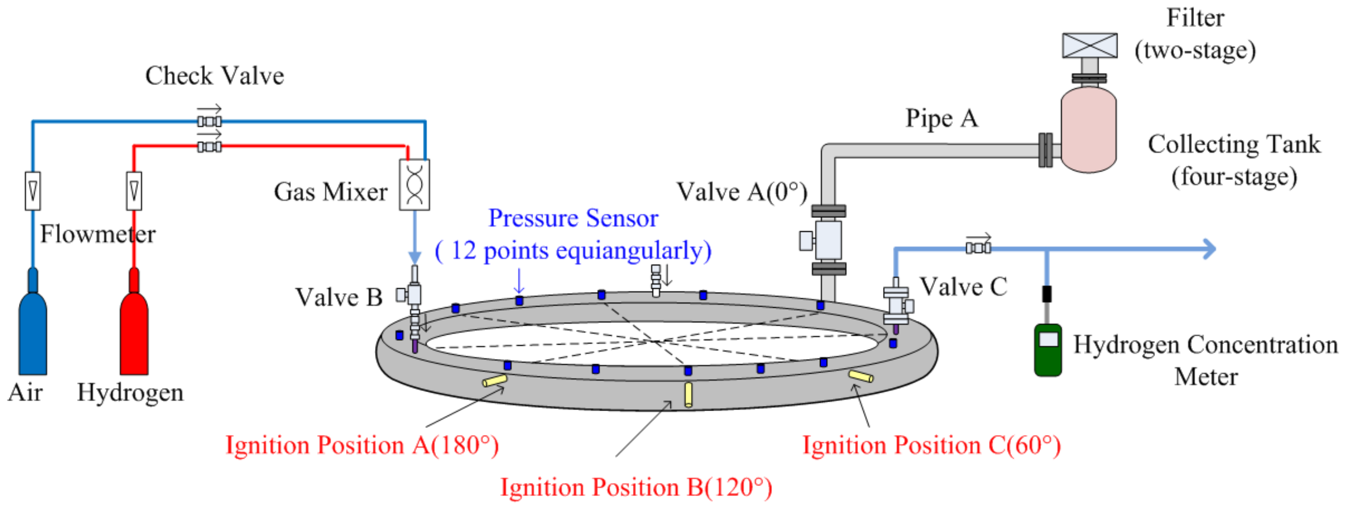


Fig. 2-3 Apparatus for annular vessel test

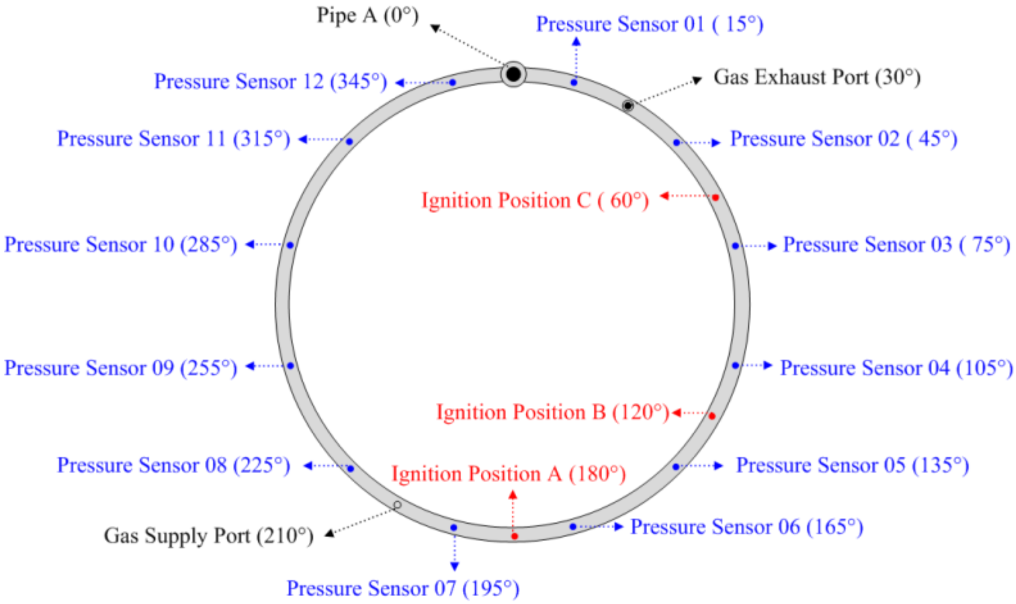


Fig. 2-4 Installed position of pressure sensor and Ignition position

Fig. 2-5 Photograph of apparatus for annular vessel test

Table 2-4 Main dimensions of annular vessel

Vessel	Annular vessel
Annular diameter	3793 mm
Pipe inner diameter	97 mm
Inner Volume	86 L
Solution volume	43 L

2.2.2 Test conditions and parameters

Table 2-5 was shown parameter of annular vessel test. Table. 2-6 was shown test conditions of annular vessel test. The parameters in this test were H_2 concentration, Ignition position and pipe A arrangement (number of bend & length).

Table 2-5 Parameters of annular vessel test

Run	H_2 concentration [vol%]	Ignition position*	Offgas line (pipe A)	
			number of bend	length [m]
21	30	A	1	2
22	30	A	1	1
23	30	A	2	2
24	30	B	1	2
25	30	C	1	2
26	20	A	1	2
27	15	A	1	2

*Ignition position A : 180 degree as against valve A (0 degree)
Ignition position B : 120 degree as against valve A
Ignition position C : 60 degree as against valve A

Table 2-6 Conditions of annular vessel test

H_2 concentration [vol%]	Gas flowrate[L/min]		
	Air	H_2	Total
30	14	6	20
20	16	4	
15	17	3	

3. Result and Discussion

3.1 Small scale test

3.1.1 Result

Fig. 3-1 is shown ARF of small scale test (Run 1). Almost of mist released from test vessel were collected at exit position of valve 1 and test chamber. This tendency was the same by all test results. Thus, the following ARF is used in the sense of total value of all (exit of valve, test chamber and filter).

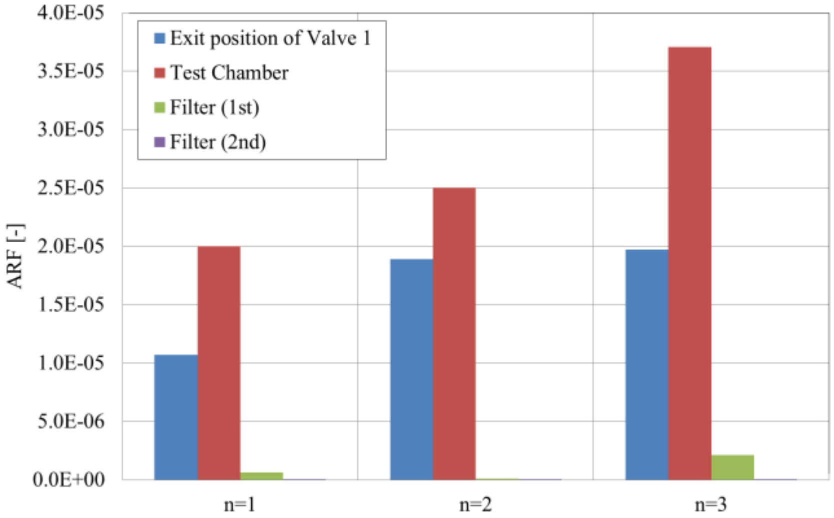


Fig. 3-1 ARF of small scale test(Run1)

Table. 3-1 is shown ARF of small scale test. We carried out each tests three times repeatedly in consideration of the unevenness. However, dispersion of data were small. In this test condition, All of ARF in small scale test were between 6×10^{-5} to 1×10^{-4}.

Table 3-1 ARF of small scale test

Run.	ARF[-]			
	n=1	n=2	n=3	Average
1	3.1×10^{-5}	4.4×10^{-5}	5.9×10^{-5}	4.5×10^{-5}
2	2.6×10^{-5}	1.7×10^{-5}	2.1×10^{-5}	2.1×10^{-5}
3	6.1×10^{-6}	5.5×10^{-6}	6.9×10^{-6}	6.2×10^{-6}
4	3.7×10^{-5}	6.2×10^{-5}	4.5×10^{-5}	4.8×10^{-5}
5	3.2×10^{-5}	2.7×10^{-5}	2.2×10^{-5}	2.7×10^{-5}
6	1.2×10^{-5}	2.1×10^{-5}	1.2×10^{-5}	1.5×10^{-5}
7	2.0×10^{-5}	2.9×10^{-5}	1.5×10^{-5}	2.1×10^{-5}
8	5.4×10^{-5}	9.0×10^{-5}	6.3×10^{-5}	6.9×10^{-5}
9	8.9×10^{-5}	8.0×10^{-5}	1.1×10^{-4}	9.2×10^{-5}
10	6.5×10^{-5}	7.3×10^{-5}	4.8×10^{-5}	6.2×10^{-5}
11	9.8×10^{-6}	8.5×10^{-6}	1.1×10^{-5}	9.7×10^{-6}
12	1.0×10^{-5}	1.1×10^{-5}	1.4×10^{-5}	1.2×10^{-5}
13	1.7×10^{-5}	2.1×10^{-5}	2.2×10^{-5}	2.0×10^{-5}

(a) Dependence of Pressure

Fig. 3-2 is shown ARF of Run1 ~ 3. And Fig. 3-3 is shown ARF of Run4 ~ 6. ARF in the same pressure is about the same, although volume of solution is different from Run1 ~ Run3 in Run4 ~ Run6. In addition, the ARF increases for pressure linearly.

(b) Dependence of Cerium Nitrate concentration

Fig. 3-4 is shown ARF of Run1, Run7 and Run8. According to result of Run1(140 gCe/L, 1.26 g/L), Run7 (280 gCe/L, 1.50 g/L) and Run8(14 gCe/L, 1.02 g/L), ARF decreases linearly due to the increase in concentration of Cerium nitrate.

(c) Dependence of pressure holding time

(From reaching target pressure to opening timing of valve 1)

Fig 3-5 is shown ARF of Run1, Run9 and Run10. As pressure holding time becomes long, the ARF also increases, but the growth rate gradually becomes small.

(d) Dependence of Vessel dimension

Fig. 3-6 is shown ARF of Run1, Run4 and Run11 ~ 13. The result of Run1 and Run4 are used by Vessel No. 1. Run11 ~ 13 are used by Vessel No. 2. If surface area of solution is same condition, the deeper the solution depth is, the more the ARF decreases. And also, the more the solution volume is, the more the ARF decreases. If solution depth is same condition, the larger the surface area is, the more the ARF increases.

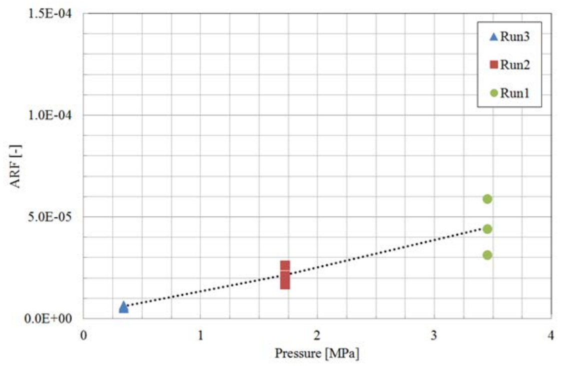


Fig. 3-2 Relationship between ARF and pressure in small scale test

(Solution: 350cm^3)

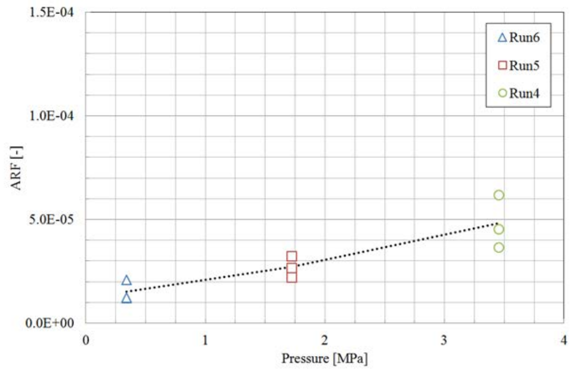


Fig. 3-3 Relationship between ARF and pressure in small scale test (solution: 100cm^3)

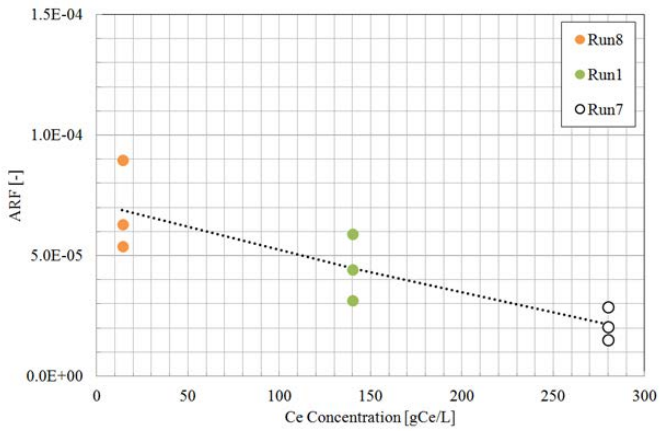


Fig. 3-4 Relationship between ARF and concentration of Cerium nitrate in small scale test

