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Implementation and Effectiveness of Actions Taken at Nuclear Power Plants Following the Fukushima Daiichi Accident

IAEA

International Atomic Energy Agency

IMPLEMENTATION AND EFFECTIVENESS OF ACTIONS TAKEN AT NUCLEAR POWER PLANTS FOLLOWING THE FUKUSHIMA DAIICHI ACCIDENT

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FOREWORD

Since the accident at the Tokyo Electric Power Company (TEPCO) Fukushima Daiichi nuclear power plant in March 2011, rigorous engineering and human performance assessments have been conducted to evaluate the current technical and organizational status of operating nuclear power plants with respect to plant design, configuration and operation. These assessments have aimed at identifying potential vulnerabilities — particularly to external events — that could affect the safety and/or margin of safety, as well as associated improvements in design robustness and programmatic/procedural effectiveness in the light of the lessons from the accident.

Although design and operation philosophies and regulatory/legal structures may differ across Member States, a majority of these assessments recommended similar actions and implementation plans. They also put forward two common objectives: that all Member States implement the corrective, preventive and protective actions adequately and in a timely manner, and that they ensure the effectiveness and sustainability of the actions over the long term. However, between Member States — and even between nuclear power plants within a Member State — there have been differences in the priorities, importance and implementation schedules of the corrective/preventive actions and procedural changes, and the approaches to confirm their effectiveness and durability.

Since the accident, and particularly since the publication of the IAEA report The Fukushima Daiichi Accident, there have been requests from Member States with nuclear power plant operating organizations (utilities) for periodic updates on effective methods and good practices in implementation and ensuring the sustainability of the actions implemented. This publication describes the challenges and needs of Member State nuclear power plant operating organizations/utilities that have been implementing and maintaining post-Fukushima efforts. It discusses the actions taken (or to be taken), good practices and effective solutions to issues relating to implementation, verification, qualification and maintenance, and to measuring and maintaining the effectiveness of those actions. It also presents examples of decision making for implementation and follow-up policies, programmes and procedures to ensure sustainability over the long term.

The objective of this publication is to provide Member States with an understanding of the actions taken to resolve the issues encountered, including technical, operational and economic challenges, as well as methods and concerns relating to future plans and strategies for maintaining and sustaining the actions. This publication is not intended to endorse or to invalidate actions that have been taken or planned, or the reasons for and justification of those actions.

The IAEA is grateful for the generous contributions of many Member States, and to all the contributors, particularly M. Franovich (United States of America), L. Gilbert (Canada), P. Jouy (France), A. Omoto (Japan), M. Powell (United States of America) and J. Taylor (United States of America) for technical discussions and contributions, and N. Barkatullah (Australia), S. Kidd (United Kingdom) and A. McDonald (United States of America) for the consultations and input on economics. The IAEA thanks those countries and organizations that participated in the survey or that shared their experiences in a series of IAEA meetings held since the first actions were implemented at their nuclear power plants.

The IAEA officers responsible for this publication were A.N. Kilic of the Division of Nuclear Power and A. Van Heek of the Division of Nuclear Planning, Information and Knowledge Management.

EDITORIAL NOTE

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1. INTRODUCTION

1.1. BACKGROUND

Following the accident at TEPCO's Fukushima Daiichi Nuclear Power Station in March 2011, rigorous engineering and human performance assessments have been conducted to evaluate the current technical and organizational status of operating nuclear power plants with respect to plant design, configuration and operation. The main objectives of the assessments were to identify potential vulnerabilities, particularly to the extreme events, and associated improvements in design robustness and programmatic/procedural effectiveness in the light of the available lessons learned from the accident.

The results of assessments conducted by regulatory bodies, nuclear power plant operating organizations and international, regional and industry collaboration entities in the IAEA Member States have led to a set of short and long term corrective and preventive actions to be taken at the nuclear power plants. These actions have ranged from purchasing/installing additional (mobile or permanent) equipment to permanent design and programmatic/procedural changes, in order to cope with events that are beyond the plants' design bases. They also have led to changes in regulatory requirements or guidance to ensure and improve safety and revised programmatic and procedural guidance for safe, reliable, efficient and long term operation of nuclear power plants with robustness against what had been learned from the accident.

Three years after the accident, the Contracting Parties to the Convention on Nuclear Safety, at the 6th Review Meeting in March–April 2014, reported an update on the implementation of safety improvements and plant upgrades based on the lessons learned from the accident. These physical and programmatic improvements and upgrades mainly consisted of [1]:

— "The introduction of additional means to withstand prolonged loss of power and cooling;
— The enhancement of power systems to improve reliability;
— The re-evaluation of site specific external natural hazards and multi-unit events;
— The improvements of on-site and off-site emergency control centres to ensure protection from extreme external events and radiation hazards;
— The strengthening of measures to preserve containment integrity;
— The improvement of severe accident management provisions and guidelines" [1].

Although the design and operation philosophies and regulatory/legal structures and requirements may differ across Member States, most of the recommendations from these assessments comprised similar actions and implementation plans. The decision making on the implementation of these actions aimed at completion of improvements based on the priority and the resource capabilities of the operating organizations which considered immediate, medium term and long term implementation. While the consensus has been established on the essentiality of timely and correct implementation of corrective and protective actions and on the assurance of their long lasting effectiveness and sustainability, there have been differences in the priorities, importance and implementation scope and methods of these corrective/preventive actions and procedural changes. Furthermore, there are large variations in the costs and schedules of changes made (or planned), as well as sustaining the level of assets and margins gained by these implementations from one country to another, or even from one unit to another in the same utility or site.

There have been requests to the IAEA from the Member States, especially from the nuclear power plant operating organizations (utilities), for reviewing and sharing effective methods and good practices in the implementation, as well as the sustainability, of those actions. These requests particularly noted the dissemination of learnings regarding the lessons learned from implementation of actions as to the decision making, establishment basis for prioritization and challenges faced and solutions implied to handle those challenges — i.e. the sharing of 'lessons learned from the lessons learned' —. Particularly, the operating organizations have been interested in:

- What did the other owners/operators need/have to do?
- What are the other owners/operators doing?
- What are the issues for the owners/operators in prioritization and implementation of the actions?
- What are the challenges and how to overcome them?
- What have the observed results been (good, not so good, to be determined later with more operating experience, etc.), especially on their impact versus value (e.g. costs versus benefits) of actions?
- What are the future needs and challenges and how they are being considered and addressed?

The operating organizations in the Member States also requested assistance in dissemination of measures and strategies for sustaining those actions for the rest of the remaining or planned nuclear power plant lifetime reliably and economically.

The IAEA's Department of Nuclear Energy followed up on these requests to provide a platform for an information and experience exchange among the nuclear power plant operating organizations, specifically to:

- Collect and share international experiences and lessons learned, as well as to exchange views on methods and strategies for implementation of post-Fukushima actions;
- Exchange reasoning and methods in establishing strategies and guidance for verification of the effectiveness of Member States and nuclear power plants' assessments and actions;
- Identify and address continuing challenges of the nuclear power plant operating organizations' in the Member States in implementing and maintaining post-Fukushima efforts;
- Provide assistance in the form of periodic and topical publications to share updated information in order to communicate most recent developments;
- Strengthen the international networking of specialists involved in the past, current and future implementation of corrective/preventive actions in response to the Fukushima Daiichi accident.

Moreover, IAEA's Department of Nuclear Safety and Security has collected and disseminated a description of safety improvements at existing nuclear power plants as identified and implemented by regulatory bodies, technical support organizations, licensees and designers [2].

Starting with the initial fact-finding mission and the report of it [3], IAEA has been holding meetings to learn and disseminate the causes and consequences of the accident that include periodic technical meetings with the experts from the Member States' organizations to follow the actions being taken in response to the lessons learned (Fig. 1).

FIG. 1. Initial IAEA evaluations, investigations and the areas of interest (circa 2011) SFP — Spent fuel pool, DCS — Dry cask storage, BDBE — Beyond design basis event, DiD — Defence in depth, SAM — Severe accident management, PAM — Post accident monitoring.

Based on the information collected from the Member States, the implementation schedules for the completion of all (or vast majority of) actions showed that 2018–2019 is a common milestone; although some long term actions need further analyses, research/confirmation test, development and validation of analytical models, etc., and will be implemented after this milestone, as long term as 2030. For example, in France, some actions will be completed by 2030 which is the time for analysis and outage for fifty nuclear power plants in periodic safety reviews (PSRs), i.e. the 'Phase 3' implementation of Fukushima actions on the third and the fourth PSRs of 1450 MW(e) and 1300 MW(e) designs, respectively.

Accordingly, the IAEA considered publishing an overall examination report of the actions and their effectiveness during this period. Therefore, it seemed necessary and appropriate to publish a report at or around the common milestone, in the form of a TECDOC, with an objective to explore, collect, discuss and address challenges and needs of the Member State nuclear power plant operating organizations which are implementing and maintaining post-Fukushima efforts. This report that is a collection and dissemination of challenges and needs of the Member State organizations which are implementing and maintaining post-Fukushima efforts, is considered to be an initial list of, and experience with, the actions taken (or are to be taken), good practices and effective solution to the issues in implementation, verification, qualification and maintenance, as well as measuring and maintaining effectiveness of those actions for ensuring sustainability in the long term.

1.2. OBJECTIVE

This publication aims to disseminate practices, as well as challenges and their effective solutions that have been experienced by the operating organizations, in the implementation, verification, qualification and maintenance and measuring effectiveness of those actions for ensuring sustainability in the long term. It aims to provide collective guidance, based on current knowledge and experience, for the decision making and implementation of post-Fukushima actions in the operating nuclear power plants in Member States in order to strengthen the international networking of specialists involved in analysis and in the implementation. It particularly intends to present experiences and lessons learned in implementation of action by:

— Summarizing views on methods and strategies for implementation;
— Providing ideas for developing better strategies and guidance for effective implementation from the experience of others;
— Sharing ideas and methods for the verification of effectiveness and maintaining/improving value;
— Identifying the challenges and needs of operating organizations which are implementing and maintaining post-Fukushima efforts going forward.

The intended users of this publication are the owner/operating organizations (utilities) involved in the decision making regarding post-Fukushima actions or in their implementation and sustainability in the operating nuclear power plants.

Although the utilities with operating nuclear power plants are the main audience, the content of this publication may be an interest to the entities in Member States building (or are planning to build) nuclear power plants. The users of this publication may also include the organizations that are designing, providing, constructing, installing, maintaining, modifying, and regulating the nuclear generating units.

1.3. SCOPE

This publication describes actions that have been taken (or are to be taken) at the nuclear power plant and sites by the operating organizations in response to the results and finding of evaluations, conducted by the nuclear industry and the governments, of the lessons learned from Fukushima Daiichi accident.

As such, all the actions that have been taken off-site, such as actions to increase regulatory effectiveness, to improve national emergency response organization, governmental structure etc. are not included in this publication.

In other words, the scope of this report only focuses on the actions of, for and by the nuclear power plants and sites that are owned or controlled by the owner/operating organizations of the plant(s).

The publication discusses reasons that actions were deemed needed or required, decision making for their implementation including the scheduling and planning. It further provides a list of issues that have arisen during the implementation and operating organizations' solutions for them, including the consideration of effectiveness and value/impact of associated actions including the procurement and maintenance of equipment and services and the programmes and procedures. Particularly, the publication covers the following for action taken (or are to be taken):

— Reason, i.e. drivers, bases, justification for implementing (or not implementing) specific actions;
— Schedule and underlying reasons/justifications for prioritizing (or deferring) specific actions in time, i.e. immediate, medium or long term implementation;
— Methods and concepts, i.e. how to more quickly and more effectively implement actions;
— Challenges encountered and resolution of the issues;
— Costs of, and other impacts from, the actions, including the comparison of differences in cost and reasons for these differences;
— Assessments of benefits and other value, including trade-off, multi-purpose utilization and credit of particular actions;
— Verification and validation of effectiveness, functional sustainability of physical changes, methods of preserving assets and anchoring of programmatic and procedural changes.

The cost of the socioeconomic impact of the consequences of an accident is not within the scope of this publication. It is recognized that such post-accident costs can be significant. However, they are not herein discussed, rather the actual accrued costs of improvements by/for the nuclear power plant operating organizations to increase robustness are presented.

This publication is not intended to judge, to endorse or to invalidate the actions that have been taken (or planned) or reasons for and justifications of those. It rather is to provide Member States with an understanding of the actions and practiced resolutions of encountered issues, including technical, operational and economic challenges, as well as methods and concerns with the future plans and strategies for maintaining and sustaining the actions.

The considerations and aspects provided in this publication are not comprehensive lists of all needs, challenges and solutions but rather provide key concepts that could be taken into account in the process, as a minimum, based on the current operating experience and technical and administrative fundamentals.

Further, this publication is not detailed and prescriptive guidance to implement and sustain post-Fukushima Daiichi accident actions in nuclear power plants. It is rather a collective list providing major modifications to plant and human factors. The report provided is supplemented by specific examples of nuclear power plant actions from operational experience, as well as good practices and lessons learned. The IAEA, however, does not take responsibility for the completeness, correctness and applicability of those examples for specific cases which require users' efforts to validate and verify.

1.4. STRUCTURE OF THIS PUBLICATION

The main body of this publication is divided into 11 Sections including the introduction in Section 1 and the observations and conclusions in Section 11. Three Appendices provide the survey questionnaire that were sent to, and collected from, operating organizations and the breakdown of surveyed plants. One Annex of this publication provides the reader with the sequence of events during the Fukushima Daiichi accident. A glossary of specific terms and a list of abbreviations are also provided for the reader's reference at the end of the publication.

Section 2 provides an 'executive summary' of this publication regarding the actions that have been taken (or are to be taken) at the nuclear power plant and sites by the operating organizations in response to the results and finding of evaluations conducted on the lessons learned from Fukushima Daiichi accident.

Section 3 provides an overview of some assessments and the conclusions of those that eventually reflected on the actions that have been implemented (or are to be implemented) at the operating nuclear power plants. It further discusses the reasons, necessities and decisions for operating organization commitment to implement such actions in their plants.

Sections 4 to Section 6 present the detailed actions in each category, namely physical modifications, analytical changes and human and organizational enhancements, respectively, as reported by the nuclear power plant operating organizations. The discussions in these sections include: the reasons, i.e. drivers, bases, justification for implementing (or not implementing) for specific actions; schedules and underlying reasons/justifications for prioritizing (or deferring) specific actions in time, i.e. immediate, medium or long term implementation; challenges encountered and resolution of the issues; costs/impact of actions. The topics discussed in these sections are not a complete list of all actions, as they will differ depending on the nuclear power plant location, design, configuration, size, age (including the vintage of technology), operation and maintenance practices, effectiveness and extent of existing programmes, etc. Rather, they are the common impacts/issues/solutions that have been observed and collected from the experience in decision making and implementations, as well as those that can be anticipated based on the latest knowledge and technical fundamentals. Thus, if an action is not mentioned in this publication, it does not mean that that specific action was not taken.

Section 7 and Section 8 discuss, respectively, reported responses in the survey on:

— Verification of effectiveness, functional sustenance of physical changes, methods of preserving assets and anchoring of programmatic and procedural changes;
— Merit of actions as to their utilization not only for beyond design basis event (BDBE)/beyond design basis accident (BDBA)[1] response and severe accident management (SAM), but also support normal plant operation and maintenance.

Section 9 presents a review and study of costs/impact of actions including the comparison of differences in cost and reasons for the differences, as well as the benefit/value.

Section 10 provides a set of open actions that need further work, such as scientific and technical research or development, and lastly all is concluded in Section 11 by providing some observations from the implementation of post-Fukushima actions at the nuclear power plants, i.e. 'lessons learned from lessons learned' from Fukushima Daiichi accident aftermath.

This publication can be used as a general database on understanding the post-Fukushima actions taken (or that are to be taken) by the operating nuclear power plants. Users who would like to compare their actions with the actions taken by other utilities worldwide,

[1] The IAEA terminology BDBA is used interchangeably with BDBE, as defined in the Glossary.

particularly regarding the solutions of challenges during the implementation of post-Fukushima modifications and the associated costs, could refer to Section 3 through Section 6. Section 7 provides an overview of measures for sustaining the effectiveness of action to establish extent and effectiveness of the actions. In combination with previous sections, of which the main audience is technical and operational staff who involved in deciding or implementing, the user may then follow the aspects to maintain actions.

As the decision for — and the implementation of — an action involves the considerations of its costs and benefits and its value/impact, discussions in Section 8 and Section 9 may provide approaches for the conduct of such analysis and decision making.

It is recommended that users refer to sources for the information provided in this publication for expansion and utilization of the topics discussed herein.

2. EXECUTIVE SUMMARY OF ACTIONS AND THEIR EVALUATION

Nearly a decade has passed since the events unfolded at the TEPCO's Fukushima Daiichi nuclear power station in Japan, during which the institutions in Member States have evaluated the lessons learned from the accident. From the causes and sequences of the accident and their explorations and derivations, nuclear power plant operating organizations (utilities) searched for and identified vulnerabilities, if any, and implemented actions for prevention of core damage, large radioactive release and mitigation of their consequences with the goal of improving safety and ensuring the protection of the people and the environment.

The implementation of resulting actions had been scheduled in short, medium and long terms typically set as less than five years, 5–8 years and more than eight years, respectively. Therefore, at nearly all plants, the completion of all (or a vast majority of) actions marked the end of 2019 as a common milestone. Accordingly, IAEA considered publishing an overall status report on the actions that have been taken by operating nuclear power plants and on the validation of their effectiveness at this milestone. This report also intends to collect, explore and address challenges and needs of nuclear power plant operating organizations which are implementing and maintaining post-Fukushima efforts in the Member States. In other words, this publication reports on the 'lessons learned from the lessons learned', i.e. have been learned by the industry from the response to lessons learned from the Fukushima Daiichi accident.

In preparation of this report, a survey was conducted by the IAEA requesting first-hand information and knowledge from the nuclear power plant operating organizations regarding their actions taken (or are to be taken) as to their:

- Drivers of actions;
- Basis/justification for implementing (or not implementing), as well as prioritization (or deferral) of specific actions;
- Challenges encountered and resolution of the issues;
- Costs/benefit (value/impact) of actions including the comparison of differences in cost and reasons for the differences, including trade-off, multipurpose utilization and crediting some actions, etc.

Furthermore, the IAEA has periodically collected experiences and challenges of operating organizations in the implementation of actions, in a series of meetings and by continuing literature search.

Accordingly, this publication was particularly prepared to disseminate information on:

- Good practices and lessons learned from the implementation of these actions;
- Encountered challenges and their resolutions during the implementation of these actions;
- Effectiveness and advancement of these actions in terms of value/impact (cost/benefit) on a long term horizon.

2.1. DRIVERS AND ACTIONS OBSERVED

The results of assessments conducted by the regulatory bodies, nuclear power plant operating organizations and international, regional and industry collaboration entities in the IAEA Member States have led to a set of short, medium and long term corrective and preventive actions to be taken at the nuclear power plants to ensure and improve safety, reliability and performance.

The primary drivers of actions were new or revised regulatory requirements (some of which also included public involvement and opinion); however, utilities' own initiatives — or joint initiatives with other utilities, industry groups and/or vendors — played a significant part in the decision on actions. The latter drivers assessed particularly those actions that are taken (or were justifiably not taken) based on technical analysis benefit/detriment or for value/impact. In a few cases, the actions were based on prescriptive requirements from the regulatory body, for which requirements are set in a deterministic approach and without value/impact analysis by the utilities. However, several operating organizations and regulatory bodies followed probabilistic approaches supplementing the deterministic approach and analysis, where applicable and adequate.

The actions taken by the operating organizations at and for their nuclear power plants in all Member States occurred in three areas: physical changes to plants; method and modelling changes to analyses; and, human, organization and programme changes, particularly in coping with events that are beyond the plants' design bases.

2.1.1. Common actions taken by the operating organizations

Although the design and operation philosophies and regulatory/legal structures and requirements may differ across Member States, a majority of the recommendations from these assessments comprised similar or identical actions and implementation plans. The studies and improvement areas that were common across operating organization actions, included:

— Estimation of external hazards and plant response (e.g. evaluation of external events, particularly earthquake and flooding, confirmation of plant capabilities and identification of gaps in protection against these events, particularly prolonged loss of power and loss of cooling events induced by natural hazards).

— Procurement of mobile or permanent equipment for power and water provision, the use of commercial grade provisions to cope with BDBAs/BDBEs, the use of a phased approach to store and deploy those (permanently placed, on-site and/or off-site mobile).

— Installation and modification of permanent designed systems, structures and components (SSCs), such as the installation of spent fuel pool cooling and monitoring systems, flood proofing structures, adding permanent power and water sources for alternative supply, backup ultimate heat sink, hydrogen monitoring and treatment system, etc.

— Revision of severe accident management concept and its guidelines including those for the low power and spent fuel pool event mitigation.

— Improvement of emergency response regarding organization, human and equipment capabilities. Installation/modification of protected and habitable emergency response centres, etc.

2.1.2. Differences in actions taken by the operating organizations

Conversely, different design and operation philosophies and regulatory/legal structures and requirements across Member States resulted in different decisions to identify, assess and implement (or not to implement) actions. It was particularly observed that main differences arose from:

- Regulatory and industry framework driving the actions, for example:
  - Natural hazard frequencies to consider in the evaluations;
  - Use of probabilistic and deterministic assessment;
  - Ultimate goal (i.e. achievement goal) of action, e.g. prevention of core damage, prevention of confinement function, prevention of large release, amount of time without outside support, etc.;
  - Requirement for redundancy, independence, classification and qualification of post-Fukushima SSCs, such as requirement for independent emergency power and core cooling in hazard protected structures with dedicated power supply and instrumentation.
- Scientific assessment on the benefits/detriments of a particular action in value/impact towards the achievement goal (e.g. adoption of filtered venting, hydrogen control, ex-vessel retention, multi-unit consideration).
- Public opinion of the accident (and/or, at large, of nuclear power) which influenced regulatory and industry actions in response, particularly imposition/demonstration of extensive safety margins and analytical conservatisms for gaining public confidence and acceptance.
- Utilities' ownership of assets, i.e. single or multi-unit site, fleet or different technologies, and combined assets as industry group, association, cooperation or joint ownership.
- Utilities' long term strategies and goals and multi-dimension decision approach, particularly:
  - Strategic utilization of the actions based on the value/impact analyses in terms of operational and economical added value and advantage to the activities in the remaining operating lifetime.
  - Meeting the minimum requirements versus gaining safety, design and operational margin for performance and production.

2.2. SCHEDULES FOR ACTIONS

Since in the immediate aftermath of the accident, the participants in IAEA's international expert meetings and technical meetings have repeatedly pointed out and a general agreement has been established on that it is essential to have the Member States' actions are correctly and timely implemented with their effectiveness and sustainability is preserved in the future. However, there have been differences in the priorities, significance and implementation scope and methods of these corrective/preventive actions and procedural changes.

The schedules for the implementation of these actions aimed at completion of improvements based on the priority and the availability of resources as well as the special plant status (i.e.

refuelling/maintenance/major outages, no construction near a unit while it is operating, etc.) to implement actions.

As mentioned above, the drivers of actions and the capabilities of the operating organizations which considered immediate, medium term and long term implementation were:

- Immediate actions were taken within a year of the Fukushima Daiichi accident and mainly included the compensatory (mostly physical) actions in place in response to early report of the accident's preliminary consequences prior to thorough evaluations of causes of accident and assessments of plant specific external hazards and vulnerabilities, particularly for earthquake, tsunami, flooding. Therefore, selection of actions was very specific to the early findings of accident causes and consequences rather than the optimal means for plant specific hazard prevention, protection and mitigation based on the comprehensively established strategies. For example, these actions included the procurement and on-site storage of mobile power and water supplies that would be used to power equipment and provide means of core heat removal in a Fukushima-like event, i.e. when a prolonged loss of power and heat sink occur.

- Medium and long term actions were decided to be taken based on further analysis of the accident causes and consequences as reflected on the specific plant site, design and configuration, such as the 'stress test' in Europe, plant hazard assessments through walkdowns and review of design bases in the USA, safety assessments in Japan, etc. They were also supported by applicable and reasonable methodologies to be used in assessments, e.g. deterministic, probabilistic, value/impact and benefit/detriment based, and determination of achievement goal and associated coping strategies, e.g. prevention of core damage, protection against the failure of confinement function, prevention of large release, amount of time without outside support, provision of margin, minimizing or eliminating cliff edge effects and potential vulnerabilities.

Main reasons to differ between the medium and the long term actions taken (or are to be taken) by the operating organizations were based on various factors, such as: the first available opportunity to perform physical plant assessments, particularly in multi-unit sites; the scarcity and availability of expertise; differing scientific and professional opinion on and further need for research and exploration to reach consensus; additional facts and findings from Fukushima Daiichi accident; findings and learnings from in the plant specific assessments, or new research and development on the accident consequences and proposed mitigation. Also noted by some utilities, the limitation of financial and human resources deferred some of the action to the long term implementation schedules.

It should also be noted that several operating organizations stated that, now, all stakeholders have a more composed (as some of them referred, 'calmer'), more integrated perspective and a better informed view and more knowledge and experience. This state of knowledge and experience that have been gained on the accident cause/consequences, as well as the effectiveness of immediate (i.e. in short term, within months of the accident) actions, are making the operating organizations reconsider and re-evaluate their immediate actions as to their value/impact, cost/benefit and effectiveness in the longer term.

As such, some long term actions are still in progress, some of which will extend into the next decade and some have already been integrated into other programmes and processes such as long term operation (LTO), PSR or major modification projects.

2.3. COST OF ACTIONS

The reported costs for post-Fukushima action by the operating organizations varied considerably based on the responses to the IAEA survey and on the research of the reported values in literature from as low as US $20 million to as high as US $1000 million (US $1 billion) per unit. It should be noted that this range depends on how each owner accounted for a particular action in corporate budget and programmes. For example, in some plants several actions were included with LTO/plant life management actions. Conversely, in some plants, actions from other programmes were bucketed together with post-Fukushima programme actions. Furthermore, some plant had already implemented modification prior to the accident (as early as a decade before the accident) that were determined to implement by other plants following the accident.

These substantial differences mainly depended on country specific conditions, e.g. regulatory framework, public opinion, etc. Although these conditions of the particular countries made a significant difference, it should also be noted that within a country where the regulatory and industry framework apply equally to all plants there, the cost differed from one plant to another based on the individual plant or the utility specific conditions, such as the nature of the reactor site, the type and age of reactor and the corporate strategies of operating organizations in decision making for actions, such as:

- Assessment and justification implementing (or not implementing) a particular action or using different options in response to a particular lesson learned (e.g. solving the redundancy and independence of provisions for BDBAs/BDBEs buy either procuring and storing commercial grade equipment in large quantities and diversity or purchasing higher grade and ensure operability and functionality by protected storage, programmatic maintenance and testing, etc.);
- Utilities' ownership of assets, long term strategies and goals, added value and advantage of utilization of the improvements on operational activities in the operating lifetime and choice to meet the minimum requirements versus gaining safety, design and operational margin for performance and production, which included, for example[2]:
  - Use of mobile equipment when installed equipment such as emergency diesel generator (EDG) is not available (performance based cost saving action);
  - Placing mobile BDBA/BDBE mitigation equipment in the plants' operating limits and conditions (OLCs).

Overall, excepting Japan where expenses are far higher (around US $1 billion per unit), the average costs of completed post-Fukushima actions in the nuclear power plants in the Member States lie in a similar range. In the background of outlying high cost in Japan, which should be noted, there have been other factors for the large deviation from other countries, such as specific regulatory standards on natural hazards, e.g. earthquake from unidentified sources, as a country which is prone to those events. Also, impacting the cost in Japan was the high demand for contractors for modifications, i.e. when every utility is seeking for restart their own units at the same time as a priority.

By comparison with other capital expenditure, such as actions implemented to allow reactors to extend license term, major plant modifications, actions from periodic safety reviews, major equipment replacements, refurbishment, etc., the post Fukushima costs were noted as not

[2] Noting that these examples required a review and approval by the regulatory body.

particularly high (in comparison, replacement of major components, such as steam generators (SGs) typically cost US $500 million or higher per unit). As such, a typical operating organization is expecting to pay three times as much on non-Fukushima related expenditure for LTO. As aforementioned, it is quite difficult to distinguish the costs of LTO and Fukushima related actions because most of them are linked in budgeting and accounting of other plant improvements. For example, in France, within ten years of the Fukushima accident, the measures mentioned in the stress test reports, will be deployed with a cost of 70 million euro per unit. At a later stage, in response to the requirements issued by the regulatory body in January 2014, some improvements have to be continued with the fleetwide life extension programme with a substantial financial investment (about 50 billion euro).

2.4. MAIN OBSERVATIONS, CHALLENGES AND FURTHER POTENTIAL ACTIONS

Post-Fukushima actions by (and for and of) the nuclear power plants demonstrated that the implementation of actions in order to respond to the lesson learned from the accident resulted in increased innovation (concepts, designs, ideas) and progression (thinking and strategizing long term). It also took nearly all the operating organizations out of complacency that was based on a belief in superior technology and on a very high comfort in human and organizational capabilities, which was pointed out in the conclusions drawn by the IAEA (as noted by, then, IAEA Director General Y. Amano in the foreword of his report [1]):

> *"A major factor that contributed to the accident was the widespread assumption in Japan that its nuclear power plants were so safe that an accident of this magnitude was simply unthinkable…"*

and:

> *"There can be no grounds for complacency about nuclear safety in any country*," [1].

This conclusion was also reached by other inquiries by other organisations, such as the investigation by the National Diet of Japan which emphasized that:

> *"The accident was as much a consequence of the complacency and hubris"* [4].

The response to the IAEA survey and the presentations and discussions among the experts from operating organizations in the IAEA Member States during periodic IAEA technical meetings (in 2012, 2014 and 2017), conferences and expert meetings showed a continued commitment from the experts to prevent and mitigate severe accident and willingness to learn from global practices with humility. They also showed that in order to increase public trust and confidence in operating organizations' commitment to prevent and be ready for events similar to the Fukushima Daiichi accident.

However, the operating organizations have recognized several challenges both in the implementation of actions in response to Fukushima Daiichi accident and in the future progress and sustainability. The challenges in post-Fukushima actions were brought up by the operating organizations included:

- Preservation of information, knowledge, experience and competence gained by the implementation of actions, particularly those that are not subject to the existing quality assurance programme requirements, such as design and documentation control;
- Development of analytical methodologies such as realistic damage descriptor for fragility assessment of SSCs, reliable and comprehensive single and multi-unit risk assessment, evaluation of human reliability in harsh environments including multi-hazard, multi-unit accident environment;
- Development of methodology for complementary use of probabilistic and deterministic approach and advanced plant analysis tools, particularly for severe accidents that is necessary to better predict accident progression as well as model the uncertainties and assumptions, in order to prepare guidance, training and simulators;
- Method and programme development for qualification, testing, maintenance and protection of post-Fukushima SSCs from credible hazards, particularly for seismic considerations and harmonization of regulations in terms of qualification standards and on redundancy requirement;
- Need for clear definitions and conditions of application, e.g. for BDBE/BDBA, DEC, DBA, severe accident, etc., and associated level of defence in depth (DiD) (as to transition from Level 3 to Level 4 to Level 5), particularly involving and provoking requirements of redundancy, independency, physical separation, etc.;
- Diversity in the hazard frequency for different hazard to consider concurrently or consequentially for determining actions, for example ranging from 10^{-4} per year for flooding to 10^{-7} per year for hurricane within the same country.
- Delays in technology developments and the long duration that it takes to deploy them, particularly owing to process for qualification and regulatory acceptance, review and approval;
- Acceptable and adequate methods and programmes to ensure availability, operability, maintenance and sustainability of severe accident equipment.

Overall, the operating organizations have implemented (or identified and scheduled the implementation of) numerous actions in response to lessons learned from the Fukushima Daiichi accident and from the derivations and exploration of those towards improving the plant robustness against extreme events and create/increase safety margins.

3. NECESSITY FOR POST-FUKUSHIMA ACTIONS BY NUCLEAR POWER PLANT OWNER/OPERATING ORGANIZATIONS

Since the accident at the Fukushima Daiichi on 11 March 2011, there have been many analyses of its causes and consequences by the organizations in the IAEA Member States, as well as detailed considerations of its implications for nuclear safety. These analyses were conducted by the regulatory bodies, nuclear power plant operating organizations and international, regional and national entities as well as industry groups, such as nuclear operator, regulator and research associations, technology owner groups, academia and other authoritative or advisory entities. As a result of these analyses of the accident and lessons learned — and further derivation and reflection of those — needs for identification of corrective and preventive actions to be taken at the nuclear power plants were determined.

Some of those actions have been incorporated in governments' regulatory requirements, rules and/or guidance, while some have become the expectations, directions and instructions of nuclear power plant operating organizations (utilities), their industry associations, technology owner groups and reactor designers. New or expanded regulations and operating organization's directions to prevent same or similar accidents, and to mitigate consequences should they occur, necessitated structured and integrated implementation of those actions at, by and for the nuclear power plants. This chapter provides an overview of some assessments and the conclusions of those that eventually reflected on the actions that have been implemented (or are to be implemented) at the operating nuclear power plants. It further discusses the reasons, necessities and decisions for operating organization commitment to implement such actions in their plants.