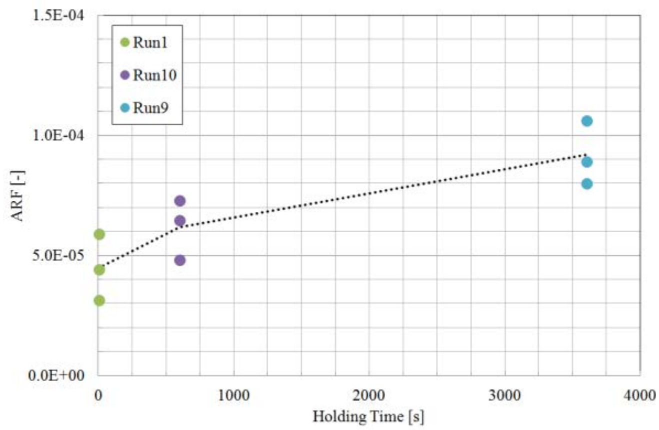


Fig. 3-5 Relationship between ARF and pressure holding time in small scale test

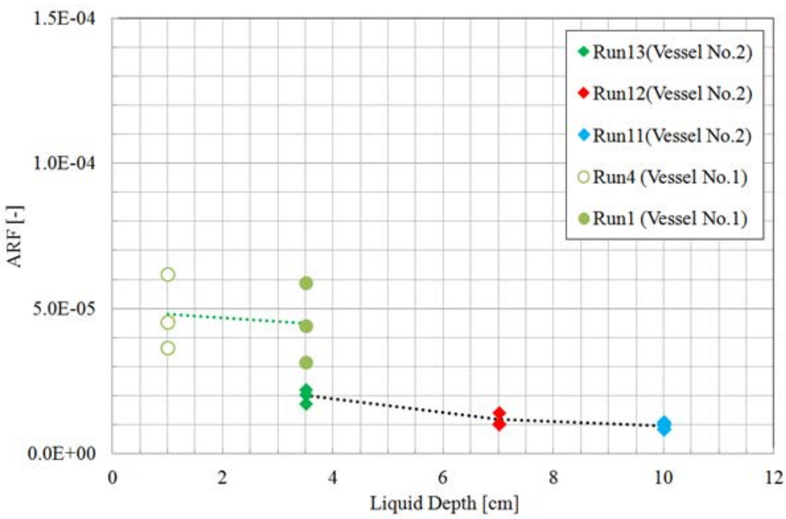


Fig. 3-6 Relationship between ARF and vessel dimension in small scale test

3.1.2 Discussion

3.1.2.1 Summary of small scale test

We show that the summary of small scale test result as follows.

- The ARF increases in proportion to pressure.
- The ARF decreases in proportion to Cerium nitrate concentration.
- The ARF increases as pressure holding time becomes long.
- If surface area of solution is same condition, the deeper the solution depth is, the more the ARF decreases. And also, the more the solution volume is, the more the ARF decreases.

3.1.2.2 Comparison of small scale test result with literature [2]

Fig. 3-7 is shown comparison of small scale test result (Run1~ Run3) with literature [2]. The ARF of Run1~ Run3 were smaller than literature [2] by 1~2 orders of magnitude. We showed below a probable cause as difference in ARF.

(a) Test vessel dimension

Geometry information of test vessel in literature [2] is unknown except volume is $800cm^3$. As a shown in Fig. 3-7, it is thought that one of cause of difference small scale test and literature [2] is vessel dimension. Because ARF is affected by surface area and solution depth. We used 2 kinds of test vessels in small scale test. But, we have not been able to elucidation yet relationship between vessel dimension and ARF, because there are few results. We continue research to elucidate relationship between vessel dimension and ARF.

(b) Gas flowrate

It is necessary to be shorten pressure rising time in the small scale test, and to simulate the pressure fluctuation at the time of the hydrogen explosion, because the pressure fluctuation by the actual hydrogen explosion progress quickly. If gas flowrate is increased to make pressure rising time shorten, the amount of released aerosol might be detected higher than actual case. Because, splash of the solution is generated due to increasing gas flowrate. In small scale test, we identified a surface condition as gas supply flowrate beforehand, and we set max gas flowrate 20L/min & pressurization time 1min (refer to Table. 3-2).

(c) Pressure holding time

As a shown in Fig. 3-5, The ARF increases as pressure holding time becomes long. Literature [2] does not be described about the pressure holding time. But, As a report of literature[1], it is described that dissolved gas has reached to solution - vapor equilibrium basically in this test condition. Therefore, It is expected that enough pressure holding time is secured as for the test of the NUREG report [1] [2]. In consideration from this, it is thought that one of cause of difference small scale test and literature [2] is pressure holding time.

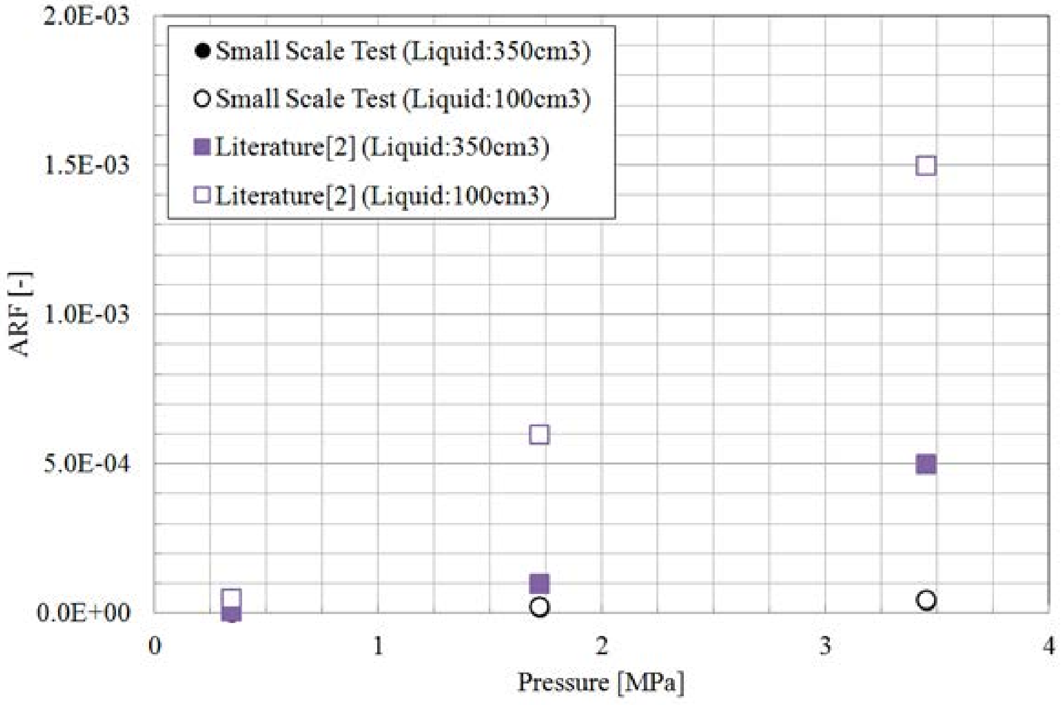


Fig. 3-7 Comparison of small scale test result with NUREG report [2]

Table. 3-2 Relationship gas flowrate and solution surface condition

gas flow rate [L/min]	solution surface condition	
28		ruffling solution
65		splashing solution

3.2 Annular vessel test

3.2.1 Result

Fig. 3-8 is shown ARF of Run21. Most of the aerosol released from annular vessel attached to pipe A. Thus, the following ARF is used in the sense of total value of all (Pipe A, mist trapped vessel and filter).

Fig. 3-9 and 3-10 are shown pressure sensor profile at Run21. The value of high pressure peak was increased with distance from ignition position. The maximum pressure was 1.1 MPa which was near point of offgas outlet pipe.

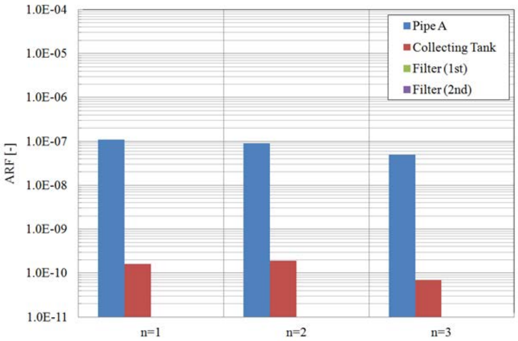


Fig. 3-8 ARF of annular vessel test (Run 21)

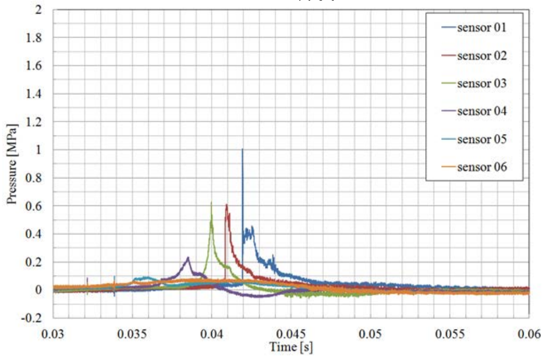


Fig. 3-9 Pressure profile at Run21

(Sensor No.1~6)

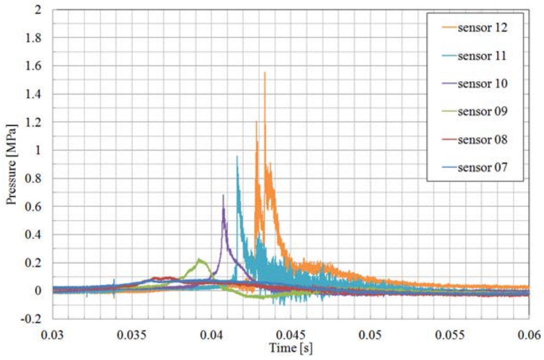


Fig. 3-10 Pressure profile at Run21

(Sensor No.7~12)

Table. 3-3 is shown ARF of annular vessel test. The dispersion of test result was bigger than small scale test, but all results of ARF were 10^{-10} ~ 10^{-6} order which were smaller than a small scale test.

Table 3-3 ARF of annular vessel test

Run.	ARF[-]		
	n=1	n=2	n=3
21	1.1×10^{-7}	9.0×10^{-8}	4.9×10^{-8}
22	1.8×10^{-9}	2.1×10^{-9}	2.7×10^{-8}
23	6.5×10^{-10}	1.7×10^{-10}	3.1×10^{-9}
24	1.9×10^{-6}	5.6×10^{-8}	7.2×10^{-7}
25	1.5×10^{-7}	2.6×10^{-8}	6.8×10^{-8}
26	6.0×10^{-6}	2.9×10^{-6}	4.4×10^{-6}
27	5.3×10^{-8}	1.4×10^{-8}	1.6×10^{-8}

(a) Dependence of pipe A arrangement (number of bend & length)

Fig. 3-11 is shown ARF of Run 21~23. Run 21is the highest in the result of ARF, and Run 23 is the lowest. We have not yet understood it about the reason of this tendency, but it is expected that the result of the ARF is affected by the off gas outlet pipe arrangement.

(b) Dependence of ignition position

Fig. 3-12 is shown ARF of Run 21 and Run 24~25. Each results of ARF were 10^{-7} order. We confirmed that the result of ARF was not influence at the ignition position

(c) Dependence of H_2 concentration

Fig. 3-13 is shown ARF of Run 21 and Run 26~27. The result of ARF in Run 21(30 vol %) was 1.0×10^{-7}. On the other hand, The result of ARF in Run 26 (20 vol %) was 5.0×10^{-6} which was higher than Run 21. We have not yet understood it about the reason of this tendency, but it is expected that the result of the ARF is affected by H_2 concentration.

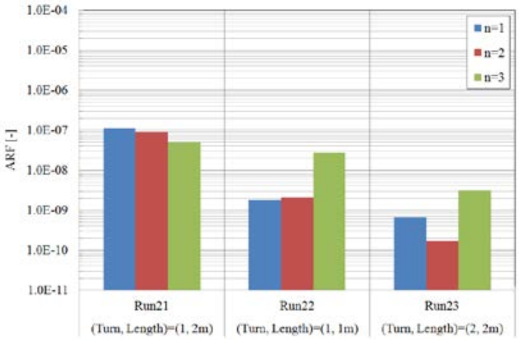


Fig. 3-11 Relationship between ARF and pipe A arrangement in annular vessel t

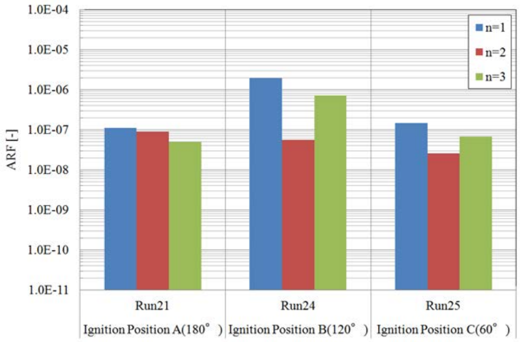


Fig. 3-12 Relationship between ARF and ignition position in el test

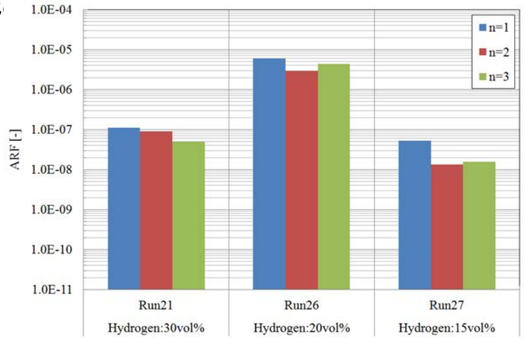


Fig. 3-13 Relationship between ARF and hydrogen concentration in annular vessel test

3.2.2 Discussion

3.2.2.1 Summary of annular vessel test

We show that the summary of annular vessel test result as follows.