3.1. THE FUKUSHIMA DAIICHI ACCIDENT SYNOPSIS

Although it is not in the scope of this publication, it is prudent to provide a brief recall and synopsis of the accident that occurred at the Fukushima Daiichi nuclear power plant, which initiated all the subsequent investigations, analyses, evaluations and the identification and implementation of actions based on the lessons learned from it as discussed in Ref. [1] (minimally paraphrased):

> "The Great East Japan Earthquake on 11 March 2011 was caused by a multi-segment fracture of the plate under the North American tectonic plate by about 500 km in length and 200 km wide, which caused a Magnitude 9.0 earthquake and a tsunami which struck a wide area of coastal Japan."
>
> "The earthquake and tsunami caused great loss of life and widespread devastation in Japan. More than 15 000 people were killed. Roughly 2500 people are still missing and more than 6000 people were injured. Considerable damage was caused to buildings and infrastructure, particularly along Japan's north-eastern coast."
>
> "At the Fukushima Daiichi nuclear power plant with six units, operated by TEPCO, the earthquake caused damage to the electric power supply lines to the site, and the tsunami with the estimated height of 15m caused substantial destruction of the operational and safety infrastructure on the site. The combined effect led to the loss of off-site and on-site electrical power and loss of heat sink (ocean). This resulted in the loss of the cooling function at the three operating reactor units as well as at the spent fuel pools. The other

four nuclear power plant sites along the coast were also affected to different degrees by the earthquake and tsunami. However, all operating reactor units at these plants were safely shut down."

"Despite the efforts of the operators at the Fukushima Daiichi nuclear power plant to maintain control, the reactor cores in Units 1–3 overheated, the nuclear fuel melted. Hydrogen was released from the reactor pressure vessels, leading to explosions inside the reactor buildings in Units 1, 3 and 4 that damaged structures and equipment and injured personnel. Radionuclides were released from the plant to the atmosphere and were deposited on land and on the ocean. There were also direct releases into the sea." [1].

Figure 2 illustrates the overall summary of accident sequence. The detailed sequences of the events concerning the fundamental safety functions are also provided in Annex I, for the readers' reference.

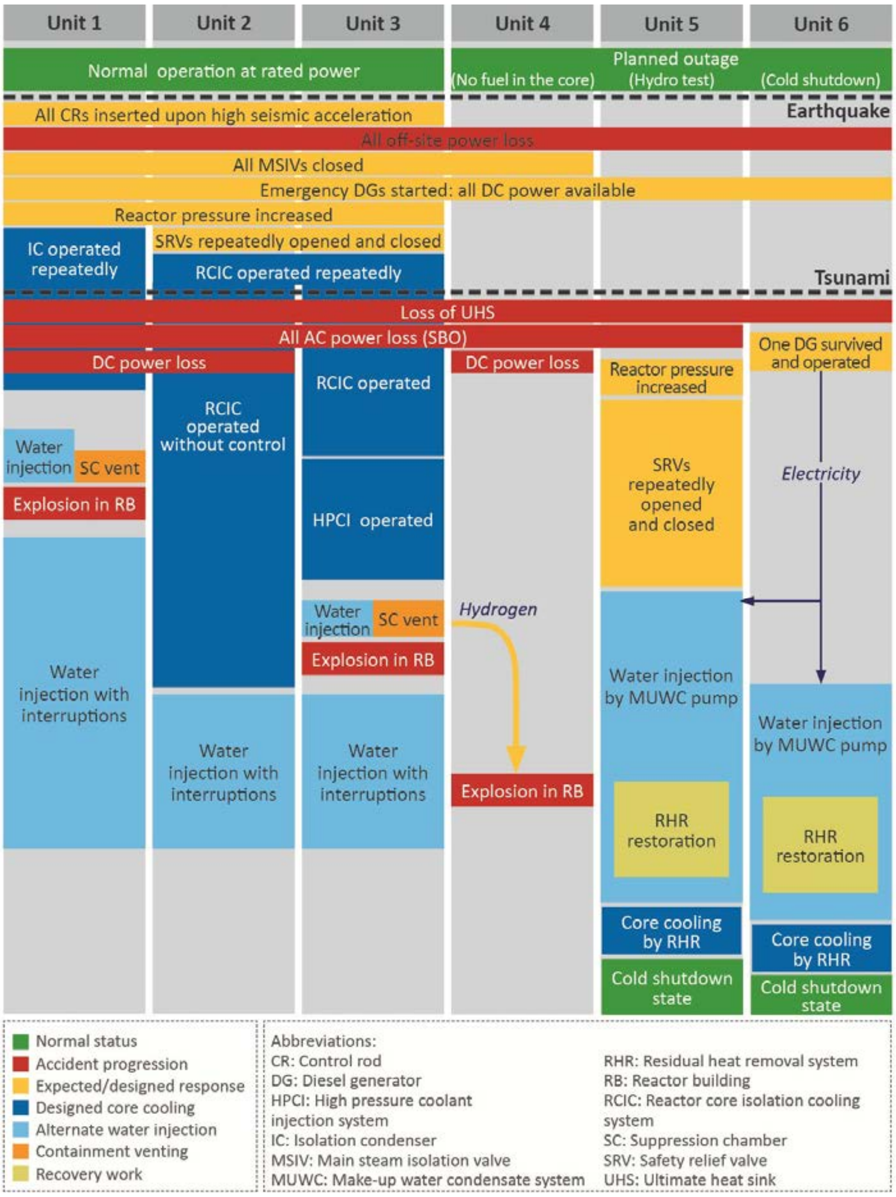

FIG. 2. The sequence of accident in Fukushima Daiichi units [1].

Also, as described in Ref. [1]:

> "People within a radius of 20 km of the site and in other designated areas were evacuated, and those within a radius of 20–30 km were instructed to shelter before later being advised to voluntarily evacuate. Restrictions were placed on the distribution and consumption of food and the consumption of drinking water. [Today,] many people are still living outside the areas from which they were evacuated." [1].

Following stabilization of the conditions of the reactors at the Fukushima Daiichi nuclear power plant, work to recovery and preparation for their eventual decommissioning began. Efforts towards the recovery of the areas affected by the accident, including remediation and the revitalization of communities and infrastructure have been performed since 2011 [1].

3.2. LESSONS LEARNED

In order to ensure safety of public and environment in nuclear electricity generation, one needs to consider two main elements: technology (technical and physical capabilities of the facility, methods for analyses, understanding of phenomena, etc.); and, human and organizational aspects (behaviours, commitments, leadership, programmes, processes, procedures, etc.). Every time after a major event or accident, investigations and the associated lessons learned reports touch upon the role of both technical and non-technical elements in the event, both in their successes and failures.

Following the accident, engineering and human performance assessments have been conducted to determine the causes, consequences, lessons learned and possible and reasonable preventive — and if identified, corrective — actions. The results of those assessments pointed out some specific areas for taking actions as the existing technical and human/organizational status of operating nuclear power plants with respect to plant design, configuration and operation, including the programmes, processes and procedures, were evaluated in the light of the available lessons learned from the accident.

There have been many reports on the Fukushima Daiichi accident causes and consequences starting with the report from the IAEA's International Fact Finding Mission conducted immediately after the accident, in May 2011 [3]. In the following months and years, Japanese and other national, regional, international organizations as well as various industry and technology entities conducted investigations, reviews and evaluations on the accident some of which identified and reported the lessons learned to be considered for applicability to other nuclear installations around the world. As these investigations found, the Fukushima Daiichi accident revealed a variety of lessons learned in both technical (e.g. design fundamentals, physical configuration, etc.) and non-technical (i.e. behavioural, organizational, institutional, societal, etc.) aspects.

The IAEA Director General's report, The Fukushima Daiichi Accident [1], that was published in 2015 provided a compiled list of observations and associated lessons learned from the accident. This report was prepared with participation by 180 experts from 42 Member States (with and without nuclear power programmes) and several international bodies performing comprehensive analysis of the accident at the plant, site, country, region and worldwide levels. It was based on the observations, collected information and the reported results of the IAEA's review missions to the Fukushima Daiichi nuclear power plant (and other power plants in Japan), TEPCO and Japanese government agencies, as well as the discussions and findings of experts during the meetings organized by IAEA and other bodies, such as a series of IAEA international expert meetings, technical meetings and action committees. The report also reviewed all other investigations, studies and evaluations by other entities and compiled and elaborated on their lists of lessons learned with the consideration of confirmed fact, the updated and new (i.e. after the completion of earlier investigations) findings. Due to its recent, comprehensive and consensus nature, the list of lessons learned provided in Ref. [1] will be the primary basis of the discussions in this publication.

As it is the scope of this report, the lessons learned related to technical and human/organizational aspects, particularly focusing on those that are directly applicable to the operating organizations, are discussed hereafter.

3.2.1. Lessons learned in technical aspects

Based on the reviewed and assessed plant response to the event(s) in the Fukushima Daiichi nuclear generating units, key technical observations mainly involved:

- Inadequacy of the evaluations of site-specific external natural hazards and multi-unit interfaces and insufficient preparedness for extreme external events;
- Not being able to withstand prolonged loss of power (i.e. station blackout (SBO), extended loss of alternating current (AC) power (ELAP)) and core and equipment cooling, e.g. loss of core flow or loss of ultimate heat sink (LUHS));
- Insufficiency of reliability and availability of electrical power systems;
- Inadequacy of on-site and off-site emergency control centres to ensure protection from extreme external events and radiation hazards;
- Inadequacy of design and operational measures to preserve containment integrity and prevent/minimize radiological releases;
- Insufficiency of the response and management of severe accident monitoring and mitigation means, including the provisions and guidelines associated with them.

Upon those observations and their evaluations, a list of conclusions and lessons learned were provided in Ref. [1] which can be summarized as follows:

- “The assessment of natural hazards needs to be sufficiently conservative. The consideration of mainly historical data in the establishment of the design basis of nuclear power plants is not sufficient to characterize the risks of extreme natural hazards. Even when comprehensive data are available, due to the relatively short observation periods, large uncertainties remain in the prediction of natural hazards.

- The safety of nuclear power plants needs to be re-evaluated on a periodic basis to consider advances in knowledge, and necessary corrective actions or compensatory measures need to be implemented promptly.

- The assessment of natural hazards needs to consider the potential for their occurrence in combination, either simultaneously or sequentially, and their combined effects on a nuclear power plant. The assessment of natural hazards also needs to consider their effects on multiple units at a nuclear power plant.

— Operating experience programmes need to include experience from both national and international sources. Safety improvements identified through operating experience programmes need to be implemented promptly. The use of operating experience needs to be evaluated periodically and independently.

— The DiD concept remains valid, but implementation of the concept needs to be strengthened at all levels by adequate independence, redundancy, diversity and protection against internal and external hazards. There is a need to focus not only on accident prevention, but also on improving mitigation measures.

— Instrumentation and control systems that are necessary during BDBAs need to remain operable in order to monitor essential plant safety parameters and to facilitate plant operations.

— Robust and reliable cooling systems that can function for both design basis and BDB conditions need to be provided for the removal of residual heat.

— There is a need to ensure a reliable confinement function for BDBAs to prevent significant release of radioactive material to the environment.

— Comprehensive probabilistic and deterministic safety analyses need to be performed to confirm the capability of a plant to withstand applicable BDBAs and to provide a high degree of confidence in the robustness of the plant design.

— In case of an accidental release of radioactive substances to the environment, the prompt quantification and characterization of the amount and composition of the release is needed. For significant releases, a comprehensive and coordinated programme of long term environmental monitoring is necessary to determine the nature and extent of the radiological impact on the environment at the local, regional and global levels." [1].

3.2.2. Lessons learned in human and organizational aspects

After each of the major nuclear incidents, reports almost always pointed out that deficiencies and failures observed cannot be addressed solely by the technical preventive and corrective actions without identifying and addressing the underlying human and organizational factors that had been a part of the cause of the event. The IAEA report on Fukushima Daiichi accident [3] stated that the accident showed that it is necessary to apply an integrated and systemic approach taking the interactions and interfaces between technology and people/organization into account in order to identify plant's vulnerabilities better and more adequately.

Reports on the Fukushima Daiichi accident (e.g. Refs [4], [5]) pointed out deficiencies in non-technical aspects, i.e. behaviours, attitudes, commitments, management styles, programmes, processes, procedures, etc. that were parts of the culture for safety and culture for leadership of the individuals and organizations. Particularly observed were the pitfalls of human and organizational characteristics, such as group think, complacency, beliefs and assumptions in decision making at the execution and administration levels. The IAEA report [1] pointed out, for example, the risk of flood triggering a nuclear accident was outside the belief of experts and leaders based on a complacent assumption that design of nuclear power plants and safety measures in place were adequate, even conservative and robust, to easily withstand such external events of low probability and high consequences.

Based on the reviewed and assessed human and organizational response to the event(s) in the Fukushima Daiichi nuclear generating units, key lessons learned in human and organization aspects mainly involved [1]:

- "Accident management provisions need to be comprehensive, well designed and up to date. They need to be derived on the basis of a comprehensive set of initiating events and plant conditions and also need to provide for accidents that affect several units at a multi-unit plant.
- Training, exercises and drills need to include postulated severe accident conditions to ensure that operators are as well prepared as possible. They need to include the simulated use of actual equipment that would be deployed in the management of a severe accident.
- In order to promote and strengthen safety culture, individuals and organizations need to continuously challenge or re-examine the prevailing assumptions about nuclear safety and the implications of decisions and actions that could affect nuclear safety.
- A systemic approach to safety needs to consider the interactions between human, organizational and technical factors. This approach needs to be taken through the entire life cycle of nuclear installations.
- In preparing for the response to a possible nuclear emergency, it is necessary to consider emergencies that could involve severe damage to nuclear fuel in the reactor core or to spent fuel on the site, including those involving several units at a multi-unit plant possibly occurring at the same time as a natural disaster.
- The emergency management system for response to a nuclear emergency needs to include clearly defined roles and responsibilities for the operating organization and for local and national authorities. The system, including the interactions between the operating organization and the authorities, needs to be regularly tested in exercises.
- Emergency workers need to be designated, assigned clearly specified duties, regardless of which organization they work for, be given adequate training and be properly protected during an emergency. Arrangements need to be in place to integrate into the response those emergency workers who had not been designated prior to the emergency, and helpers who volunteer to assist in the emergency response." [1].

3.3. IDENTIFIED NEEDS FOR ACTIONS

The lessons learned from the causes and consequences of the accident, as well as the derivation, extrapolation and applicability of those to other similar initiators, conditions and circumstances, triggered wide range of methodical considerations and assessment by the nuclear power plant operating organizations, regulatory bodies and technology owners to determine needed actions. Although the variety of concerns and considerations for what is needed for a specific plant was wide, they could be abstracted into two main groups to determine the needs for action: nuclear safety and emergency preparedness and response [1]:

— The nuclear safety needs for action were concerned with addressing:
  - Vulnerabilities of the plant to external (and internal) events that are not considered in the design and assess to be reasonably anticipated in plant lifetime;
  - Application of the DiD concept not only in prevention and protection but also, equally, in mitigation;
  - Fulfilling the fundamental safety functions under rare or extreme events and conditions;
  - Management of BDB and severe accidents;
  - Regulatory effectiveness[3];
  - Human and organizational factors and interactions between people, organizations and technology under abnormal and extreme situations.

— The needs for action in emergency preparedness and response were concerned with being prepared for the response to a possible nuclear emergency, for example:
  - Emergencies that could involve severe damage to nuclear fuel in the reactor core, to spent fuel, or to both, on the site;
  - Emergencies involving several units at a multi-unit site (possibly occurring at the same time as a natural disaster) which could result in disruption at the site and of the local infrastructure;
  - Protection of emergency workers who need to perform duties under extreme conditions during a severe accident;
  - Systems, communications and monitoring equipment for providing essential information for both on-site and off-site responses need to be able to function under severely harsh conditions and circumstances;
  - Availability, functionality, habitability and operability of facilities where the response will be managed (e.g. on-site and off-site emergency response centres) under a full range of emergency conditions.

Consequently, the following compiled list of technical aspects needing improvement in the nuclear power plants was generally identified[4]:

— Assessment of extreme natural hazards in terms of source location, magnitude and probability, etc.;

— Preparedness for prolonged SBO/ELAP and LUHS including the supply capability, capacity and adequacy of electricity and water;

— Preparedness for LUHS including the consideration of alternative heat sink;

— Application of DiD concept for appropriate prevention and mitigation as well as for reliable assessment of safety margin in BDBE/BDBA conditions and uncertainties;

[3] The regulatory effectiveness discussed in this publication is mainly the effective self-regulation of the operating organizations and the industry. The regulatory bodies' needs for action and actions taken by them for their programmes, processes, procedures and organizations are not in the objective and scope of this report.

[4] The terminology used in this list is a compilation of survey responses. As such, some terms are not exactly the IAEA terminology but rather harmonized (see Glossary for the IAEA definitions). This also attest to one of the challenges noted by the operating organizations as to unclear definitions, for example, of BDB conditions (see Section 2.4).

— Fulfilment of safety functions in beyond design basis (BDB) conditions (including to what extent and how);
— Application of probabilistic and deterministic safety analyses in a comprehensive, combined and complimentary manner to confirm the capability of a plant to withstand applicable BDBEs/BDBAs, and if should they occur, to provide a high degree of confidence in the robustness of the plant design;
— Risk assessment methodologies (e.g. for the assessment of extreme external and multiple hazards, events involving multi-unit/multi-source situations, human reliability in harsh conditions and in execution of planned or unplanned actions in a complex and devastated environment);
— Containment integrity under core melt and reactor vessel conditions for suppression and retention of fission products for preventing or minimizing early and/or large release to the public and the environment;
— Continual and periodic reassessment of design basis for protection from changing external hazards and strategies to maintain, restore or improve safety, design and operational margins;
— Assessment and enhancement of off-site and on-site power distributions system reliability;
— Definition and determination of design basis internal and external events, design extension conditions (DEC) with or without core melting, requirements for severe accidents and DEC, applicability and adequacy of PRA/PSA in analyses of DEC, etc.
— SAM under damaged infrastructure by initiating, concurrent or dependent extreme natural (and man-made) events in and around the site and conditions and events involving multiple units at one site;
— Instrumentation and control systems that are necessary to remain operable during BDBEs/BDBAs/DEC in order to monitor essential plant safety parameters and to facilitate plant operations, e.g. instrumentation for water levels of reactor in boiling water reactors (BWRs), containment (e.g. drywell, sump, cavity) and spent fuel pool;
— Monitoring and communication methods, programmes and equipment that can survive extremely harsh conditions;
— Development of new technologies, such as:
  - In-vessel and ex-vessel debris retention and cooling capabilities;
  - Materials and designs for low or no discharge from the reactor, reactor vessel and containment, such as low leakage reactor coolant pumps, high temperature seals;
  - Fission product and hydrogen control and suppression, including accident tolerant fuel as a candidate for replacement of zircaloy cladding;
  - Protection, prevention and mitigation systems that are independent of electric power, (e.g. improved passive cooling for containment and SG, or upgrading to higher safety categories);
  - Remote sensing and handling tools (e.g. robots, drones, etc.) during and after an accident that may incapacitate human interference.

Similarly, the human and organizational issues needing improvement by nuclear power plants were generally identified as:

— Culture for safety such as questioning attitude, raising concern, group think, learning culture, complacency;
— Training, exercises and drills including postulated severe accident conditions to ensure that operators are as well prepared as possible;
— Communication protocols at the time of emergency;
— Delineation of responsibility in emergency (on-site and off-site);
— Methods for decision making in an organization, such as by institutionalizing a risk management committee and installing risk manager position, deployment of independent check or critique system;
— Systemic approach to safety by considering the interactions between human, organizational and technical factors;
— Institutionalized oversight and independent review and advise;
— Continuous training and education of leaders and staff on nuclear safety;
— Method for qualification of experts for assigned duties following competence mapping;
— Procedure changes such as in severe accident management guidelines (SAMGs) for use of mobile equipment and for load shedding from extended service of battery;
— Resources allocation (i.e. emergency team, communication, drill, etc.).

3.4. DECISION MAKING APPROACHES TO DETERMINE ACTIONS TO BE TAKEN

There have been many possible ways to characterize the approaches in deciding on the post-Fukushima action implementation strategies, such as by their drivers; by the extent of risk information is utilized; the balance of value and impact; or various combinations of these. The decisions on actions and their implementation were generally based on following common steps:

— Definition of problem (i.e. identification of gaps, vulnerabilities, margins, etc.);
— Governmental considerations (e.g. socio-political environment, national energy policy regarding energy security, reliability and acceptability, public opinion);
— Regulatory consideration (i.e. new or revised requirements, compliance verifications, conditions on operating license, etc.);
— Utility consideration (e.g. time and windows of implementation, practicality in retrofitting, economic feasibility within the remaining plant life, public and industry peer image, positive side effects, economic capability);
— Options and strategies to confirm (deterministically, probabilistically, or both) and solve (by, for example, technical and/or administrative measures, coping tools and strategies, permanent or temporary means, etc.) the problem;
— Justification of preferred solutions and their benefits and effectiveness (what can go wrong, what should go right, understanding risk, etc.) by:
  - Deterministic considerations (i.e. assurance of safety functions in design basis accidents (DBAs) and DECs, including phenomena analysis, DiD, safety margin, etc.);

- Probabilistic considerations (e.g. likelihood, risk, consequences, cliff edges for DEC);
- Assessment of value (risk reduction, safety and operational margin gain, public trust increase, etc.) versus impact (necessary and available resources, time and opportunity for implementation, such as shutdown for modifications, sequential and/or simultaneous work, particularly in fleets or multi-unit sites).

In the review of decision making in the Member States, decision making has generally not avoided various biases arising from cognition (cognitive bias) of decision maker (or of the organization that makes decision) or other external environment, such as from political or public pressure. Examples of such biases could be ‘probability neglect’ or ‘loss aversion’ (consequence-driven decision). In other words, if the decision is about a very low probability of occurrence and/or with high uncertainties, decision process and criteria are consequence-driven, namely if the consequence is significant, a strategy for prevention or mitigation might be justified because of this significance.

Therefore, there have been some divergences in decision making and the resultant actions by the operating organizations among the Member States — even among the plants in the same country or between each plant in one fleet, utility or site. This diversity in decisions making approaches on which action to take, and when to take them, are primarily seen owing to the differences in:

— Drivers (regulatory, industry, peer, self);
— Achievement goals (e.g. safety goal, such as prevention of core damage, prevention of release, or target core damage frequency (CDF) and/or large early release frequency (LERF) in limitation of radiological consequences, amount of time without outside support, sufficiency and adequacy of safety and operational margins, etc.);
— How deterministic and probabilistic assessment is used (solely or complimentary, in sequential or confirmatory manner);
— The extent to which the result of value assessment is taken into consideration by the operating organizations, such as considerations for additional potential benefits from particular actions beyond meeting minimum regulatory requirements;
— Possibility of cooperation with other operating organizations.

It is also recognized that even within the same country, same regulatory framework and same utility, differences in the approaches to scoping and implementation of actions also existed, particularly depending on:

— Specific location and associated hazards for individual plants;
— Plant’s technology and age;
— Corporate strategy to maximize safety and performance benefits from the actions, i.e. multi-purpose use, trade off specifications, etc.;
— Remaining capacity and licensed life of plant operation, such as in cases of small and old plants, premature shutdown of plant was preferred rather than investing on massive modifications. (For example, small units in Japan such as Shimane-1 (a 460 MW(e) unit with 41 years of operation), Genkai-1 (a 559 MW(e) unit with 39 years of operation) or Fessenheim-1 and Fessenheim-2 units (both 880 MW(e) with 39 and 40 years of operation, respectively) in France);

— Plant shutdown decisions made independent of the Fukushima actions, e.g. based on other economic, technical or regulatory circumstances.
  For example, a U.S. plant (619 MW(e), 49 years of operation, a single-unit site, part of a fleet) was granted a license extension to continue operation until 2029, but its owner/operating organization announced in 2010 (prior to the Fukushima Daiichi accident) that it would close the plant and cease operation in December 2019 for economic reasons [6]. The plant was one of the NPPs that were subject of the Fukushima Daiichi related order issued by the regulatory body, United States Nuclear Regulatory Commission (USNRC), for installation of severe accident capable venting system (wetwell, latest by June 2018 and drywell latest by June 2019). However, the USNRC agreed to plant's request for a waiver from installation considering the plant closure schedule, allowing the plant to avoid the vent installation until January 2020 [7]).

The nuances were also due to methodologies used to identify areas for improvement and to determine value/impact. As such, the justification and decision for selecting a certain strategy/action was generally based on the following decision criteria:

— Process of decision making was appropriate and transparent, including whether the decision criteria are clearly defined;
— Identification of shortfalls in prevention or mitigation analyses on the age of the nuclear power plant and time for deployment is comprehensive;
— Effectiveness of the strategy in performing the intended function is confirmed by model calculation and/or testing;
— Uncertainties can be quantified and evaluated;
— Aggregated actions, including newly selected options, combined with other safety functions, can meet the achievement goal (or safety and performance goals) in an integrated manner;
— Strategy is superior and/or add other benefits/values to the safety and performance when compared to potential alternatives;
— Effectiveness to reduce risks is balanced with the cost of implementation (reasonable cost) and the action is 'reasonably practicable' which needed definition and guidance (for example, Western European Nuclear Regulators' Association (WENRA) provided the regulator bodies and operating organisations with a guidance [8] on the application of 'reasonably practicable' safety improvements to existing nuclear power plants to explain the Article 8a requirements of the EU Nuclear Safety Directive on, *"timely implementation of reasonably practicable safety improvements to existing nuclear power plants"* [9]);
— Consequence of the decision in meeting regulatory, social and utility's business environment at large.

Figure 3 shows sets of considerations that were input to the decision making. The weighing of these factors differed from one utility to another, even one unit in one site to another.

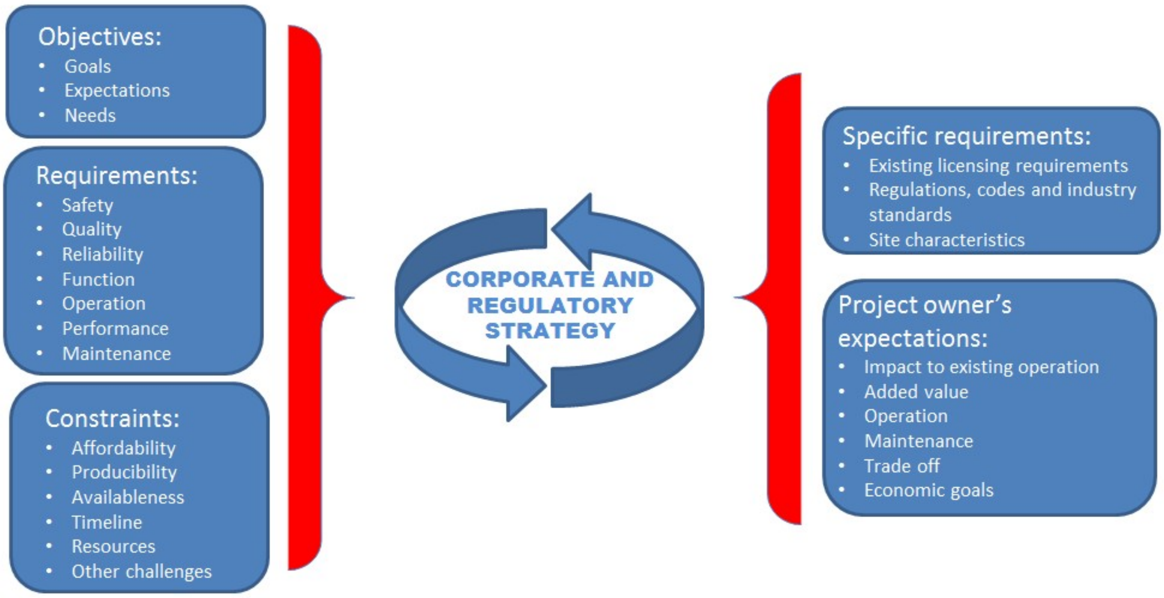

FIG. 3. An example of decision making for the actions that are reasonably practicable and for prioritization and value-impact by balancing the effectiveness to reduce risks with cost of implementation.

In order further to provide practical examples and methodology, the following sections illustrate sources of variations, i.e. nuances in drivers, achievement goals, integrated risk informed decision making and value/impact assessment.

3.4.1. Drivers

As mentioned above, actions in response to the lessons learned from the Fukushima Daiichi accident have primarily had three types of drivers: regulatory requirements; industry (e.g. self-regulatory, technology ownership, etc.) initiatives; and plant operating organization's strategies and conditions.

From the survey results, a clear majority of actions were driven by regulatory requirements, while some were taken by the individual plant owners and/or as suggested by the association of operating organizations and technology owner's groups. In addition, in some cases, political and sociologic environment that supports or opposes nuclear energy have affected the scope and extent of the actions taken, as well as their implementation schedule and cost/benefit analyses. As such, in a few cases, public (as stakeholder or in form of societal opinion) has played role in driving the extent of actions by the operating organizations to establish public comfort and confidence.

It should also be noted that, following the accident and its continued consequences, there has been a keen interest in (and scrutiny of) utilities' initiatives as a bottom-up approach by various stakeholders, particularly the public. This resulted in a lot of focus on ever more elaborate characterisation of vulnerabilities and strengths, e.g. BDBE/BDBAs and hazards, to regain public trust on the utilities. To accomplish this, continual context-setting and informing were needed from both regulation and operation perspectives.

3.4.1.1. Regulatory driven actions

The sources of differences among countries is particularly arising from the variation in the regulatory landscape (e.g. if, what and how new requirements are added); and the achievement goals, such as target CDF and/or LERF in limitation of radiological consequences, that were typically set (or had to be agreed by) the regulatory bodies. These could be mainly observed in regional approaches, for example:

— In Europe, the safety for continued operation was confirmed and, under the framework of 'stress test' [10] that was established and applied to all European countries by regulatory organizations. By responding to the descriptive stress test requirements, the European nuclear power plants identified the areas for improvement by individual plant evaluations. Although the framework and requirements applied to all plants in Europe, there were differences in decision for actions owing to the ultimate goals and end states among the plants/countries in response to the stress test evaluations. For example, French concept and strategy of *noyau dur* ('hardened safety core') aimed at providing large margins in prevention and protection, and deeper defence in mitigation [11] by several major additions to the design and configuration of their plants some of which have also been integrated in plant life management and long term operation goals and strategies. Other plants in Europe, meanwhile, established evolutionary measures, against the potential vulnerabilities identified during the stress test, aimed at providing robustness to the existing configurations and maintaining margins). Apart from the deployment of mobile equipment for additional means for power and water supply, physical plant modifications were considered and are implemented with a medium and long term view, as well as a part of other previously considered (or being implemented) plant improvement programmes, such as those for plant life management, license renewal.

— In the USA, where the regulatory body initially confirmed the safety of plants for continued operation, the actions were determined for short, medium and long term implementation based on the recommendations of a task force established by the regulatory body. The methods and approaches for assessment and implementation of actions were based on individual plant hazards and also driven by industry and technology owner initiatives, that have been agreed with the regulatory body. The main theme of US actions was with a main goal to prevent core melt under all situation, so majority of actions taken were specific to protect the reactor core, with some enhancements to prevent release where a potential vulnerability was identified in the Fukushima Daiichi accident evaluations (e.g. requirement for reliable hardened vent system in BWRs with Mark I containment design)[12, 13]. Furthermore, emergency response aspect was added in the assessments.

— In the Republic of Korea, similar to the European and US approaches, the safety of plants for continued operation was confirmed and a mixed regulatory approach built upon a government led safety review by addition of the stringent aspects of aforementioned international approaches, i.e. European and US, as well as periodic safety review process, was pursued. The ultimate goal of the basis for action was aiming at a universal implementation with deeper DiD and provision of increasing margins. On-going license renewal efforts for older plants were also coupled with the safety assessments. Additionally, due to the proximity to the accident state, public involvement in the decision making played an important factor, and public input (as well as non-governmental and/or non-technical expert reviews) was added in the decision making process described in Ref. [14]. It should also be noted that multi-unit considerations were also a more relevant factor in the safety assessment compared other regions, due to specific conditions such as having one site with eight units.

— In Japan, on the other hand, being the accident state, immediate shutdown of operating plants until a comprehensive safety assessment is performed was the approach. The restart of those were contingent upon complete evaluation and 're-licensing' in accordance with the new regulatory requirements set to eliminate recurrence of similar accidents. The ultimate goal was prevention of core damage as well as early release, as such the actions taken have covered, prevention and protection of those. It also necessitated additional action for mitigation of release. Also, different from the other Member States, post-Fukushima modifications were decoupled from plant operating license extension. Furthermore, plant modifications for assurance of safety in the case of terrorist attack are being made in consideration of nexus between safety and security, since such event would create similar conditions as the Fukushima Daiichi accident as to causing extended loss of power and loss of heat sink. This move was linked with incorporation of IAEA standards for security into domestic rules and regulations [15].

Overall, decision making by all regulatory bodies in driving the scoping and implementation of actions (as determined or agreed) were based on deterministic approaches. Particularly owing to the lack of standard and comprehensive PRA models that are approved and reliable for risk informed decision making, a significant part of new regulatory requirements and acceptance criteria in the Member States were established deterministically. In some cases, however, a combination of deterministic and probabilistic approaches was utilized with the intention of securing reasonably appropriate DiD, as well as minimizing/eliminating potential impacts of Fukushima related activities on normal, safe and reliable plant operations (for example, in consideration of risk and consequences from Fukushima related modification/construction in one unit while other units in the same site are operating. It has also been used to utilize the actions taken to reduce operational risk and enhance performance, as discussed in Section 7.

Also, in the Member States where a defined value/impact analyses process existed, such as those under a backfit rule, e.g. one in the USA, a cost/benefit analysis was consulted as an input to regulatory reviews. During the establishment of regulatory acceptance criteria and review methods, the safety was assured based on the significance of impact and value of the proposed changes for continued operations of the plants. These also supported the regulatory acceptance of action scope and schedules. One of the exceptions to such process was Japan, where the plants were to remain until a comprehensive and deterministic safety assessment is performed. This prompted a need to define acceptance criteria for review and approval of applications for restart of temporarily shut down plants. However, it had to be done without an established methodology for value/impact analysis, such as the backfit process, as well as the absence of a standard PRA/PSA that could be utilized in the assessment and reduction of risk [16].