- It is expected that the result of the ARF is affected by the off gas outlet pipe arrangement.
- ARF was not influence at the ignition position
- ARF is affected by H_2 concentration.

3.2.2.2 Comparison of annular vessel test result with small scale test result

Fig. 3-14 and Fig. 3-15 are shown pressure sensor profile at Run 26 & 27. We confirmed very sharp max pressure peak at Run 21(ca. 1.1MPa). On the other hand, we confirmed broad max pressure peak at Run 26(ca. 0.1MPa). A pressure fluctuation was hardly seen in Run 27.

We could confirm the ARF increases in proportion to pressure at small scale test. But, we couldn't confirm the same tendency in annular vessel test. It is not clear from these results whether the cause that is mismatched for the ARF tendency between small scale test and annular vessel test is due to test vessel dimension or difference of releasing mechanism.

While several challenge remains in this test, we could confirm that ARF provided in hydrogen explosion test (using test equipment simulated gas phase part of the annular vessel) was smaller than small scale test.

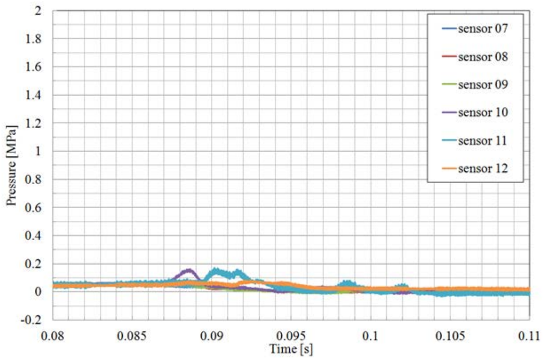


Fig. 3-14 Pressure profile of annular vessel test (Run26)

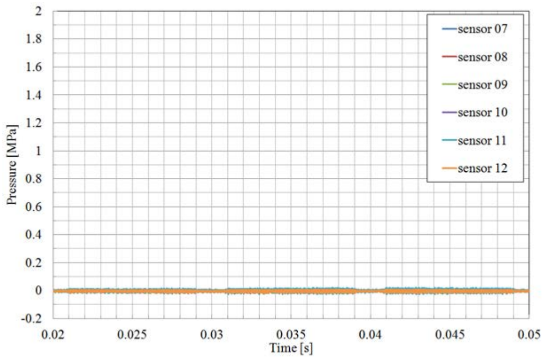


Fig. 3-15 Pressure profile of annular vessel test (Run27)

4. Conclusions

In conclusion, ARF in experiments is far smaller than result of previous report "NUREG/CR-3093". Test equipment is simulated the vessel in the reprocessing plant, so we get the useful data for evaluating environment assessment in hydrogen explosion. In the future, we continue the research for applying these data to another type vessel.

However, there are still many challenges about elucidation of the release mechanism at the time of the hydrogen explosion accident. We hope that the experiment on ARF in hydrogen explosion accident is carried out in other institutions.

5. References

[1] U. S. Nuclear Regulatory Commission, "Nuclear Fuel Cycle Facility Accident Analysis Handbook", NUREG/CR-6410(1998)

[2] U. S. Nuclear Regulatory Commission, "Aerosols Generated by Releases of Pressurized Powders and Solutions in Static Air" NUREG/CR-3093, PNL4566 (1983)

OECD-NEA CSNI Workshop
15-18 November 2016, Aomori City, Japan

Development of Standard Procedure for Consequence Analysis of Criticality Accident in Fuel Cycle Facilities

Y. Yamane and H. Abe
Japan Atomic Energy Agency
2-4 Shirakata Shirane, Tokai-mura, Naka-gun
Ibaraki, Japan 319-1195

yamane.yuichi@jaea.go.jp

ABSTRACT

After the experience of the accident at the Fukushima Dai-ichi Nuclear Power Plants of Tokyo Electric Power Company (TEPCO), Japan's nuclear safety standard for reprocessing plant has been renewed and the standard now requires accident management measures and the assessment of their effectiveness. Newly defined severe accidents of reprocessing plant include criticality, evaporation to dryness of high active liquid waste (HALW), fire of organic solvents, and explosion of hydrogen generated by radiolysis which accidents occur beyond DBA conditions. For such accidents, the assessment of accident management measures needs studies on accident scenarios and source terms.

Not an overestimation but the best estimation is required to maximize the total safety of both the public and workers on an accident management because an overestimated dose of them may lead a wrong decision for the accident management plan. This paper summarizes the issues for the best estimation in criticality accident consequence analysis and proposes a new method to estimate source term in criticality accident.

Unique characters of criticality accident in nuclear fuel facilities are described in association with the best estimation of public and worker's dose. Described are the unique characters such as;

- *Its safety is fully considered as Design Basis Accident*
- *Radioactive materials are created in the accident*
- *Workers' dose by direct radiation of neutron and gamma-ray should be considered*
- *Hydrogen and oxygen gases are created and will be a source term of their explosion*

In the light of those characters, this paper proposes a procedure to estimate source term in criticality accident by utilizing five-component equation described in DOE handbook. In the procedure, the latest methods to estimate MAR, Material At Risk, is summarized. Then the issues on estimating other components such as DR, Damage Ratio, ARF, Airborne Release Fraction, RF: Respirable fraction, *LPF, Leak Path Factor, are discussed as well as remaining issues for the best estimation of source term.*

1. INTRODUCTION

Prevention of criticality accident is one of the most important and basic requirements for nuclear fuel facilities. In Japan, such facilities were designed along with the idea of Defense in Depth to prevent the supposed accidents including inadvertent criticality. For the accidents supposed as the design basis accident (DBA) and siting evaluation accident (SEA), it was confirmed that the estimated public dose were lower than a criteria determined for the public health. For such deterministic approach to the estimation of the accident effect, an overestimated value of the public dose is acceptable if it is considerably lower than the criteria.

After the experience of the accident at the Fukushima Dai-ichi Nuclear Power Plants of Tokyo Electric Power Company (TEPCO), Japan's nuclear safety standard for reprocessing plant has been renewed and the standard now requires accident management measures and the assessment of their effectiveness. For example, newly defined severe accidents of reprocessing plant include criticality, evaporation to dryness of high active liquid waste (HALW), fire of organic solvents, and explosion of hydrogen generated by radiolysis which accidents occur beyond DBA conditions. For such accidents, the assessment of accident management measures needs studies on accident scenarios and source terms, and graded approach based on probabilistic risk assessment (PRA) has been recommended by Atomic Energy Society Japan[1)]. In the graded approach, the key point is the estimation of accurate value of the public dose and the best estimation is required other than the overestimation.

Not an overestimation but the best estimation is also required to maximize the total safety of both the public and workers on an accident management because an overestimated dose of them may lead a wrong decision for the accident management plan. For example, the second plan might be chosen if too much dose of workers is expected during the best action to minimize the public dose.

This paper summarizes the issues for the best estimation in criticality accident consequence analysis and proposes a new method to estimate source term in criticality accident, which method is expected to be useful for the graded approach.

Unique characters of criticality accident are described in association with the best estimation of public and worker's dose. Described are the unique characters such as;

- Its safety is fully considered as Design Basis Accident
- Radioactive materials are created in the accident
- Workers' dose by direct radiation of neutron and gamma-ray should be considered
- Hydrogen and oxygen gases are created and will be a source term of their explosion

This paper explains the characteristics of criticality accident in nuclear fuel cycle facility and, in the light of those characters, proposes a procedure to estimate source term in criticality accident by utilizing five-component equation described in DOE handbook[2)]and NUREG/CR-6410[3)]. In the procedure, the latest methods to estimate MAR are summarized at the first. Then the issues on estimating other components such as DF, ARF, RF, LPF are discussed as well as remaining issues for the best estimation of source term.

2. BACK GROUND OF CRITICALITY ACCIDENT

The word of "Criticality" is defined in the glossary of IAEA[4)] as "The state of a nuclear chain reacting medium when the chain reaction is just self-sustaining (or critical), i.e. when the reactivity is zero. Often used, slightly more loosely, to refer to states in which the reactivity is greater than zero." And the word of "Criticality Accident" is defined in ANS-8.1 as "The release of energy as a result of accidental production of a self-sustaining or divergent neutron chain reaction."[5)]

Twenty two cases of the criticality accident that occurred during process operation were reported[6)], in which 21 occurred in solution fuel or slurry. The most of them occurred during about 10 years around 1960. The typical condition was the failure of mass-control with highly enriched uranium solution in a cylindrical tank[7)8)]. The number of fissions released in the almost cases were between 10^{15} and 10^{18}. So far, 4×10^{18} in Idaho Chemical Process Plant (ICPP) has been the worst case.

A criticality accident in solution fuel can be categorized into two types depending on whether it reaches boiling state or not. For small reactivity insertion, the temperature of the solution fuel might not reach boiling point. It reaches, however, boiling point if large reactivity is inserted or the fuel solution has positive or very small negative feedback reactivity against to temperature increase. It has been reported that dilute plutonium nitrate solution has such unique property[9)].

In non-boiling case, the solution fuel becomes subcritical mainly due to the temperature feedback negative reactivity, for which a typical profile of power, fission energy released per unit time, is shown in **Fig.1**. After the reactivity insertion, the power profile shows exponential increase and decrease. Then, it shows gradual decrease which is kept by the decay of delayed neutron precursors. This peak and following decrease in the power profile is called "power burst." After the burst, the power reaches a steady level called "plateau." If the solution fuel container has no forced cooling system, the level of the plateau is very low and it is considered that the criticality accident has temporally terminated. In JCO criticality accident, the precipitation tank had a forced cooling system and the power level in plateau was kept high corresponding to the power of the cooling system[10)]. To terminate the criticality, The operation needed were turning off the cooling system, cutting the pipe to drain the cooling water, then throwing neutron poison such as Boron into the precipitation tank.

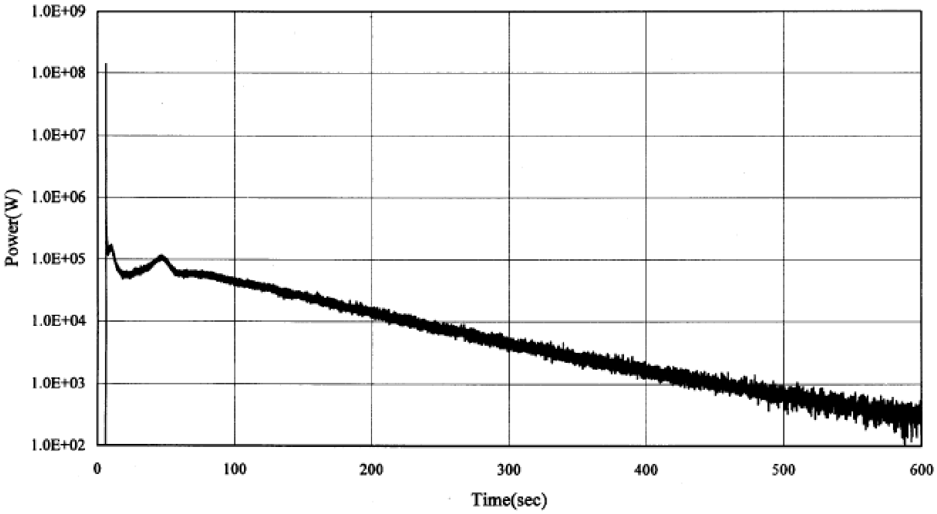


Fig.1. Typical power profile obtained in transient criticality experiment (R166 PW mode 1.5$) (reprint from ref. 11) [11].

In boiling case, the solution fuel is kept critical mainly due to the balance between the inserted reactivity and void feedback reactivity. After the reactivity insertion, the power oscillates around a high average level, which is determined by temperature increase and radiolytic gas void until the temperature of the solution fuel reaches boiling point. During the boiling, boiling void gives the dominant feedback reactivity and the power keeps a steady level called plateau. The loss of neutron moderator due to the evaporation of water by the boiling terminates the criticality. High power of cooling can cause to increase the power level of plateau. If the fuel solution is in a closed system, the return of the vaporized water to the solution fuel due to condensation can lead a long time criticality with a high power. In 1959, occurred a criticality accident in which 800Lit. of uranium nitrate solution had been boiling for 15 – 20 min. and it terminated after a half of the solution had vapored. In that case the number of fission was 4×10^{19}, that is the maximum in history[6].

3. CHARACTERISTICS

Criticality accident has unique characteristics which should be taken into account for the best estimation of the public and worker's dose.

- It has been fully considered to prevent criticality accident along with the idea of the defense in depth such as prevention, mitigation, etc. That makes it difficult to consider the scenario because the severe accident is defined to occur in the condition beyond DBA such as a huge earthquake in which S-class earthquake resistant container could be deformed to leak the nuclear fuel solution.

- Radioactive materials are created in the accident and short life nuclides are important to estimate the dose of public and workers. Nuclides are in a variety of chemical forms such as organic and inorganic substances, gas and aerosol, etc.
- Worker's dose by direct radiation of neutron and gamma-ray should be considered. Such dose even for the neighborhood of the site became an issue in JCO criticality accident in 1999.
- Hydrogen and oxygen gases are created and will be a source term of explosion.

4. CONSEQUENCE ANALYSIS METHODS

From the historical background and unique characters of criticality accident, the highest priority is given to solution fuel for the development of consequence analysis method because hydrogen, which is good moderator and reflector of neutron, is contained in solution fuel more than in solid and powder fuel and the minimum critical mass of solution fuel is the lowest among them. Its shape depends on the geometry of the container. The radioactive materials can easily transfer from the fuel solution to the air.

A criticality accident occurs when the state of nuclear fuel materials changes from a subcritical state to a critical state by initiating event, which is called "reactivity insertion." In the initial state, the fuel materials are under criticality safety control, in which controlled are one or more of parameters such as mass, concentration and geometry. Then one or more initiating events change one or more such controlled parameters to give rise to criticality accident. It should be studied in detail that the reactivity insertion, which has an important role on the power profile and fission yield.

This paper describes the consequence analysis method to mainly the criticality accident in solution fuel initiated by the change of its geometry. This scenario is simple because one fuel material can be critical by itself. The reactivity insertion by the mixture of fuel materials requires more than two solutions. The scenario here has a wide range of applications such as the transfer of solution fuel from a container under the criticality safety control to one not under such control by the failure of operation or the malfunction of apparatus, the leak of solution fuel due to the deformation of S-class earthquake resistant container by a huge earthquake beyond design basis, etc.

In a fuel reprocessing plant, there are a lot of fuel materials in a variety of states of matter and many initiating events can be supposed to happen. The application of the methods to the criticality accident initiated by the event other than geometry change is also mentioned as needed.

4.1 Five component equation

A simple equation has been developed to express Source Term (ST) as a function of 5 components as follows[3];

$$ST = MAR \cdot DR \cdot ARF \cdot RF \cdot LPF$$

where *MAR* : material at risk(in mass or volume), *DR* : damage ratio(-), *ARF* : airborne release fraction(-), *RF*:

Respirable fraction(-), *LPF* : leak path factor(-). When this equation is used to estimate *ST* of an accident, each factor is commonly not a function of time. So it is too simple to perform the best estimation for the severe accident, while it is tentatively utilized here to estimate source term because there no better method at the moment. Key points such as the treatment of short life nuclides and remaining issues are also described and the analysis method of the dose by direct radiation is noted later.

4.2 MAR

MAR is estimated in 3 steps: (1) criticality calculation, (2) estimation of the number of fission, (3) estimation of radioactive materials production. In (1), the effective neutron multiplication factor, k_{eff}, kinetics parameter such as prompt neutron life time, etc., and reactivity temperature and void coefficients are calculated for the target fuel material based on the atom number density and geometry of the material which has the cross section of neutron reaction of fission, absorption and diffusion. With those data, the number of fission released by the time, *t*, is estimated by using methods like one-point kinetics in (2). In (3), estimated is the quantity of radioactive materials produced in the criticality accident.

(1) Criticality calculation

At the first, neutron multiplication factor, k_{eff}, inserted reactivity and insertion ratio, kinetics parameter such as prompt neutron life time, effective delayed neutron fraction, βeff, are calculated. The criticality of the nuclear fuel is evaluated based on the criteria described in the criticality handbook[12)].

A typical method for criticality calculation is Monte-Carlo calculation. Some codes such as MCNP[13)] and SCALE[14)] in USA and MVP[15)] in Japan have been developed and used. Those codes can precisely calculate the k_{eff} of the materials in very complicated geometry, while it takes a time appropriate to reduce statistical error.

The kinetics parameters are calculated by deterministic method in common because that is free from statistical error. Some codes such as SRAC[16)] and DANTSYS[17)] have been developed and used. They are used with a simple geometry such as a cylinder which simplifies the actual geometry

Reactivity temperature coefficients are calculated in following steps; 1) calculate k_{eff} at some values of temperature between room temperature and boiling point, 2) calculate reactivity difference of each k_{eff} value from one at room temperature, 3) fit a line or quadratic curve to the points to obtain reactivity coefficients. For reactivity void coefficients, the same procedure but void ratio (%) is used instead of temperature.

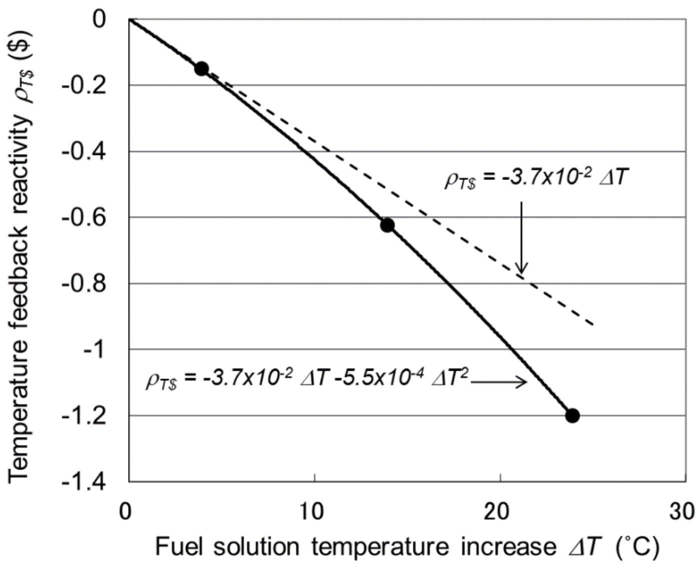


Fig.2 An example of the calculation of reactivity coefficients by fitting a curve to the points (reprint from ref. 18)

For reactivity insertion ratio, which is needed for kinetics calculation, a number of the values of k_{eff} are calculated at the times during the insertion. In the case that criticality is achieved by the inflow of additional solution fuel, for example, such calculations are done at some solution levels each of which corresponds to a time during the reactivity insertion. The each value of k_{eff} is converted to the reactivity difference from the initial value and a line or curve is fitted to them. Because the way of reactivity insertion affects the power profile, the order of approximation in the fitting depends on the required accuracy. Instantaneous reactivity insertion, i.e. the all reactivity is inserted instantaneously, can be used to estimate the maximum value of the power and fission yield. For the case that concentrated solution fuel flows into water or dilute solution fuel, k_{eff} is calculated at some solution fuel concentrations and levels. For the case that the geometry changes in course of time, the same procedure is used to calculate k_{eff}.

(2) Estimation of number of fission

(i) Fissions in power burst

After nuclear fuel reaches criticality, it is common that the power increases and decreases rapidly to form a peak called “power burst” in its profile due to the increase in the fuel temperature. In metal fuel, the power profile simply shows rise and fall. In solution fuel, the fuel temperature must not beyond boiling point and oscillation in the power profile may be observed, that is due to radiolytic gas void. It has been observed that many criticality accidents terminated after power burst.

For the estimation of the number of fission and/or the power profile, developed are a lot of method such as kinetics analysis, pseudo steady method, simple equation, analytical solution of one-point kinetics equation.

Kinetics analysis is the best in general versatility. There are many codes which solves one-point kinetics equation[19)-24)] and neutron transport equation[25)]. A lot of those codes are developed for solution fuel and there are some for metal or powder fuel. As an example of one-point kinetics code, AGNES code[19)] is described briefly below. Using total reactivity ρ(-), effective delayed neutron fraction β_{eff}(-), neutron generation time Λ(s), decay constant of the i-th delayed neutron precursor λ_i and neutron source power S(W/s/m^3), the power density P(W/m^3) and the i-th delayed neutron precursor density C_i(m^{-3}) satisfies following one-point kinetics equations;

$$\frac{dP}{dt} = \frac{\rho - \beta}{\Lambda} P + \sum_{i=1}^{6} \lambda^i C^i + S, \qquad \frac{dC^i}{dt} = \frac{\beta^i}{\Lambda} P - \lambda^i C^i,$$

where

$$\rho = \rho_0 + \rho_T + \rho_V,$$
$$\rho_T = \alpha_{T1} T + \alpha_{T2} T^2,$$
$$\rho_V = \alpha_{V1} V + \alpha_{V2} V^2,$$

ρ_0: excess reactivity, ρ_T: temperature feedback reactivity, ρ_V: void feedback reactivity, α_{xn}: coefficients of feedback reactivity (x=T, V, n=1,2).

The following thermal balance equations are solved in three regions: fuel (j=1), wall of tank (j=2), cooling material (j=3).

$$V_i (\rho C_p)_j \frac{\partial T_i}{\partial t} = \gamma_j P V_i + (hA)_{i-1}(T_{i-1} - T_i) - (hA)_i (T_i - T_{i+1}) \qquad \textit{for j=1,2,}$$

$$V_i (\rho C_p)_3 \frac{\partial T_i}{\partial t} = \gamma_3 P V_i + (hA)_{i-1}(T_{i-1} - T_i) - \varpi C_{p3} (T_{out} - T_{in}) .$$

where V_i: volume of the i-th region (See ref 19). The power and temperature are calculated by solving numerically those equations.

For the case that the solution fuel reaches boiling state, suitable is pseudo steady state method[26)27)], which can calculate average power and energy profiles during the transient state where power oscillation due to radiollytic gas void exists. The calculation procedure is as follows;

i) calculate void ratio *f*(-) from inserted reactivity ρ_{in}(t),

$$f = \rho_{in} / \alpha_f$$
$$aQ = f(1 - f)$$

ii) calculate power Q(W) from *f*(-),
iii) calculate temperature increase ΔT (K) from Q(W),
iv) calculate void ratio *f*(-) from total reactivity ρ_t,
repeat ii) through iv)

$$\frac{d\Delta T}{dt} = KQ$$

$$f = (\rho_{in} - \alpha_T \cdot \Delta T)/\alpha_f$$

where α_f: void reactivity coefficient.

The equations have been developed[28] in which averaged power and energy are a function of time, which equation has been derived by solving approximately the system of equations of the procedure of the pseudo steady state method. The simple equations have been developed[29)-34] to estimate the maximum value of the number of fissions under restricted condition in fuel concentration and/or geometry. The analytical solution of one-point kinetics equation can be used to easily obtain almost the same result as that of kinetics code under the condition such that there is no strong forced cooling[35,36].

(ii) Fissions in plateau

In the criticality accident in solution fuel, it reaches a steady boiling state called "plateau," when the fuel temperature reaches the boiling point. While there was no boiling, in the case of the JCO criticality accident in 1999, a high power continued for about 20 hours, which is also called plateau. The high power was understood to be equivalent to the cooling power of the forced cooling jacket surrounding the precipitation tank in which the nuclear fuel solution became critical. That indicates the power balancing with cooling power is kept in the plateau, and forced cooling can make the plateau power higher. The plateau power is thought to be low for the criticality accident in metal fuel in the air because the cooling power of natural convection of air is not so high.

For the power in plateau with boiling of uranium or plutonium nitrate solution, $Q=7.96fV$ is proposed, where *V*: the solution volume[27].

(3) Estimation of radioactive materials

The quantity of radioactive materials such as Cs and I is calculated based on the estimated number of fissions. For such purpose, ORIGEN[37] and SWAT[38] codes are used. There are tables in DOE handbook that lists the materials produced by 10^{19} fissions. But the calculation specific to own scenario should be done and the materials being contained before the occurrence of the criticality accident should be taken into account for accurate estimation.

4.3 Other factors

For the criticality accident in solution fuel, conservative value is proposed in DOE handbook and NUREG/CR-6410: DR = RF = LPF = 1 for gas and aerosols; ARF = 1 for noble gas, 0.05 for Iodine, etc. as shown in **Table 1**. Those values are not the best for the ideal estimation, and it is expected that much better values would be proposed in future. In the case of metal or powder fuel, it can be expected that almost all fission products except noble gas remain in the fuel. Radiolytic gas, which can be a driving force to take out radioactive materials from the container, is not produced in solid fuel. So, the source term can be small at the criticality accident in metal fuel. In powder system, it is known that thermal decomposition of Zinc stearate, which is used as lubricant for mixture of powders, produces some gases which may be such a driving force[39)].

4.4 Exposure to direct radiation

Air dose rate of neutron and gamma-ray from the fissions can be calculated by using Monte-Carlo or transport calculation codes such as ANISN[40)] with the intensity of the radiation source or the amount of source term, the distance between the source, nuclear fuel, and the width of isolator such as concrete. The intensity of the radiation source as a function of time can be calculated by using one-point kinetics code.