The drivers were also defined either descriptively or prescriptively which differed from one Member State to another. Two particular experiences (both with deterministic approach) for selection of post Fukushima strategies with different requirement approach could illustrate such cases in Europe and Japan:

— In Europe, given that prolonged station blackout (PSBO) combined with LUHS, which had a very low probability of occurrence, was the contributing cause of the Fukushima Daiichi accident, the approach in Europe depended much on deterministic (stress test[5]) approach as proposed by WENRA [10]. There, the importance of resilience against beyond design basis external events was recognized and this led to the consideration of coping strategies for when the design basis envelope is exceeded. Deterministic approach, particularly the stress test, was recognized as a useful tool to determine if and where a cliff edge exists (and to explore how to extend the distance to the identified cliff edge) with due consideration of probability of occurrence. The results of this test confirmed the justification for continued operation of European nuclear power plants. Yet, the test identified many points for safety improvements, such as filtered venting systems, on-site and off-site storage of mobile equipment that may help prevent and mitigate the consequence of severe accident, as well as a standard method for calculating risks arising from earthquake and flood, installation, among others. Particularly for containment, measures to be taken for equipment, procedures and accident management guidelines were identified during the peer review process [17], addressing the depressurization of the primary circuit in order to prevent high pressure core melt, the prevention of hydrogen explosions and containment overpressurization.

— Whereas, in some Member States, such as Japan, the actions were driven by prescriptive regulations, where all the available units were shut down immediately (or soon after the accident) and their restart was contingent upon conformance to the newly established restart criteria. The new regulatory evaluation and new design basis requirements for restart had to be retrofitted in the review of existing fleet which resulted in extensive efforts to reconstitute and revise the existing design to meet the new requirements. For example, the determination of design basis earthquake (DBE) and tsunami resulted in a challenging case. As the regulation for new nuclear power plants did not allow installation on an active fault, if a fault beneath the existing reactor building was found to be active, or a shattered zone, the subject existing plant would not be authorized to operate unless a reliable probabilistic fault displacement hazard analysis (PFDHA) were to be used for further evaluation on safety. Furthermore, considering that the current IAEA guidance for seismic hazard evaluation [18] requires comparisons with similar structures for which historical data are available should be used in this determination. Also, new tsunami review guide in Japan assumes significant large earthquakes observed in the Pacific Rim [19].

3.4.1.2. Industry driven actions

Although the most plant specific actions were driven to comply with the requirements and requests by the regulatory bodies, a coordinated industry’s response, for scope and extent of actions, also drove not only the implementation methods and tools, but also some additional actions. These efforts were particularly in determination of methods and strategies to better comply with regulatory requirements, as well as for further DiD in compliance. Noted ones were coordinated through various international/interregional groups, such as pressurized

[5] European stress test was ordered by the European Union for over 140 nuclear power plants in Europe, in accordance to the specification prescribed by European Nuclear Safety Regulators Group (ENSREG). The order was carried out by the owner/operating organizations which was reviewed by national regulatory bodies followed by peer reviews by other EU countries. The finalized reports were endorsed by ENSREG in April 2012. The assessment covered three topical areas: Initiating events (earthquakes, flooding and extreme weather conditions); loss of safety systems (issues related to loss of power or ultimate heat sink, or a combination of both, as a consequence of any event); and severe accident management. Stress tests were also carried out in some non-EU countries such as Russia and, as an initial stage, in Japan.

water reactor (PWR), BWR and Canadian deuterium uranium reactor (CANDU) Owners Groups (PWROG, BWROG and COG, respectively) and/or WANO; while some were imposed by the national associations, e.g. INPO and Nuclear Energy Institute (NEI) in the USA, Federation of Electric Power Companies (FEPC) and Japan Nuclear Safety Institute (JANSI) in Japan, etc. For example:

— The general assembly of the WANO in October 2011 approved a set of wide-ranging commitments to nuclear safety at the organization's first major meeting after the accident at Fukushima Daiichi. INPO's lessons learned report [5] was instrumental for WANO to develop the its actions, which are implemented through 12 projects: self-assessment; emergency planning; severe accident management; on-site fuel storage; emergency response planning; design bases; corporate peer reviews; equivalency of other organization's peer review; early notification strategy; visibility and transparency; increasing frequency of WANO peer reviews; and assessment process. While respecting autonomy in each regional centre, regional crisis centres were established on March 2013 and internal consistency between four regions were sought to increase.

— In the U.S., the industry developed Diverse and Flexible Coping Strategies (FLEX) [20] — a program that aimed providing additional safety and emergency response equipment at each nuclear power plant in the USA — to the 2011 Fukushima Daiichi accident in Japan. This industry initiative was based on the initial response of the industry (noting that, immediately following the accident, the industry's self-regulating organization, INPO, issued three event reports [21–23] which later combined and expanded in Ref. [24] requiring the assessment and implementation of actions to improve the capabilities for and manage extreme events, i.e. BDBEs and ELAP).
Per FLEX strategy, which was reviewed and endorsed by the regulatory body [25], each nuclear power plant site had to determine and acquire portable equipment for providing power and water to keep the reactor cool and stable based on the coping strategy of preventing core melt until a longer term solution is in place. The portable equipment is stored in on-site protective storage buildings with mobility to key locations in the plant. Additionally, in a collaboration of operating organizations, two national response centres provide additional equipment and resources that can be dispatched to any U.S. nuclear power plant within 24 hours under the FLEX strategy. INPO also established a common operator training on response in accordance with the established FLEX strategy. Overall, the industry response covered, among others [26, 27]:

  - Defined external hazards, challenges and considerations including warning time;
  - Assumed initial conditions;
  - FLEX capabilities and storage requirements;
  - FLEX strategies and timing, phased approach;
  - DiD concept;
  - Procedure integration;
  - On and off-site resources and capabilities;
  - Mitigation strategy assessments.

— Technology owner's cooperation groups have provided direction and drive to implement some actions for the technology fleet's operating organization response. For example, PWROG has led the efforts to implement actions for the global PWR fleet owner/operator response [27] to provide a proven and consistent means to meet the post-Fukushima regulatory requirements (primarily based on the US response to Fukushima Daiichi accident and on the U.S. regulatory body, USNRC, review and approval) generically for technologies and topics (e.g. Refs [28–30]). Also, for European and Japanese PWRs, cafeteria projects by the PWROG intended to provide cost effective solutions for addressing region specific requirements. Such global operating experience sharing and benchmarking, as well as periodic technical meetings, workshops and maintenance programs, was to provide efficient implementation by the PWR operating organizations, particularly in 21 identified consensus safety improvements, including:

- SAMG update to incorporate filtered vents;
- Hydrogen transport methodology outside of containment;
- Hardware options for reactor coolant system (RCS) depressurization during severe accidents;
- Combined PWROG SAMG upgrade for all nuclear steam supply system (NSSS) technologies (i.e. Westinghouse, Combustion Engineering, Babcock & Wilcox) to make SAMGs more consistent and usable, issued January 2016 [31];
- Analytical bases for RCS response to ELAP;
- Development of NSSS specific FLEX core cooling strategies.

3.4.1.3. Nuclear power plant owner/operator self-driven actions

The post-Fukushima strategies, actions (or no actions) and schedules were ultimately dependent on the decisions of individual plant operating organization. These decisions on implementation were made with consideration of site and plant specific conditions, while mainly being in line with coordinated action through various groups as mentioned in Section 3.4.1.2 and being in compliance with the regulatory requirements.

However, in some cases, the owner/operator organizations reported that they proactively initiated and implemented actions based on the decision made at the plant management and/or fleet management levels (for example, nuclear power plant management may choose a strict compliance path or an operationally focused path. At the fleet level, the decision could be made to have consistent implementation, equipment and procedures at all fleet sites and units).

These own decisions by owner/operator organizations were made in accordance with the corporate strategy and vision, regardless of whether there is a regulatory and/or industry self-regulatory requirement being imposed.

Examples of nuclear power plant owner/operator self-driven actions included:

— In France, Electricité de France (EDF) decided to install French ‘ultimate diesel generator’.

— In the USA, several individual plants implemented solutions in response to post-Fukushima lessons learned by considering the management of operational margin and risk in addition to the requirements by the regulatory body while deciding on actions to implement. Such efforts required valid and approved PRA models and USNRC review and approval of margin and risk management. For example:

  - One of the modifications considered at some plants was to install diverse methods of making up to the spent fuel pool (SFP) by providing two pathways to transfer water from the condensate storage tank (CST) through a portable pump. By pre-deploying the portable pump during an outage and making the connections, this adds redundancy to the SFP makeup system (which is a critical system during a refuelling outage), improves the SFP inventory safety functions success paths and provides additional DiD without adding any additional cost.

- In another nuclear power plant, for the alternative success path for SG water makeup, a higher capability portable diesel generators (DGs) and larger capacity pumps than minimum needed capacity was decided to be installed in order to contribute to non-accident plant response to the failures of safety equipment (in this case, EDGs) that could happen during normal operation and would require plant shutdown to repair. The utilization of the temporary equipment, installed by the post-Fukushima actions, the unit would have had to be shut down until the EDGs are fixed and put back in service. The contribution from the post-Fukushima equipment (mainly the portable DGs and SG pumps) to the risk could allow the plant to remain online during repair effort.

3.4.2. Achievement goals

Achievement goals are generally set at the major thresholds for the consequences to decide on the strategy for prevention, protection and/or mitigation, i.e. prevention of core damage; protection and/or control of confinement function and structure; and the mitigation of consequence of confinement function failure.

Although prevention and protection have always been given a paramount importance, mitigation of consequence is equally important and socially expected. This consensus was reached in the lessons learned from the Fukushima Daiichi accident [1] given that the accident showed that large release of radioactivity leads to a social disruption and relocation of residents[6]. As a result, the achievement goal in post-Fukushima era tends to be harmonized to prevent/limit core damage, e.g. in the US approach where the FLEX strategy had an objective “to establish an indefinite coping capability to prevent damage to the fuel in the reactor and the SFPs and to maintain containment functions by plant equipment and FLEX equipment” [Section 1.3 of Ref. 20], and/or to prevent/limit large release from the containment. In this context, two following contexts are to be noted:

— The European Commission amended nuclear safety directive [9] to mandate the applicants for a license for the construction of a new power or research reactor should demonstrate that the design limits the effects of a reactor core damage to within the containment.

— The Vienna Declaration on Nuclear Safety [32] called for:

> “New nuclear power plants are to be designed, sited, and constructed, […], should an accident occur, mitigating possible releases of radionuclides causing long term off site contamination and avoiding early radioactive releases or radioactive releases large enough to require long term protective measures and actions.” [32].

[6] Both Chernobyl and Fukushima Daiichi accidents released large amount of radioactivity resulting in relocation of residents and change in the social situation, noting that the time for relocation was very different between Chernobyl and Fukushima Daiichi due to existence of actinide in the former.

Furthermore, for practical purpose, regulatory bodies in some countries such as Finland, Canada and Japan have a goal of release limit to around 100 TBq of Cesium-137 (around 1 per cent of Fukushima release) with an associated probabilistic target. Canada had set a level to limit Iodine release, as well.

These ultimate achievement goals are linked with overall safety goals, e.g. CDF or LERF, which have traditionally focused on avoiding or minimizing harmful effect of radiation on human body. Thus, the goals were set in the context of fatalities of humankind (acute and latent cancer) and to limit health effect by exposure to radiation by nuclear accident to a very small fraction (around one in one thousand) of the societal health risk to which everyone is exposed. However, as for health effect, Fukushima Daiichi accident has shown many cases of fatality of hospital patients in the process of evacuation and psychological illness observed among evacuees or relocated residents, even though there was no acute fatality by radiation nor will be no discernible increase of latent cancer fatality (Refs [1, 33]). Several reports on the accident, including that of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) [34] concluded that the health impact on people experiencing prolonged period of evacuation (or relocation) by degraded quality of life and mental health among would be more serious compared to the predicted health effect of radiation. (Ref. [34] described the latent effect of radiation among them as indiscernible).

Additionally, some nuclear power plant operating organizations decided on further achievement goals that supplemented the safety goals, for example on increasing safety margins and DiD and provision of additional operational margins and operational risk reduction, as discussed in Section 7.

3.4.3. Consideration of costs and benefits

In decision making for prioritization, planning and scheduling actions, the operating organizations also had to consider the associated costs and added values/benefits. Various institutions or national safety authorities have tried to 'ballpark' the cost of the actions to be implemented in the immediate aftermath of the accident. These initial estimates had wide range of uncertainties and expected variations at the plant and utility level based on the preliminary scope and extent of the actions and unestablished requirements for their implementation, as well as the supply chain arrangements. As such, it varied from one country to another. For example:

— In Japan, a cost estimate of US $316 million per unit was foreseen immediately after the Fukushima accident [35]. However, in the process of establishing final requirements of the new Nuclear Regulatory Authority (NRA), it was a very preliminary estimate impossible to assess the full requirements. As the additional requirements for restart of plants were established, this initial estimate was an underestimate. For example, in the cases where significantly high concrete seawalls have had to be built to protect plants from tsunamis, the level of expenditure was estimated to approach US $1 billion per unit.

— In the USA, following the report from the Fukushima Near Term Task Force (NTTF) [36], the industry tried to develop an initial strategy for nuclear power plant safety modifications. The cost of those safety enhancements was estimated US $23.5 billion (for 104 reactors, at that time, so roughly US $225 million per unit). It appears that this included many items that were subsequently not required by the USNRC, so it can only be described as an overestimate since the average costs estimates were later revised to be US $4 billion (in 2015 dollars) for 100 reactors (US $40 million per unit, excluding the hardened containment venting system (HCVS) for the BWR Mark I & II which was estimated to be US $10-20 million per unit) based on the industry input [37].

— EDF in France initially estimated an extra US $11 billion bill for 58 reactors to cover necessary post-Fukushima enhancements (in addition to expenditure of some US $45 billion to achieve continued operation of all the units beyond 40 years) [38]. This equated to approximately US $190 million per unit. The measures mentioned by EDF in the stress test reports were to be deployed within ten years of the Fukushima accident with an estimated final cost of nearly US $80 million per unit. At a later stage, in response to the requirements issued by the regulatory body, Autorité de sûreté nucléaire (ASN), in January 2014 ("*noyau dur*" or hardened safety core), some improvements have to be continued with the fleetwide LTO program with an important financial investment (about US $55 billion).

— In Brazil, Eletronuclear was looking to spend US $200 million on stress tests and modification work at the Angra Nuclear Power Station (two units) [39].

There were also some variations in the projected expenditure for each plant to explore better alternatives to provide similar benefits with lesser costs. Such variation was particularly dependent on the reactor type and consideration of plants' competitiveness in electricity market (where operating organizations could carry out the minimum mandated enhancements) or a regulated jurisdiction (where cost recovery in electricity rates may be possible for additional actions above and beyond the minimum requirements).

In the Member States and fleets with a large number or type of reactors (e.g. in France, USA, where fleets use PWR reactor commonly and in abundance) also created some scale economies in refurbishment and long term operation which would be a consideration of the recovery of expenditures. For those countries operating other reactor types, by contrast, the level of costs would likely to be lower. Some of this, however, may be because (in Canada and the United Kingdom) it was expected that many of the reactors have a more limited future operating life.

On the benefits side, an assessment of options for the safety enhancement modifications and how they could be utilized within the bounds of the member state regulations/requirements to provide tangible operational benefits to the site via reductions in the site outage risk profile, outage duration, online risk profile, etc. For example, any critical path limitations in a plant's current outage configurations can be reviewed to determine if the safety enhancements/modifications and strategies could be used to address such limitations, and ultimately reduce outage time (e.g. procurement of high grade portable equipment for that purpose to recover costs) (see Section 7).

Standard methodologies for value/impact assessment that are also available in some reports, such as Ref. [40], were referred in some cases.

3.5. DECIDED ACTIONS TO IMPLEMENT

The overall results of assessments conducted by nuclear power plant operating organizations in Member States have led to a set of short, medium and long term corrective and/or preventive actions for their facilities and processes. They identified potential vulnerabilities and gaps to reach achievement goals and suggested actions for implementation in improving design robustness and programmatic/procedural effectiveness in the areas of, for example:

— Re-evaluation of site specific external natural hazards and credible events that would create multi-hazards and/or impact multi-units;
— Introduction of additional permanent and/or mobile means to withstand prolonged loss of power and core (and SFP) cooling;
— Enhancement of plant electrical power systems to improve reliability and availability;
— Improvements of emergency response to protect workers and the public including on-site and off-site emergency control centres/structures to ensure protection from extreme external events and radiation hazards;
— Strengthening of measures to preserve containment integrity;
— Improvement of severe accident management provisions and guideline.

Applicability of these actions can be classified in three major categories, that in most cases overlap and complement each other, in implementation areas:

— Hardware (i.e. physical changes to the facilities);
— Analysis (i.e. method and modelling changes, improvements);
— Human and organization (programmatic and institutional revisions).

The first two categories were related to hard skills such as engineering, i.e. hardware, software and component, while the last category dealt with the soft skills.

Following three sections present the detailed actions in each category, as reported by the nuclear power plant operating organizations, and discuss their:

— Reasons, i.e. drivers, bases, justification for implementing (or not implementing) specific actions;
— Schedules and underlying reasons/justifications for prioritizing (or deferring) specific actions in time, i.e. immediate, medium or long term implementation;
— Challenges encountered and resolution of the issues;
— Costs/impact of actions.

They are followed by Section 6, Section 7 and Section 8 which discuss, respectively, the reported:

— Verification of effectiveness, functional sustenance of physical changes, methods of preserving assets and anchoring of programmatic and procedural changes;
— Benefit/value assessment, including trade-off, multi-purpose utilization and credit of particular actions;
— Costs/impact of actions including the comparison of differences in cost and reasons for the differences.

4. SIGNIFICANT PHYSICAL CHANGES TO PLANTS

Following the Fukushima Daiichi accident investigations and evaluations, nearly[7] all nuclear power plants in the Member States implemented (or are implementing) physical changes to their facilities by either procuring and installing equipment and components or modifying existing SSCs. These physical changes are addressing the following elements by applying engineering and technological means:

- Identified vulnerabilities of the plant to external (and internal) events that are beyond what is considered in the original design or that are discovered during evaluations as new hazards;
- Provision of additional robustness, diversity, redundancy and margin where it is prudent and expected;
- Application of the DiD not only in prevention and protection but also, equally, in mitigation;
- Provision of additional assurance and alternatives to fulfil fundamental safety functions under rare or extreme events and conditions and BDBAs and severe accidents.

Particularly, these changes are (or being) implemented in response to the following lessons learned and recommendations [1]:

- "DiD concept remain valid, but implementation of the concept needs to be strengthened at all levels by adequate independence, redundancy, diversity and protection against internal and external hazards.
- There is a need to focus not only on accident prevention, but also on improving mitigation measures.
- Instrumentation and control systems that are necessary during BDBEs/BDBAs need to remain operable in order to monitor essential plant safety parameters and to facilitate plant operations.
- Complex scenarios involving consequential or independent occurrences of multiple external hazards affecting multiple units located on a site and, possibly, multiple nuclear power plants at different sites in the same region need to be considered in accident scenarios and actions to be taken.
- Robust and reliable cooling systems that can function for both design basis and BDB conditions need to be provided for the removal of residual heat.
- There is a need to ensure a reliable confinement function for BDBAs to prevent significant release of radioactive material to the environment." [1].

In addressing these lessons learned with the most optimized physical modifications, a comprehensive understanding and evaluation of a nuclear power plant's design and licensing basis, credible hazards and the achievement goals and strategies were necessary to:

[7] A few plants have reported that due to the limited funding some of the actions have been deferred to after 2019.

— Reach an informed decision on the need for and extent of physical modifications;
— Confirm the capacity and capability of the design and configuration for DEC and DiD;
— Plan and implement design features or modifications to meet regulatory requirement and achieve needed capabilities beyond what is required;
— Continue operations of new SSCs in a nuclear power plant, safely, reliably and efficiently, including the adequate programmes, process and procedures and human capabilities.

This process included an adequate identification and understanding and a subsequent comprehensive evaluation of internal and external hazards (reassessment of existing hazards and investigation of new ones that were previously considered low probability and high consequences) and vulnerabilities/strengths against them using structured assessment methods, for example described by IAEA in Ref. [41]:

— Existing design, operation and maintenance specification of plant SSCs;
— Margins in existing design and safety analyses;
— Existing, new or revised regulations;
— Applicable codes and standards;
— Existing and new operating procedures and training;
— Needs for instrumentation and human resources for special controls and monitoring.

The scope and extent of this evaluation depended on many variables including operating experience, plant and system design, regulatory environment, methods and expertise, corporate strategy, etc. For example, internal fire hazard had been comprehensively assessed and addressed by many nuclear power plants prior to the Fukushima Daiichi accident based on the operating experience. However, some plants had not performed a hazard evaluation of internal fire event prior to the accident, and consequently, added such assessment to the list of potential hazards to be considered and addressed as post-Fukushima action.

Resulting physical changes that are installation (new) and modification (revised) of SSCs, commonly, fell into the following categories [1, 5]:

— Structures to enhance protection against extreme external events;
— On-site AC and DC (direct current) power supply and electrical power distribution systems;
— Off-site power supply systems;
— Core cooling systems;
— Systems for spent fuel protection;
— Containment or reactor building protection;
— Backup ultimate heat sinks;
— Instrumentation and control systems;
— Emergency management systems and mitigation of severe accidents;
— Post-accident management systems;
— Protected function and storage of equipment, buildings, structures for emergency management;
— Plant protection systems.

These areas are discussed one by one in this chapter. It should be noted that these discussions are based on the survey response from nuclear power plant operating organizations and the topics discussed in this section may not be a complete list of all actions. It should also be noted that a complete list of physical changes will differ depending on the nuclear power plant location, design, configuration, age (including the vintage of technology), operation and maintenance practices, effectiveness and extent of existing programmes etc. Therefore, the physical plant changes discussed herein are the common impacts/issues/solutions that have been observed and collected from the operational experience, as well as those that can be anticipated based on the latest knowledge and technical fundamentals.

4.1. INSTALLATION OR MODIFICATION OF STRUCTURES TO ENHANCE PROTECTION AGAINST EXTREME EXTERNAL EVENTS

The safety of nuclear installations including the site related aspects needs to be reassessed during their lifetime from the beginning of design to the end of operation. This reassessment is necessary owing to new and additional knowledge, hazards, regulations, practices and experience. These reassessments may also consider the revised/updated national and international standards related to external event assessments, methods, tools that are used in original site evaluation and plant design — which need to be periodically updated and revised upon scientific and technical developments, advance engineering practices, as well as using lessons learned from occurred extreme natural external events . In this regard, the role of national and/or international peer reviews and exchange of operating experience is an important tool to assess and enhance safety against the hazards that may arise during the operational life of a plant (Volume 2 of Ref. [1]).

Another lesson learned regarding the extreme internal and external events was the necessity of making assumptions of complex scenarios and application of adequate conservative estimations at the site evaluation, design and different operational stages in relation to the potential occurrence of extreme external events of very low frequency but with high safety consequences. In response to this lesson which emphasized to consider extreme natural events with a low probability of occurrence but with the potential to produce severe consequences for the plant safety in the plant design phenomena, the nuclear power plants in the Member States reviewed their current design basis, verified and validated plant design and configurations, identified and quantified existing design and operation margins against the extreme natural events. The goal of the analysis was primarily to avoid cliff edge effect for hazards with a higher level than design basis. Also, in some Member States, the consideration of combined effects of natural external and internal hazards also became a regulatory requirement.

These lessons learned also expanded the protection from and prevention of situations beyond the design basis in plant operating organizations' (as well as regulatory bodies') decision on the scope and extent of making physical changes. In several cases, the plant operating organizations performed analyses (themselves or with some support from national and international scientific and technical support organizations) with consideration of DiD criteria with a comprehensive common cause failure (CCF), beyond the regulatory requirements. These analyses are used to identify potential weaknesses of critical systems and to suggest possible improvements. An operating organization responding to the IAEA survey, for example, noted the efforts that were voluntary, beyond the regulatory requirements:

“Reassessment of the safety cases against seismic, external flooding and weather hazards to check whether we could do more to enhance resilience, focusing on bottom line plant and security of integrity of the reactor pressure boundary, particularly for the Advanced Gas-cooled Reactors (AGRs), because the deployable backup equipment (DBUE) is based on securing natural circulation cooling.”

The following sections summarize specific consideration and actions that have been performed for enhancing protection against the extreme natural events. These sections also discuss common issues faced by the nuclear power plant operating organizations and how they manage them.

4.1.1. Seismic events

All nuclear power plants in the Member States took a hierarchical approach to consideration and assessment of their plants’ robustness and potential vulnerabilities to the extreme seismic events that were highlighted in the lessons learned from the Fukushima Daiichi accident. This approach consisted of:

— First, review of seismic state and identification new seismic hazards and associated vulnerabilities for beyond design basis seismic conditions that is based on the ‘most recent’ ground motion response spectrum (GMRS), by rigorous reviews, such as plant walkdowns, SSC design basis reviews and configuration status, i.e. determine whether the nuclear power plant configuration is bounded by the existing design basis.

— Then, screening of newly identified seismic hazards and vulnerabilities and assessing the impact on the inherited safety and operational acceptance criteria and margins for determining the requirement and necessity, as well as no action due to remaining bounded by existing analysis with adequate margins. An operating organization responding to the IAEA survey described an example of the evaluation process as follows:

  “As a part of the FLEX project for the nuclear power plant, two tanks were identified as key sources of backup water, whose seismic margins needed to be quantified. These were the reactor make-up water tank (RMWT) and CST. The RMWT, being a non-safety-related tank, was designed to the plant’s operating basis earthquake (OBE) level of 0.13g, whereas the CST, being a safety-related tank, was designed to the plant’s safe shutdown earthquake (SSE) level of 0.25g. Both tanks were evaluated using state-of-the-art methodology to determine their true seismic margin over and above their respective design basis values. These evaluations accounted for soil-structure interaction (SSI) effects between the tanks and the underlying soil media. High confidence low probability of failure (HCLPF) values were then calculated representing the seismic margins associated with each tank.”

— Finally, evaluating (or revising/expanding evaluation) discovered seismic hazards and vulnerabilities that deemed necessary and required to analyse to determine the scope of corrective and preventive actions, and their value/impact and timeliness of implementation for detailed planning and prioritization.

— Additionally, if hazards and margin assessment and following evaluation necessitate, performing seismic probability risk assessment (SPRA) — utilizing qualified PRA models — to determine extent of actions and reassessing risk/consequence and value/impact.

Based on the survey response from nuclear power plants, all performed seismic reviews. Some plants reported that they determined that the most recent GMRS is still bounded by the existing design (i.e. no modification was needed or required) or some minor seismic analysis was needed. On the other hand, seismic margin assessments in most of the plants identified numerous SSCs to have further evaluation to demonstrate functional capacity and capability during extreme seismic events as a result of updated input and assumptions in the assessments (e.g. new seismic curves being higher than original design). These SSCs particularly concerned those that enable the control (to put plant in a safe and stable state) of a prolonged SBO/ELAP situation based on their functional analysis, and they typically included:

— Primary system SSCs;
— I&C systems;
— Relief valves;
— Steam generators (in PWR plants);
— Condensate storage tank and other water sources;
— SFP detailed seismic analysis, including the structural assessment at high temperatures (for example, as one survey responder stated:

  "*the regulatory body required the evaluation of the structural response of the SFP structure to temperatures in excess of the design temperature, including an assessment of the maximum credible leak rate following any predicted structural damage"*).

  Another survey response explained the analytical steps that were followed for a detailed SFP analysis included:

  "The first involves analysis to determine the time to reach the inner concrete design temperature limit, the second is a stress analysis to determine the response of the SFP concrete to temperatures that exceed the upper temperature limit of the concrete. The results of these analyses are then used to determine the time available to begin providing emergency water makeup to the pool, to assist with procedural aspects of emergency SFP cooling (e.g. steaming and intermittent top-up, continuous flow/overflow) and to determine if additional measures are required to provide additional cooling."

Several survey respondents indicated that they qualified a PRA model for their plants and some utilized those models to perform a SPRA. Particularly, all nuclear power plants in one

Member State a SPRA was required for those nuclear power plants in the construction or the design stages.

These assessments and evaluations determined several physical modifications, for example:

- Enhancement of earthquake resistance of the SFP and fuel transfer canal, cranes, gantries, bridges;
- Seismic enhancements to support passive flow paths;
- Civil/seismic resilience enhancements of safety systems and housing buildings and structures, e.g. fire stations, dry risers to enable a water supply to charge face level (buffer store cooling);
- Implementation of an automatic reactor trip in the event of an earthquake;
- Performing seismic margin assessments of tanks which are the key sources of primary and backup water;
- Enhancement of earthquake resistance of the containment venting filter (e.g. sand filter);
- Strengthening seismic resistance of Quality Class B and C SSCs;
- Installation of automatic reactor trip on seismic conditions, which necessitated the evaluation of an automatic reactor trip in the event of an earthquake to determine whether it is required, beneficial or neither.

4.1.1.1. Drivers and reasons

Seismic walkdowns were done first on a plant initiated basis or on an industry initiative basis, in immediate response to the accident, and any issues found during these initial walkdowns were immediately corrected by most of the nuclear power plant operating organizations. Later, a more descriptive walkdown and seismic assessment for BDB earthquake was driven by the regulatory requirements in all Member States (results of which were to be submitted to the regulatory bodies). Accordingly, all utilities had to assess their seismic hazards using the 'most recent' GMRS. It should be also noted that, in some regulatory frameworks, conduct of a seismic PSA is required even if a seismic margin assessment (SMA) has been done and a probabilistic seismic hazard analysis (PSHA) was required for new nuclear power plants.

Furthermore, in some nuclear power plants, the owner/operator organizations, took the initiative to check whether they could do more than the regulatory requirement to further enhance resilience. For example:

- In some PWRs, enhancement of earthquake resistance of the containment venting filter was implemented although keeping existing system which had limited resistance to seismic hazards as is would be acceptable by the safety requirements since the utility has already planned to implement a new containment cooling system that would minimize the need for containment vent;
- For a nuclear power plant where the GMRS was bounded by the SSE, a SPRA may not be required; however, some nuclear power plants performed a SPRA to qualify the PRA model for per the regulatory guidance provided by the regulatory body, as nuclear power plant owner/operator initiatives;
- For the AGRs, further focus was provided on bottom line plant and security of integrity of the reactor pressure boundary since the DBUE is based on securing natural circulation cooling. As stated by a survey responder:

"Reassessment of the safety cases against seismic, external flooding and weather hazards to check whether we could do more to enhance resilience, focusing on bottom line plant and security of integrity of the reactor pressure boundary."

4.1.1.2. Challenges and resolutions

The biggest challenge for the seismic walkdowns was that the inspection of equipment could only be accessed during an outage. This challenge was resolved by better planning of inspections that would systematically delay parts of the walkdown and perform them during the outage.

Challenges in the performance of seismic assessments and analyses reported by the nuclear power plants included:

- Determining the magnitude of BDB earthquake and GMRS (this challenge resulted in extensive discussions between the operating organizations and regulatory bodies regarding to the seismic criteria, in some cases still in discussion. In order to ensure adequate margin and prevention, one approach was to ensure that there is no cliff edge, which necessitated extensive PRA (for example, in France, EDF performed 20 000 cases to perform a probabilistic (statistical) approach for determining the cliff edge effect), or improvements in the existing seismic probabilistic safety assessment (SPSA), as tried in Japan [41]. Typically, nuclear power plants chose to use a '1 in 10 000 years event' for the magnitude of BDBE.
- Applicability of existing conventional and conservative methods when, seismic coefficient, *Ss*, level rise and reaching a determined criterion by both the regulatory body and licensees for the level of conformity to the requirements.
- Prioritizing the first unit in a fleet to perform assessments and modification which necessitates the identification of most vulnerable unit.
- Ensuring the impact and value of seismic reinforcements the design by analyses which contain significant analytical uncertainties which may require accounting validity of evaluation by adding various test and examinations.
- Determination of trip setpoint and sensor locations to manage the risk of untimely or spurious reactor trip was a challenge for the plants that installed an automatic reactor trip on seismic conditions.
- Finding and hiring qualified technical support which can be handled by resource sharing by other nuclear power plants/utilities (particularly the availability of the seismology, geology and geophysics experts that were needed).
- New methods for fault identification.

It was also noted that the time that it took to establish advisory group/committee as parts of the regulatory and industry initiatives imposed a challenge for the schedule. The survey responses from the U.S. NPPs indicated long preparation time for a structured process that is referred to as the Senior Seismic Hazard Analysis Committee (SSHAC) process [43–45], which may take up to three years if it is performed from the beginning, was a challenge. This schedule was also driven by the availability of experts, as mentioned in the last bullet above. Industry organizations that already had a generic SSHAC process with possibility of incorporating plant specifics helped reduced this time. For example, pre-dating the Fukushima Daiichi accident (2003–2005 timeframe), the utilities in the USA participated in the development of a generic SSHAC analysis (performed by Electric Power Research

Institute (EPRI) [46, 47] in response to the regulatory requests [48−50]. There, the issue was specifically applicable to the Central and Eastern US plants owing to new higher seismic hazards identified by [then] updated data and models requiring further study and analysis for those plants. Since the generic SSHAC only performed for the Central and Eastern US plants, in response to Fukushima Daiichi accident related regulatory orders, the Western US plants had to perform the SSHAC process from the beginning, causing extended schedules — primarily due to difficulties in finding experts for the panel(s) and finding available time of the experts in those panels for scheduling the panel meetings in timely manner.

4.1.1.3. Schedule

As aforementioned, the seismic walkdowns include the inspection of equipment and equipment anchorages that could only be accessed during an outage. In some nuclear power plants, completion of seismic walkdowns took two consecutive outages owing to inaccessibility of some equipment during power operations and/or inclusion of additional items resulting from the assessment of first round of walkdowns.

The schedule of seismic hazard assessment depends on whether or not a plant site has already had a SSHAC process performed for their site. As discussed in the previous section, performing a SSHAC process may take up to three years if it is performed from the beginning.

4.1.1.4. Approximate cost

Cost of physical modifications that are associated with seismic event evaluations varied widely depending on the extent and scope of modifications which was also a function of vintage and the conformance codes and standards. For example, in France, the cost for seismic enhancements widely differs between 900 MW(e) 1400 MW(e) design plants since modern units have larger margins and less retrofitting to meet the codes and standards.

As the modifications ranged from small fixes (adding bolts and hangers) and hardening of large components (such as tanks, e.g. condensate storage, reactor water storage, reactor water makeup storage tanks) and structures. Hardening and modification of structures ranged from screening, evaluating, retrofitting existing structures, to full new system design and construction (e.g. French hardened safety core approach), so did the costs. Even within the retrofitting, the cost varied based on the required process or method. For example, when applying American Society of Civil Engineers (ASCE) standards [51, 52], the need for Tier 1 screening, Tier 2 evaluation or Tier 3 evaluation and retrofit changes (see Glossary) impacted the cost of modifications significantly.

Also, it was noted that, there were significant cost associated with the initial plant walkdowns, in addition to the cost of physical modifications. Per the survey responders, the approximate cost to complete the plantwide seismic walkdowns was typically in the range of US $300 000 to $500 000 per nuclear power plant unit.

Analytical work may cost US $100 000 to US $2 million per analysis depending on the extent of analyses. Additionally, the average cost for a SPRA is in a range of approximately US $3.5 million to US $8 million per nuclear power plant unit. These costs were dependent on the extent of performing additional or new analysis/evaluation, particularly fragility analyses for SSCs, as well as change in the DBE spectrum, etc. Cost of seismic hazard assessment depends on whether a plant site has already had a SSHAC process performed for their site. If not, the cost for the SSHAC process is in the range of US $10–$14 million.

4.1.1.5. Lessons learned from decision, implementation and strategies for long term sustainability

It is a lesson learned by the difficulties encountered to quantify the robustness of nuclear power plant's seismic aspects that it may be more effective and efficient to reassess them in response to new information, knowledge, hazards and practices arising during the nuclear power plant's operational life, for example, as part of periodic safety reviews or other as new information, knowledge, tools or methods become available. This lesson learned was also noted in regulatory evaluations and reviews (both prior to and after the Fukushima Daiichi accident) following the updates and changes in seismic hazards data. For example, in the USA, the estimated risk from seismic events at some NPP sites might have increased from previous estimates [53−55]. The USNRC acknowledged in a post-Fukushima Daiichi accident fact sheet that:

> "Seismic hazard estimates at some current Central and Eastern U.S. operating sites may be potentially higher than what was expected during design and previous evaluations, although there is adequate protection at all plants" [55].

This was also confirmed and handled accordingly by the utilization of updated methods in Japan in the aftermath of Fukushima Daiichi accident and the response of Kashiwazaki-Kariwa NPP during the Niigata-Chuetsu-Oki earthquake [56, 57] as well as the response of North Anna NPP during the August 2011 earthquake.

4.1.2. External flooding events[8]

Volume 2 of Ref. [1] notes:

> — "There is a need to use a systemic approach in dealing with the design and layout of SSCs for effective protection against flooding hazards.
> — There is a need to act effectively and promptly in implementing upgrading measures to maintain the defence in depth concept of an installation and to ensure the performance of safety functions when an original dry site becomes a wet site during its operational life as result of a reassessment of the flooding hazards at the site (i.e. having a potential for higher flood levels than the main plant grade level)." [1].

As stated in IAEA publications (such as Refs [41, 58 and 59]), external flooding events include floods due to tides, tsunamis, seiches, storm surges, precipitation, waterspouts, downstream dam forming or upstream dam failures, snow melt, landslides into, water bodies, channel changes and work in the channel. Increase of water levels in an NPP site, up to the point that water starts affecting safety related systems, may compromise the performance of the fundamental safety functions and start an accident sequence. Floods upstream in the river basin may carry large amounts of debris and items accumulated on the riverbank that could

[8] Note that this section discusses only external flooding since nearly all Members States had conducted detailed internal flooding risk assessments prior to the Fukushima Daiichi accident. However, for the NPPs in some Member States which had not conducted a detailed room-by-room internal flooding risk assessment, re-reviewed and re-evaluation of their internal flooding risks by indoor pipe or tank rupture was a part of their post-Fukushima accident actions.

impact the cooling water intake structures. Furthermore, flooding can inhibit plant ingress/egress for critical personnel, materials and vehicles that would be used for mitigation.

Similar to the seismic hazards, all nuclear power plants in the Member States took a hierarchical approach to consideration and assessment of their plants' robustness and potential vulnerabilities to the extreme external flooding events that were highlighted in the lessons learned from the Fukushima Daiichi accident. This approach consisted of:

- First, review of flood protection state and identification new flooding hazards and associated vulnerabilities for beyond design basis flood conditions that is based on the 'most recent' data and criteria, such as: Probable maximum flood (PMF) which is developed based upon the probable maximum precipitation (PMP); local intense precipitation (LIP); tsunami, hurricane, storm surge, etc., as applicable. These reviews included plant walkdowns, flooding design basis reviews and configuration status to determine whether the nuclear power plant configuration is bounded by the existing design basis.
- Then, screening of newly identified hazards and vulnerabilities and assessing the impact on the inherited safety and operational acceptance criteria and margins for determining the requirement and necessity, as well as no action due to remaining bounded by existing analysis with adequate margins.
- Finally, evaluating (or revising/expanding evaluation) discovered flooding hazards and vulnerabilities that deemed necessary and required to analyse to determine the scope of corrective and preventive actions, and their value/impact and timeliness of implementation for detailed planning and prioritization.
- If hazards and margin assessment and following evaluation necessitates, performing flooding PSA to determine extent of actions and reassessing risk/consequence and value/impact.

For some nuclear power plant sites where the calculated LIP accumulation depths at entrances to safety related structures were higher than the inlet elevations of some doors and hatches for limited durations, the potential pathways for water intrusion into potentially affected buildings/structures through gaps in doors and hatches were evaluated for each unit. A room-by-room internal flooding analysis of these potentially impacted areas of the plant was performed to assess the potential impact to these key SSCs when water enters several buildings through door thresholds and gaps in hatches. This room-by-room internal flooding analysis determined there are no adverse effects on key SSCs based on existing permanent passive plant features and the room-by-room internal flooding analysis.

In some Member States, the extended analysis of external flooding included consideration and evaluation of dam failures close to the nuclear power plant, particularly based on the latest information on development, i.e. upstream levees, dams, land use, impacts on the flooding frequency and maximum flood level (e.g. in the USA, Missouri River Flooding in 2010 and 2011 [60–63]), were required by the regulatory bodies.

Also, for the Member States with NPPs on a seacoast, risk of flooding caused by high winds and storm surges had also been assessed regarding to the damages to the flood defences observed in earlier events (e.g. hurricane Katrina, in 2005, extratropical storm Martin, in 1999 [64, 65], etc.). As such, prior to the Fukushima Daiichi accident, many NPPs had been evaluating and implementing protective actions based on the Blayais event that occurred in December 1999. For example, in France (the event state), by the time of the Fukushima Daiichi accident, upgrades were implemented in 22 NPP sites. All the planned work to enhance the safety of the installations following the Blayais NPP flooding was completed in 2014 with a total cost estimated around US $120 million [66].

Based on the survey response from nuclear power plants, all performed flooding reviews, some determined that the most recent data is still bounded by the existing design (i.e. no modification was needed or required) or some minor flooding analysis was needed. In most of the nuclear power plants the flooding margin assessments identified some SSCs to have further evaluation to demonstrate functional capacity and capability during extreme flooding events, for example due to tsunami height, storm surge or LIP (including some of the dry sites) exceeding the original design considerations.

Several survey respondents indicated that they qualified a PRA model for their plant and some utilized these models to perform a flooding PSA.

In some plants, it was later determined that there may exist vulnerabilities that are more important for established strategies for mitigation of other events (for example wind versus flood [67]. Particularly, some mobile equipment, when deployed, and/or stationary plant SSCs that are credited in the severe accident coping strategies, e.g. atmospheric dump valves (ADVs), were identified as vulnerable to combined event impact that required further re-considerations of coping actions and/or evaluations of vulnerabilities of such equipment.

Furthermore, during the time of the implementation of actions, there were events or close-call incidents that revealed unidentified vulnerabilities [68].

These assessments and evaluations determined several physical modifications and associated mitigation measures and controls, such as procedural, administrative, manual actions, etc., in accordance with the accident management strategies (for example, in FLEX strategy [69], as discussed in Section 3.4.1.2), including:

— For protection of the site from an extreme external flooding hazards, building new or modifying flooding barriers that included one or more, the following:
  - Tsunami wall/seawall (e.g. building a new wall, raising the existing wall or reinforcement of the shingle bank on the seaward side of the site);
  - Floodwalls (in some cases, the plants built a floodwall around the entire site);
  - Dikes, levees, embankments and fill structures;
  - Weirs and berms;
  - Flood control reservoirs/channels and drainage;
  - Structures for undertow protection, such as reservoirs.

— For improved detection of external flooding hazard, installation of various monitoring devices, some of which are also utilized for crediting advanced warning time [70], such as:
  - Tsunami/flood observation facilities;
  - Tide level measuring instrument at ground;
  - Water level measuring instrument at seawater intake pit;
  - Surveillance cameras to monitor site status.

— For protection of the building and the SSCs inside the buildings from an extreme external flooding hazards, installing new or modifying flooding measures that included one (or more) of the following:
  - Water tightening common building features, e.g. doors/windows;

- Seals for waterproofing building penetrations, e.g. electrical and mechanical penetrations and floor drains, as well as conduits, pathways, underground pipes that connect rooms (including those that had been cut out, abandoned) where those are determined — by field inspections or design history reviews — to be potentially affected by an 'external' flood event;
- Raising the elevation of equipment;
- Installing dam-boards in the perimeter or low points for building protection;
- Strengthening flooding resistance of SSCs, including those with the Safety Class 2 and 3 (as defined in Ref. [71]);
- Purchase portable submersible and sump pumps dewatering affected rooms or compartments;
- Waterproofing/sealing room-to-room connecting penetrations (conduits, pathways, underground pipes including those that had been cut out, abandoned);
- Installing leakage detectors;
- Installing check valves on drain lines;
- Purchasing and storing portable water barriers, such as aqua berms, sandbags, water absorbing flood stopper bags, etc., noting that these require advanced warning for deployment.

4.1.2.1. Drivers and reasons

Flooding walkdowns were done first on an industry initiative basis, in immediate response to the accident [72]. In this preliminary walkdowns, most of the issues found were immediately corrected by almost all NPPs.

Later, a more descriptive walkdown and assessment for beyond design basis flooding was driven by the regulatory requirements in all Member States (results of which were to be submitted to the regulatory bodies). All utilities had to assess their external flooding hazards using the 'most recent' data and criteria available.

In most Member States these assessments also included the 'cliff edge' considerations in accordance with their national and regional regulatory requirements which were in harmony with the IAEA Safety Standards, including the provision of adequate margin, as Paragraph 4.48A of IAEA Generic Safety Requirement (GSR) Part 4 [73] requires:

> "Where practicable, the safety assessment shall confirm that there are adequate margins to avoid cliff edge effects that would have unacceptable consequences" [73].

Furthermore, in some plants, the owner/operator organizations, took own initiative to check whether they could do more to enhance resilience, for example:

— Purchasing portable submersible pumps for dewatering affected rooms or compartments;

— Installing leakage detectors in the rooms or compartments;

— Installing tsunami/flood observation facilities, tide level measuring instrument at ground, water level measuring instrument at seawater intake pit, surveillance cameras, etc.;

— Enhancement of hazard prevention communication tools and methods (e.g. enhanced communication with national seismic, meteorological, hydrological and marine organizations) for advanced warning time (some plants also took credit for warning time in their response plans as approved or allowed by the regulatory bodies);
— Internal flooding protection provision, such as waterproofing electrical and mechanical penetrations between the rooms and compartments.

4.1.2.2. Challenges and resolutions

The following challenges were reported by the survey responders when implementing the actions for external flooding event:

— Determining the magnitude of maximum flood (e.g. 500-year flood or 10 000-year flood, etc.) (e.g. determined criterion between regulatory bodies and licensees for the level of conformity).
— Difficulty in determining design basis and the likelihood/evaluation of reaching cliff edge in lack of updated/most recent information which may result in artificially restrictive (i.e. conservative) evaluation, or conversely, finding out the original design basis is non-conservative based on recent information.
— Finding and hiring qualified technical support which can be handled by resource sharing by other nuclear power plants/utilities.
— Complexity of characterisation of the coastal flooding hazard particularly in combined probability of hazards (for example, some nuclear power plants reported that the detailed review of the coastal flooding licensing basis revealed an oversimplification in the combined probability assessment of still water height and wave height required extensive and time intense determination).
— Determination of ‘cliff edge’ which may be dependent on the initiating event. Since electric equipment submerged by flooding generally lose their function, flooding tends to create cliff edge, just as was the case of flooding of electric equipment room in Fukushima Daiichi Units 1–4. The issue is the likelihood (and uncertainties) of reaching this cliff edge to justify precautionary actions.
— Principally, if the required coping strategy equipment, if protected and deployable, could be used to mitigate extreme flooding, including ensuring the operability of devices (on or off site) in case of rough transport conditions and identification of areas that would be accessible and available for placement of equipment in case of degraded conditions owing to the flood water in nuclear power plant site and platform (e.g. portable submersible pumps for dewatering affected rooms or compartments).
— Ensuring the ‘dry site’ (for areas of important facilities) when the original design was not;
— Prioritizing the first unit in a fleet to perform assessments and modification which necessitates the identification of most vulnerable unit.
— Inspection and maintenance programmes for penetration seals.
— Initial licensing and the latest update being prior to a guidance for determining PMP, LIP, coastal flooding, tsunami height, etc.
— Taking manual action in combination with warning time for mitigation in lieu of physical changes with enhance natural occurrence procedures and programmes This goes into training and exercises for timing and staffing.

4.1.2.3. Schedule

In most nuclear power plants, flooding walkdowns were immediately performed in short term response to the accident and any issues identified during this initial preliminary walkdowns were immediately (within a year) corrected by most of the nuclear power plants. Later, more detailed flooding assessments through more descriptive guidelines that were provided by the regulatory bodies were executed, including detailed walkdowns and room-by-room assessments, in all Member States results of which were to be submitted to the regulatory bodies. Results of the findings were placed in medium term schedules with the exceptions of a few where more rigorous physical modification had to be performed, for example:

- In Japan, a requirement to ensure to avoid or minimize external flood risks was a part of restart of a nuclear unit, while, in some Member States, similar requirement was reviewed and applied in case-by-case basis from the results of site-specific flooding re-evaluation under new criteria.
- In one Member State, reanalysis of dam failure and associated need for reinforcement of a dike (water wall) resulted in temporary shutdown of four units at one a site until appropriate physical modifications are implemented.

4.1.2.4. Approximate cost

The approximate cost to complete flooding walkdowns was around US $300 000 to US $350 000 per nuclear power plant unit.

Analytical work may cost as low as US $100 000, and as high as US $2 million per nuclear power plant unit depending on the extent of analyses. Additionally, the estimated average cost for a flooding PRA was approximately US $3.5 million per nuclear power plant unit.

Construction costs of tsunami wall ranged (depending on the wall height) from US $200 million to US $1.3 billion (US $1.3 billion for Hamaoka tsunami wall with initial estimate of US $900 million [74]).

4.1.2.5. Lessons learned from decision, implementation and strategies for long term sustainability

Regarding the maintaining reliability of the mitigation equipment, several NPPs noted that maintaining the integrity of flood seals is an issue, that still needs to be determined.

Another lesson learned — from the difficulties encountered to quantify the robustness of nuclear power plant’s flooding aspects — was, similar to the seismic aspects, that it may be more effective and efficient to reassess them periodically in response to new information, knowledge and practices arising during the plant’s operational life. The quantification would have been less challenging, for example, if it had been a part of periodic hazard and vulnerability reviews or other as new tools or methods become available. Moreover, this lesson learned was noted from another angle in Ref. [75] which identified the fact of existence of inevitable differences between an analysis and those when it is repeated after a large time span, as one of the key lessons learned in seismic hazard evaluations. Ref. [75] noted that when utilizing state-of-the art tools, methodologies and latest knowledge and data, it would be nearly impossible to obtain the same quantitative results as those that were generated by tools and methods used in the initial design and licensing studies (for example, as discussed in Ref. [63]) “some of which were performed 40 to 50 years ago.” [75].

This challenge with alignment of original analyses with current situation necessitated a long, careful and detailed review/assessment of proposed requirements by the regulatory bodies who were the drivers to re-evaluate the hazards using updated hazard information and the latest methodologies. In some cases, these reviews resulted in removal of *"reference to the re-evaluated hazards, allowing licensees to address them within their mitigating strategies in a flexible and appropriate manner"* [76]. Removal of such requirement enabled the regulatory body to *"directly align the reasonable protection standard"* [76].

Nuclear power plants in some Member States have identified that a plant does not always need to implement plant modifications to mitigate the impacts/effects of flooding. These plants have developed guidance on evaluating manual actions associated with successfully carrying out an external flood response strategy. A successful manual response strategy is one which can be implemented successfully by properly trained crews in an organized pre-planned manner under the expected flooding/environmental conditions. In one Member State, the industry initiated and has developed specific guidance [69] on how to evaluate and credit flooding response manual actions in accordance with the guidance provided by the regulatory body soon after the Fukushima Daiichi accident [37]. Particularly noted in that guidance is the provision of conditions for taking credit, specifically for warning time, for the demonstration of taking manual actions in flood response.

4.1.3. Other applicable extreme external events

While the external events that were investigated immediately after the accident primarily focused on earthquake and tsunami (flooding), subsequent evaluations in mid-term expanded the scope all potential external events. It has also been clear that, depending on the location of the plant, the critical external hazards, for which the plant could be vulnerable, differed.

Therefore, in addition to the seismic and flood assessments in BDB conditions, those plants that have other plausible external events in their design basis have reported that they reviewed and assessed their applicability and extent to the beyond design basis. They also evaluated the spectrums of other low frequency, high consequence occurrences that need to be considered in the evaluation of plant vulnerabilities to those, particularly in relation to causing ELAP. These included, but were not limited to:

— High wind and tornado events;
— Off-site fires (e.g. forest, brush, chemical fires);
— Extreme temperature (high and low) events;
— Off-site and on-site landslides and avalanches due to extreme precipitation;
— Geomagnetic storms.

4.1.3.1. High wind and tornado events

In addition to the beyond design basis seismic and flooding events that were particularly observed occurrences in Fukushima Daiichi NPP, application of lesson learned included the extreme wind, hurricane and tornado events (which have been generally anticipated and considered hazards for the design basis of nuclear power plants) as a BDB condition (i.e. those that may exceed the design basis criteria and potential damage to the plant SSCs.)

High wind (i.e. tornado, hurricane, sustained high wind, etc.) events were reassessed by plant operating organizations and/or the regulatory bodies (integrated assessment to determine what requirements are necessary) to identify vulnerabilities to the potential occurrence of extreme wind events with the potential to produce severe consequences for the plant safety in the plant design phenomena. This was particularly necessary for the plants' on-site and off-site facilities to store mobile equipment in the construction of such buildings with resistance to various wind loads as to their availability and accessibility in complex scenarios.

As such, the nuclear power plants in the Member States reviewed their current design basis, verified and validated plant design and configurations, identified and quantified existing design and operation margins against the extreme wind events (such as resistance up to 100 m/s and potential missiles generated by such wind speeds).

High wind walkdowns typically generated a high wind equipment list and operator actions that are associated with this equipment. Particularly as part of the walkdown, the SSCs that are dependent on off-site power for operation identified for prioritizing the fragility evaluation of these.

The assessments necessitated physical changes in some nuclear power plants, in particular, hardening the structures by construction of a metal housing/blockage structures (e.g. steel housing, steel walls or nets, etc., including the new structures built for the storage of mobile equipment (see Section 4.12.6) constructed to be resistant to various wind loadings) to protect the critical SSCs, such as valve house of CST, for high wind (or tornado) missile protection.

In one case, high wind analysis required the NPP to construct new draft cooling towers due to the finding that existing cooling towers can be damaged by extreme wind velocities [77].

The challenges encountered in the implementation of actions for robust protection against high wind external events included:

— Setting maximum wind speed and design wind speed considering standard tornado, implementation of missile analysis;
— Limitation on installing area of outdoor facility and storage (necessitated to install the facility for heavy equipment in an area where tornado would not cause impact on the important facilities);
— As mentioned in Section 4.2.1, in some plants, it was later determined that there may exist vulnerabilities that are more important for established strategies for mitigation of other events (for example wind versus flood);
— In some cases, it was found that essential water storage tanks (condensate storage, reactor water storage, boric acid storage, etc.) were not missile protected which necessitated compensatory actions, such as:
  - Drilling on-site wells to compensate for loss of inventory of tanks that are determined to have high wind risk (which also had its own challenges as the poor water quality of the groundwater requiring deeper well drilling with special drilling tools and methods or necessitating water reclamation);
  - Physical enhancement of such water storage systems such as hardening by steel plate enclosures, installation of tornado protection steel nets which in most cases were challenging in terms of the determination of the strength and configuration of those measures. In some plants; however, steel nets had already been implemented not related to the lessons learned from the Fukushima Daiichi accident.

Schedule for these accident). However, some national regulatory bodies provided specific schedule based on the timing and duration of their evaluation and decision. For example, in the USA, the actions to be completed by the time of the issue of the regulatory order, which was five years after the accident, the latest.

Cost of actions widely changed from unit to unit, particularly based on the selection of actions (e.g. hardening of buildings by steel plate cover, covering with steel nets, drilling on-site wells, etc.). As actions were primarily mid-term (i.e. to be completed within eight years of the example, approximate cost to harden a typical CST by using steel plates that are plug-welded together, was approximately US $8 million. Section 4.12.6 provides further costs of a storage building that is designed for high wind.

4.1.3.2. Extreme low temperature events

Extreme low temperatures, particularly when combined wind, rain and snow, can inflict physical damage (including collapse) on power and water supply structures at site. It can also damage off-site power infrastructure, such as causing overhead lines and towers to collapse by the excessive weight of ice build-up on them heightening the threat of collapse in strong winds as one example shown in Fig. 4.

FIG.4. Damage caused by glaze ice on the 400 kV powerline Beričevo-Divača in Slovenia, February 2014. Courtesy of ELES (Slovenian transmission network operator) [81].

Such an extreme external event from ice and snow, the ‘ice storm’, can be caused by super-cooled rain, in combination with strong wind, freezes in contact with trees or structures, rapidly forming a thick layer of ice. In the last couple of decades, extreme ice storms, where the ice thickness was beyond what was anticipated in the design of transmission system elements, have occurred (for example, in Canada and North East USA in 1998, Germany in 2005 and Slovenia in 2014 [78–80]. In all these events a large number of transmission towers were damaged by exceeded design. For example, in the 1998 Canadian event, a 70 to 90 mm-thick ice formation on overhead lines was far greater than the thickness that was anticipated in the design concept and criteria or defined in national or international standards [78].

Extreme ice and snowstorms would also impact the site ingress/egress, and movements within the site, for longer periods inhibiting plant personnel, equipment and material transfers — as well as repair and recovery of off-site power infrastructure, resulting in an ELAP.

In addition to the ELAP, extreme low temperature and ice storms may cause LUHS owing to the ice clogging and ice blocking of cooling systems (or due to the formation of frazil ice, which was observed as early as 1990s [82]) or due to blockage of flow path as a result of freezing pipes. The freezing of pipes may increase the risk of equipment damage, pipe breaks and probability and frequency of other plant hazards (e.g. internal flooding) which would be affecting multi units at one site, or multiple sites in a region. Furthermore, the freezing of instrumentation lines or pipes/lines will render the SSCs needed for monitoring and mitigation of the consequences.

Therefore, the plants that are located in the regions that are predicted to experience extreme low temperature and associated events particularly beyond design basis heavy snow fall and ice storms. Typically, the regions in scope of reassessments were determined based on the climageography, such as located above specific latitude and/or historic weather data. For example, in the USA, the industry proposed (which was endorsed by the regulatory body) extreme snow fall vulnerability assessments to be performed by NPPs that are above the 35th parallel and ice storm assessments to be based on a database that summarized ice storms that occurred between 1959 to and 1995 [20]. As such, the US industry required snow, ice, and extreme cold assessments to be conducted by all US NPPs except those are located in Southern California, Arizona, the Gulf Coast and Florida (noting that, extreme low temperature events have been a design basis event for NPPs, for example, in Scandinavia).

Those nuclear power plants re-evaluated the frequency and consequences of such events and assessed the vulnerabilities for the maximum predicted loadings (ice, snow, wind, etc.) beyond their design basis, based on the lessons learned about the ELAP and LUHS event and their consequences leading to some actions that were implemented. The following is a sample list of physical changes reported by some plants as a result of their evaluation for extreme low temperatures (In addition to other actions, such as installation or modifications against the ELAP and LUHS that are discussed in Sections 4.2 through Section 4.7, as well as the purchase of vehicles and associated assessments that are further discussed Section 4.12.7):

- Installation of a heating, ventilation and air conditioning (HVAC) system and individual space heaters in the storage buildings or for temperature control (e.g. to maintain temperatures in the buildings about 50°F);
- Diverting warm water from the essential service water to the inlet of the intake structure to prevent icing;
- Installation of area heaters (e.g. gas heaters) in essential equipment rooms and around safety related SSCs (for example, boric acid tank (BAT) room) to prevent freezing;
- Strengthening structures for heavy snow and ice loads;
- Extension of pipes for deeper intake as protection against the formation of ice on the surface of the cooling water source from lakes, ponds, etc.);
- Taking credit in analyses/assessments for advanced warning time for heavy snow and/or extreme low temperatures;
- In the event of extremely low temperatures, add to daily plant surveillance (e.g. increasing the frequency and scope of existing surveillance of air exposed essential water lines.

4.1.3.3. Extreme low water or high temperature events

An extreme low water condition of the cooling water source could affect the safety related water supply for safe shutdown and for keeping the plant in a safe shutdown condition (typically for several days). Such low water conditions can occur with or without an advance warning and may lead to a gradual or rapid drawdown of heat sink water that could directly or indirectly result in LUHS. For example, draining or depletion away from intakes and other service water structures could cause permanent damage to the safety related ultimate heat sink (UHS) pump owing to air ingestion via vortex formation or cavitation due to inadequate net positive suction head.

The extreme low water events that required reassessment for LUHS by the NPPs generally included:

— Failure of downstream dams, including:
  - Concrete dams and embankments that are not seismically qualified and/or that were not in the original design basis;
  - Older large dams that were built based on significantly different codes, standards and design criteria and methods than today, which could be considered as inadequate;
  - Beyond design basis failure of seismically qualified dams).
— Failure of reservoirs/ponds/impoundments that are the heat sinks for NPP by safety design;
— Seiche in large lakes (low water phase);
— Low water level due to long lasting drought due to extreme long term low temperature and/or low precipitation.

Depending on the achievement goals and thresholds put in place by the regulatory requirements and/or industry expectations, the actions taken varied.

For example, in the USA where the NPPs established their FLEX strategies to prevent core melt during BDBE, regulatory assessment of the effects of a dam failure downstream of an NPP, particularly investigated the need for NPP actions against failure of seismically-qualified downstream dams. As discussed in Ref. [83], the reason for focusing the assessment only on those dams was that they are considered ‘robust’ and any other ‘non-robust’ dam failures (e.g. of non-seismic dams, embankments, or impoundment reservoirs) had to be addressed by the NPPs, under the regulatory body’s order to modifying licenses with regard to requirements for mitigation strategies for beyond-design-basis external events [84]. Accordingly, the industry guidance [20] directed the NPPs to establish coping strategies without relying on non-robust SSCs, as Ref. [83] stated that:

> “a non-seismic dam would not be considered robust and, therefore, licensees would develop strategies to use other sources of water [in accordance with FLEX strategies]. The scenario evaluated under the Order is an extended loss of AC power and a loss of normal access to the ultimate heat sink, coupled with a beyond-design basis external event.” (emphasis added) [83].

It was stated by the regulatory body that such scenario would bound *“all single scenarios that would result in a dam failure, including a random (sunny day) dam failure”* [83].

Although, the regulatory body evaluated 'non-seismically robust downstream dam' separately, it noted that: "*if a downstream dam or downstream impoundment was categorized as seismically-qualified, the NRC staff would consider the structure to be robust and would not evaluate its failure under the Order"* [83]. The USNRC eventually concluded that: *"additional regulatory actions related to random failures of downstream dams cannot be justified"* [85].

However, the actions taken had a common purpose of providing additional on-site and/or nearby water sources, but the means to obtain and deliver water through to remove decay heat differed. Some of the actions taken by NPPs against low water level of cooling water source included:

— Drilling groundwater deep wells for backup ultimate heat sinks;
— Installing additional deep well pumps for existing back up water source;
— Purchasing/installing submersible pumps;
— Installation of active cooling tower(s) for recycling and cooling water supply;
— Securing arrangements with the owner/operators of upstream dams to release water to maintain the water level of the UHS when a downstream dam fails;
— Procuring and installing stationary and portable connections, pipes/hoses, as well as reliable and adequate on-site water sources, e.g. tanks.

The challenge in the implementation of these action included ensuring good quality of water from deep wells or shallow brackish water remaining in the ultimate heat sink when water level decreases. These necessitated compensatory measures such as purchasing/installing water treatment systems and/or deep drilling (which also had its own challenges regarding the drill damages).

4.1.3.4. External fire events

External fires (e.g. fires affecting the site and originating from nearby forest, wild brush or grass) has the potential to initiate loss of off-site power (LOOP) owing to direct damage to the conductors and insulators on the overhead lines supplying off-site power. Also, smoke from fires occurring near an overhead line can cause repeated arcing faults on an overhead line because the ionized air in the smoke can become a conductor of electricity Such fires can also disrupt the ability of critical personnel and emergency responders to access plant and equipment for a long time causing an ELAP with impairment of operator actions.

A fire spreading fast towards the site with wind, can not only threaten the site directly (including potential sparks and embers) but also results in heavy smoke effecting plant ventilation/filtration systems and habitability (e.g. smoke with particulates plugging the filtration system). Furthermore, some NPPs evaluated the potential hazard from the use of fire extinguishing planes (low flying in hazardous flight conditions) which may increase the risk of plane crash combined with fire. Also, if there are chemical facilities or substances on the path of fire (or they are the originator of the fire), the effects of aerosols, corrosive and toxic substances needed to be considered in addition to the smoke, particulate and thermal impact.

— Develop plans for emergency access and response plans for the timely repair and restoration of electrical power grid system, in case of nearby wildfires;
— Control of vegetation under and near overhead lines and on and around the plant ingress/egress routes;

— Administrative measures may be implemented to constraint the flying routes;
— Establishing a forest fire protection belt (such as deforestation and/or concrete slabs) around the NPP site, dimensions of which were explicitly specified in regulations in some Member States, such as in Japan where the regulations recommended actions for minimizing/eliminating fire hazard up to 10 km from the power station [86];
— Establishment of natural fire brigade and construction of facility for the firefighters and equipment;
— Building firewalls.

Regarding the firewalls however, it should be noted that, in some Member States, the fire evaluations following the Fukushima Daiichi accident, included both external and internal fire assessments in NPPs. In many Member States, internal fire assessments had been required prior to the accident (e.g. between 2000 and 2009. In those countries NPPs had already conducted walkdowns and PRAs for internal fire and their associated actions were completed. However, NPPs in several Member States, particularly where such assessments were voluntary initiatives, had not performed and/or completed extensive internal fire evaluations. Those NPPs completed comprehensive internal fire assessment and took actions as a part of their post-Fukushima actions. As expected, those actions included performing internal fire hazard determination, plant walkdowns, deterministic and probabilistic analyses resulting in plant modifications, such as installation of firewalls, installation of fire detection systems, addition of fire extinguishing pipelines, and relocating some SSCs, such as motor operated valves (MOVs), cables and circuits.

4.1.3.5. Landslides or avalanches

Earthquake, heavy precipitation or volcanic activity could result in landslides, mudslides and avalanches in mountainous or sloped areas. Landslides caused by earthquakes or dip slope sliding have been anticipated external event to be considered for the NPPs adjacent to slopes (e.g. in Japan, Republic of Korea and Taiwan, China), since such events could directly affect the SSCs at site [87, 88]. For example, one plant identified the switchyard, the CST and the auxiliary building which were located outside and exposed to direct impact of the landslide as the SSCs that are likely be damaged by the adjacent slope failure.

Also, as it happened in Fukushima Daiichi accident, more common impact on NPPs in areas that are subject to earthquakes and avalanches, a landslide/avalanche at a further distance from the site generally would damage off-site power transmission system, e.g. overhead lines, towers, underground cables and major substations, which would cause LOOP. Furthermore, such external events would inhibit transportation and movement of critical personnel and materials where system equipment is damaged or submerged. In most cases, it is likely to take days (even weeks) to repair or replace, and thus, resulting in an ELAP.

Landslides were also considered as the initiator of combinations of other external events that are discussed earlier in this section, such as:

— Landslide generated tsunamis;
— Formation of a natural dam (and its subsequent failure) after a catastrophic landslide or large rock fall which cause flooding upstream;
— Landslides that enter a reservoir reservoir/pond/impoundment that is the heat sinks for NPP by safety design causing or increasing the risk of overtopping downstream embankment.

The assessment of landslides by NPPs were based on PSHA and typically included and identification of instable slopes by geological and geotechnical assessments and seismic fragility analysis of slopes.

Regarding the avalanches, following the reassessments of the risk of beyond design basis avalanches, all the reporting NPPs excluded (screened out) an avalanche as an extreme external event necessitating further detailed deterministic and/or probabilistic evaluations. (This should not be understood as all world's NPPs screened out avalanche as an initiating event, noting that this publication reports on the NPPs that responded to the IAEA survey and the publicly available information).