The analytical solution of one-point kinetics can be used to estimate the peak power in the burst of the criticality accident in solid, powder and non-boiling solution fuel[41)]. In the case that the burst with power oscillation due to continuous insertion of reactivity, an equation is proposed between average power Q and radiolytic gas void ratio f[26)];

$$f(1-f) = \frac{\left(1-0.36C_N^{0.45}\right)\left(2.25-0.2C^{0.33}\right)+0.15+0.17C_N^{0.7}}{S \cdot v_\infty} 2.82 \cdot Q \,,$$

where C: fuel concentration, C_N: acid molarity, S: cross section of tank (m^2), v_∞: rising velocity of a bubble.

Table 1 Release Fraction of Various Chemical Classes from Heated Spent Fuel
(reprint from ref. 2).

Group #	Group Name	Rep. Ele.	Elements in Group	ARF
1	Noble Gases	Xe	Xe, Kr, He, Ne, Ar, Rn, H	5E-1
2	Alkali Metals	Cs	Cs, Rb, Li, K, Fr, Na	2E-1
3	Alkali Earths	Ba	Ba, Sr, Mg, Ca, Ra, Be	3E-2
4	Halogens	I	I, F, Cl, Br, At	5E-2
5	Chalogens	Te	Te, S, Se, O, Po, N	7E-2
6	Platinoids	Ru	Ru, Rh, Pd, Os, Ir, Pt, Au, Ni	2E-3
7	Transition Metals	Mo	Mo, V, Cr, Fe, Co, Mn, Nb, Tc	3E-2
8	Tetravalent	Ce	Ce, Ti, Zr, Hf, Th, Pa, U, Np Pu	4E-4
9	Trivalent	La	La, Al, Sc, Y, Ac, Pr, Nd, Pm, Sm, Eu, Gd, Tb, Dy, Ho, Er, Tm, Yb, Lu, Am, Bk, Cf	6E-4
10	Main Group I	Cd	Cd, Hg, Zn, As, Sb, Pd, Tl, Bi	4E-3
11	Main Group II	Sn	Sn, Ca, In, Ag	4E-3
12	Boron	B	B, Si, P, C	6E-4

5. REMAINING ISSUES

Many issues are remaining to be solved from the view point of applying the five component equation to the best estimation of the criticality accident effect as the dose of the public and workers.

In DOE handbook, DR = RF = LPF = 1 and some values are given to ARFs. The experimental data of recent years suggest smaller values for those factors, while much more data are needed to determine much accurate value of the factors. It is expected that much accurate values of the factors will be proposed by improving experimental data and studying the data to develop evaluating models.

5.1 Temporal change of MAR

Most of the criticality accidents in solid fuel and non-boiling solution fuel temporarily terminate at the end of their power burst and the power becomes very low in several or several ten minutes. Some radioactive materials such as I-133 are created by going through decay series and that may take more than several ten minutes. For example, it is reported that I-133 increases in the air above the fuel solution a few hours after the power burst[42)] as shown in **Fig. 3**. In a decay series, some nuclides can be gas or aerosol. It must be considered that noble gases don't react to other material and are not trapped by filter, and some material such as Xe-138 (half life: 14.08 min.) could become aerosol from gas in the outside of the building.

5.2 Driving force

The transport of radioactive materials needs a driving force, without which they are thought to remain in the building. It is important to clarify the behavior of such driving force to determine the value of ARF and LPF. In criticality accident, radiolytic gas, NOx gas and vapor can be a driving force. Water vapor becomes the main driving force in the case of boiling. The data of radiolytic gas and NOx obtained in TRACY experiments are reported[43)]. But more data are needed to develop an estimation model.

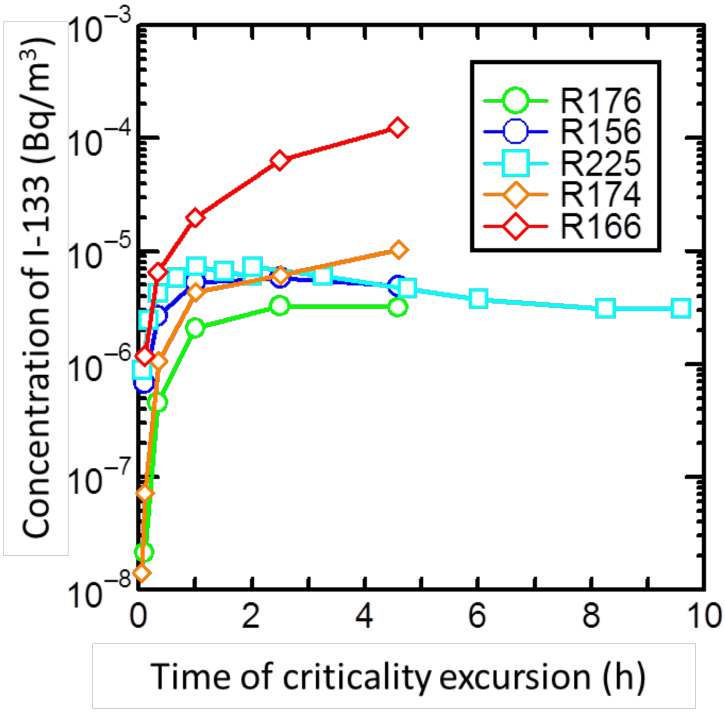


Fig. 3 The concentrations of I-133 in the air are different from each other depending on the experimental condition such as the excess reactivity (reprint from ref. 42).

5.3 ARF and RF

The rate of the transport of the radioactive materials from the fuel solution to the air may depend on the condition such as their concentration in the solution and may change in course of time due to the chemical reactions in the solution. **Figure 4** shows that the release rate of I-133 changes in course of time and are different from each other.

In the criticality accident in solution fuel, the bubbles of radiolytic gas, etc. break on the solution surface to make the mist of radioactive materials. It can be supposed that ARF becomes higher in boiling state than in non-boiling state, while that has been not confirmed experimentally. RF depends on the size of mist particle or aerosol, and the aerosol originated from noble gas is thought to be easily inhaled, RF ≈ 1, because its size must be very small.

5.4 LPF and DR

Among important radioactive materials, there are the data of Cs with high temperature in reactor accidents. It has not been studied whether such data are applicable to the accident in fuel reprocessing plant. As for iodine, the adhesive property of I_2 is known. It is reported that in TRAY experiment I_2 and organic Iodine species such as CH_3I were produced and about 60% of collected iodine was CH_3I[43]. It is known that CH_3I is not adhesive and can easily transport to the outside of the container. The ratio of such chemical species should be taken into account by setting the value of DR properly. It is supposed that there are the vapors of water and nitrate acid in boiling state. However, it is not experimentally confirmed whether the chemical form of iodine may change or not in such condition. As described in the previous section, experimental data imply LPF < 1 for iodine, but much experimental studies are needed to determine the accurate value of LPF. The adhesive property of Ru should be taken into account and the data obtained from the studies on the evaporation to dry out accident can be utilized.

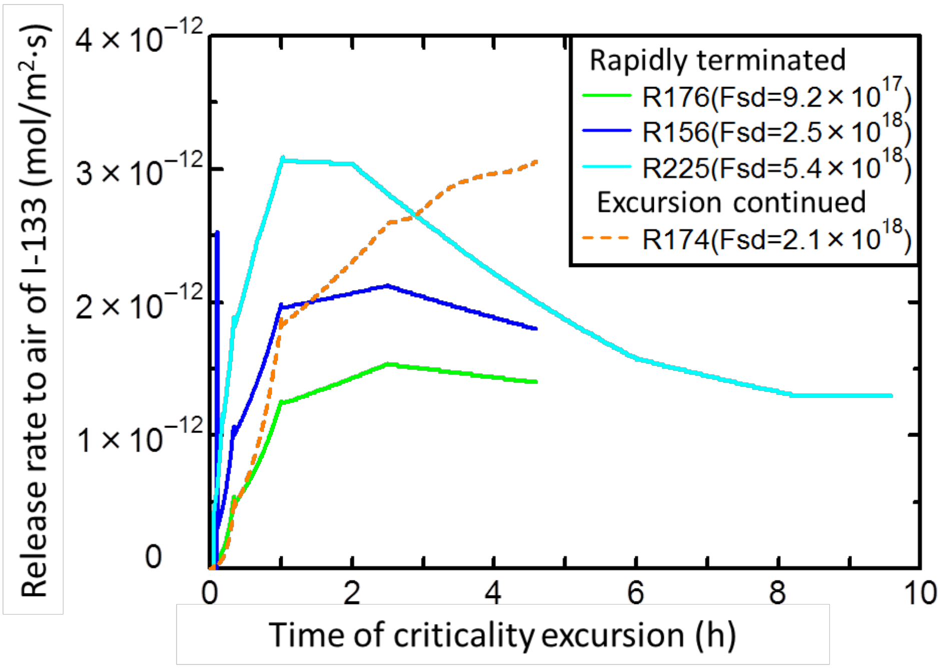


Fig. 4 Release Rates of I-133 are different from each other due to the experimental condition such as the excess reactivity (Fsd: Total number of fission)(reprint from ref. 42).

5.5 Hydrogen gas

In the criticality accident in solution fuel such as uranyl nitrate solution, hydrogen gas is produced because the radiation decomposition of water produces hydrogen and oxygen. Hydrogen gas is a source of the both fire and explosion accidents, and the production of hydrogen gas is important to evaluate the total effect of the accident. It is reported that the G-value of hydrogen molecule obtained in transient criticality experiments is between 0.6 and 1.8[44)-45)]. Much more data under various experimental conditions such as the fuel concentration, acid molarity, etc. are needed to develop a model for the accurate estimation of the hydrogen gas production in criticality accident.

6. SUMMARY

The application of five component equation to the criticality accident as the severe accident in nuclear fuel facility is developed and the remaining issues are explained. The source term is thought to be overestimated by the method, but it is currently the result of the best estimation. It is expected that the value of the factors are improved by taking into account the characteristics of criticality accident.

7. ACKNOWLEDGEMENT

The authors wish to thank Dr. Satoshi Sakurai for the helpful comments on the paper.

REFERENCES

1) The Severe Accident Study Working Group for Nuclear Fuel Cycle Facilities Phase-I Report, http://www.aesj.or.jp/~recycle/sawg/sawg_report_final.pdf, (2014).

2) DOE-HDBK-3010-94, "AIRBORNE RELEASE FRACTIONS/RATES AND RESPIRABLE FRACTIONS FOR NONREACTOR NUCLEAR FACILITIES," (2013).

3) Science Applications International Corporation, "Nuclear Fuel Cycle Facility Accident Analysis Handbook," NUREG/CR-6410, (1998).

4) https://www.iaea.org/ns/tutorials/regcontrol/intro/glossaryc.htm

5) ANSI/ANS-8.1-2014, Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors, (2014).

6) McLaughlin TP, Monahan SP, Pruvost NL, Frolov VV, Ryazanov BG; Sviridov VI. A review of criticality accidents 2000 revision. LA-13638; 2000.

7) Nakajima K. Application of simplified methods to evaluate consequences of criticality accident using past accidentdata.InternationalConference onNuclearCriticality Safety ICNC 2003; pp. 171–176; Tokai-mura; 2003.

8) K. Yoshida, H. Abe, Y. Yamane, S. Tashiro and K. Muramatsu, "Research on the State-of –art of Probabilistic Safety Assessment for Non-reactor Nuclear Facilities (II), JAEA-Research 2007-002, (2007)[in Japanese].

9) IRSN

10) Atomic Energy Society Japan JCO accident investigation board, "JCO Criticality Accident; Clarification of its whole picture," (2005)[in Japanese].

11) K. Nakajima, et al., JAERI-Data/Code 2002-005, 2002

12) Nuclear Criticality Safety Handbook (English Translation), JAERI-Review 95-013, (1995).

13) MCNP Monte Carlo Team. MCNP-a general Monte Carlo N-Particle transport code, version 5. La-UR-031987, Los Alamos National Laboratory; 2003.

14) B.T. Rearden and M.A. Jessee, Eds., SCALE Code System, ORNL/TM-2005/39, Version 6.2, Oak Ridge National Laboratory, Oak Ridge, Tennessee (2016). Available from Radiation Safety Information Computational Center as CCC-834.

15) Nagaya Y, Okumura K, Mori T, Nakagawa M. MVP/GMVPII: general Monte Carlo codes for neutron and

photon transport calculations based on continuous energy and multi-group methods. JAERI 1348; 2005.

16) Okamura K, Kaneko K, Tsuchihashi K. SRAC95;General purpose neutronics code system. Tokai(Japan): Japan Atomic Research Institute; 1996, JAERI-Data/Code 96-015.

17) Alcouffe RE, Baker RS, Brinkley FW, Marr DR, O'Dell RD, Walters WF. DANTSYS : A diffusion accelerated neutral particle transport code system. New Mexico (USA): Los Alamos National Laboratory; 1995, LA-12969-M (1995).

18)Y. Yamane, Improvement in estimation of first peak power based on non-linear temperature feedback reactivity in criticality accident with instantaneous reactivity insertion, J. Nucl. Sci. Technol. 52, No.11, p.1425, 2015.

19) Nakajima K, Yamane Y, Miyoshi Y. A kinetics code for criticality accident analysis of fissile solution systems: AGNES2. JAERI-Data/Code 2002–004; 2002.