4.1.3.6. Volcanic activities

Volcanic activity can affect the NPP in several folds: Seismic impact, impact from mud and/or lava flow, impact from atmospheric shockwaves; and impact from volcanic ashfall. The evaluations of seismic impact were evaluated as discussed in Section 4.1.1 and the mud flow, lava flow and atmospheric shockwaves were considered as extreme external events for NPPs located in the immediate area of the volcano. More common impact identified by several NPPs was the impact from ashfall as more potential volcano related hazard which will constrain the functions of SSCs, particularly those for mitigation measures, such as portable equipment and vehicles [89].

The measures taken by the NPPs that identified ashfall as anticipated extreme event included:

- Purchasing portable mitigation equipment and vehicles equipped with (or ensuring that they can be equipped with) combustion air filters;
- Designing and installing equipment storage structures with protection from ashfall;
- Locating critical portable or permanent power and water supply equipment (such as diesel generators and diesel pumps) inside the buildings that are protected from ashfall;
- Provision of alternate connection points for portable equipment in case that the ashfall necessitates the relocation of the mitigation equipment or vehicles;
- Cooperation with geological survey organizations that monitor volcanic activity to provide days or months advanced warning time for volcanic eruption.

Challenges in the implementation included:

- Difficulties in setting ash concentration in the air for the filter choke analyses;
- Determining travel time of ash from volcano to the site for different wind and precipitation conditions for early site preparation;
- Determining reliable strategies for periodic filter changes for the equipment and vehicles.

4.1.3.7. Electromagnetic disturbance

Electromagnetic disturbance events, whether natural or man-made, are typically included in the design basis of NPPs, when applicable. However, following the Fukushima Daiichi accident, such external events have been rereviewed for BDB conditions, particularly regarding prolonged SBO/ELAP and their impact on mitigation equipment and deployed vehicles.

Geomagnetic storms can cause instability and disruptions in the electrical power grid system by creating large currents to flow in the upper atmosphere near the earth's poles inducing currents (geomagnetically induced current) in the surface of the earth; particularly near to the polar regions or suitable conducting structures. Although the effects of such storms are likely to be greater nearer the poles, the effects may spread to larger areas owing to the, the electrical power grid systems, that span large geographical surface with long distance overhead lines along the direction of the currents in the upper atmosphere, acting as conductors [90]. These storms are generally most intense around a peak in the sunspot cycle (with an approximate frequency of 11 years) with the potential effects on electricity systems that include the saturation of the cores of transformers and the overheating of ground connection conductors or resistors. For NPP, this translates to an ELAP owing to the susceptibility of main transformer to core saturation and thermal damage or the potential collapse of transmission system — that can result in a widespread electrical power grid system blackout for an extended duration (from a week up to several months). For example, in the infamous event caused by a solar storm, the main transformer at the Unit 1 of Salem NPP in the USA failed during the 13 March 1989 storm when overheating melted low voltage service connections and insulation discoloration. Furthermore, almost exactly six months later, a second solar storm damaged the main transformer of Unit 2 at the very same plant [91, 92].

Similar impacts occur if the electromagnetic disturbance is man-made, such as a nuclear detonation in the upper atmosphere that creates an electromagnetic pulse (EMP). This extreme external event, commonly referred as high altitude electromagnetic pulse (HEMP), can result in a powerful, damaging electromagnetic field which can affect large scale areas of electrical power grid system and resulting in ELAP at NPPs.

Furthermore, such events may significantly hamper critical infrastructure, such as telecommunication and transportation system interruptions and disarrangement of the operations of fuel suppliers, preventing or delaying the timely provision of off-site resources needed to cope with ELAP.

The design of NPPs considers the effects of electromagnetic interference (EMI) and radio frequency interference (RFI) which is anticipated to be initiated by plant equipment, e.g. a high voltage switchgear, wireless systems, etc. Also, the interference effects of portable telephones, electronic devices, radios, or natural phenomena (electromagnetic pulse by lightning), or intentional means, such as intentional electromagnetic interference (IEMI) devices are considered in the design, as such interferences can cause damage to sensitive electronics.

The overall design basis considerations for all these electromagnetic disturbance events followed similar regulations, codes, standards and equipment qualification in all Member States. However, the consideration of such effects in BDB conditions varied among the NPPs in Member States, particularly the equipment qualification and protection of mobile or permanent equipment, that are acquired for mitigation purposes, against the electromagnetic disturbances. For example, in Sweden, regulatory body required independent core cooling to be able to perform under electrical disturbances from conduction and electromagnetic radiation. Therefore, Swedish NPP constructed a new building to house independent power supply equipment, for new independent feedwater injection, SFP make-up and RCS volume control systems, with electromagnetic disturbance protection (both the structure and cables with appropriate shielding) [93]. On the other hand, in the USA, a prevention approach was used, such as establishing a 'NO RADIO ZONE' in the areas, for example, where the SFP instruments are located that could be susceptible to electromagnetic interference [94].

4.2. INSTALLATION OR MODIFICATION OF AC POWER SYSTEMS

Most of the problems in the Fukushima Daiichi event arose from the total and extended unavailability of electrical power, a scenario which had not been anticipated in the plant design. This situation was generally shared by most of today's operational reactors, as consideration of the total loss of electrical power has not been a design requirement. It became important to implement short and long term solutions for this challenge at existing nuclear power plants.

In response to lessons learned on on-site and off-site AC power supply and distribution systems which emphasized the consideration of phenomena with a low probability of occurrence but with the potential to produce severe consequences for the plant safety, in the plant design. Consequently, in all Member States, it became a regulatory requirement to establish mitigation strategy for prolonged SBO combined with LUHS to include additional mobile cooling features, in one form or another. In most plants, even for BDB situations, DiD criteria had to be considered and a comprehensive CCF analysis needed to be performed based on DiD criteria to identify potential weaknesses of critical systems like the on-site and off-site AC power supply and distribution systems. Considerations of low probability events with high consequences suggested possible improvements which could add further costs, as one NPP reported that the average cost per unit for the electrical power modifications, i.e. safety enhancements, (which included both the 480 volt and 4160 volt) was US $6.2 million per unit.

4.2.1. Adding low voltage AC power suppliers

A low voltage (400–600 V-AC) diesel generator could be used to repower low voltage safety bus, which typically powers the battery chargers and other critical loads (e.g. fuel transfer pumps, high pressure reactor coolant injection pumps/charging pumps, battery room exhaust fans, control room lighting, boron injection pumps, etc.). Installing additional low voltage generators (medium capacity, i.e. about 100 kW) could provide electricity for the instrumentation, pools, lighting, ventilation system.

A low voltage generator can also be used to power the busses for an external (mobile) high pressure reactor coolant injection pump (See Section 4.6.1 and footnote 65).

During prolonged SBO event, they could also be used to provide power to critical reactor control instrumentation and monitoring loads, secondary control area (SCA) services, main control room (MCR) and control equipment room (CER) selected lightings, selected emergency core cooling isolation valves and emergency filtered air discharge system (EFADS), as well as to the equipment that are used to establish normal flow paths to ensure a heat sink for containment (e.g. containment air coolers) and in the management of water in the reactor building (e.g. emergency coolant injection recovery system) in CANDU plants.

In most nuclear power plants, additional mobile or permanent low voltage diesel generators were purchased by the nuclear power plant site. The decision of the nuclear power plants about the storage and availability varied to be on-site, off-site or both. In some nuclear power plants, mobile low voltage diesel generators were purchased and stored at the site (or at close proximity of it with transportation means), while in some nuclear power plants they were stored at a regional centre that is shared by multiple sites with established transportation agreements. In some cases, the mobile low voltage generators that are stored at the regional centres are 'defence in depth' to what the nuclear power plants have stored on-site.

Nearly all nuclear power plants installed a connection for low voltage AC power supply and most of the nuclear power plants also provided a backup connection point. Following is a list of common procured items and associated actions taken for low voltage generators:

- Purchased portable, low voltage diesel generators to repower low voltage safety bus to repower battery chargers and other critical loads, e.g. fuel transfer pumps, charging pumps, battery room exhaust fans, control room lighting, etc. (the low voltage generators can repower the busses for a positive displacement charging pump, not necessarily a high pressure centrifugal RCS injection pump, depending on a specific plant design);
- Identified/installed required primary connection for low voltage generator;
- Identified/Installed backup connection for low voltage generator (this backup connection was required in case there was no access to the primary connection due to the extreme external event);
- Procured low voltage AC cables and cable trailers which, in some cases, included 'quick connect' cables.

4.2.1.1. Drivers and reasons

The fundamental reason for such additional equipment was to improve DiD and improve redundancy and diversity against ELAP by accomplishing simple means to make significant changes in prevention, protection and mitigation.

In all Member States, installation of these types of equipment, to some degree, was required by the regulatory body, while the type, quantity, storage facilities, etc., decided by the owner/operator organizations at the plant, fleet or industry level. For example, power rating depended on the strategy which determines the loading, i.e. which equipment to be used in the timeline of the mitigation strategy.

4.2.1.2. Challenges and resolutions

The following challenges have been encountered when deciding and installing additional low voltage diesel generators:

- Qualification and classification of the mobile generators;
- Acceptance/permissibility of commercial grade, non-qualified, non-quality assured mobile generators by regulations;
- Permissibility of a non-safety source to power (i.e. connected to) a safety related busses (including 'source switch' scheme to enable power supply by the normal on-site and off-site power source or the ultimate power source, e.g. ultimate diesel generator);
- Inevitability of inside building connection which necessitated extra engineering to determine the path, curve radius, length of cables and associated costs;
- Protection of structures/buildings/facilities housing and storing these mobile generators;
- Maintenance and testing to ensure reliability, operability, availability and usability when needed;
- Unexpected additional plant modifications to avail connection points to mobile AC power supply;
- How to identify the scenario of beyond design basis, or the conditions of the scenario to determine size of generator, location to move them and connection points;

— Installation of seismic interface for power supply;
— Storage structure enhancement according to regulatory requirement with seismic resistance and waterproof design;
— In cases of systems with greater than 480 V-AC, there may be a need for additional (mobile) step-up transformer;
— For passive PWRs where the passive safety systems are designed to perform safety function without Class1E AC power supporting, integration of active DiD cooling measures power supply demand with the original passive safety systems for plant cooling.

The resolution of issues differs from one Member State to another, even further from one nuclear power plant to another within the same Member State:

— Non-Q class is selected based on the agreement and approval by the regulatory bodies;
— In order to maintain the design basis with respect to separation of safety and non-safety equipment, adequate isolation means, e.g. breakers were installed. Adequate isolation either existed with spare breakers or was installed (breakers) so that the plant design basis was maintained;
— Purchasing large quantities of non-safety mobile low voltage generators in order to ensure availability and operability, and therefore eliminate the need for safety qualification and classification of the mobile generators;
— Placing mobile generators in a maintenance programme to ensure reliability and usability. (For example, in one Member State, standardized maintenance templates to be utilized by all nuclear power plants and utilities were prepared as an industry initiative).

The implementation schedule was within one to five years of the Fukushima Daiichi accident, and the cost associated with adding a low voltage portable diesel generator varied with the NPP.

The unit cost variations for low voltage portable diesel generator were mainly due to loading selection criteria (i.e. the kW loading for the portable generators which, based on the survey response, varied from 100 kW to 800 kW owing to the mitigation strategies employed and the equipment needed and powered for the strategy). The unit cost, of course, also dependent on the universal availability of spare parts, whether the portable generator is an off-the-shelf item or custom made, etc.

Similarly, for the plant modifications to allow for connectivity of the low voltage portable generators, the costs are highly dependent on the strategies developed and whether the strategies rely on external to the building connections or connections internal to the building. The use of external versus internal connections drives the selection of items, such as the length of permanent plant conduits and cables, the length of portable cable runs, bend radius concerns, etc., all of which impact the cost.

In addition, NPP Member States did not track the costs in a similar manner which makes it difficult to compare costs. For example, some Member states, tracked the total electrical modification costs and did not differentiate between low voltage connection modifications and medium voltage connection modifications.

As a rough pricing, the cost for a low voltage generator ranges from approximately US $50 000 for a 100-kW portable DG to US $300 000 for an 800-kW portable DG.

When the costs for other materials, e.g. cables, cable trailers, connection modifications, breakers, storage facilities, etc. and labour, a typical added cost to modify the facility for a low voltage generator is calculated to be US $100 000 to US $200 000 per unit.

4.2.2. Adding medium voltage AC power supplies

A medium voltage (4–7 kV) generator could repower medium voltage bus, which typically powers equipment necessary to restore plant safety functions, primarily motor driven emergency feedwater pump(s), residual heat removal (RHR) pump, certain high pressure centrifugal RCS injection pumps, etc., for relatively longer times. For example, a 4 kV/3 MW generator could provide AC power to plant systems and may ensure longer period, e.g. up to 72 hours, without necessitating the reestablishment of off-site power during an ELAP. Therefore, 2 MW medium voltage generators were commonly selected by the plants, while the 3 MW medium voltage generators were less common or specially customized to extend the supply time.

In some nuclear power plants, it is also considered to utilize medium voltage diesel generators for site risk reduction measures as they could be credited for normal operational activities to reduce core damage frequency when their availability, functionality and operability are assured. For example, one plant has found measurable success in crediting portable/mobile equipment for the purpose of configuration work management where the mobile SFP makeup pumps and mobile medium voltage AC power suppliers are used to maintain safety functions (e.g. decay heat removal capability and power availability) during a refuelling outage (see Section 8).

For many nuclear power plants, additional mobile or permanent medium voltage diesel generators were purchased while in some nuclear power plants, permanent and stationary diesel generators, such as ultimate emergency diesel generator, were purchased and installed per decision by the nuclear power plant owner/operator organizations.

In some NPPs, mobile medium voltage diesel generators were purchased and stored at the site (or at close proximity of it with transportation means). In some Member States, nuclear power plants participated in making medium voltage generators available at/by a national response centre(s) for additional back up with ability of providing them, when/if needed, to multiple sites with established transportation agreements. This decision was based on whether the medium voltage generator would or would not be required for the first 24 hours of the event. If they are not required for the first 24 hours, those NPPs choose not to purchase and store medium voltage generators at site. Rather, they established plans to have them supplied from the national response centres, with the arrangement to ensure arrival to the site no later than 24 hours after the declaration of ELAP. For example, the US NPPs needing medium voltage generator earlier than 24 hours, purchased medium voltage generators and store them at the site as FLEX Phase 2 equipment.

Some NPPs chose to install a connection for medium voltage AC power supply, and furthermore, also provided a backup connection point in case of the primary connection point becomes inaccessible during an event.

Common procured items and associated actions taken for medium voltage generators included:

— Portable medium voltage diesel generators to repower safety medium voltage bus, which repowers equipment necessary to restore plant safety functions;

— Installation/identification of primary, backup or alternate connections for medium voltage generator;
— Installation/identification of connection for medium voltage generator;
— Procured medium voltage AC cables and cable trailers.

4.2.2.1. Drivers and reasons

The fundamental reason for such additional equipment was to improve DiD and improve redundancy against the extended loss of AC power in restoration and preservation of key safety functions.

In most Member States, installation of these types of equipment was required by the regulatory body, while the type, quantity, storage facilities, etc., decided by the owner/operator organizations at the plant or fleet level.

4.2.2.2. Challenges and resolutions

The following challenges have been encountered when deciding and installing additional medium voltage diesel generators:

— Qualification and classification of the generators (not only for use and storage but also transportation, e.g. the proven resistance of mobile generators to shaking during the transport from storage to connection location due to road conditions);
— Maintenance and testing to ensure reliability, operability, availability and usability when needed;
— Permissibility of a non-safety source to power a safety related bus;
— Additional permanent plant modifications to avail connection points to mobile AC power supply, for example incompatibility of connections with the generators purchased by regional response centres or uneasiness of installation location and procedure of new connections;
— Price of medium voltage diesel generators with capacity to provide reliable power for long term (e.g. up to 72 hours);
— Extensive fuel transport and/or storage requirements for long term operation;
— Necessity of obtaining special permits to use some public roads to transport medium voltage diesel generators owing to their very heavy weight exceeding the allowed tonnage of the roads;
— Necessity of large structures/buildings/facilities to be constructed to house or store diesel generators;
  - Protection and qualifications of structures/buildings/facilities housing and storing these mobile generators;
  - Constructability of structures, e.g. availability of space at the site to place structures or mode of operation of other units at multi-unit sites to allow construction activities, such as potential interference or obstacles with overhead power lines when operating heavy construction equipment;
  - Longer and intermittent schedule to construct.

The resolution of issues differs from one Member State to another, even further from one nuclear power plant to another within the same Member State. Some of the solutions practiced by nuclear power plants for the challenges listed above include:

- In order to maintain the design basis with respect to separation of safety and non-safety equipment, adequate isolation means, e.g. breakers, were installed.
- Procurement and storage of adapters for incompatible connections with the generators purchased by regional response centre equipment.
- Purchasing large quantities of non-safety mobile medium voltage generators in order to ensure availability and operability, and therefore eliminate the need for safety qualification and classification of the mobile generators.
- Storing the equipment in away locations that would not require the same qualification due to less extreme conditions than what the site would be anticipated to experience.
- Placing mobile generators in the maintenance programme to ensure reliability and usability.
- Installation of permanent safety class diesel generators within a seismically qualified building and justifying cost as additional safety margin and risk reduction. In the Member States which implemented this modification, it was decided to install permanent and stationary safety class diesel generators, typically within a seismically qualified building, in their mid-term (i.e. completion before 2018-2019) or long term (i.e. completion up to 2025) action plans (e.g. new air cooled EDG in Japan, ultimate emergency diesel generators in France, new safety class DG in Slovenia, etc.) Until then, those NPPs chose to purchase and medium capacity diesel generators as temporary mobile equipment.

The acquisition/installation of medium voltage diesel generator was scheduled as medium term (i.e. within one to five years of the Fukushima Daiichi accident. However, the schedule is longer (5–10 years) for stationary/permanent medium voltage diesel generator (such as ultimate diesel generator in some Member States).

The surveyed NPPs stated that a medium voltage mobile 2 MW diesel generator costs approximately US $1 million. The costs for other materials, e.g. cables, breakers, storage facilities etc., and labour, beaded another US $1–1.5 million to the cost of modification.

Altogether, a typical cost of a set of low and medium voltage power source modifications (i.e. generators, connectors, adapters, connectors, labour, etc.) was approximately US $3–4 million per train).

4.2.3. Provision of other A/C power supplies

In addition to low and medium voltage diesel generators, the several nuclear power plant owner/operator organizations purchased/installed specific AC power sources. These included (in a graded approach):

- Procuring various portable small generators, e.g. non-quality class, commercial grade diesel generators for redundancy and as power supply to mobile small pumps, such as submersible pumps, stored on-site;
- Purchasing additional mobile emergency diesel generator (e.g. 120 VAC) for powering the small cooling system pumps (on-site and off-site storage);

— Purchasing additional mobile emergency diesel generator for 15 days of redundant AC supply (off-site storage);
— Installation of gas turbine generators;
— Installation permanent and/or stationary diesel generators some of which includes a housing building that is resistant to the external events;
— Procuring various portable small generators to power portable ventilation equipment;
— Procuring various portable small generators to power mobile boration equipment.

Just for illustration purposes, a typical cost for the non-Q class portable generators (e.g. 120 VAC) ranged from a few thousand US dollars (for example, a 4.3 kW mobile generator is approximately US $2000) to over US $10 000 (for example, a 20.4 kW mobile generator is approximately US $12 000).

In some nuclear power plants, additional mobile gas turbine generators were purchased as a decision by the owner/operator organizations at the plant or fleet level. The fundamental reason for such additional equipment was to improve DiD and improve redundancy against the ELAP. In most Member States, installations of these gas turbine generators were completed within a year of the Fukushima Daiichi accident.

The following challenges have been encountered when deciding and installing, for example, additional gas turbine generators:

— Qualification and classification of the equipment and associated structures housing them;
— Storage, maintenance and testing, in some cases finding qualified personnel to perform those;
— Ensuring reliability, operability, availability and usability when needed;
— Additional plant modifications to avail connection points to gas turbine generators, including seismically qualified connection.

As aforementioned, some nuclear power plant operating organizations decided to install permanent and/or stationary diesel generators some of which includes a housing building that is resistant to the external events (e.g. seismically qualified) with its own quality class power distribution and vent system. Price for such stationary safety class diesel generator cost ranges from US $1 million to US $10 million; however, when the costs for housing structures included, the total cost may exceed US $40 million per nuclear power plant unit. This high cost of an additional permanent safety class diesel generator within a seismically qualified building was justified by the operating organizations who chose to take such action by the provision of substantial safety margin and risk reduction.

4.2.4. Provision of support structures for enhanced A/C power sources

In order store and prevent procured mobile AC power sources from external events and other degradation mechanisms, all nuclear power plant owner/operator organizations had to build on-site (or near site) structures/buildings/facilities to support the approved coping strategies where the support equipment needed at the beginning of the ELAP event. This was accomplished by some units in few Member States and by modifying existing on-site structures/buildings, instead of building new ones.

In addition, in some Member States regional or central response centres that also consisted of one or more protected storage structures/buildings/facilities and supporting equipment for dispatch transportation and site layout were established. These centres are used to store equipment and personnel that are typically required after the first 24 hours of the event. In some Member States, these structures were protected, while in some, they were unprotected due to the selection of the site and associated hazards.

Here it should also be included that some nuclear power plants who choose to install stationary AC power supplies, also built structures/buildings/facilities to house them.

These support structures are further discussed in detail in Section 4.12.6.

4.3. INSTALLATION OR MODIFICATION OF DC POWER SUPPLY SYSTEMS

The Fukushima Daiichi accident shown the vitality of DC power supply for plant safety, as it is needed for instrumentation and control and also provides AC power from inverters to a small number of essential components including essential reactor control/monitoring, lighting, and motorized equipment, e.g. MOVs, loads needed for prevention and protection actions. As noted in Ref. [1]:

> "Nuclear power plants are equipped with on-site DC and additional backup AC power sources (i.e. gas turbine generators or diesel engines) to withstand a station blackout for a limited period of time, varying between 4 hours and 72 hours. The determination of the SBO coping period is based mainly on the time that it would take to restore AC power sources to the nuclear power plant and the capacity of the available measures. During that time, equipment such as DC batteries, DC/AC inverters and other available secondary backup AC sources (e.g. gas turbines or diesel generators) is used." [1].

Also coming from one of the lessons learned in Fukushima Daiichi nuclear power plant, where the DC power supply design was robust against single failures, e.g. by having multiple DC busbars per unit, the physical layout of the associated switchgear and the batteries made the system vulnerable to CCF by flooding, which was discussed in Section 4.1.2. This highlighted that the actions may be needed to enhance the design and layout of electrical equipment enough to avoid the possibility of CCFs from credible BDBE/BDBAs and associated measures to be taken in case of identified vulnerabilities accordingly.

Even though the batteries inherited in the plant configuration were available, a prolonged SBO beyond the anticipated SBO coping period could result in depletion of these batteries when their mission time is expired (e.g. in the case of Fukushima Daiichi nuclear power plant's Unit 3). For increasing the robustness of DC power supply systems, three particular measures were found beneficial for situations in which only DC power backed by batteries is available but needed for a prolonged time:

— Having additional redundant or independent stationary high capacity batteries available;
— Establishing capability to replace or recharge batteries with mobile equipment;
— Providing physical and administrative measures for extending battery mission time.

Therefore, the nuclear power plant owner/operator organizations and regulatory bodies considered these measures in response to the lessons learned and took actions to ensure the availability of DC power supplies for the time span when they are needed (i.e. time between the total loss of AC power and its restoration either by designed AC power systems or the mobile supplies discussed in Section 4.2). These actions included some or all of these three measures depending on the nuclear power plants coping strategy during a prolonged SBO resulting from beyond design basis external event (BDBEE) with the consideration of capabilities and capacities of equipment and human resources, as well as potential site conditions.

4.3.1. Adding battery capacity

To bridge the gap of power loss between total loss of AC power and portable AC generator deployment, the batteries need to supply power to essential reactor control/monitoring, lighting and equipment loads. Based on the existing battery capacity and anticipated time required to restore AC power, the nuclear power plants evaluated whether there is a need to add high capacity with longer mission time. Some nuclear power plant units which responded to the survey indicated that they took actions to increase the available battery capacity in various manners. These included:

- Installing on-site stationary DC batteries;
- Installing backup redundant on-site stationary DC batteries;
- Purchasing additional portable DC batteries to be stored on-site or near site;
- Purchasing additional portable DC batteries to be stored in and transported from an off-site facility;
- Re-evaluation of battery room heat-up and impact on the battery life;
- Re-evaluation of battery room hydrogen concentration;
- Purchase and installation of stationary or portable ventilation equipment (fans, ducts, etc.) for controlling battery room temperature and hydrogen concentration. (see Section 4.12.4);
- Performing a DC load shed to extend battery life (discussed in Section 4.3.3);
- Establishing procedural controls for battery room temperature and hydrogen.

4.3.1.1. Drivers and reason

Provision of additional battery capacity was a regulatory requirement unless there is a proven strategy to being able to manage the essential functions with the existing battery capacity (i.e. demonstration of capacity with load shedding) or to recharge batteries.

4.3.1.2. Challenges and resolutions

The following challenges have been reported by the nuclear power plants responding to the survey:

- How to identify the scenario of beyond design basis, or the conditions of the scenario to determine size and capacity of batteries as the determination of battery size depends on the time needed to deploy AC generators. This time is dependent on the accident scenario that would include consideration of capabilities and capacities of equipment and human resources, as well as potential site conditions. Therefore, the battery capacity determination is accompanied with best estimate accident sequence and timing assessments, as well as administrative controls for extending the battery life (see Section 4.3.3).

— Qualification and classification of the batteries, particularly the portable batteries.
— Maintenance and long term storage to ensure reliability, operability, availability and usability when needed.
— Protection of structures/buildings/facilities housing and storing these mobile batteries.
— Additional batteries may result in increased battery room temperature requiring new or revised battery room heat-up evaluation and subsequent portable ventilation system modifications or provisions (see Sections 4.1.3 and 4.12.4).
— Current battery technology is well understood and in the industry standards. Although newer and better technology (e.g. Li) exists they cannot be easily used owing to significant investment to qualify them.

NPPs adding more battery capacity completed the action within from one year up to five years of the Fukushima Daiichi accident.

4.3.2. Recharging batteries

Per the design of electrical system in nuclear power plants, the batteries are charged by low voltage AC low voltage safety bus [96] for extended power supply to instrumentation, some control and instrumentation protection (noting that the extended power supply can be accomplished without batteries but it would be much more difficult). In case of prolonged SBO, alternative low voltage (400–600 VAC) AC power supply could be used to repower low voltage safety bus, which typically powers the battery chargers [97]. Therefore, the nuclear power plants took following physical and administrative actions to provide AC power to the station battery chargers:

— Purchasing portable, low voltage (400–600 VAC) diesel generator to repower safety low voltage bus to repower the battery chargers (see Section 4.2.1);
— Identification/installation connection for low voltage generator to the battery chargers;
— Preparation of procedures for recharging batteries, e.g. establishing procedural controls for battery room temperature and hydrogen monitoring and control;
— Purchase and installation of stationary or portable ventilation equipment (fans, ducts, etc.) for controlling battery room temperature and hydrogen concentration. (see Section 4.12.4);
— Adding administrative controls (i.e. procedures) to start battery room exhaust fans prior to resuming operation of battery recharges to ensure removal of hydrogen.

4.3.2.1. Drivers and reasons

Provision of available battery capacity by charging for ELAP was a regulatory requirement.

4.3.2.2. Challenges

In addition to the challenges and resolutions discussed in Section 4.2.1.2 for low voltage AC power supplies, determining administrative measures for recharging batteries — with respect to the anticipated scenario of BDBE/BDBA or the conditions of the scenario to establish procedural steps — was a particular challenge. Other reported challenges included:

— Control of hydrogen generated during charging (solution: engineered systems to simply opening doors);
— Issues with charging from non-safety related source;
— Recharging actions required much more difficult steps than adding batteries; however, it can be done with precise timing and opportunity in the demonstrable strategies.

This action was scheduled to be completed within 1-5 years of the Fukushima Daiichi accident.

Typically, the total cost of improving battery chargeability during an ELAP is in the range of US $750 000 to US $1 million per nuclear power plant unit in addition to the purchase of low voltage diesel generators (see Section 4.2.1.4) and the modifications for installation of connections to the battery chargers. This cost range depends on the strategies implemented and whether the connections are internal or external to the building. One survey responder noted: *“Once the low voltage (400–600 VAC) diesel generator modification was completed; the additional action was to set the administrative controls”.* The cost of improving battery rechargeability is included in the cost of the modifications associated with low voltage AC supply (see section 4.2.1.4).

4.3.3. Extending battery life cycle

As mentioned earlier, most nuclear power plants are equipped with stationary batteries to withstand a station blackout for a limited period of time, varying between 4 and 16 hours, although some newer designs may provide DC power longer time, up to 72 hours [97]. Whether additional redundant or independent stationary high capacity batteries are made available (Section 4.3.1) or capabilities to replace or recharge batteries with mobile equipment are provided or not, as a DiD measure, extending the available battery lifecycle is beneficial to increase autonomy of DC power sources for a longer duration during a prolonged SBO event.

Therefore, most nuclear power plants re-evaluated their battery duration time considering the essential loads to be powered by DC power sources after a load shed and the conditions under which the batteries will have to perform. These evaluations lead to several physical and administrative measures for extending battery mission time in case that battery capacity is limited or the SBO situation is prolonged to support the anticipated coping strategy. The actual battery mission time needed for the coping strategy at a site is dependent upon the individual station's specific coping strategies. For example, the followings are a set of responses received in the survey:

— Performed assessment of battery duration under the prolonged SBO conditions. Engineering calculations are utilized to provide a reasonable engineering basis to demonstrate that the batteries after a load shed will have sufficient coping strategy mission time to maintain power to key instruments until deployment of on-site portable coping strategy equipment.
— Considered feasibility of power supply without load shed and then with load shed (of non-essential loads including non-essential Class 1E loads from the batteries.
— Performed tests to support analysis that batteries would provide power during load shedding.

In some cases, the regulatory bodies performed testing of typical plant batteries to prove that battery availability beyond 24 hours can be achieved [98].

4.3.3.1. Drivers and reasons

The actions for determining the available battery capacity and taking physical and administrative measures to extend the battery life, particularly extending the availability of power for key I&C needed in accident management actions following a loss of all AC power, was a regulatory requirement in all Member States. However, the minimum required mission time varied, for example, from eight to 16 hours without load shed and 16 hours to 72 hours with load shed, depending on the consideration of existing design basis and approved coping strategies.

4.3.3.2. Challenges and resolutions

The following challenges were faced in confirmation of the battery life extension measures and how they resolved:

- Determining a reliable method for evaluating battery duty cycle duration after load shed that is applicable to BDBE. Eventually, agreements between the regulatory bodies and the NPP operating organizations were reached for an acceptable approach for evaluating battery duration after load shed by previously established methods for DBA, such as the method described in Institute of Electrical and Electronics Engineers (IEEE) standard [99]. It was noted in the survey response that some NPPs, such as those in the USA, argued that the IEEE standard may not be applicable to BDBEs and provided alternative methods based on the efforts by the industry (nuclear power plants owner/operating organizations, and technical support organizations) determined [100]. The application of these methods were agreed by the regulatory body [101]; however, this imposed a requirement for the NPP operating organizations to demonstrate that the battery manufacturer's discharge curves support the duty cycle duration assumed in the plant specific evaluation.
- Regulator inquiry on how the batteries would respond in a load shed condition which required tests to be performed to support analysis that batteries would provide power during load shedding.
- In some cases, the battery life was limiting and required very short period of time for completing time sensitive actions associated with the coping strategy. In such cases, either physical modifications, such as light-emitting diode (LED) lighting to reduce loads or installing permanent AC source to minimize the operator actions that were needed.
- The guaranteed capability of Class I batteries to support key or essential electrical equipment is found to be too short, in some cases less than one hour, although it is recognized that some services may last much longer. This duration is short compared to other essential supply capabilities and gives little time to restore AC power when one considers the site conditions during an accident. Under such conditions, the deployment of portable generators may take too long, in some cases no less than two hours. This low battery margin requires measures for prompt debris removal and portable transport and connection as well as the associated evaluations and exercises to ensure that these actions can be completed on time.

— The increased temperature in the battery room will shorten the long term battery life; however, it will also increase the short term efficiency. These competing effects are due to the impact of temperature on the battery's chemical reaction rate, as well as type of the battery that determines the efficiency and resistance. Typically, the lower temperatures reduce the efficiency of charging and discharging affecting the battery run and discharge times, while long time operation at high temperatures shortens the battery life. Therefore, these competing effects of battery room temperature was needed to be considered and analysed in 'room heat-up calculations' had to be performed to confirm effectiveness and durability during extended battery mission time.

The analysis costs for evaluating battery duration/duty cycle are in the range of US $50 000 to US $150 000. This does not include the testing of the batteries. Similar costs should be added for the room temperature assessments which varies depending on the calculation method used, i.e. from simple hand calculations to very detailed computer modelling.

4.4. INSTALLATION OR MODIFICATION OF ON-SITE ELECTRICAL POWER DISTRIBUTION SYSTEMS

The electrical power distribution system being the backbone linking sources of power supply to components and controls which are essential for the safe handling of the plant implies a high reliability requirement. The design of the electrical power distribution system was robust against single failures. However, it was learned from the Fukushima Daiichi accident that the physical layout of the associated switchgear could make the systems susceptible to CCF by flooding. Moreover, flooding of switchgear room had been identified as one of the events that contribute to CDF during the individual plant examination of external events (IPEEE) for severe accident vulnerabilities performed in the U.S. in 1990s, long before the Fukushima Daiichi accident (for example, since the pilot programme on internal flooding assessment performed at the Unit 1 of Surry NPP in 1993 [102], a large risk of core damage there by the flooding of switchgear rooms disabling their electricity distribution capabilities has been long recognized).

Susceptibility of the switchgear to flooding, once again, was identified during the external flood assessment. However, it is very substantial modification to relocate switchgear (or other on-site electrical power distribution systems). Therefore, using flooding PRA results, the actions against the flooding of such systems focused on the protection of existing SSCs against the external flood, including the measures of utilization of mitigation equipment and vehicles. For example, in the USA, the vulnerable NPPs decided to include special mitigation measures in their FLEX strategies [103−105]. Not surprisingly, one of those NPPs was Surry NPP which revaluated LIP flood hazard and determined that it results in flooding of the emergency switch gear room and the battery rooms rendering both AC and DC emergency electrical power distribution system unavailable.

Based on the provision of reduced risk by power connectivity between several units in a multi-unit site, the other enhancements of on-site electric power distribution systems included:

— Installing permanent interconnections between the units;
— Establishing strategies and procedures to set up temporary or manual interconnections.

4.5. INSTALLATION OR MODIFICATION OF OFF-SITE POWER SUPPLY

A loss of off-site power is anticipated and taken into account in the plant design — as is common practice — by the provision of EDGs to provide backup power supply. Still, it is desirable to keep the probability for a LOOP event reasonably low. In the design of the plant, account needs to be taken of power grid–plant interactions, including the independence of and number of power supply lines to the plant, in relation to the necessary reliability of the power supply to plant systems important to safety.

This was accomplished in the design of the Fukushima Daiichi nuclear power plant featuring redundant off-site power supply lines; however, with a low degree of independence. The consequences of low degree of independence of off-site power sources reinforced view that the off-site power supply of nuclear power plants can be strengthened not only by sufficient redundancy (e.g. multiple lines), but also by sufficient diversity —for example by being able to connect each unit to different lines which use different pylons or underground cables, leading to different substations.

The overall vulnerability against external LOOP events that may lead to ELAP, can be reduced by enhancement of the electrical power grid system and its interface with the NPP, for example by strengthening the seismic design of the transmission lines and the associated switching devices, by arranging priority restoration of off-site power, additional diverse and independent off-site power sources and connections, such as black start of collocated or nearby gas or hydro plants.

Based on the survey results, there have not been significant modification in this area, other than the report from Krsko NPP which has an extra 400 kV transmission line for off-site power supply, which still was installed before the implementation of post-Fukushima actions. Reports to the IAEA, however, included plans for:

- Installation of dedicated off-site power line and generation source;
- In the case of multiple units in the site, connecting more than three transmission lines;
- Improvement of reliability of off-site power connection.

4.6. INSTALLATION OR MODIFICATION OF CORE COOLING SYSTEMS

A significant amount of effort has been expended to provide a variety of approaches for enhancing core cooling capabilities during a BDBE/BDBA. The approaches vary significantly based on reactor design (PWR or BWR), reactor state (operating or shut down), and regulatory requirements. This section describes modifications for direct RCS injection, decay heat removal via secondary systems (e.g. through SGs), criticality control (necessary to account for positive reactivity added during RCS cooldown or Xenon decay) and RCS leakage control via the primary pump seals.

To adequately cool the reactor core, two fundamental physical requirements exist: (1) a heat sink is necessary to accept the heat transferred from the reactor core to the reactor coolant in the RCS; and (2) sufficient RCS coolant is necessary to transport heat from the reactor core to the heat sink via natural circulation in the RCS. The RCS coolant should be replenished with borated coolant in order to maintain the reactor in a subcritical condition.

4.6.1. Adding/modifying core cooling injection and coolant inventory preservation and makeup capabilities

Primary system injection is necessary for several aspects of accident management with an objective of minimizing RCS inventory loss and maximize and sustain makeup water (also see Section 4.6.4, for maintaining subcriticality objective):

- For PWRs and pressurized heavy water reactors (PHWRs) starting from an operating condition, there will be slow reactor coolant leakage through pump seals and other sources. Also, once the secondary heat sink is depleted, the primary circuit will heat up and additional inventory will be lost through pressure relief valves. Establishing a means of injecting from a high head, low flow pump can mitigate the inventory loss and provide other criticality management functions as well (see Section 4.6.4). This is accomplished through the establishment of connection points (in some cases a primary and secondary connection), and then a high pressure, low flow pump.
- For PWRs and PHWRs starting from a shutdown condition with the primary system open, the only way to remove decay heat is through injection into the primary system via a low head, high flow pump. This is accomplished through the establishment of connection points and then a high flow, low pressure pump.
- For BWRs, injection of water into the core, where it then boils and the steam is removed to the wetwell or isolation condenser, is the primary means of removing decay heat. Typically, a steam driven reactor core isolation cooling (RCIC) system provides this function, but the pump cannot run indefinitely, so a backup means of injecting using a portable pump has been implemented. This is accomplished through the establishment of connection points and then a high flow, low pressure pump.

Establishing a backup means for primary injection was a regulatory requirement. Most of the NPPs implemented an approach with portable means and established strategies and procedural steps, while some have added additional injection capabilities with hardened, permanent, self-powered equipment. The means for core cooling injection and coolant inventory preservation that were deployed in the NPPs based on the survey results included:

- New low pressure primary injection pump (borated water).
- New stationary injection pump.
- Modification of existing emergency cooling pump for cooling.
- Procure portable pumps to inject into the RCS for shutdown modes if the plant is in an outage and the SGs are not available (different pumps for different shutdown modes, e.g. reactor pressure vessel head (RPVH) is on or off, etc.).
- Install primary connection into the RCS for injection during shutdown modes. Pump typically ought to be deployed and ready to inject during shutdown modes. Connection points are needed to allow a portable pump to inject borated water directly into the RCS for once through core cooling if the plant is in an outage and the SGs are not available.
- Install secondary connection into the RCS for injection during shutdown mode in case of primary connection being inaccessible (particularly in the US plants where the rules that were set by the industry [20], and endorsed by the regulatory body, required a primary and alternate connection point).
- Install additional AC power independent high pressure makeup system, e.g. high pressure alternative cooling system (HPAC) or turbine driven auxiliary feed pump (TDAFWP) using steam produced by the decay heat removal.

— Portable suction booster lift pumps to provide 26 feet of suction lift to low pressure medium flow pumps or low pressure/high flow pumps (on-site and/or off-site storage).
— Procure portable pumps to inject into the RCS for inventory makeup and boron injection.
— Install primary connection for RCS makeup control.
— Install backup connection for RCS makeup control in case of primary connection being inaccessible.
— Establish a connection to allow for suction from a protected borated water source.
— Portable water treatment equipment for RCS makeup water (see also Section 4.11.3).
— Portable water storage bladders for treated water for RCS injection.
— Purchase of large capacity pumping trucks.
— Purchase of fire engines with injection capabilities.
— Procurement of mobile pumpers (i.e. fire trucks) as mitigation equipment to deliver water to SG, SFP and potentially other systems, as required. Such vehicles were selected as an effective option owing to value/impact consideration. For example, one survey responder noted:

> *"The station has procured mobile pumpers (fire trucks) as the means to provide emergency water to key systems and equipment in the event of a station blackout. Procurement is complete and it consists of 5 fire trucks two of which would be dedicated to A block, two to B Block with one spare. A fuel truck has also been procured which is capable of delivering fuel to the pumpers as well as the portable generators. Dry hydrants are being installed to ensure a positive drafting source. The dry hydrants will extend below the water line up to an elbow that is above grade. Emergency responders will then make a horizontal connection between the dry hydrants and the pumper. This strategy minimizes the chance for a bad joint resulting in problems obtaining a vacuum and water supply for the truck. A secondary water source will be from each of the stations "boat docks" which are located outside of the two security fences. In this configuration, the pumpers are to be placed on the docks to draft water for the stations."*

The NPP further explained the reason for selection of fire trucks as:

> *"The selection of fire trucks was made for several reasons. First, the ownership and accountability for fire protection equipment are clear, residing with Emergency Protective Services (EPS), who are already highly familiar with their operation. Fire trucks are highly mobile and can be deployed quickly where required. They also have very high flow capacity and are capable of providing water at a sufficiently high rate to supply multiple systems simultaneously. There are five trucks in total: two trucks are required for each station and the fifth is a backup for the other four. Two trucks are reserved for one station because one truck is responsible for supplying water to both IFBs, while the other truck will supply water to the boilers. The*

availability of the backup truck also allows Bruce Power to respond to other events occurring on site whenever the other four trucks are in use. The pumps have been overdesigned to compensate for the head pressures required to draw water from the lake and push to the upper elevations of the stations. The primary water supply will be from the Condenser Cooling Water (CCW) outfall."

- The installation of permanent fire hose 'quick connect' couplings that will enable emergency water to be added to the heat transport system of each unit. The design and fabrication of a temporary fire hose connection which will enable emergency water to be added to the heat transport system of each unit using an alternate location.
- Gas (nitrogen) supplies for re-pressurisation of primary circuit and establishing cooling by natural circulation (AGR plants), as stated in Ref. [106]:

  "No feasible resilience enhancements to on-site stocks of carbon dioxide could be identified and accordingly a different approach was adopted. It was recognised that to repressurize a reactor, gas pressure support equipment (including gas stocks) may be required; particularly during outage conditions. Off-site DBUE [Deployable Back-Up Equipment] therefore includes sufficient nitrogen stocks to suitably repressurize a reactor and thus promote cooling by natural circulation."

- Primary heat transport system (PHTS) connection/moderator connection (CANDU plants).
- Adding low leakage seals or seal injection systems to minimize/limit/prevent RCS coolant leakage (PWR plants), as well as the considerations of additional benefits, such as in reduction of risk in fire, flood induced SBO scenarios.
- Purchasing and installing air compressors or compressed air tanks to open/close/cycle air operated valves (AOVs) identified that could be needed used for decay heat removal and in reactor auxiliary systems (e.g. letdown or reactor water cleanup system valves) in successful implementation of the strategies with compressed air is used.
- Performing 'best estimate' and/or realistic analyses for determining the timing for RCS water injection in potential scenarios.
- Some Member States, the nuclear power plants had to perform seismic fragility studies to demonstrate seismic ruggedness of alternate water sources (e.g. permanent tanks);

The additions and/or modifications to systems utilized for water injection to RCS for core cooling have been implemented in most countries as a medium term action, i.e. within 1−5 years of the accident. For those NPPs who planned to build hardened and permanent SSCs, the project schedule was typically long term with completion dates were between 8−10 years.

For the NPPs who set coping/mitigation strategies based on portable and temporary equipment, such as those in the US that utilized the FLEX strategies (particularly, FLEX Phase 2 strategy), portable equipment had to be procured and delivered to each NPP prior to the final compliance date agreed with the regulatory body. It should be noted that for the FLEX Phase 3 equipment at the national SAFER response centres (NSRCs) in the USA, the equipment had to be procured and delivered by the fall of 2014 to declare the NSRCs operational.

4.6.1.1. Challenges

The challenges that were encountered during the plant modifications for primary injection means included:

— How to identify the scenario of beyond design basis, or the conditions of the scenario which will determine identification of connection points and portable equipment layout that are easily and most likely accessible and protected from external events for a substantial variety of scenarios.
— Identifying injection paths and connections that:
  - Are not affected by the extreme external events, for example the tie-ins being restricted to only seismically qualified paths or not;
  - Minimize use of AC/DC power and affected, e.g. flow diversion due to CCF (or failure mode), equipment, such as MOVs, on these path(s);
  - Are not impacted by control logic failures of active components.
— Finding locations for primary and backup injection points downstream of active components such that control logic failures do not impact water injection.
— Longer time needed for the installation of connections points and tie-ins because the work can be performed only during an outage.
— Protected source of borated water that would be available following the extreme external event. This required measure to enhance protection of existing borated water source or building an additional protected one.
— Ensure a positive drafting source that necessitated a connection, particularly for pumping vehicles such as fire engines, to allow for suction from a protected water source and portable suction booster lift pumps to provide 26 feet of suction lift to low pressure/medium flow pumps or low pressure/high flow pumps were purchased with on-site and/or off-site storage.
— Storage of mobile equipment that required protected storage structures/buildings/facilities which in some nuclear power plants required high level seismically, e.g. SL-2, as well as waterproofing them. Also, in one Member State, as noted in the survey response, the elevation was a requirement to build storage facilities (as well as the locations where they are deployed during mitigation actions) as 5 m above the design basis flood (DBF).
— Storage of mobile equipment in cold weathers which necessitated temperature-controlled storage buildings or the provision of freeze protection for pumps and hoses (as an example, the freeze protection in one NPP is provided by maintaining flow in the pump/hose through the use of a minimum flow line controlled at the distribution manifold by the operator).
— Necessity to procure different pumps since different plant configurations require different pumps, particularly in shutdown configurations:
  - Shutdown mode configurations require different pumps when the RPVH is on when the RPVH is off;

- It is possible to use the same pump for, say, RCS or SG water make-up under certain shutdown configurations, while that pump can only be utilized for one of those purposes under a different shutdown configuration.

— Some nuclear power plants assessed that the time for time-sensitive manual actions during some shutdown conditions (e.g. RCS reduced inventory) may not have adequate margin. Therefore, in such cases, operating organization decided to pre-deploy and keep portable pumps during the duration of shutdown (for example, during the refuelling/maintenance outages when the RCS inventory is reduced, or even during the entire outage) to improve time margin and reduce outage risk.

— Determining the timing for RCS water injection in potential scenarios.

— Extreme low temperatures could result in ice formation on the body of water (e.g. lake, reservoir, etc.) used as a water source, necessitating capabilities to break ice for establishing suction for cooling system.

— For passive cooling design nuclear power plants, integration of active DiD cooling measures with the original passive safety systems for plant cooling;

— Qualification and classification of the mobile pumps, hoses, connections (not only for use and storage but also transportation, e.g. the proven resistance of mobile pumps to shaking during the transport from storage to connection location due to road conditions) which was handled by purchasing extra quantities with both on-site and off-site storage, for example purchased backup equipment one for each unit on site, and then one more as a backup at off-site storage.

— If UHS water is to be used as a makeup source, some additional measures may need to be taken to assure that the equipment can utilize the water.

— Bladders can fail and cause consequential events (e.g. flooding). A solution for this lesson learned was to replace the bladders with plastic or metal tanks.

4.6.1.2. Lessons learned from decision, implementation and strategies for long term sustainability

The challenges listed above and their resolutions lead to several lessons learned:

— Ensuring the permanent connection points meet all the codes and standards applied to the plant systems to which they are connected and the temporary makeup components such as hoses and other components not permanently connected to the system meet lesser commercial grade requirements;

— Performing a plant habitability assessment to demonstrate location of tie-in points would be accessible owing to post-accident conditions;

— Installing tie-ins as close as possible to the entrance to the plants so that personnel do not have to transverse through the plant;

— Installing a seismic interface for cooling water makeup;

— Installing a primary tie-in on an existing seismically qualified path and an alternate which does not need to be on a seismically qualified path.

The cost for a typical high pressure injection pump to be used in RCS injection, such as a pump with 1600 pounds per square inch, psi, (app. 11 MPa) pressure and 60 gallons per minute (app. 14 m^3/h) flow, is US $17 000 to US $22 000, while the cost of installing low leakage RCS pump seals would vary depending on selection of the vendor (e.g. original equipment manufacturer (OEM) or new vendor) or seal type, number of reactor coolant pump (RCP) per unit, change in seal control bleed off (e.g. criteria, procedural steps, etc.) and can range from US $50 000 to US $5 million per pump (i.e. US $15 million for a nuclear power plant unit with three loops).

4.6.2. Adding steam generator injection capabilities for PWRs and PHWRs

In PWR and PHWR technologies, the SG removes heat from the primary system during normal operation and shutdown (decay heat) in support of cooling the core. The water inventory in the SG is essential to remove heat and water injection to the SG is necessary for several aspects of accident management with an objective of maximizing and sustaining makeup water for maintaining core cooling:

- Feedwater the steam discharge processes work together for heat removal from the reactor. By almost all PWR/PHWR designs, the available pumps to inject water in the plant design are typically diverse (AC power driven and steam driven) auxiliary (emergency) feedwater pumps.
- In case of ELAP, the steam driven pump will inject, as long as the steam to drive the pump and make up water last. The latter one necessitates a sustained source of water. If the steam to drive steam-driven pumps is not adequate, that would necessitate other means to inject water to the SGs.

Similar to the core cooling injection, establishing a backup means for secondary system injection was a regulatory requirement for PWRs and PHWRs. Most of the NPPs implemented an approach with portable means and established strategies and procedural steps, while some have added additional injection capabilities with hardened, permanent, self-powered equipment. The means for SG water injection and SG inventory preservation that were deployed in the NPPs based on the survey results included:

- Identify and install permanent water supply for RCS heat removal by the secondary system (groundwater wells, ponds, etc.).
- Add a second storage tank for condensate water, resistant to extreme natural hazards, e.g. a hardened CST with automatic transfer based on loss of power to main CST level instrumentation (low suction pressure).
- Install new stationary injection pump.
- Procure portable high pressure pumps to inject into the SG for RCS cooling (Modes 1–4).
- Establish a connection to allow for suction from a protected and backup water sources.
- Install primary connection into the SG for RCS heat removal.
- Install backup connection into the SG for RCS heat removal in case of primary connection being inaccessible.
- Provide make-up flow path of water to the turbine driven auxiliary feed pump in the event of loss of the primary source, i.e. CST.
- Re-evaluate turbine driven emergency feed pump room heat-up.
- Procure water treatment equipment for SG makeup water (see also Section 4.11.3).
- Procure water storage bladders for treated water for SG injection.

— Procurement of mobile pumper (fire trucks) as mitigation equipment which take water from identified/arranged water sources and deliver it to the SG and potentially other systems as required.
— Install permanent fire hose ‘quick connect’ couplings that will enable emergency water to be added to the heat transport system of each unit. The design and fabrication of a temporary fire hose connection which will enable emergency water to be added to the heat transport system of each unit using an alternate location.
— Perform seismic margin assessments of tanks which are key source of backup water.
— Perform RCS cooldown analyses using ‘best estimate’ methods to optimize deployment time margins.

Portable SG makeup pumps selected by utilities cost in the range of US \$35 000–US \$50 000 per pump. In some Member States, at least one extra pump was required, i.e. three pumps for a two-unit site.

The additions and/or modifications to systems utilized for water injection to RCS for core cooling have been implemented in most countries as a medium term action, i.e. within 1–5 years of the accident. For those NPPs who planned to build hardened and permanent SSCs, the project schedule was typically long term with completion dates were between 8–10 years.

4.6.2.1. Challenges and resolutions

The challenges that were encountered during the plant modifications for primary injection means included:

— Identifying the scenario of BDBE/BDBA progress and the cooling system, units and site conditions which will determine identification of connection points and portable equipment layout that are easily and most likely accessible and protected from external events for a substantial variety of scenarios.
— Identifying injection paths and connections that are not affected by the extreme external events, for example the tie-ins being restricted to only seismically qualified paths or not which can be resolved by:
    - Ensuring the permanent connection points meet all the codes and standards applied to the plant systems (i.e. classification/qualification) to which they are connected and the temporary makeup components such as hoses and other components not permanently connected to the system meet lesser commercial grade requirements;
    - Performing a plant habitability assessment to demonstrate location of tie-in points would be accessible owing to post-accident conditions;
    - Installing tie-ins as close as possible to the entrance to the plants so that personnel do not have to transverse through the plant.
— Identifying injection paths and connections that provide available, adequate and effective injection in different scenarios.
— Identifying and necessary protected primary and backup source of water that would be available following the extreme external event. This required measure to enhance protection of existing borated water source or building an additional protected one.
— Some nuclear power plants assessed that the time-sensitive actions during particular shutdown conditions (e.g. SGs are available) may not have adequate margin. Therefore, in such cases, operating organization decided to pre-deploy and keep portable pumps

during the duration of shutdown (for example, during the refuelling/maintenance outages, or even during the entire outage) to improve time margin and reduce outage risk.

— Ensure a positive drafting source that necessitated a connection, particularly for pumping vehicles such as fire engines, to allow for suction from a protected water source and portable suction booster lift pumps with on-site and/or off-site storage.
— Ensure that control logic failure of active components do not impact the ability of water injection by installing the tie-ins (both primary and backup) downstream of active components;
— Storage of mobile equipment in protected storage structures/buildings/facilities which in most nuclear power plants required earthquake and flood resistance provisions.
— Deciding on the level of qualification and classification of the mobile pumps, hoses, connections (not only for use and storage but also transportation, e.g. the proven resistance of mobile pumps to shaking during the transport from storage to connection location due to road conditions) which was handled by purchasing extra quantities with both on-site and off-site storage.
— Determining the appropriate design for the equipment, particularly portable ones, considering needed pressure, flow losses in temporary and long hoses, backpressures, etc. which required extensive thermohydraulic analyses. This challenge was handled by some NPPs by performing RCS cooldown analyses using 'best estimate' methods to optimize deployment time margins.
— Depending on the plant design and siting as well as the RCS cooldown analyses, pump ought to be deployed and ready to inject typically within 6 to 8 hours following initiation of the event, providing backup to the turbine driven aux feed pump.
— Procedural changes for extending preservation of source water inventory, such as minimizing the cycling of auxiliary feedwater (AFW) valves by setting them up at an optimal position determined by thermohydraulic analyses.

4.6.2.2. Lessons learned from decision, implementation and strategies for long term sustainability

From the challenges and their resolutions a few ideas were noted by the NPPs replying to the survey:

— Best estimate cooldown analyses using realistic methods may provide optimization of deployment and increase time margins. As an example, one operating organization responding to the survey provided the amount of time margins as:

> *"Best estimate analysis provided increase in time needed to refill CST (e.g. from 16 hrs to more than 32 hrs, and when combined with seismic fragility analysis of other available and cross-connected water sources, that might provide water up to 72 hours)."*

— *'N+1'* capability had to be satisfied for certain Member States, i.e. three pumps were required for a two-unit site, unless a plant strategy calls for one pump feeding two units through a common header (a few nuclear power plants).

4.6.3. Adding alternate means to relieve steam

The pneumatically operated valves are necessary for decay heat removal from the primary system. The operation of these valves needs pneumatic support, which is typically compressed air or nitrogen that provides the motive force. Therefore, most nuclear power plants purchased and installed additional compressed air or nitrogen equipment, such as air compressors, air tanks or nitrogen bottles, to operate (open/close/cycle) AOVs that were identified needed for decay heat removal, with back up compressed air supply. Also, in some designs, these included power operated relief valves (PORVs) and/or safety relief valves (SRVs), etc. which are air operated. For example, PORVs to relieve main steam system and steam generator pressures, such as those at Krsko NPP [107].

In some Member States it was a regulatory requirement; however, the type, quantity and location of compressed air and nitrogen supply equipment were determined in accordance with the mitigation strategy.

Cost of an air compressor depends on the needed header pressure; however, based on survey, a typical air compressor cost around US $45 000.

4.6.4. Adding capability to maintain the core subcriticality

The primary method used by the nuclear power plants (particularly PWR plants) to maintain and/or ensure core subcriticality (i.e. keeping reactor shutdown) under BDBE conditions was boron injection to provide sufficient negative reactivity, as the plant is cooled down and Xenon decays, based on the plant's specific configuration and analysis. Therefore, the NPPs took actions to ensure long term availability of means for boron addition to the injection systems that were discussed in Section 4.6.1. The plant's portable RCS water injection systems would take suction from a tank to inject boron as needed by the plant's mitigation strategies. Availability and sustainability of shutdown margin determines the implementation of plant modifications and the equipment needed.

Therefore, the nuclear power plants procured portable mobile boration equipment (in addition to the procurement/installation of RCS injection measure listed in Section 4.6.1). Also, some nuclear power plants (for example, the BWR plants in Japan) choose to install backup/alternative anticipated transient without scram (ATWS) mitigation systems/facility to ensure function of alternative control rod insertion, function of alternative recirculation pump trip and boric water injection system.

The installed/modified equipment and their associated systems and analyses included:

- Perform cooldown thermohydraulic analysis for given cooldown strategies (including the considerations for RCP seal leakage and control rod worth) to determine if and when boron injection is needed;
- Perform seismic margin assessment of tanks which are the main source of backup borated water;
- Investigation of alternative boron injection sources (for example, boron injection from the accumulators or safety injection tanks (SITs) providing that instrumentation for volume/level indication is qualified and procedural action to isolate the tanks before nitrogen injection occurs);
- Mobile equipment to mix boric acid for either a continuous injection or a batch injection, or some combination of these;

— Electrical connections to repower permanent plant pumps (e.g. charging pumps) which are connected to a seismically robust source of borated water (e.g. refuelling water tank (RWT)).

Cost for a mobile boration system was dependent on whether a continuous injection system or a batch system is purchased. A continuous injection mobile boration system provides flexibility for other uses (e.g. outages) and a batch system can only batch in limited quantities, can be labour intensive and may need other support equipment (e.g. its own dedicated portable generator, etc..) which is an additional cost. Also, the cost for a batching system needed to consider the additional support equipment required (e.g. the need for a separate portable generator to power the heaters and mixers, etc.), as well as the additional support personnel that are needed to support the mixing/batching operations.

4.6.4.1. Challenges and resolutions

The challenges that surveyed NPPs reported that they faced during the plant modifications for means to ensure core subcriticality during BDBE included:

— Identifying the timing of when boron injections may be required in the BDBE progression;
— Identifying injection paths and tanks to store the mixed boron solutions;
— Determining support equipment and that are needed to support the mixing/batching operations.

4.6.4.2. Lessons learned from decision, implementation and strategies for long term sustainability

Credit can be taken for boron injection from alternative sources, such as accumulators/SITs, if the SIT instrumentation for volume indication is qualified (and procedural action to isolate the tanks before nitrogen injection occurs). This can be demonstrated analytically to provide some negative reactivity.

The boric acid may also be transferred to one of the plant's storage tanks for later use as RCS makeup.

4.7. INSTALLATION OR MODIFICATION OF SYSTEMS FOR SPENT FUEL PROTECTION

IAEA's Fukushima Daiichi Report [1] assessed the performance of spent fuel pools and associated actions during the accident sequence. Based on the events, there were lessons learned from two aspects:

— The SFP cooling and make-up capabilities were lost upon LOOP, owing to the not being powered by emergency AC power from the EDGs when the normal AC power is lost[9];
— The additional loss of DC power which disabled the monitoring systems resulted in loss of the SFP water temperature and level indications, such that operators could not monitor those.

Therefore, the industry actions addressed the protection of SFP (including AGR buffer storage tubes as discussed in Ref. [106]) against both the loss of cooling and makeup capabilities and the loss of indications.

4.7.1. Adding alternate measures for spent fuel pool cooling and preservation of coolant inventory

In order to address maintaining and/or restoring SFP cooling and inventory the actions that were taken included the provision of alternate or backup water injection systems, alternate cooling mechanisms (including surface spray cooling and makeup) and measures to prevent loss of water inventory due to potential occurrences, such as damage to SFP structure or sloshing as a result of seismic event.

For establishing alternate water injection systems, most of the NPPs implemented an approach with portable means and established strategies and procedural steps considering the lack of access to the SFP and the equipment areas due to physical damage to the structures and/or radiological conditions. Furthermore, some NPPs have added additional injection capabilities with stationary, permanent, hardened, and/or self-powered equipment some of which are located away from the SFPs. In some Member States, depending on the external event, the existing coping strategies with unusual (i.e. man-made) events that are provided for DiD were utilized in the extreme external events coping strategies. For example, in the USA, previous (i.e. pre-Fukushima) improvements for mitigating strategies for loss of large areas established the utilization of existing fire headers to provide water for spray coverage for spent fuel pool [108, 109]. The same methods were also considered and applied in response to the lesson learned from Fukushima Daiichi accident regrading SFP cooling and makeup.

The means for water injection and inventory preservation that were deployed in the NPPs, based on the survey results, included:

— Perform study of the behaviour of the fuel and water in SFP, e.g. ‘time to boil’ analysis, in prolonged SBO situation. (‘Time to boil analysis’ for SFP is also used to demonstrate that there is no need for temperature measurement if a reliable level measurement is provided).
— Perform seismic structural integrity analysis for SFP.
— Perform analysis of the loss of SFP water inventory due to sloshing during a seismic event. The amount of water sloshed out of the SFP depends on the wave motion inside pool.
— Identify and install permanent water supply for SFP cooling and makeup (groundwater wells, ponds, etc.)

[9] Alternative cooling via RHR system, which was powered by the emergency AC power; however, the UHS was also incapacitated by the tsunami damage of the intake structure and the loss of power [1].

- Add a second storage tank for condensate water, resistant to extreme natural hazards, e.g. a hardened CST with automatic transfer based on loss of power to main CST level instrumentation (low suction pressure).
- Portable water treatment equipment for SFP makeup water (see also Section 4.11.3).
- Portable water storage bladders for treated water for SFP injection.
- Portable water polishing system along with filters and cation exchangers to remove the radioactive particulates like Cs and Sr in order to mitigate/prevent high radiation levels in the fuel building due to SFP water activity.
- Install new stationary injection pump.
- Procure portable pumps to inject into the spent fuel pool.
- Establish a connection to allow for suction from a protected and backup water sources.
- Install primary connection into the spent fuel pool for makeup and cooling.
- Install backup connection into the spent fuel pool for makeup and cooling in case of primary connection being inaccessible.
- Modify for branch piping as siphon breaker in SFP.
- Arrangements for gravity feed.
- Procurement of mobile large capacity pumper (fire engines, pumping trucks) with pumping and spraying capabilities from high elevations as mitigation equipment. Injection (or pool spray) water will be provided by means of mobile pumpers (fire trucks or portable diesel pumps) which draws water from the dedicated water source delivers it to the SFP.
- Acquirement of mobile heat exchanger that can be connected to the spent fuel pool.
- Installation of permanent spray system around the SFP (and/or on adjacent building roofs) with provisions for quick connection from different sources of water.
- Acquirement of portable spray oscillating monitor nozzles and hoses to provide a spray over the entire SFP.
- Installation of permanent fire hose ‘quick connect’ couplings that will enable emergency water to be added to the SFP of each unit. The design and fabrication of a temporary fire hose connection which will enable emergency water to be added to the heat transport system of each unit using an alternate location.
- Installation of sloshing curbing modifications based on the analyses of sloshing during a seismic event.

In the USA, for example, adding alternate measures for spent fuel pool cooling and preservation of coolant inventory was a regulatory requirement (as one of the options to be provided by the licensees). There, the industry guidance [20], specified three acceptable approaches consisting of three separate capabilities for the SFP cooling strategies. This approach uses a portable pump to provide the capability for:

“1) makeup via hoses on the refuelling floor capable of exceeding the boil-off rate for the design basis heat load; 2) makeup via connection to spent fuel pool cooling piping or other alternate location capable of exceeding the boil-off rate for the design basis heat load; and 3) spray via portable monitor nozzles from the refuelling floor using a portable pump” [20].

During the event, the licensee selects the method to use based on plant conditions.

The challenges that surveyed NPPs reported that they faced during the plant modifications for maintaining and/or restoring SFP cooling and water inventory during BDBE included:

— Potential for and prevention of SFP building internal flooding due to condensation of steam that accumulates inside the building as the water in the SFP and make up boils (feed and boil off operation) to relieve the steam (if any) to establish a vent path (e.g. opening the fuel building roll up door). This modification may be a part of the SFP cooling strategies, noting that this may be a time sensitive action to open vent path since the conditions in the spent fuel building may not permit performing this action later. Also, this requires consideration of scenarios regarding early radiological release, as well as special conditions such as freshly unloaded core to the SFP and fuel loading patterns in the SFP.
— Storage of mobile equipment that required protected storage facilities.
— Qualification and classification of the mobile pumps, hoses, connections (not only for use and storage but also transportation, e.g. the proven resistance of mobile pumps to shaking during the transport from storage to connection location due to road conditions) which was handled by purchasing extra quantities with both on-site and off-site storage, for example purchased backup equipment one for each unit on site, and then one more as a backup at off-site storage.
— Injection pump typically ought to be deployed and ready to inject within a certain time after the event that needs to be supported by ‘time to boil analysis’ for spent fuel, i.e. SFP heat-up by loss of SFP cooling function or reduced water level due to loss of injection function and evaporation, to ensure that top of fuel remains submerged, water level is maintained to provide proper shielding and the spent fuel remains subcritical.
— The biggest challenge for establishing capabilities for the SFP cooling spray strategy was to determine the ability to access the SFP area or surrounding under accident conditions. This requires identifying scenarios of BDBE/BDBA progress and units and site conditions which will determine identification of spray points and portable equipment layout that are easily and most likely accessible and protected from external events for a substantial variety of scenarios. Consequently, this necessitates performance of a plant habitability assessment to demonstrate location of spray points would be accessible owing to post-accident conditions.

The average cost of an SFP makeup pump is US $43 000. The total cost for provision of SFP cooling and makeup measures ranges in US $250 000 – US $750 000 depending on the types and number of pumps procured, complexity of modification for connections and the extent of the ‘time-to-boil’ studies (which typically costs between US $10 000 and US $45 000) based on various scenarios.

4.7.2. Adding spent fuel pool status indications

As discussed in Ref. [1], the lack of knowledge on the status of the SFP (regarding the actual water level and temperature, as well as the radiation monitors) by the plant operators and plant personnel hindered effective prioritization of emergency actions.

In nuclear power plants, several parameters are monitored for various indications of SFP status either by safety or non-safety related instrumentation as discussed in Ref. [110]:

> "Some plants have safety-related power to the SFP instrumentation, but the instruments themselves are in most cases not safety related. Other plants have neither safety-related power supplies for the instrumentation, nor instruments that are safety-related. In general, the plants have no redundancy for the SFP instrumentation" [110].

These instruments include, but not limited to:

— Some type of SFP water level instrumentation (generally measuring narrow range but varies among the plant designs) is provided to ensure SFP cooling and shielding from the spent fuel during normal operations;

— SFP temperature is monitored by temperature instrumentation with indicators (or alarms) either in local panels or in control rooms to observe the SFP heat-up/cooling;

— Pressure (e.g. of pump discharge) and flow of cooling systems to monitor their function and performance;

— Radiation levels are also monitored around the spent fuel area generally by fixed area radiation monitors (mostly donned with alarms) to protect workers in the area, particularly during fuel movement and as an indicator of SFP water level decrease that reduce shielding;

— SFP water activity is regularly checked by on-line or manual sampling for radioactive release and corrosion of spent fuel or subcriticality (typically applicable to PWR plants);

— Most plants also have leak detection systems to monitor potential SFP leakage;

— In AGR plants, buffer store temperature is monitored by the coolant gas temperature thermocouples which are fitted to the fuel assemblies.