20) Basoglu B, Yamamoto T, Okuno H, Nomura Y. Development of a new simulation code for evaluation of criticality transients involving fissile solution boiling. JAERI Data/Code 98–011; 1998.

21) Kato R, Nakamura T, Senda Y. The code CREST to simulate criticality accident power excursion in fuel solution. Proceedings of Safety Margin in Critical Safety; 1989 Nov 26–30; San Francisco.

22) Mather DJ, Shaw RJ. CRITEX – A computer program to calculate criticality excursions in fissile liquid systems. SRD R 380; 1984.

23) Dodds HL, Westfall RM. SKINATH – A computer program for solving the reactor point kinetics equations with sample thermal hydraulic feedback, ORNL/ CSD/ TM – 210; 1984.

24) Gmal B, Weber J. FELIX–A computer code for simulation of criticality excursions in liquid fissile solutions. In: Ebert K, Ammon RV. editors. Safety of the nuclear fuel cycle. Deerfield Bch: VCH Publishers; 1989.

25) PainCC, deOliveiraCRE, GoddardAJH, UmplebyAP. Transient criticality in fissile solutions compressibility effects. Nucl Sci Eng. 2001;138:78–95

26) T. Schulenberg, J. D¨ohler, "Heating-up, nucleation and boiling of a critical solution of fissile material," Int. J. Multiphase Flow, 12[5], 759 (1986).

27) Nakajima K, Yamamoto T, Miyoshi Y. Modified quasisteady-state method to evaluate the mean power profiles of nuclear excursions in fissile solution. J Nucl Sci Technol. 2002;39(11):1162

28) Y. Yamane, "Average time profile of energy release fluctuated due to radiolytic gas void in criticality accident," Proc. Fall meeting of Atomic Energy Soc. Japan, 2H13, (2016)[in Japanese].

29) Tuck G. Simplified methods of estimating the results of accidental solution excursions. Nucl Technol. 1974;23:177.

30) OlsenAR, HooperRL, UotinenVO, BrownCL. Empirical model to estimate energy release from accidental criticality. Trans Am Nucl Soc. 1974;19:189.

31) BarbryF, RozainJP, RatelR, ManarancheJC. Criticality accident studies in France: experimental programs and modelisation. Proceedings of International Seminar on Nuclear Criticality Safety, ISCS'87, p.423, Tokyo, Japan; 1987.

32) Nomura Y, Okuno H. Simplified evaluation models for total fission number in a criticality accident. Nucl Technol. 1995;109:142–152.

33) Nomura Y, Okuno H, Miyoshi Y. Validation of simplified evaluation models for first peak power, energy, and total fissions of a criticality accident in a nuclear fuel processing facility by TRACY experiments. Nucl Technol. 2004;148:235–243.

34) Duluc M. New improvement in simplified methods of estimating the number of fissions during a criticality accident in solution. Proceedings of Topl. Mtg. Nuclear Criticality Safety Division (NCSD2009;2009Sep13–17; Richland, Washington (DC).

35) Y. Yamane, "Approximate expression of energy released at 1st power peak in a criticality accident based on non-linear temperature feedback reactivity," Proc. Spring meeting of Atomic Energy Soc. Japan, J25, (2015)[in Japanese].

36) Y. Yamane, "Energy released in criticality accident by instantaneous reactivity insertion to system with non-linear temperature feedback," Proc. Spring meeting of Atomic Energy Soc. Japan, 1O08, (2016)[in Japanese].

37) S.B. Ludwig, A.G. Croff, "Revision to ORIGEN2 – Version 2.2", Transmittal memo of CCC-0371/17, Oak Ridge National Laboratory (2002).

38) T. Kashima, K. Suyama and T. Takada," SWAT4.0－The Integrated Burnup Code System Driving Continuous Energy Monte Carlo Codes MVP, MCNP and Deterministic Calculation Code SRAC," JAEA-Data/Code 2014-028, 2014.

39) Y. Yamane, et al., "Material Data for Criticality Transient Phenomena Analysis of MOX Powder System – Data of MOX and Zinc Stearate Powders- (Contract Research), JAERI-Data/Code 2006-021, (2006).

40) Ward W. Engle, Jr.: " A Users Manual for ANISN: A One Dimensional Discrete Ordinates Transport Code with Anisotropic Scattering," K-1693, 1967.

41) Yamane Y, Improvement in estimation of first peak power based on non-linear temperature feedback reactivity in criticality accident with instantaneous reactivity insertion, Journal of Nuclear Science and Technology, 2015 Vol. 52, No. 11, 1425–1435.

42)S. Tashiro and H. Abe, "Study on Release Behavior of Radioiodine From Fuel Solution under Criticality Accident Condition," JAERI-Technology 2015-044,(2015)[in Japanese].

43) H. Abe, et al., "Study on the Release Behavior of Radioactive Iodine Spices and Noble Gasses from the Fuel Solution under Simulated Nuclear Criticality Accident (Contract Research)," JAERI-Tech 99-067, (1999)[in Japanese].

44) F. Y. Barbry, D. Mangin, H. Revol, C.R.A.C., Recapitulation of Experiment Results, Rapport S.E.E.S.N.C. No. 116, Department de Sˆuret´e Nucl´eaire, Commissariat a l'Energie Atomique, (1973), [in French].

45) R. Yoshida, Y. Yamane and H. Abe, "Evaluation of Hydrogen Gas Production at Transient Criticality in Uranyl Nitrate Solution," Proc. Fall meeting of Atomic Energy Soc. Japan, 2G22, (2016)[in Japanese].

Experimental Evaluation of Release and Transport Behavior of Gaseous Ruthenium under Boiling Accident in Reprocessing Plant

Naoki YOSHIDA, Shinsuke TASHIRO, Yuki AMANO, Kazuo YOSHIDA, Yuichi YAMANE, Hitoshi ABE
Nuclear Safety Research Center, Japan Atomic Energy Agency.
yoshida.naoki@jaea.go.jp

Abstract

The "Evaporation to Dryness due to the Loss of Cooling Functions" (EDLCF) of highly-active liquid waste (HALW) was newly defined as one of the severe accidents in Japan's nuclear safety standard for the reprocessing plant. Studies on accident scenarios and their source terms have led to an increased need for the development of accident management measures and the assessment of their effectiveness. Previous studies have shown that ruthenium was released at a greater rate than other elements because it formed volatile species such as ruthenium tetroxide (RuO_4). In addition, ruthenium isotopes, ^{106}Ru and ^{103}Ru, have radiotoxicity. Accordingly, the accident management measures require the experimental information on the release and transport behavior of the gaseous ruthenium (Ru(g)). This paper summarizes our experimental results on the characteristics of Ru(g) in the EDLCF. Two kinds of tests have been conducted: the “cold airborne release factor test (ARF-tests)” and the “leak path factor test (LPF-tests)”. The ARF-test aimed at evaluating Ru(g) release behavior by using a non-radioactive simulant of HALW. The LPF-tests were conducted for evaluating Ru(g) transport behavior by using flow a reactor with gaseous RuO_4.

The following results were obtained:

I. The ARF-test showed that Ru was released from the simulant of HALW between 120 and 300 °C and the cumulative release rate of Ru was 8.8%.
II. It was also found that two peaks of the ruthenium release were found around 140 °C and 240 °C.
III. The LPF-test showed that the majority of Ru(g) (approximately 70%) was deposited on the transport pathway under the water vapor atmosphere without HNO_3, whereas most of the rest were transported as aerosols.
IV. Under the water vapor atmosphere with HNO_3, Ru(g) passed through the transport pathway without deposition.

These results suggested that the behavior of Ru(g) in the accident was complicated depending on the accident conditions. This work includes the results of the experiments carried out under the agreement among JAEA, Japan Nuclear Fuel Ltd. and Japan Nuclear Energy Safety Organization.

1. Introduction

A series of studies for safety improvement of fuel reprocessing facilities, which consists of the studies for risks of serve accident, the revision of the whole concept of security and the improvement for defense in depth, is urgent issues in Japan in consideration of the lessons learned from the Fukushima-Daiichi accident. The licensing standards were further strengthened, in which countermeasures against severe accidents were newly required as regulatory items. The severe accidents at fuel cycle facilities were defined as serious accidents that occur under conditions exceeding design basis accidents.

Due to the heat release from radioactive decays of fission products in the highly-active liquid waste (HALW), a shutdown of the tank cooling system resulted from the loss of offsite power and failure of accident responses may lead to the Evaporation to Dryness due to the Loss of Cooling Functions (EDLCF).[1] The EDLCF wasn't considered as the design basis accident in the safety evaluation of Rokkasho reprocessing plant because temperature rise of the HALW would be very slow and some effective countermeasures could be most probably taken until boiling. However, for example, if sufficient accessibility for taking the countermeasures can't be kept, the EDLCF could be induced. In this way, an occurrence frequency of the EDLCF is considered very low but effects caused by the EDLCF on the public dose is recognized large because of a large amount of radioactive inventory at the HALW tank. From this point, the EDLCF was defined as one of the severe accidents in fuel reprocessing plant in the new licensing standards.[2]

It is expected that the EDLCF would induce the release of radioactive materials with H_2O and HNO_3 vapor from HALW.[3] It was assumed that Ru was released at a rate greater than other elements because it formed gaseous ruthenium (Ru(g)). In addition, Ru has radiotoxicity for from its isotopes of ^{106}Ru and ^{103}Ru.[4] Accordingly, the experimental information on the release and transport behavior of the Ru(g) were valuable for the accident management measures for EDLCF.

We recognized the EDLCF as one of the most important accident on the risk evaluation for fuel reprocessing plant and have been acquiring data for evaluating release and transport behavior of radioactive materials under the EDLCF before the Fukushima-Daiichi accident. In 2009-2014, we conducted a joint research[1] under the framework agreement with Japan Nuclear Energy Safety Organization and Japan Nuclear Fuel Limited. In this work, we focused on the release and transport behavior of Ru in the EDLCF.

In JAEA, experimental studies for source term data development of Ru(g) has been conducted at the EDLCF in reprocessing plants.[1,5–7] In this paper, we introduce our recent works and the data acquired in the joint research including the experimental results of the release and transport behavior of Ru(g) in the two kinds of tests; a cold small scale test for evaluating the airborne release fraction (ARF-test)[5], and a cold engineering scale test for leak path factor (LPF-test)[1].

2. ARF-Test (Reference 5)

In the ARF-test using a non-radioactive simulant of HALW (s-HALW), the release behaviors of Ru(g) were investigated under the simulated EDLCF accident condition. The contents about ARF-test in this paper correspond to a part of the contents in the reference 5.

2.2. Experimental of ARF-test

2.2.1. Preparation of HALW simulant

A non-radioactive s-HALW containing 28 elements was prepared on the basis of the composition of HALW in a reprocessing plant (**Table 1**).[1,8] The s-HALW was prepared by dissolving the nitrates of each element in a nitric acid aqueous solution (2.0 mol/L). The characteristic feature of the s-HALW was a relatively high composition ratio of lanthanide elements (approximately 40% at a molar ratio). It is important to note that the molar ratio of Ru, which was focused on in this study, was approximately 10%.

Table 1 Composition of s-HALW

Element	Concentration (mol/L)	Element	Concentration (mol/L)
H	1.96	Ag	2.09×10^{-3}
P	3.89×10^{-3}	Cd	3.50×10^{-3}
Cr	7.00×10^{-3}	Sn	2.40×10^{-3}
Fe	2.59×10^{-2}	Sb	7.10×10^{-4}
Ni	3.37×10^{-3}	Te	1.59×10^{-2}
Rb	1.55×10^{-2}	Cs	6.60×10^{-2}
Sr	3.30×10^{-2}	Ba	3.23×10^{-2}
Y	1.78×10^{-2}	La	2.96×10^{-2}
Zr	1.80×10^{-1}	Ce	1.00×10^{-1}
Mo	1.20×10^{-1}	Pr	3.00×10^{-2}
Mn	5.00×10^{-2}	Nd	9.63×10^{-2}
Ru	9.10×10^{-2}	Sm	1.80×10^{-2}
Rh	1.45×10^{-2}	Eu	3.47×10^{-3}
Pd	4.53×10^{-2}	Gd	9.00×10^{-2}

2.2.2. Experimental apparatus and procedure

The schematic diagram of the apparatus for ARF-test was shown in Figure 1. This apparatus consisted of a cylindrical reactor vessel (Pyrex glass), a pair of condensers with a 3-way valve, gas absorbent (2 × 150 ml of 1 mol/L NaOHaq.), and a vacuum pump. The s-HALW (200 ml) was heated up to 400 °C under a constant air ventilation rate of 0.1 NL/min. The rate of temperature increase was derived from the calculation of thermal-hydraulic analysis of boiling accident by using MELCOR code (initial volume of

HALW; 120 m^3, (1.43×10^5 kg), heat generation density of HALW; 5 W/h, partitioning model of facilities were referenced from the application for designation of reprocessing business for the Rokkasho reprocessing plant).[9,10] The vapor from the reactor was collected by using condenser (Liebig condenser, 12 mmI.D × 30 cm) every 5 minutes. The condensers are used alternately and washed with 1 mol/L HNO_3aq. Aerosols and gaseous compounds which passed through the condenser were collected on the air filter (quartz fiber filter, QR-100) and gas absorbent. The filter residues on the air filters were eluted by using elution agent (18 mmol/L potassium persulfate in 0.2 mol/L potassium hydroxide). The amount of each element released from the s-HALW was analyzed by inductively-coupled plasma mass spectrometry (ICP-MS). The concentration of nitric acid was analyzed by an alkalimetric titration method. NO_x gasses were measured by using a portable gas analyzer (SHIMADZU NOA-7000).