During the accident conditions, when the SFP cooling is lost for a long time, the water level decreases due to evaporation. In such conditions, a reliable and operable SFP water level indications (typically by a wide range level instrumentation, e.g. down to the top of the fuel assemblies, depending on the heat load of the SFP and the extent of boiling), temperature indications and radiation monitors are essential, as one of the lessons learned from the Fukushima Daiichi accident. Therefore, all plants implemented actions to install or to improve SFP status instrumentation that would maintain or restore their functions under the accident conditions, although the type and extent of modifications varied among plants, such as:

— Installation of new/redundant SFP water level measurement instrumentation in fuel building (i.e. for the spent fuel pool) and new water level measurement instrumentation for refuelling pool;
— Installation of SFP water level instrumentation qualified for seismic resistance and harsh conditions (e.g. 100% humidity environment);
— Installation of passive/self-powered monitoring systems for water level, temperature and radiation levels in addition to redundant systems;
— Enhancement of existing SFP water level instrumentation to impose seismic resistance and 100% humidity environment;
— Relocation of electronic parts of the instrumentation to outside of SFP to prevent from the effects of harsh environment;
— Provision of redundant instrumentation with power supply during ELAP qualified for DBE;
— Installation of redundant radiation measurement equipment;
— Installation of temperature measurement instrumentation, which, in most nuclear power plants, necessitated a 'time-to-boil' study for SFP to demonstrate that there was no need for temperature measurement with the provision of a reliable water level measurement instrumentation and an accurate SFP heat load analysis.
— Procurement of portable (hand-held or deployed) temperature, level and radiation monitors;
— Stationary cameras and visible level markings (visual inspection from the control room or in fuel building) and temperature monitoring;
— Installation of hydrogen detectors in areas near the SFP.

The main reason for such differences was the variation in regulatory requirements. Some regulatory bodies required, for example, independent and redundant power source and/or seismic or environmental qualifications, while some agreed with the use of derived parameters (for example using level estimation by time to boil analysis that could eliminate need for temperature measurement) or with the alternative methods for measurement and their qualification and maintenance. For example, in the USA, the regulatory body issued an order in March 2012 [111], requiring all U.S. nuclear power plants to install water level instrumentation in their SFPs. It was also required to monitor at least three distinct water levels: normal level; low level but still with enough biological shielding; and minimum level (i.e. near the top of the spent fuel rods where more water should be added without delay). In response, the industry prepared a guidance on how to comply with this order [112] which was reviewed and agreed (with some exceptions) by the regulatory body [113].

The challenges that surveyed NPPs reported that they faced during the plant modifications for monitoring SFP status during BDBE included:

— Qualification of equipment for harsh environment for high temperature, humidity and radiation which was resolved by moving (or installing) the electronics outside the fuel building. Another approach was installation of venting systems (or strategies) for removal of steam (and hydrogen) from the fuel building to mitigate the effects of steam from the SFP.
— Qualification of equipment for seismic resistance which could be resolved by ensuring the permanent installation meets all the codes and standards applied to the plant

systems to which they are connected and the temporary or mobile components not permanently connected to the system meet lesser commercial grade requirements.

— For portable instrumentation, the challenge was to store and transport and layout in the fuel building which necessitated a plant habitability assessment to demonstrate that they can be deployed under adverse conditions or accident conditions.
— Time to boil analysis for SFP can demonstrate that there is no need for alternative/redundant temperature measurement if a reliable level measurement is provided.

Spent fuel pool monitoring equipment and installation typically costs in the range of US $750 000–US $1.2 million per nuclear power plant unit. This cost includes the development and installation of modification, materials and labour. When relocation of existing instrumentation from potential harsh environment during the accident, seismic qualification and testing are included, the cost can be as high as US $2 million.

4.7.3. Other spent fuel pool protection enhancements

Upgrading of fuel handling systems had already been in progress prior to the Fukushima Daiichi accident for seismic resistance as well as for eliminating single point vulnerabilities. No additional requirements were added from the lessons learned in Fukushima Daiichi accident; however, the event highlighted the essentiality of being able to place fuel in a safe position during external events. In order to provide DiD and redundancy to nuclear power plants fuel handling systems, redundant manual systems for handling and placing spent fuel in a safe position were installed with appropriate procedures in some nuclear power plants. Also, some nuclear power plants installed modification for enhancement of earthquake resistance of the SFP and fuel transfer canal (see also Section 4.1.1), including gantries, bridges and cranes.

One Member stated reported that, although the enhancement of spent fuel handling system was not a regulatory requirement, nuclear power plant operating organization chose to implement modifications for enhancement for DiD.

There were also some actions taken regarding the SFP arrangements considering that storing spent fuel in certain way of SFP loading, based on some studies [114]. Such actions were considered to enhance DiD by *"further reducing the likelihood of fuel assemblies overheating in the event of substantial SFP damage"*, as stated in Ref. [114]). The strategies that were discussed in Ref. [115] included: "*Storing spent fuel in a more favourable loading pattern"* and *"directly offloading fuel from the core into dispersed patterns in the SFP"* [115].

Other enhancements also included installation of hydrogen detectors in areas near the SFP, hydrogen vents in buildings housing SFP, hydrogen igniters/recombiners in SFP area, for hydrogen protection.

4.8. INSTALLATION OR MODIFICATION FOR CONTAINMENT COOLING

The containment or reactor building is the final barrier of protection for radiation control following an accident. As discussed in Ref. [1], to prevent significant release of radioactive material to the environment, a reliable confinement function, such as by containment or reactor buildings, needs to be provided for BDBAs. The confinement function is assessed:

> "to ensure that all possible hazards are considered in the design of equipment intended to maintain the integrity of the confinement system" [1].

In a BDBE/BDBA, if no other heat sinks are present, the containment will act as a heat sink and absorb the decay heat generated by reactor core. Without any means of removing the heat from containment, the containment/reactor building can be challenged by overpressurization due to steam or other non-condensable gases. There are other containment failure mechanisms during a severe accident including failure due to hydrogen ignition, failure due to corium melting through the basemat, and failure due to containment bypass conditions (e.g. the rupture of a steam generator tube).

This section discusses installation of new equipment or modifications made to existing equipment to enhance the capabilities of containment to withstand challenged due to the BDBE/BDBA. Also, adding means to maintain structural integrity of containment building against ex-vessel phase is later discussed in Section 4.12.7.

4.8.1. Adding means of containment cooling

In response to the lesson learned from the Fukushima Daiichi accident, some plants have installed alternate (mobile or stationary) containment cooling systems, while some chose to strengthen the ability of existing system in addition to installation of alternate system, as DiD.

Alternate (mobile or stationary) containment cooling systems measures taken by the nuclear power plants included:

- Installing an 'ultimate containment cooling system' for residual heat removal without opening (venting) the containment building enhancement to mitigate severe accidents. This design enables to flood reactor sump before core melt (as a measure to have similar impact of core catcher for ex-vessel phase instead of retrofitting a core catcher) and requires a means of moving water out and back into containment and a heat exchanger outside of containment, or a heat exchanger inside of containment and a means of providing cooling water. Therefore, this modification included a low pressure primary injection pump with water source (to maintain pressure and subcriticality) for ultimate containment cooling system and a large mobile heat exchanger that is to be transported from the national response centre, deployed within two days for extreme external events and connected to primary, secondary or backup UHS.
- Installing a stationary alternative low pressure injection pump.
- Purchasing portable low pressure pump and or fire extinguishing pump.
- Installing hardened containment vent from wetwell and/or drywell (BWR type reactors).
- Creating gravity-fed reactor vessel external cooling by creating a flow path to flood the reactor cavity, supplied from the localisation tower (from the bubble trays) to the floor of the containment floor, and then through the gap around the reactor vessel (some VVER type reactors). It should be noted that such modifications were planned long before the Fukushima Daiichi accident and some had already been installed in VVERs in Finland and Hungary [116].
- Installing a severe accident sprinkler system that uses the water accumulated in the containment floor through the sumps and returning it to the localization shaft by (some VVER type reactors).

Actions to strengthen the ability and capacity of existing containment cooling systems included:

- Self-cooling modification of conventional pump;
- Purchasing larger capacity pump;
- Performing containment compartment heat up analyses.

Following challenges were reported by the surveyed NPPs in deciding and implementing the action related to adding alternate means of containment cooling:

- Deployment of mobile means from outside centres in extreme hazards situations and conditions;
- Qualification of equipment for severe conditions (particularly for equipment located inside containment building and for those located outside containment and subject to severe external events);
- Need for extensive analysis to determine the potential effects of creating a path connecting inside containment to outside by connecting ultimate containment cooling system;
- Difficulty in maintaining the integrity of connections and hoses against the harsh and potentially overwhelming conditions that can induce leakage;
- Large capacity mobile heat exchangers necessitate utilization of a large capacity pump that is connected to ultimate heat sink by mobile means for containment cooling.

4.8.2. Adding means to depressurize containment by venting

Under an accident scenario where steam or other non-condensable gases build up inside of containment, pressure will rise to the point where the containment will fail and create an uncontrolled path. This can be prevented by installing a filtered release path from containment. Containment filtered venting systems (CFVS) have been around for several decades, with the first major wave of CFVS systems being installed in response to the Chernobyl accident. For BWR plants, in some Member States' regulatory framework, HCVS or CFVS from drywell, wetwell, or both had already been required for some types of BWR designs. Similarly, some PWR designs originally included (or later added) filtered or unfiltered containment vent system for severe accident management [117].

After seeing the difficulties with depressurizing the containment vessel and the environmental consequences of an unfiltered release during the Fukushima Daiichi accident, many more countries decided/required to install CFVS systems or to improve existing systems, such as converting them to HCVS. CANDU plants, as well as some PWR plants, were also required by the regulatory bodies to install passive containment venting systems.

The primary concerns for filtered venting are the corium aerosols and the iodine (both elemental and organic iodine). Several CFVS and HCVS designs have been used or existed in the marketplace prior to the accident, and additional enhancements were made to these designs while other designs were also explored and brought on to the marketplace following the accident [116].

Method used by NPPs to decide whether a CFVS is an improvement typically followed a process of:

— Evaluation of the CFVS benefit under different DEC scenarios by:
  - Predicting the conditions for pressure reaching the maximum design limit, as well as its possibility;
  - Assessing the possibility of preventing containment failure by using containment venting systems;
  - Evaluating the key requirements for CFVS.
— Selection of the CFVS types that are possible for installation at NPPs which considered different designs, such as sand filter, venturi scrubber, wet/dry filter, etc.;
— Preliminary evaluation of the CFVS efficiency of each possible types under different accident sequence.

According to the response to the survey, the operating organizations noted that the drivers for CFVS installation were both regulatory and political in some cases. Most countries ended up installing CFVS to provide additional assurance that the environmental concerns could be managed during a severe accident. However, there were a few countries that identified that the CFVS did not have an appropriate benefit, and the resources (human and financial) that would have been spent on a CFVS could be rather spent on activities to prevent the severe accident (i.e. to prevent the core melt which would mean there would be no need for containment protection against pressurization).

Such variation in cost/benefit assessment due to differences in the cost of containment venting system hardening or installing filtered systems which varied among Member States mainly owing to the differing labour costs. For example, the survey responses from the U.S. NPPs reported that a HCVS from wetwell in a BWR cost typically around US $12 million, while making it severe accident capable adds around US $2 million. This is the total cost of the development and installation of modification and labour and parts that include the monitoring and measurement instrumentation. For a VVER-1000 plant, the hardening of existing CVS (e.g. replacement of air ducts, upgrading discharge line, etc.) cost US $200 000, while the installation of CFVS (dry filter or venturi scrubber) cost US $6 million [118], which eventually was decided that CFVS did not have an appropriate benefit.

Also, several NPPs used a staged approach, first hardening the existing venting systems as a mid-term action, followed by the installation of CFVS (owing to the need of a longer timeline for installation because the unit will need to be in an outage state, as well as the results of filter efficiency determination studies).

Actions taken by the NPPs to support containment depressurization included one or more of:

— Installation of hardened containment vent from wetwell and associated monitoring systems and venting procedures (some BWR plants).
— Installation of hardened containment vent from drywell and associated monitoring systems and venting procedures (some BWR plants).
— Provision of passive containment venting system or CFVS (CANDU, and some PWR and some BWR plants).
— Installation of larger relief capacity to increase the shield tank overpressure protection (CANDU plants). This modification was due to crediting the shield tank as the final barrier for in-vessel retention of the molten core, noting that the shield tank was originally designed as a biological shield with low (smaller) overpressure capability. Shield tank overpressure protection, stated by the survey, *"will allow the existing water in the shield tank to act as a heat sink thereby providing additional time to take suitable*

mitigating action by deploying emergency moderator makeup and/or shield tank makeup. Shield tank overpressure protection also reduces the containment overpressure transient that can occur following shield tank rupture, thereby reducing the possibility of containment failure".

— Enhancement of earthquake resistance of the containment venting filter (sand filter) (some PWR plants). This modification was owner/operating organization's decision as a DiD, as it was not included in regulatory requirements due to the provision of a new ultimate cooling as indicated by one Member State in the survey.
— Enhancement of the capability of the wetwell HCVS during ELAP conditions by providing a supplemental compressed gas connection to the wetwell HCVS isolation valves that are protected from severe external hazards owing to their location. This modification was to provide a means of operating the valve control solenoids for wetwell HCVS isolation valve from outside of containment during a loss of AC, DC and normal control air supply system.
— Initiation of research and benchmark on the effectiveness of containment vent systems and utilization of new technologies to filter gaseous iodine (See Section 9.1).

The challenges encountered by NPPs in evaluation, decision and installation of containment depressurization included:

— Choosing a design with a relief capacity that would not challenge normal operation of the reactor due to spurious operator as a result of transients.
— Ensuring the effectiveness of containment vent filters which necessitated on-going research and benchmark testing of new technologies to filter iodine gas. Making this technology available for ready for industrial use but not for nuclear.
— Discussions are needed to agree on decision making criteria, safety goal and authority to make venting decision.
— Additional requirements by the regulatory body during implementation in-progress, resulting in complication in design and delay in implementation schedule. For example, in the United States, the hardened vents for some BWRs were initially ordered by the regulatory body to provide a method for removing heat from containment that would be available following a beyond design basis event. However, the regulatory body later added new requirements for the hardened vents to be severe accident capable. This added additional complexity to the design and resulted in an approximately 2-year delay for implementation.
— Determining the adequacy and applicability of new filter technologies, such as new vent filter technologies that would increase the effectiveness of filtering iodine gas (for which the research and development is continuing).

4.8.3. Adding hydrogen mitigation means

Fukushima Daiichi accident highlighted the essentiality of monitoring and controlling hydrogen in the plant buildings and compartments, particularly in the containment and reactor buildings. Also, the plant evaluations showed that it is also beneficial to monitor hydrogen in equipment rooms, such as in battery rooms which were discussed separately in Section 4.3. This prompted nuclear power plant owner/operator organizations and regulatory bodies to assess hydrogen monitoring, control and mitigation by performing hydrogen analysis and assess the effectiveness of measures in maintaining the function during severe accident conditions. These assessments lead to the regulatory requirements as to monitor and control hydrogen during severe accident conditions. In response, following actions at the nuclear power plants:

- Installation of active hydrogen recombiners;
- Installation of ignitors with backup power supplies;
- Installation of passive autocatalytic hydrogen recombiners (PARS) to avoid explosive atmospheres if active recombiners are disabled;
- Installation of equipment to open blowout panel manually;
- Installation of redundant detectors for measuring hydrogen concentration;
- Installation of instrumentation to measure hydrogen recombiner temperature.

Main challenge in taking actions on hydrogen monitoring and control equipment installation was the necessity for heavy engineering, including seismic assessment, inside containment which necessitated good work planning and longer schedule of implementation since the work would be phased over two (or more) outages.

4.9. INSTALLATION OR MODIFICATION OF BACKUP ULTIMATE HEAT SINKS

The UHS is one of the most critical systems as it provides a medium to remove and discharge core decay heat (residual heat), and thus, its failure may lead to core degradation. Furthermore, in most NPP designs, the UHS is utilized to cool safety related plant equipment (e.g. emergency diesel generators, air compressors, intermediate component cooling systems, containment cooling fans) loss of which could lead to safety related equipment failure which subsequently could lead to core or containment building damage. As a lesson learned for the UHS integrity from Fukushima Daiichi accident, prevention of loss of the UHS is important for removing the decay heat and to maintain the safety functions of SSCs.

Regarding the design of heat transfer to the UHS, the Requirement 53 of IAEA Safety Standards Series No. SSR-2/1 [15] requires that:

> “The capability to transfer heat to an ultimate heat sink shall be ensured for all plant states” [15].

and highlights that:

- “Systems for transferring heat shall have adequate reliability for the plant states in which they have to fulfil the heat transfer function. This may require the use of a different ultimate heat sink or different access to the ultimate heat sink.”
- “The heat transfer function shall be fulfilled for levels of natural hazards more severe than those considered for design, derived from the hazard evaluation for the site.” [15].

The reliability of the systems is provided by measures including the redundancy, diversity and physical separation. In the design of such systems, natural and human induced events are taken into account and the possible diversity in the ultimate heat sinks and in the storage systems, from which fluids for heat transfer are supplied, is considered. Following these lessons earned, the NPPs evaluated:

- Providing additional diverse and redundant water sources, such as alternate tanks or wells on the site and reservoirs, lakes, aquafers, etc. in the vicinity;
- Providing additional direct dissipation of heat to the atmosphere (such as air cooled cooling tower);
- Eliminating or minimizing the dependency of safety related components on water for cooling;
- Providing alternate means for UHS operation, e.g. for supplying power and water from portable equipment;
- Maintaining the integrity of UHS structures during extreme external events by various means.

As such, the actions taken primarily included the modification of existing ultimate heat sink and/or adding backup ultimate heat sink, including:

- Adding ultimate water source (groundwater wells, reservoirs, ponds, dams, etc.) as the backup ultimate heat sink (BUHS);
- Adding large capacity mobile heat exchangers;
- Installation large capacity pumps for transfer of water from the backup ultimate water source;
- Adding back up air cooled cooling towers;
- Modifying existing cooling tower basin to enable to provide backup ultimate heat sink;
- Installation permanent water treatment plant to provide additional time until replenishment is provided (in case of large infrastructure damage around the site hampers replenishment);
- Adding portable connections, pumps, hoses for alternate water sources;
- Adding power supplies and portable connections for operating pumps, heat exchangers, etc., for example, to rely on portable DGs to power the existing RHR system, as DiD;
- Making buildings that house the SSCs of UHS watertight to prevent against flooding;
- Reanalysis of the adequacy of existing water supplies for DEC.

The reported challenges encountered by NPPs in enhancement of the reliability of UHS included:

- Poor water quality of the groundwater which required deeper well drilling with special drilling tools and methods.
- Deep drilling resulted in drill damages that deeper drilling should be optimized with consideration that the price will increase per length while the water quality improves. As an example, one survey responder indicated that the cost of establishing groundwater well for BUHS (in case of deeper well drilling due to quality of water) is approximately US $1 million in addition to the cost of procuring and installing stationary and portable connections, pipes/hoses and the supporting analysis for water supplies.

— Extreme low temperatures could result in ice formation on the body of water (e.g. lake, reservoir, etc.) used as a water source, necessitating capabilities to break ice for establishing suction for cooling system.

— Mobile heat exchangers strengthen the ability of heat transfer to the ultimate heat sink but necessitates utilization of a large capacity pump and connections to ultimate heat sink by mobile means, and therefore, not only they become costly[10] but also weight and size and shielding are challenges to be resolved. Also, long lead times for procurement which can take up to two years. Modular constructability (e.g. plate design heat exchangers) could provide easier ability to add capacity.

4.10. INSTRUMENTATION AND CONTROL IMPROVEMENTS

For instrumentation and control aspect, the main lessons learned from the Fukushima Daiichi accident was that critical instrumentation needs to continue to operate in order to monitor essential plant safety parameters and to facilitate plant operations during severe accidents. Particularly, the equipment important for providing important information about the reactor, SFP and radiological status needs to be resistant to severe accident conditions [1]. An important lesson from the accident noted in Ref. [1] was that CCFs can occur due to extreme external events resulting in complete loss of the instrumentation of the plant, and thus, important equipment needs to be invulnerable to CCFs during such events. Critical instrumentation needs to be designed and maintained so that the plant is equipped with adequate hardware provisions in order to fulfil the fundamental safety functions as far as is reasonable for BDBEs and severe accidents. A guidance on hardware provisions is further described in IAEA Safety Standards Series SSG-54 [119].

One of the most significant aspects of the accident at the Fukushima Daiichi NPP was the progressive loss of all I&C, leaving the operators with few, if any, indications to assist them to understand, monitor and control the situation. Therefore, loss of indications to the operators, strongly affected the decision making process during the accident progression. Reference [1] elaborates:

> "For example, with instrumentation available in Units 1–3, the operators would understand what was happening in the reactors and would be able to take action to further mitigate the consequences of the event. Similarly, operator awareness of the real condition of the SFP in Unit 4 would have enabled them to know that, in spite of the explosion, there was no immediate threat from the fuel in the SFP in Unit 4" [1].

Reference [1] further recommended potential actions to ensure availability of essential indicators for accident management as:

> "Special attention should be devoted to the I&C as an essential component for effective AM [accident management]. The plant parameters needed for AM measures should be identified and be available from the instrumentation. Dedicated instrumentation that is qualified for the expected environmental conditions is the preferred method to obtain the

[10] The modification costs to the primary system of a PWR for a mobile heat exchanger connection are approximately US $5 million per unit. In addition, the cost of a heat exchanger ranged from US $1 million to US $3 million depending on the capacity.

necessary information. The ability to infer important plant parameters from local instrumentation or from unconventional means should also be considered. The need for the development of computational aids to obtain information where parameters are missing or their measurements are unreliable should be identified and aids developed accordingly" [1].

4.10.1. Critical safety parameter measurements

The assessment of failures and degradation of critical safety parameters during the Fukushima Daiichi accident mainly focused on key BWR parameters of reactor vessel water level and pressure, drywell and wetwell pressures which is a small subset of large amount and type of instrumentation that are discussed in Ref. [120].

Understanding that the lack of indications on many plant parameters prevented the plant operators from having a clear understanding of core cooling, containment status or the status of safety systems, the NPPs conducted comprehensive evaluations of the critical safety parameters, their measurement and associated instrumentation under extreme external event and severe accident conditions resulting in one or more of following actions:

- Improvements of instrumentation (safety parameter) system for seismic resistance;
- Installation of redundant instrumentation to detect vessel break (by core melt);
- Addition of indications for hardened containment vent support;
- Purchase of additional mobile measuring units for temperature, pressure, radiation;
- Development of derivation/estimation methods for parameters by measuring and/or trending alternative parameters, as well as observing instrument response to operator/equipment actions;
- Survivability assessments for equipment and instrumentation for severe accident management;
- Instrumentation reliability and resilience enhancements to provide monitoring of critical reactor parameters when normal instrumentation has failed, e.g. continuous emergency monitoring system (CEMS);
- Installation of backup reactor and plant monitoring systems to provide alternate system for monitoring and communication to off-site monitoring, such as the deployable communications and instrumentation system (DCIS).

NPPs reported following challenges in the identification, improvement and installation of I&C systems and equipment that are needed for critical safety parameter monitoring during extreme external event and severe accident:

- Identifying conditions for survivability assessments for equipment and instrumentation under severe accident scenarios;
- Selection of scenario for decision on reliability limits and criteria;
- Qualification and classification of the equipment (seismic resistance, harsh condition resistance, etc.);
- Identifying parameters and alternate parameters to observe, trend and derive to get status indications and accounting overall plant conditions;

— Anticipation and incorporation of human and organizational factors (HOF) elements under accident conditions;
— Technical challenges for establishing secure reliable links for the communication of critical parameter data and information to off-site emergency centres.

Another challenge that was handled differently in different Member States was the quantification of the level of confidence in monitoring critical parameters, as well as the demonstration of provision and sufficiency of indications (for example, whether any available single channel may be sufficient to perform monitoring). It required a definition of 'quantification' which was provided by the regulatory bodies in some Member States, for example, as reported by a survey responder from Canada on the regulatory requirement:

> *"A quantified reasonable level of confidence that the means (e.g. equipment and instrumentation) necessary for severe accident management and essential to the execution of SAMGs will perform its function in the severe accident environment for the duration for which it is needed."*

4.10.2. Spent fuel pool measurements

In addition to the lack of understanding of core cooling, containment status or the status of safety systems, inadequate information about the actual conditions in the SFPs, caused diverting attention from other accident management efforts. For example, operators took priority actions to add water to the SFP owing to misinformation about the water level there.

Therefore, all plants implemented actions to install or to improve SFP status instrumentation that would maintain or restore their functions under the accident conditions, although the type and extent of modifications varied among plants as discussed in Section 4.7.2.

4.10.3. Improvements to plant protection systems

Some NPPs have already had automatic reactor trip on seismic conditions in their existing design and configuration. Some other NPPs considered adding automatic reactor trip on seismic conditions and took the following actions (see also Section 4.1.1):

— Evaluation of an automatic reactor trip in the event of an earthquake to determine whether it is required, beneficial or neither;
— Implementation of an automatic reactor trip in the event of an earthquake, if found required or beneficial.

The challenge with retrofit installation of automatic seismic reactor trip was to manage the risk of untimely or unintended reactor trip

4.10.4. Hydrogen monitoring

As discussed in Section 4.8.3, Fukushima Daiichi accident showed the necessity of monitoring and controlling hydrogen in the plant buildings and compartments. Some installed instrumentation included:

— Redundant detectors for measuring hydrogen concentration;
— Hydrogen recombiner temperature measurement instrumentation.

4.10.5. Radiation monitoring

Both on-site and off-site radiation monitors for dose rate and contamination were lost due to the loss of AC power. The action taken by NPPs included purchasing and installation of redundant radiation measurement equipment, as well as procurement of portable (hand-held or deployed) radiation monitors. Further discussions on the SFP radiation monitoring are provided in Section 4.7.2, while the actions regarding the on-site/off-site radiation measurement instrumentation are discussed in Section 4.12.3.

4.11. ENHANCEMENTS TO MITIGATE CONSEQUENCES OF SEVERE ACCIDENTS

As discussed in Section 4.8, the NPPs have implemented various actions to ensure a reliable confinement function for BDBAs to prevent significant release of radioactive material to the environment. However, during the Fukushima Daiichi accident the failure of containment venting, and the damage of the reactor building resulting from a hydrogen explosion, led to a significant release of radioactive material to the environment. While the most important objective of Level 4 DiD is the protection of the confinement for mitigation of the consequences of an accident, it was shown that this objective of Level 4 could not be successful. In case of the containment barrier fails, one mitigating action in Level 5 can be that to minimize the significant initial releases to reduce the radiological consequences on the public and environment in and around the plant and at the local, regional and global levels.

Based on this observation on the mitigating consequences of severe accident, the NPPs have implemented (or have been exploring) several actions at the Level 4 and Level 5 DiD. This section discusses the action to reduce and minimize the radiological consequences of core and/or containment failure.

4.11.1. Adding means to prevent corium melt-through of containment

One of the lessons learned from the Fukushima Daiichi accident was the monitoring, prevention and mitigation of ex-vessel phase of a severe accident. Depending on the regulatory framework and vintage of the nuclear power plant units, the actions form lessons learned varied widely across the Member States. Drivers for those actions were typically based on regulatory requests but the type and extent of the modifications were mainly decided by the NPPs based on the value/impact assessments (even if they were not included in the required safety measures). The visible actions taken by the nuclear power plants included:

— Installation of stationary alternative low pressure injection pump, having portable alternative low pressure injection pump, fire extinguishing pump, self-cooling modification of conventional pump for flooding the containment basemat;
— Installation of redundant instrumentation to detect vessel break by core melt through;
— Installation of redundant instrumentation to detect hydrogen (e.g. to measure hydrogen recombiner temperature);

— Installation of installing corium shield/core catchers in some NPPs that do not have one;
— Installation of corium spreaders in NPPs that installation of corium shield/core catchers are physically impossible which, for example, was the case for older vintage 900 MW(e) PWRs in France for which regulatory body requested to have similar (but not the same) molten core retention capabilities for all plants;
— Implementation of a passive solution flooding reactor pit (instead of a core catcher) to prevent corium melt-trough of the basemat in earlier vintage of NPPs;
— Installation of a containment ultimate cooling (heat removal) system for residual heat removal without opening (venting) the containment building enhancement to mitigate severe accidents as a measure to have similar impact of core catcher for ex-vessel phase instead of retrofitting or installation of a core catcher in existing design [11];
— In-vessel retention by flooding the reactor cavity (See Section 4.8.1).

There have been several challenges and proven solution to those as follows:

— Implementation of a passive solution flooding reactor pit (instead of a core catcher) requires extensive studies and tests using molten corium concrete interaction (MCCI) models and ex-vessel corium behaviour analyses. They also depend on the material of containment basemat, e.g. silica concrete or not, particularly concern for older vintage units.
— Redundant instrumentation to detect vessel break and hydrogen requires qualification for very high temperature, high humidity and high radiation. This necessitates survivability assessments for equipment and instrumentation for severe accident management.
— Installation of corium shield/core catchers depends on the available space and accessibility which necessitated alternative solutions to install spread areas with passive cooling for ex-vessel phase if the space available did not exist.
— Personal dose concerns in installation of redundant instrumentation to detect vessel break and hydrogen which requires work in high radiation or high contamination areas, e.g. the reactor pit.

4.11.2. Adding capability to prevent/minimize atmospheric dispersion of radioactive material

One of the lessons learned from the Fukushima Daiichi accident was to prevent or minimize radiological release during a severe accident. These actions were decided and selected by NPP operating organizations based on the Level 4 and Level 5 DiD achievement goals to mitigate (minimize/reduce) to release and dispersion of gaseous radioactive material to environment. Some of these actions taken by the nuclear power plants included:

[11] This modification typically included a low pressure primary injection pump with water source (to maintain pressure and subcriticality) for ultimate containment cooling system and a large mobile heat exchanger that is connected to the primary, secondary or backup UHS. The new system provides two days for extreme hazards that could damage the infrastructure (allowing support to be brought in by the off-site national response centre); and enable to flood reactor pit before severe accident.

— Adding filtered containment vent system (see Section 4.8.2);
— Purchase of zeolite to be utilized for the prevention of gaseous iodine release;
— Adding iodine retention system, such as the installation of sodium tetraborate baskets, in the containment to reduce the emission of gaseous iodine;
— Purchase of large capacity pump trucks, water cannon vehicles to externally spray containment, plumes or soil to supress radiological release.

The challenges with the implementation of these action reported in the survey were as follows:

— Determining the best technology for iodine filter containment vent according to given conditions/scenarios (see also Section 4.8.2 and Section 10.1.5);
— Finding available space for sodium tetraborate baskets, which are very large items, at the bottom of reactor building necessitating other solutions using different designs;
— Determining types and deployment time and locations for pump trucks, water cannon vehicles, particularly when considering multi-unit accidents and damage to the infrastructure at the site.

4.11.3. Adding measures to manage water sources and discharges

Primary and secondary system injections with available and suitable makeup water was proven to be necessary for several aspects of accident management in Fukushima Daiichi NGS, with the objectives of minimizing core cooling system inventory loss and maximizing and sustaining proper makeup water (see Sections 4.6.1 and Section 4.6.2). Similar objectives are also applicable to cooling of spent fuel and containment, as discussed in Section 4.7.1 and Section 4.8.1, respectively.

Furthermore, one of the challenges during the Fukushima Daiichi accident has been to manage (i.e. treat, store, reduce/minimize) radioactive or brackish water.

Therefore, the NPPs have implemented several measures based on these lessons learned considering:

— Water treatment is needed to maintain a source of quality water for RCS, SG, reactor pressure vessel (RPV) and SFP makeup;
— Portable and permanent water storage may be required to store or to contain potentially radioactive water pumped from contaminated areas of the facility, or to store water that has been produced from purification equipment if other on-site storage tanks have been damaged by the initiating event.

Some of these actions taken by the nuclear power plants included adding water filtration means, such as:

— Procurement of water treatment equipment (e.g. for treatment of radioactive, brackish, salty water) which is modular in nature and can be shipped, arranged and assembled on site based on the identified need of the specific site.
— Purchasing water storage bladders as an effective method to store a large quantity of water (including radioactive and/or purified) on site. Since the bladders are stored and

transported collapsed and folded, they are an effective way to rapidly deploy a mechanism for water storage.

- Portable water polishing system along with filters and cation exchangers to remove the radioactive particulates like Cs and Sr in order to mitigate/prevent high radiation levels in the fuel building due to SFP water activity.
- Several NPPs have also purchased silt fence to be deployed to prevent spread of radioactive and brackish water.

Challenges and lessons learned from the implementation of these actions and the consideration of potential strategies for long term sustainability that were reported by the survey responding NPPs noted:

- The portable water treatment equipment/skid may require their own dedicated portable generators to provide continuous power to them which is additional cost (as reported in the survey responses, a typical water treatment unit costs approximately US $1.2 million excluding the cost of additional membrane cartridges and the cost of a portable power supply);
- Water treatment component deployment depends on the final use of the filtration system. For example, SG makeup water may need parallel operation of reverse osmosis units, while the reactor makeup water could be a series operation of reverse osmosis units;
- The membrane cartridges (installed in the reverse osmosis units when they arrive at the site) determine the output flow and water quality;
- Bladders can fail and cause consequential events (e.g. flooding and radioactive release);
- Development of good water management plan in the short and medium term, considering type of water to process, recycle and/or store, would be beneficial.