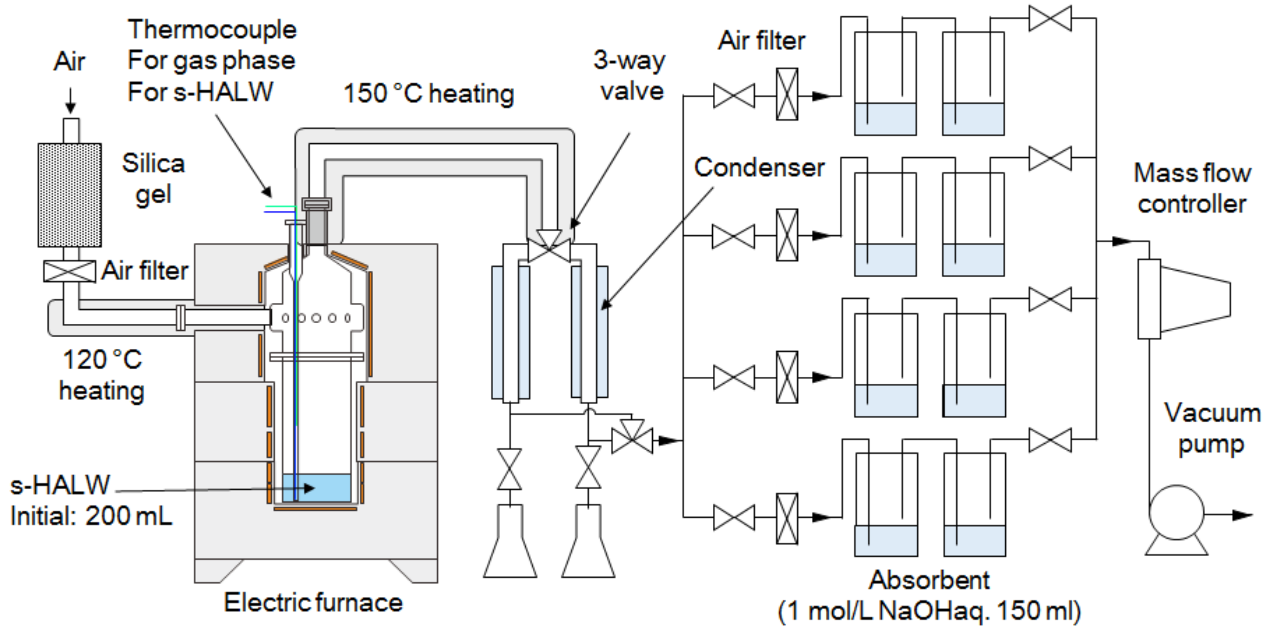


Figure 2 Schematic diagram of experimental apparatus for ARF-test

2.3. Results and discussion of ARF-test

Figure 2 shows that the collected amount of Ru in the condensate. Release amount of Ru increased from approximately 120 °C, its maximum was observed at approximately 140 °C and subsequently it decreased through 170 °C. The secondary increase of release amount of Ru was observed at 210~240 °C. The release of Ru was not observed above 300 °C. The cumulative release ratio of Ru in this experiment was 8.8%.

Figure 3 shows the volume of condensates and their acid concentration as a function of the s-HALW temperature. Most of the condensate was collected below 120 °C (172 ml). The second peak of condensate volume was observed at approximately 120~140 °C (8.5 ml) and the maximum release rate of Ru was detected in this temperature region (Figure 2). The acid concentration of condensate (5.5~8.0 mol/L) in this region was higher than the initial acid concentration of the s-HALW. At the temperature of 170~270 °C, condensate was not obtained. However, small amount of condensates were obtained at 270 °C or above and their acid concentrations were approximately 11 mol/L. The condensates in this temperature region were derived from the decomposition of lanthanide nitrate complexes.

The release rate of NOx was shown in Figure 4. The generation of NOx started at approximately 170 °C and it increased monotonically until 300 °C. This behavior partially corresponded to the second peak of Ru release at 210~240 °C. This release behavior indicated that the main part of NOx was generated from the decomposition of lanthanide nitrate complexes as was the case with condensates at 270 °C or above.[11]

We inferred the generation behavior of Ru(g) in relation to the change of s-HALW temperature, condensate volume, acid concentration, and NOx generation, as below. In the temperature region from 120 to 170 °C, concentration of HNO_3 in the condensate was high (approximately 8~9 mol/L) as shown in Figure 3. The cause for the release of Ru was presumed to be the formation of volatile Ru compounds such as RuO_4 by oxidation with concentrated HNO_3.

The temperature region of the second peak of Ru release at 210~240 °C corresponded to the temperature region of the NOx generation (170~300 °C) (Figure 4). It was reported that nitrates of Ru thermally decomposed at 150~300 °C and Ru(g) was released by thermal decomposition of ruthenium(III) nitrosyl nitrate ($Ru(NO)(NO_3)_3$).[11] Consequently, these results suggested that the nitrates of Ru were decomposed to release volatile Ru compounds. The NOx gasses were also released as byproducts and they acted as carriers gas for Ru(g).

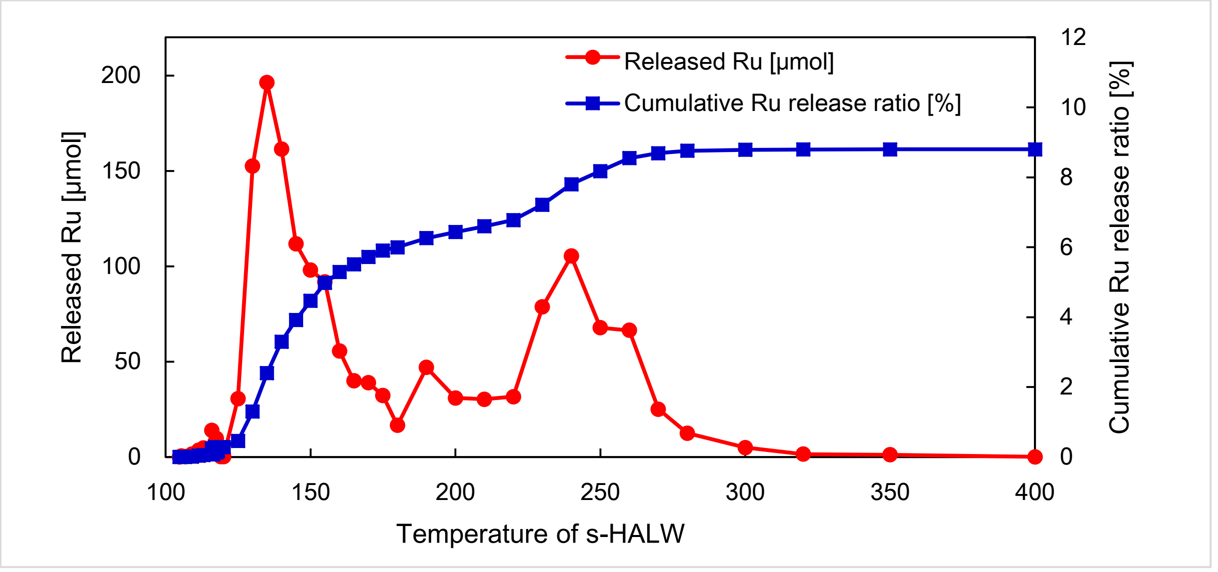


Figure 3 Released amount of Ru and cumulative amount of released Ru

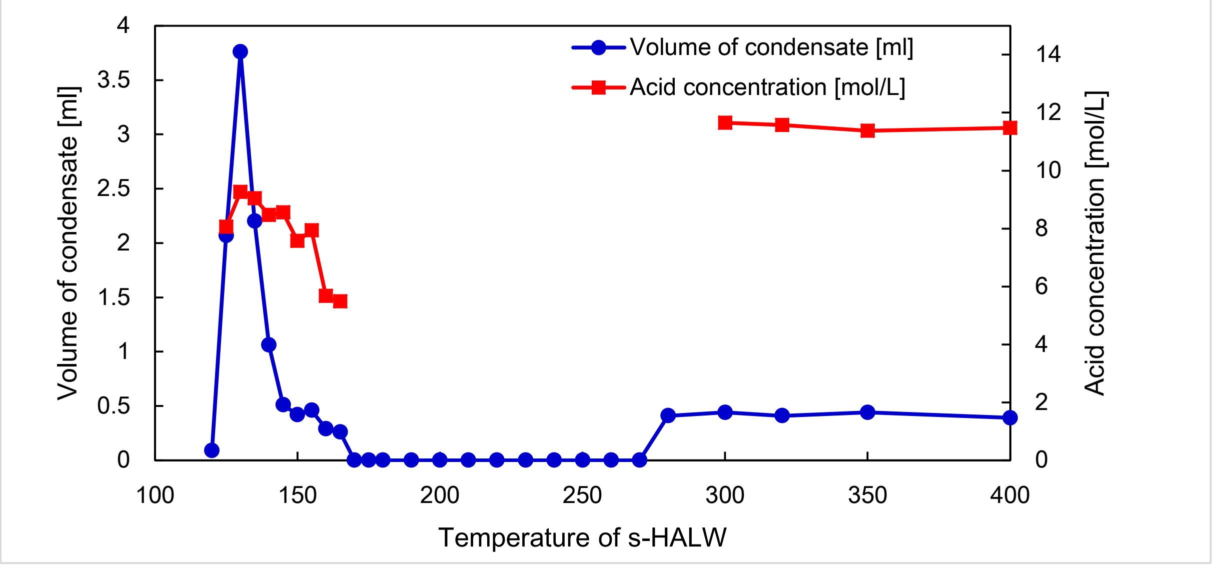


Figure 4 Volume of condensates and their acid concentration vs. temperature of s-HALW

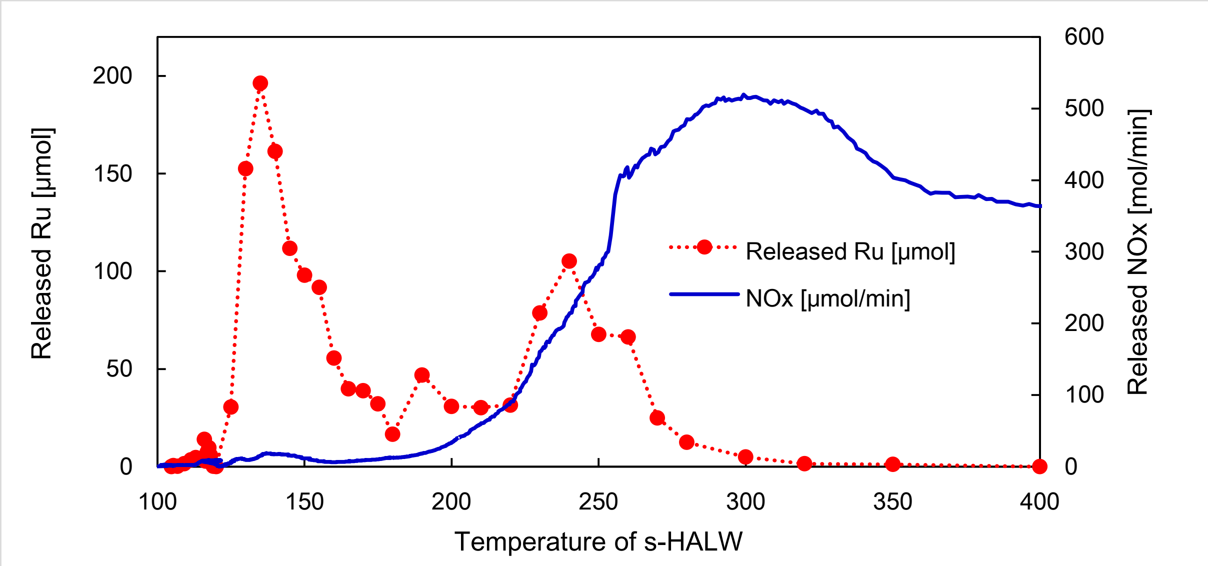


Figure 5 Comparison of released amount of Ru with NO_x release rate

3. LPF-Tests (Reference 1)

In the LPF-tests, RuO_4 was used as typical gaseous ruthenium compound and transport behaviors were evaluated under the various atmospheric conditions by using a flow reactor. The contents about LPF-tests correspond to a part of the contents in the reference 1.

3.1. Experimental of LPF-tests

3.1.1. Experimental Apparatus

A schematic diagram of the experimental apparatus is shown in Figure 5. This is a flow reactor which simulates the generation and transport of the Ru(g) at the EDLCF. The apparatus composed of four major parts: a gaseous RuO_4 generator, an ultrasonic mist generator, reaction pipes, and a ruthenium absorption system. The RuO_4 was employed as a typical volatile ruthenium compound. The gaseous RuO_4 was supplied into the reaction pipes (6 cmϕ × 20 cm × 9 parts with inlet/outlet connectors, Pyrex glass) which simulate the transport pathway of the reprocessing plant. The Ru passing through the reaction pipes was collected by a ruthenium absorption system which consists of three components connected in series: a glass filter (ADVANTEC, 86R), a Liebig condenser (35 mmφ x 300 mm, 5 °C), two gas washing bottles (filled with 300 ml of 0.1 mol/L NaOH). The transport behavior and the LPF of the Ru under the various gas phase conditions were evaluated from the comparison between the collected amounts of the Ru in each apparatus part.

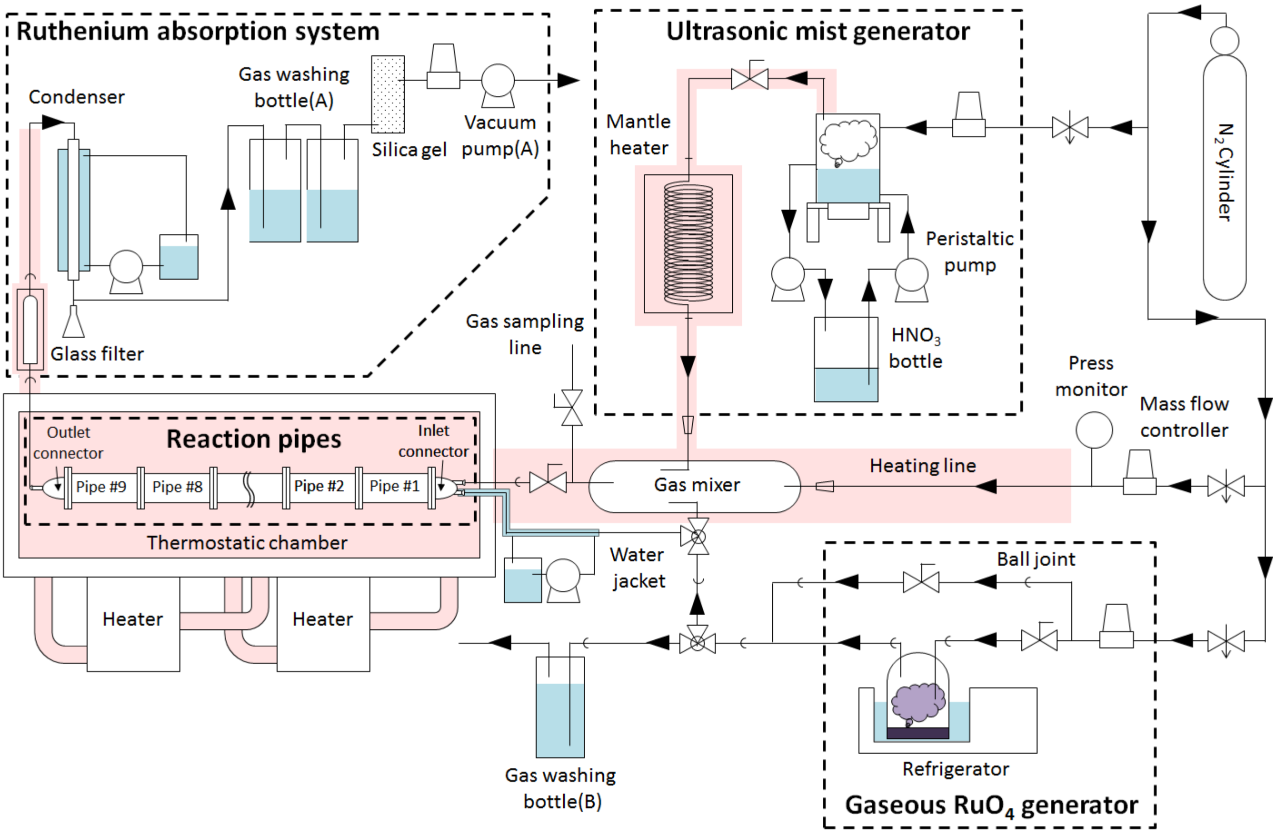


Figure 6 Schematic diagram of experimental apparatus for LPF-test

3.1.2. Definition of LPF

The LPF of Ru in this study was defined as the ratio between the total amount of supplied Ru and amount of Ru which deposited on the reaction pipes (**eq. 1**).

$$\mathrm{LPF} = 1 - \frac{\text{Amount of ruthenium which deposited on reaction pipes}}{\text{Total amount of supplied ruthenium}} \quad \textbf{(eq. 1)}$$

3.1.3. Experimental condition

Experimental conditions were shown in Table 2. The experiments with N_2 and N_2+H_2O vapor atmosphere conditions (Exp. No. 1 and 2) were performed as the control tests for the presence of HNO_3 vapor which was a characteristic of EDLCF accident. The experimental condition of Exp. No. 3 simulated the atmospheric condition of the late stages of HALW boiling in the EDLCF accident. The experimental condition of Exp. No. 4 simulated the atmospheric condition of drying-up phase of HALW in the EDLCF.

Table 2 Experimental conditions of LPF-test

Exp. No.	1	2	3	4
Atmospheric composition	N_2	N_2+H_2O	N_2+H_2O;HNO_3 (3 mol/L HNO_3aq.)	N_2+H_2O;HNO_3 (10 mol/L HNO_3aq.)
Temperature [°C]	150	150	120	150
Ru supply rate [mol/min]	7.2×10^{-7}	1.6×10^{-6}	1.7×10^{-6}	3.2×10^{-6}
HNO_3 supply rate [mol/min]	-	-	4.3×10^{-4}	6.0×10^{-4}
H_2O supply rate [mol/min]	-	1.0×10^{-2}	7.2×10^{-3}	2.2×10^{-3}

3.1.4. Experimental procedure

A typical procedure of the experiments (Exp. No. 3) was as follows: The RuO_4(g) was firstly collected in the gas washing bottle (B) (Figure 5) for evaluating the supply rate of the RuO_4(g) and amount of total supplied Ru (Table 2). The inner atmosphere of reaction pipes was substituted with the sample gas without RuO_4 for 20 minutes beforehand. The temperature of the heated lines and thermostatic chamber were set at a constant value of 120 °C. After that, the RuO_4(g) and HNO_3+H_2O vapor, which were generated by the corresponding generators, were fed into the reaction pipes with each carrier gas (N_2, 0.1 NL/min) for 20 minutes. After the completion of the RuO_4(g) supply, the HNO_3+H_2O vapor was supplied for 20 minutes to purge the remaining Ru(g). Subsequently, the N_2 gas (0.4 NL/min) was supplied for flushing of the reaction pipes. After the flushing, the reaction pipes, the glass filter, and all of other parts were washed with HNO_3aq. (1.0 mol/L HNO_3, eluent for RuO_4) then dipped into the elution agent[12] (a mixture of 18 mmol/L potassium persulfate and 0.2 mol/L sodium hydrate) more than two days to collect the Ru which deposited on the glass surface. The quantitative analysis of Ru was performed by using ICP-MS (Perkin-Elmer, ELAN DRC-e).

3.2. Results and discussion of LPF-tests

The results of the LPF tests were summarized in Table 3. The distributions of deposition amount of Ru in the LPF-tests were shown in Figure 6). The LPF values varied greatly depending on the atmospheric conditions (LPF = 3.0×10^{-3} ~ >0.99).

In the experiment with N_2 atmosphere (Exp. No. 1), the result shows quite low LPF (3.0×10^{-3}) because almost all of Ru were trapped in the reaction pipes. The deposition amount of Ru was decreased along with the transport distance and the main part of Ru was collected in the inlet connector and the pipe#1 (Figure 5, Figure 6). These results indicated that the RuO_4 was rapidly decomposed in the N_2 atmosphere at around 150 °C.

A major part of the supplied Ru was trapped in the reaction pipes (LPF = 0.29) in the experiment with the N_2+H_2O atmosphere (Exp. No. 2). However, rest of the supplied Ru were collected in the glass filter (29% of supplied Ru). The collected Ru in the glass filter cannot be eluted in the HNO_3aq., whereas these Ru were eluted and collected in the elution agent. This result indicated that chemical form of Ru collected in the glass filter was RuO_2. These investigations suggested that a part of supplied RuO_4 changed to RuO_2 particles in the gas phase.

In contrast to above results, Ru deposition was not observed (LPF > 0.99) in the experiment with N_2+H_2O+HNO_3 atmosphere (Exp. No. 3 and 4). Almost all of Ru was detected in the condensate and gas absorbent whereas a negligible amount of Ru was detected in the reaction pipes and glass filter. These results showed that the supplied RuO_4 passed through the reaction pipes as the gaseous form without deposition. These results also suggested that either the nitric acid stabilized gaseous RuO_4, or the RuO_4 reacted with the nitric acid or nitric oxides and took another chemical form such as nitrosylruthenium. However, the chemical structure of Ru(g) in the Air+HNO_3+H_2O vapor has not yet been characterized due to the difficulty in sampling. The chemical speciation of the Ru(g) in the Air+HNO_3+H_2O vapor is an outstanding issue.

Table 3 Results of LPF-test

Exp. No.		1	2	3	4
LPF		3.0×10^{-3}	0.29	>0.99	>0.99
Reaction pipes (%Ru*1)		100	71	1	1
Glass filter (%Ru)	HNO_3aq.	<0.1	N.D.	<0.1	<0.1
	Elution agent	<0.1	29	<0.1	<0.1
Condensate (%Ru)		-	N.D.	24	20
Gas absorbent (%Ru)		N.D.	N.D.	75	79

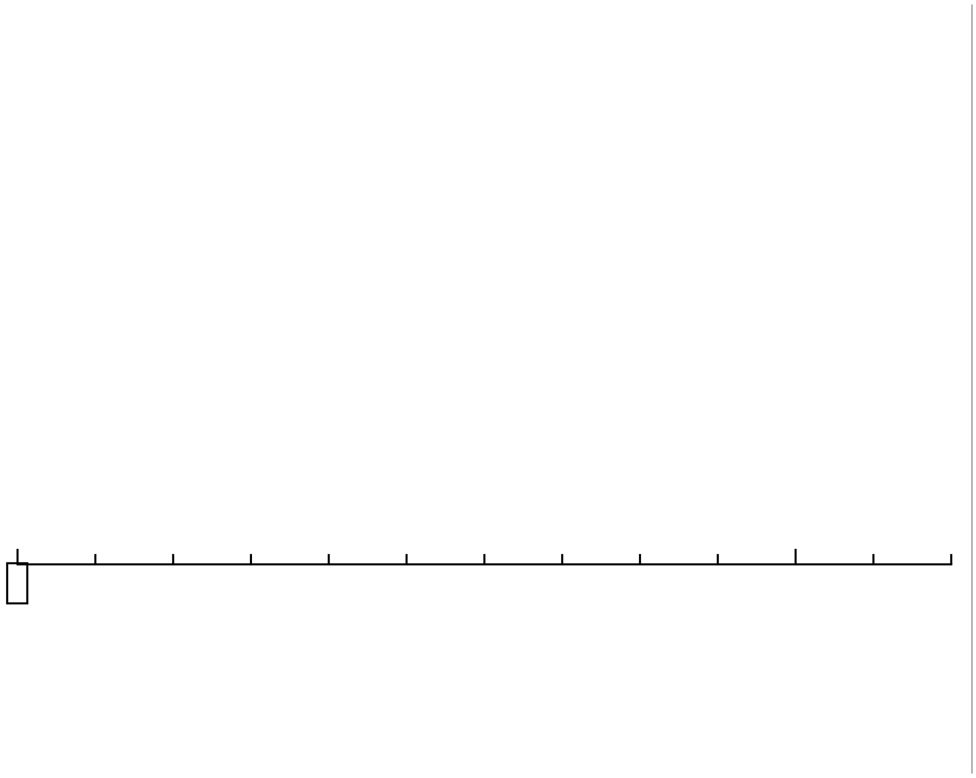


Figure 7 Distribution of deposition amount of Ru

Liquid Waste Tank of a Reprocessing Plant: Hydrogen Consumption Reaction Catalyzed by Pd Ions in the Simulated Solution. *Nucl. Technol.* **172,** 77–87 (2010).

9. Yoshida, K. & Ishikawa, J. Thermal-hydraulic analysis of boiling event of reprocessed liquid wastes with MELCOR code. *JAEA-Research* **26,** (2012).
10. JNFL. Japan Nuclear Fuel Limited, ‘ Application for Designation of Reprocessing Business ’, [JAPANESE] , 1989 (revised in 1996, 2001, and 2004).
11. AMANO, Y. *et al.* Trends of Nitrogen Oxide Release during Thermal Decomposition of Nitrates in Highly Active Liquid Waste [JAPANESE]. *Trans. At. Energy Soc. Japan* **14,** 86–94 (2015).
12. Mun, C., Cantrel, L. & Madic, C. Study of RuO4 decomposition in dry and moist air. *Radiochim. Acta* **95,** 643 (2007).