4.12. ENHANCEMENTS TO EMERGENCY MANAGEMENT SYSTEMS

Technical Volume 2 and Technical Volume 3 of Ref. [1] extensively discussed the issues faced in accident and emergency management during the Fukushima Daiichi accident. Particularly those from:

- Disrupted regional infrastructure, including serious deficiencies in communication, transport and utilities, that reduced the effectiveness of internal and external support;
- Incompatible or insufficient equipment received from various off-site sources that could not provide support or supplement the existing on-site capabilities (e.g. mismatched fittings, connectors, etc.) [12];
- Huge amount of rubble that hindered or obstructed on-site response measures and recovery actions;

[12] For example, some portable and heavy equipment was provided to the Fukushima Daiichi NPP by different external organizations, including the Self Defense Forces (SDF) and local and prefectural firefighter units, which in some cases could not be used owing to mismatch connectors [1].

— Lack of effective communication, lighting and high radiation levels that significantly extended the time needed to perform verification and control actions and the coordination of containment venting with emergency planning actions;
— Inhabitability of control room, plant buildings, including on-site and off-site emergency response centre;
— Non-functioning systems, communications and monitoring equipment for providing essential information for both on-site and off-site responses.

This section provides the actions taken based on those observations and lessons learned on emergency response and management of the accident.

4.12.1. Communication devices

The communication systems during any emergency include public address (PA) system, mobile/satellite/cable/sound-powered phones, cable facsimiles, radios (short or ultrashort wave), pagers, handsets for mobile communications standards (e.g. 2G, 3G or 4G that allows wireless access of portable electronic devices, mobile phones, laptops pads to the internet), Code Division Multiple Access/Time Division Multiple Access (CDMA/TDMA) clusters, Global Positioning System (GPS), etc., for communication of voice, text, short message, technical, environmental, radiological, personnel health data and other information. The communication system is essential during an emergency for dispatch, report, access support, command and control functions performed internally (on-site) and externally (off-site) [121].

Some of these systems provide communication between control room(s), command and control centre(s) and field vehicles and workers internally (on-site), typically through an integrated unit, such as Private Branch Exchange (PBX) system. Some equipment and media, such as voice communication, dispatch order, or transmittal of plant monitoring data, video conferencing, etc., between the plant, emergency support and external (off-site) response centres.

More importantly, nearly all of these devices require power to operate either from station AC/DC power systems — which may be unavailable during prolonged SBO — or enclosed, attached or portable batteries, which have limited charge.

In response to lessons learned from the Fukushima Daiichi accident communication issues, regulatory bodies in most Member States required nuclear power plants assess the reliability and resilience of their existing communication system capabilities. The requirements also included to have NPPs to identify potential vulnerabilities during ELAP events and/or events that could damage to external telecommunication infrastructure, including power and cell towers) to implement actions to enhance reliability and resilience of communication systems.

In general, the requirements established in Member States were descriptive (instead of being prescriptive) which allowed the operating organizations or industry chose method for enhancing reliability. For example, in the USA, communication strategies follow the industry guideline [122] which required that NPP should maintain the capability to perform critical communications during and following an event that results in an ELAP. The critical time to provide communications was determined by the industry as the first 10 hours after a BDBEE. Such critical time requirement was not defined by the regulatory order [53] which mainly instructed in general terms: *"Provide an assessment of the current communications systems and equipment used during an emergency event to identify any enhancements that may be needed to ensure communications are maintained during a large scale natural event"* [53].

Based on the assessments, for enhancing communication and to make communications more resilient to the disruptions during a severe or extreme accident in performing accident and emergency management, the NPPs have taken the following actions:

- Purchase of more communications equipment (vehicles, radios, spare batteries, chargers, hardwired telephones, cellular telephones, satellite telephones, radios, pagers, and spare batteries, etc.) same or similar to the existing ones to ensure availability and redundancy;
- Implementation of satellite technology, which included on or more of:
  - Purchasing new satellite phones (including spare batteries);
  - Installation of permanent satellite antennas mounted on each unit, in the control room, and on-site and off-site emergency centres;
  - Integration of satellite phones into the site telephone system to enable phone communication with off-site entities (e.g. between the plant and off-site government and regulatory agencies) during a prolonged SBO;
- Establishing an information technology (IT) redundancy project providing further enhancements to the emergency management centre (EMC) in the form of a stand-alone IT data centre and very small aperture terminal (VSAT) system (satellite internet connectivity);
- Setting up a suitable communication system on the site in order to manage situations involving total loss of electrical power (e.g. sound-powered telephones, Genephones);
- Upgrading the existing installed communication equipment, enhancing the robustness of the existing communication equipment to seismic and flooding events;
- Deployment of new line-of-sight radios;
- Implementation of new (or reliability enhancement of existing) PBX system;
- Installation of new or additional Genephones on key areas on site;
- Enhancement of power supply for communication equipment by enabling to power the communication equipment with portable and stationary diesel generators;
- Purchasing new mobile communications vehicles to establish/re-establish both on-site and off-site communications;
- Deployment of mobile emergency crisis centre (ECC) and Access Control Points in mobile pods as backup in case that on-site facilities are disabled;
- Connection of communication devices and data acquisition system to the integrated nuclear emergency preparedness network;
- Establishment of direct communication lines with earthquake, meteorology, hydrology and marine observation authorities;
- Initiation of research and development (R&D) for exploring more effective communication during severe conditions with limited capabilities.

To determine what is sufficient and adequate communication to ensure to work under anticipated situations was challenging for the NPPs. This necessitates continuing research to find the best and most reliable technology and communication means. For the future sustainability of communication system, it is beneficial to continue R&D for exploring more effective communication during severe conditions with limited capabilities.

Selection of the best and most reliable technology and communication sufficient and adequate communication was also a reason for the wide range of cost of communication system enhancements from one Member State to another (and from one plant to another). The lowest cost reported in the survey was around US $15 000 and the highest cost was almost US $2 million per nuclear power plant unit. The cost was primarily dependent on the plant specific communication strategy to comply with regulatory minimum requirements and industry guidance as driven by the plants communication assessment.

4.12.2. Emergency management facilities on-site or near site

Although the on-site emergency response centre (ERC) was located in the seismically isolated building, which was fitted with special features, including an autonomous electrical power supply and ventilation systems with filtration devices[13], the off-site emergency response centre, which is located 5 km from the NPP, did not have adequate protection, as discussed in Ref. [1]:

> "The activation of the emergency Off-site Centre, located 5 km from the Fukushima Daiichi NPP, was difficult because of extensive infrastructure damage caused by the earthquake and tsunami. Within a few days, it became necessary to evacuate the Off-site Centre due to adverse radiological conditions. The OFC was located within 5 km of the Fukushima Daiichi NPP and this resulted in a number of additional difficulties. For example, after 13 March, shortages of food, water and fuel occurred, because normal deliveries within the evacuation zone were suspended" [1].

Hence, one important lesson earned from the accident was that the on-site and off-site emergency response facilities need to be accessible, functional, operable, habitable and protected under emergency conditions, including radiological and environmental conditions, resulting from severe external events at or around NPPs.

Based on this lesson learned, the NPPs assessed their on-site and off-site ERCs, resulting in a range of actions from small modifications to enhance the protection of existing facilities against the severe extreme events (e.g. providing flood protection), to building new facilities that are resistant to such events. Some of the reported actions by NPPs include:

- Construction of new EMC building which is:
  - Higher civil seismic design level and consistent with the site SSE, as well as protected against DBF plus PMP condition) to manage emergency situation to be used as ERC for crisis team;
  - Fully equipped with emergency response equipment including electronic dosimetry, satellite communication, external broadcast capability and backup power.
- Enhancements to resilience of the existing emergency facilities, e.g. emergency control and alternative monitoring centres resilience to increase standing against the severe accident conditions;

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[13] This on-site building had been constructed (finished right before the accident) as a result of lessons learned from the experience of the Kashiwazaki-Kariwa NPP following the Niigata-Chuetsu-Oki earthquake in 2007, and its use enabled mitigatory actions to continue at the site during the response to the accident [1].

— New backup emergency control centre near the site emergency response centre (for example, in the nearest township (3–5 miles away);
— A mobile command centre with an independent power supply and emergency communications capability in place;
— Information technology (IT) redundancy which further enhanced the EMC in the form of a stand-alone IT data centre and VSAT system (satellite internet connectivity);
— Deployment of mobile ECC and Access Control Points in mobile pods as backup in case that on-site facilities are disabled;
— Purchase of new vehicles for emergency response to transport personnel and equipment;
— Purchasing equipment to operate in high radiation areas, such as remote controlled robots.

4.12.3. On-site and off-site detection and monitoring systems

Prompt monitoring of the environment and timely radiological assessment capabilities during an accident provide real time radiation readings and enable the verification and validation of radiological conditions. This information facilitates the awareness of existing and predicted on-site and off-site dose levels. It is, therefore, essential for accident management, as it ensures radiological protection and prevention of the plant personnel in dispatching for actions and performing on-site response and recovery activities. It is also a critical part of emergency response, for example, in developing actions and strategies to minimize the radiological impact on the public at the local and regional levels, particularly for those are at great risk and require prompt protection (e.g. in the immediate vicinity of the site, precautionary action zone (PAZ) and urgent protective action planning zone (UPZ)).

At Fukushima Daiichi NPP and its immediate vicinity, initial environmental monitoring activities had many difficulties owing to the damaged infrastructure by the earthquake and tsunami. Infrastructure damage impacts included: Destroyed or washed away monitoring devices; dangerous conditions on the roads which caused vehicles being abandoned or incapacitated due to falling into cracks, having flat tires, running out of fuel, etc.; loss of electrical power that resulted in ceasing the operation of radiation monitors that survived the natural events due to running out of backup power; loss of communication systems which hampered the conduct and communication of field operations. These resulted in incomplete or missing environmental condition data and assessments which contributed to the radiological risk to the plant personnel and the public. Particularly, unknown quantification and characterization of the amount and composition of radioactive material released to the environment and movement of the plume resulted in risky or ineffective action, such as sending personnel to the areas with unknown current or projected dose levels, evacuation of public from some areas, etc.

Therefore, ensuring the reliability, resilience and effectiveness of real time radiation monitoring (on-site and off-site) under extreme events was a lesson learned from the accident, including supply of power and communication capabilities. Accordingly, the NPPs have taken actions to enhance radiological and environmental monitoring for the protection of workers and the public, such as:

- Installation of automated station boundary radiation monitoring system;
- Installation of remote monitoring system equipment that are fully redundant to support operation during extreme infrastructure outages;
- Installation of battery operated detectors that are designed to sustain detector operation for almost a week on batteries, and augmentation of detectors with solar panel trickle charge to sustain detector operation indefinitely;
- Utilization of cellular communication systems to ensure redundancy (for example, utilization of a satellite communication service which provides a back up to cellular communication system in order to transmit data) and continued connectivity to feed almost real time (e.g. every 15 minutes) from the detector via cellular service to a fully redundant data host server, and to an analytical engine where it is analysed and presented via a web browser within the EMC;
- Delivery of forewarning systems for severe weather/sea conditions, and improvement of preparedness instructions;
- Installation of tsunami/flood observation facilities: tide level measuring instrument at ground, water level measuring instrument at sea water intake pit, surveillance camera;
- Additional monitoring vehicles, boats (for ocean-sited plants);
- Purchase of portable monitoring posts;
- Purchase of portable meteorological posts.

As an example of the implementation of radiological and environmental monitoring enhancements, a survey responder provided the details of an automated station boundary radiation monitoring system as:

> *"The monitoring system consisted of 28 offsite gamma monitors and 16 onsite detectors. These air samplers were to augment the existing station environmental tritium air monitors in order to provide more detailed data in terms of airborne and ground deposition. It should be noted that the number of installed monitors and detectors was NPPs decision that was beyond the requirement to install eight air particulate monitors (by the end of 2015)."*

The same survey responder further provided the details of remote monitoring system as:

> *"Including 44 gamma detectors (16 onsite detectors with the remaining 28 within the 10 km area around the site). The battery operated detectors are designed to sustain detector operation for up to five days and are augmented with solar panel trickle charge to sustain detector operation indefinitely."*

4.12.4. Habitability systems for control room and other emergency facilities

The Fukushima Daiichi accident highlighted the importance of protecting the locations where the emergency actions are taken, controlled and instructed, i.e. the MCR, the supplementary control room and emergency response facilities. These locations require accessibility, functionality and habitability under all plant condition, including extreme event and severe accident conditions, as specifically required in Paragraph 5.60 of the Requirement 32, Design for optimal operator performance, of IAEA Safety Standards Series No. SSR-2/1 [15]:

> "The design shall be such as to ensure that, following an event affecting the plant, environmental conditions in the control room or the supplementary control room and in locations on the access route to the supplementary control room do not compromise the protection and safety of the operating personnel." [15].

Also, the Requirements 65, 66 and 67 of Ref. [15] further list the measures to be taken for the protection of occupants of the control room, supplementary control room and emergency response facilities, respectively.

Although the MCRs at the Fukushima Daiichi NPP were designed with systems to ensure habitability, because of the loss of power, these systems ceased operation resulting in loss of habitability systems. This loss of the MCR habitability systems made it difficult for the MCR staff to monitor and control the plant and hindered necessary actions to be taken by operators. Reference [1] also adds:

> "In addition, direct radiation exposure in the MCR, after the beginning of core degradation, was not preventable due to its location. The operators inside the MCR were subjected to the added stress of exposure to radiation" [1].

In response to this observation and lesson learned, the NPPs first evaluated the habitability of existing MCRs and other control facilities under severe accident conditions arising from DEC/severe accidents, mainly focusing on, but not limited to severe external events and prolonged SBO.

These plant control facility habitability evaluations particularly included radiation, heat up and hydrogen concentration impacts (see also Section 4.3 for battery room habitability). Where a need is identified by the evaluations, NPPs implemented upgrades for strengthening the habitability of existing MCRs and other control facilities, which included:

— Construction of emergency room centre for crisis team with own HVAC system with dedicated diesel generator to provide days of sustained autonomy.
— Purchase of portable ventilation equipment (fans, ducts, etc.) to ensure MCR and battery room habitability. These portable ventilation fans and ductwork assist with habitability, equipment availability and controlling battery room temperature.
— Establishing administrative controls for exclusion of contaminated articles from the MCR.
— Installing lighting for control room supplied by a dedicated alternative AC power supply.

As reported by the survey responders, some of the challenges encountered during the decision making or implementation of actions taken to strengthen/upgrade the habitability of existing MCRs and other control facilities included:

- Construction of a new EMC or satellite CR buildings on-site while units are operating required careful work control, particular attention to the work equipment near overhead power lines;
- Storage of mobile means, e.g. radiologic protection devices, portable ventilation equipment (fans, ducts, etc.), that should be adequate for MCR access and use;
- Procedures and validations to ensure prompt action to connect generators to portable ventilation equipment;
- Procurement of portable oxygen bottles and masks (for habitability while isolated for prevention from intake of airborne radioactivity).

4.12.5. Lighting

One of the consequential human performance issues during the Fukushima Daiichi accident was that the plant personnel, including the MCR staff performed critical control or mitigation functions in total darkness upon the loss of all power at the site. When applicable and available, flashlights, car batteries and other means were utilized to overcome dark work environment. Based on this learning, the NPPs assessed the lighting conditions inside and outside buildings at the site during extreme events and prolonged SBO. As a result of these assessment enhancement action for lighting at the site were taken, including:

- Installing lighting for safety passage supplied by dedicated emergency power supply.
- Purchasing portable lighting towers (typically priced around US $5000 for each portable lighting tower) for providing area lighting for continued operation of portable and emergency response equipment and personnel. The portable light towers are self-contained units and need no support system with the exception of refuelling operations.
- Installation of lighting on the portable equipment which are powered by the generators (or batteries) of or by the equipment.

4.12.6. Protected storage for mobile equipment

To support the accident management strategies that necessitate deployment of portable equipment maintaining critical safety functions and mitigating actions, to construct a building, or multiple buildings, became necessary for:

- Storing the equipment in a housing that is protected from the applicable hazards;
- Deploying and operating them in a secure manner.

These protected buildings were placed on-site (for the equipment that is needed immediately after the event), near site or off-site (for the equipment that could be deployed later as the accident progresses). Also, in some Member States, the nuclear power plants could build one building that is protected from all applicable external events, or two separate buildings if you can show at least one would survive for all applicable external events.

The following is the list of responses received from nuclear power plants who participated in the survey:

— Construction of buildings for storing stationary and mobile power supplies;

— Construction of emergency power platforms (includes connection, large generators, storage pads and equipment) to be utilized in provision of power supply to critical equipment;

— Construction of on-site building protected from external events to store hardware;

— Construction of temperature controlled storage building to house vehicles, generators, and pumps, etc. since the plant is subject to extreme low temperatures;

— Installation of a new nearby off-site deployable backup equipment storage;

— Installation of emergency power platforms (which includes connection, large generators, storage pads and equipment) to speed up the equipment deployment;

— Setting up arrangements for deployment of DBUE from staging point to on-site laydown/deployment/staging areas.

There were some challenges on the selection of the building with appropriate properties and criteria, such as:

— Deciding on whether to construct a bunkered location on-site or non-bunkered off-site. Choose to use a storage location off-site that would not likely be exposed to the same external hazard as the plant, e.g. located on a hill to avoid flooding. The building has been assessed for high winds and seismic events beyond the design basis level.

— Using a tent facility that would protect the equipment from the outside environment but would blow away or collapse during a high wind windstorm or seismic event without damaging the equipment stored inside.

— Deciding on whether to construct one building that is protected from all applicable external events, or two separate buildings if you can show at least one would survive for all applicable external events:
  - Constructed two separated buildings and various wind load calculations to support new and existing mods;
  - Constructed single building on-site.

Certainly, the site specific hazards dictated the building decisions and the associated costs. Costs are dependent on whether a utility builds one, two, three or more buildings, or uses an existing building/structure or Sea-Land containers to store portable equipment, etc. Depending on which option chosen the costs varied. For example:

— For a large single building designed for all site applicable hazards from US $2 million per nuclear power plant unit using Sea-Land containers to US $26 million;

— Based on the input from one nuclear power plant which responded to survey stating that they constructed two separated buildings, the cost was US $750 000 for each building, including all analyses and preparation to support new and existing modifications.

4.12.7. Capabilities for clean-up of debris caused by external events

When the tsunami flooded the Fukushima Daiichi site, it covered much of it with sand, silt and debris. The mitigation and accident management actions were hindered by, among others, the presence of a huge amount of this debris, as plant personnel worked among debris and the debris block roads and deployment locations for mitigation support vehicles. Outside the site, fallen power lines, rocks, trees also adversely affected the ingress/egress of personnel and vehicles. This observation highlighted the importance of debris removal following an external event, as noted in Technical Volume 1 of Ref. [1]:

> "After a four-hour effort of searching for and locating the FP water injection port among the debris, establishing the connection and getting the fire truck to the connection, the alternative water injection from the freshwater tank to the Unit 1 reactor via the fire protection system by using fire engine started about 12.5 hours after the SBO" [1].

The earthquake and tsunami were the external events creating debris; however, as discussed in Section 4.1.3, other events can create other debris or disruption, such as heavy snow or ice.

Furthermore, following the Fukushima accident, nearly all NPPs established protection, mitigation and recovery actions some of which relies on portable/mobile equipment that needs to be transported within the site or from off-site facilities. This strongly necessitated the provision of equipment/vehicles capable of removing debris and damage that might obstruct emergency teams to deliver the emergency equipment. Availability and operability of such vehicles is very beneficial to take timely actions in order to deploy portable equipment when needed in accordance with the established mitigation strategies. Survey responders indicated that debris removal times typically are in the range of 2–6 hours depending on the site, based on drills conducted, which needs to be considered in establishing strategies and taking timely actions.

Therefore, the NPPs have taken action to ensure availability and operability of debris removal vehicles, equipment and means when and why they are needed, with an optimised cost, as a typical debris removal equipment purchase package ranged from US $400 000 to US $800 000 depending on the anticipated debris type. A data for a unit price, for example, a front-end loader for debris removal costs around US $225 000.

Accordingly, actions taken by the NPPs for debris removal equipment (and the cost of actions) varied from plant to plant depending on the potential debris type, as well as needed functions from the equipment, based on the applicable hazards and conditions:

- Common action noted by various nuclear power plants in the survey was purchasing adequate amount of vehicle and equipment for redundancy and diversity, such as front-end- and pay-loaders, bulldozers, tractors with different front attachments for different debris types, etc.;
- Purchasing snow removal equipment including redundant snow blades and miscellaneous support equipment (e.g. large trucks with snowplough attached, tractors with different attachments on the fronts, etc.);
- Procurement of accessories for vehicles, including starting battery trickle charger to maintain the batteries at full charge and diesel fuel additives to prevent fuel from gelling, to ensure start of the vehicles in low temperature;
- Several NPPs chose to use protection against LIP to prevent debris by flooding;
- Some plants also took actions related to the infrastructure and tools, such as strengthening seismic resistance of access routes; and utilization of robot vehicles.

Notable challenges with, and lessons learned from, debris removal plans and actions reported in the survey response by the NPPs, included:

— Need for readily availability of equipment requires storing the equipment on-site;
— Need for readily availability of qualified staff to operate the equipment which necessitates training EPS shift crews;
— A through debris assessment is needed to be performed to understand potential debris patterns for purchasing vehicles and equipment accordingly;
— Need for administrative controls or provisions to perform live/dead/live checks on downed power lines in and around the site;
— Overhead power lines require planning out and establishing multiple travel paths, including at least one path with no power line interference;
— Seismic resistance of access routes which require enhancement to be used after a seismic event.

4.12.8. Portable equipment refuelling capabilities

Fukushima Daiichi accident somehow showed that vehicles used for protection and mitigation actions needed refuelling which in some cases interrupted the necessary action resulting in further degradation of the accident.

Although the lessons learned from the accident made NPPs consider action for fuel supply, the primary consideration for fuelling/refuelling of equipment was the installed/purchased for post-Fukushima actions. Following the Fukushima accident, nearly all NPPs established protection, mitigation and recovery strategies which rely on portable/mobile pumps, generators, large capacity pump trucks, fire engines, water cannon vehicles, etc., as discussed so far, in Section 4. This further necessitated the provision of fuelling/refuelling of equipment/vehicles in order to deploy and operate portable equipment and vehicles. Therefore, the NPPs took actions to ensure a reliable flow of diesel fuel is available (here, the survey responders also noted that it was their preferred tactic to use the diesel fuel located on site, to the extent practical) to maintain the equipment running to prevent service disruption:

— Identifying and implementing a cost effective[14] and reliable method of transferring fuel from the plant's EDG fuel oil storage tanks (as well as other available storage tanks) to the portable diesel powered equipment;
— Purchasing a fuel delivery truck(s) for maintaining fuel inventory for the entire coping period to ensure the ability to effectively refuel all portable emergency equipment during a BDBE;
— Procuring a small (e.g. 20 kW) backup generator for pumping fuel from various storage facilities;
— Purchasing small portable diesel fuel tank with pumps that can be loaded onto small trucks for refuelling efforts.

4.12.9. Shared off-site support centres

During the Fukushima Daiichi accident, disrupted regional infrastructure reduced the effectiveness of internal and external support in delivering equipment and human resources. Furthermore, incompatible or insufficient equipment received from off-site sources could not

[14] For example, a DG fuel tie-in modification was not the cheapest (costing approximately US $1 million per unit) but it was deemed that in the long term it would be more cost effective.

provide support or supplement the existing on-site capabilities (e.g. mismatched fittings, connectors, etc.). Analysing such deficiencies, NPP operating organizations, notably in two Member States, France and the USA, considered and decided to establish central and/or regional response organizations that would timely and effectively provide necessary equipment or human resources to the NPP site(s) that are in their country. These off-site entities will provide support activities in case of nuclear emergency by dispatching equipment/staff and tackle on the disaster under harsh conditions (e.g. high radiation), in cooperation with the owner/operator organizations:

— Nuclear Rapid Intervention Force (Force d'Action Rapide du Nucléaire, FARN) in France:

  French operating organization EDF has established the crisis management organisation, FARN, shortly after the Fukushima Daiichi accident. The objective of the FARN is to maintain a competent national team with the capability of deploying additional equipment and providing human assistance at an accident site within 24 hours (noting that numerous emergency exercises that have been conducted by/from regional bases have shown that needed support could be sent to a NPP without difficulties, starting the operations on site within 12 hours after their mobilisation). FARN is also capable of complementary means of communication and radiological protections and measures in and nearly 300 people in five different off-site locations (four of them are four regional centre) to rescue all units in one site with the arrival on-site in 24 hours. This rapid intervention organisation established four regional centres at Civaux, Paluel, Dampierre and Bugey NPPs. By 2015, the FARN has a capacity that could simultaneously respond to six reactors (e.g. entire Gravelines NPP site). FARN teams would provide support the accident site organization in implementing, monitoring and maintaining [123, 124]:

  - local accident and emergency management means (e.g. with connections, fittings, fuel supply, some minor maintenance, etc.);
  - regional accident and emergency management means (e.g. pumps, generators, compressors and telecommunication, transportation and handling systems).

  Overall, the FARN teams are to support the operation teams at the accident site (noting that they could take over the activities under certain conditions).

— National SAFER[15] Response Centers (NSRCs) in the USA:

  The U.S. has established and maintains two diverse and redundant equipment storage facilities identified as the NSRCs. These NSRCs are located in two geographically separate locations. The NSRCs contain portable equipment such as portable generators, portable pumps, hoses, and other supporting equipment to be used by licensees to meet the Phase 3 requirements of the NRC Mitigating Strategies Order. Established plans, in part through contracts with Federal Express (FedEx) Custom Critical (FCC) and commercial heavy-lift helicopter operators, allow either one of the NSRCs to provide this equipment to any nuclear power plant in the United States. In the U.S., the industry has also established two redundant control centres that will be manned and operated by the organization which runs the NSRCs to coordinate the response for any such event.

  The equipment cost only for two centres located in the USA was approximately US $55 million. This cost figure did not include two building construction and rental costs, the manpower for maintenance, the utilities etc.

[15] SAFER — Strategic Alliance for FLEX Emergency Response.

Several challenges were encountered and resolved during the establishment process of NSRCs, such as:

- Setting up large and long term contract through life management partner for storage, maintenance and transport, if required. This requires a corporate executive decision.
- Need for separate contracts for each method of transportation (e.g. truck, fixed or rotary wing aircraft, etc.) which requires a clearly defined strategy.
- Setting up arrangements and ownership for deployment of DBUE from staging point to on-site laydown areas. This ownership can be assigned to an internal (or external) organization, for example in one nuclear power plant, the “Turbine Management Group” delivers the equipment to the “Forward Deployment Service”.

5. ANALYTICAL AND MODELLING CHANGES

In the nuclear power plant safety analyses, analytical models and associated input and assumptions are used to determine or demonstrate the design function performance of SSCs designed to prevent, protect or mitigate the consequences of events resulting from internal and external hazards and occurrences. These models and analyses are mainly deterministic and, in some cases, supplemented and complemented by probabilistic assessments, for those events anticipated in design basis. Furthermore, plant safety analyses are based on the first principles of the neutronics, thermohydraulics, etc., and the variables within the model algorithms, input and assumptions are used to calibrate the model predictions to known or anticipated conditions (e.g. those based on the OLCs).

Deducted from the investigation of Fukushima Daiichi accident conditions, the ability of the models and analyses to accurately simulate and assess the SSC response was impacted by the conditions that far exceeded these known conditions occurring during accident, which can be classified as DEC for several aspects:

- The initiating event exceeded what was anticipated in the design basis of Fukushima Daiichi units.
- It was a severe accident initiated from multiple extreme events caused by multi-hazards.
- It affected multiple units in parallel or in series.
- It impacted the surrounding off-site infrastructure.

Therefore, the conditions used in design basis safety analyses (as input and assumptions and to calibrate the models) would not be applicable to the ranges of parameters during a severe accident which would invalidate the applicability of the model to simulate the event, as well as the adequacy of the results of the analysis utilizing the model.

Beyond all of that, the accident was the massive CCF induced by multi-hazards. For example, a common mode failure impaired both the AC and DC electrical systems depriving the operators of almost all means of control over multiple units [1]. This CCF was further that what was typically anticipated in the earlier assessment of DECs.

Reference [1] provided some lessons learned on the modelling and analyses of DBAs and DECs together with some potential actions to be taken to address them. These included, among others [1]:

- In the site evaluation, design and operation in relation to the potential occurrence of extreme external events of very low frequency but with high safety consequences, particularly, in the assessment of natural hazards, conservative estimations need to be applied and complex scenarios need to be assumed.
- Events from multiple external hazards (consequential or independent) affecting multiple units located on a site need to be considered.
- Design basis criteria with due account taken of complex scenarios of either extreme or severe natural hazards needs to be derived with enough conservatism to ensure adequate DiD concept and to ensure reasonable safety margins.

— In the assessment process for BDBEs/BDBAs, a systematic approach, including the complementary/supplementary utilization of PSA/PRA and appropriate ‘best estimate’ models, is beneficial.
— Deterministic and probabilistic BDB safety analyses need to be comprehensive and take into account both internal and external events.
— The combination of deterministic safety analysis (DSA) and PSA needs to be used to assess factors such as cliff edge effects, realistic equipment and personnel performance, and the relative contribution of various accident sequences to the overall plant risk.
— Comprehensive probabilistic and deterministic safety analyses using ‘best estimate’ methods/techniques need to be performed to confirm the capability of a plant to withstand applicable BDBEs and to provide a high degree of confidence in the robustness of the plant design.

This section discusses a variety of analyses and models for which the methods have been, or will be, changed in the light of the Fukushima Daiichi accident, particularly focusing on four visible areas of changes in modelling and analysis are discussed:

— Modelling of external hazards (earthquake, tsunami, flooding, volcano etc.);
— Modelling of reactor accident phenomena;
— Modelling of extended loss of AC power responses;
— Modelling with probabilistic approaches.

5.1. MODELLING OF EXTERNAL HAZARDS

Following the Fukushima Daiichi accident, there have been changes to analyses and models for the evaluation of natural hazards and their consideration for risk assessment and hazard profiles of nuclear power plants. In this section, the changes models and analyses for individual natural hazards are discussed. Consideration of combined effects of natural hazards (i.e. events that one hazard triggers another hazard in sequential manner, such as an earthquake causing flooding (e.g. tsunami) or fire, are discussed in Section 5.4.1.

In nearly all Member States, regulatory bodies agreed with the use of existing methods to re-evaluate seismic and external flooding hazards for understanding of cliff edge effects. Many of those methods were developed or updated prior to the Fukushima Daiichi accident and had already been applied, for example, to support licensing of new reactors in the USA, or for non-licensing reassessments that were called ‘trial analyses’ (see Technical Volume 2 of Ref. [1] for the discussion on the ‘trial analyses’) in Japan.

However, there were some visible differences in input and assumptions for these models, such as hazard frequencies to consider[16], and in some cases in the models themselves, such as applicability of existing deterministic and conservative methods and/or the adequacy of ‘best estimate’ methods, particularly for DEC/severe accidents.

Commonly noted by Member States, there were several technical and resource challenges and lessons learned in ensuring the applicability and adequate representation of the phenomena by the revised models, including:

[16] For example, assumed hazard frequencies ranged from 10^{-4} per year (e.g. for flooding), in some Member States to 10^{-7} per year for hurricane in a Member State.

— Inevitable differences between the results of new models and the existing ones whether they have been repeated or continuously upgraded over a large time span. When utilizing state-of-the-art models, it was nearly impossible to generate the same quantitative results as those that were generated by the original design and licensing studies. This was particularly a challenge when the models contained significant analytical uncertainties requiring the validity of analyses by adding various tests and model benchmarking.

— Finding and hiring qualified analysts such as experts, modellers.

— Building PRA models, particularly in establishing reliable and representative PRA methodology for multi-hazard, multi-unit scenarios (See also Section 5.4.1).

— Keen interest in bottom-up approach from various stakeholders, with a lot of focus on ever more elaborate characterisation of beyond design basis hazards, noting that the analytical uncertainties are very significant, but the overall response to Fukushima has been top-down.

5.1.1. Seismic modelling

As discussed in Section 4.1.1, there have been many analytical assessments to decide on post-Fukushima action to increase robustness of NPPs against the BDB seismic events. In the Post-Fukushima models and analyses, mainly the applicability of existing conventional and conservative methods, for example when *Ss* level rise or the magnitude of SSE increases, were questioned. Also, determining the magnitude of BDB earthquake and GMRS (see also Section 5.4.3 also for fragility analyses) was a main and extensive part of discussions (in some cases still in discussion) between the operating organizations and regulatory bodies regarding to the seismic criteria.

On the other hand, the frequency for the magnitude of BDBE had a consensus by nearly all Members States as 1 in 10 000 years event. Furthermore, in some Member States, the nuclear power plants had to reassess both the seismic source characterization (see Glossary) and the ground motion characterization (GMC), while in some, it was deemed to existing assessments are validated. Such assessments were supplemented by the a SPRA which varied from one country to another. For example:

— In order to ensure adequate margin and prevention, one approach was to ensure that there is no cliff edge, in France, EDF performed 20 000 cases to perform a probabilistic (statistical) approach for determining the cliff edge effect). Also, the additional margin of 50 per cent was added to the SSE criterion.

— In Japan, considered improvements to existing SPSA focused on the removal of some over and 'unnecessary' conservatism and the refinement of success criteria for plants systems in response to seismic event. Tsunami caused by and earthquake was also being tried in SPSA improvement [41].

— In the USA, NTTF recommendation on seismic event specified a probabilistic approach for re-evaluating design ground motion, GMRS [36]. Based on the results of the GMRS comparing with SSE, USNRC determined the necessity of further actions performing by licensee (e.g. if the GMRS is greater than SSE, a seismic PRA results should be submitted), For the nuclear power plant, if the GMRS is bounded by the SSE, a SPRA may not be required. However, a nuclear power plant can perform a seismic PRA to qualify their PRA model for per the regulatory guidance provided by the regulatory body [125].

— In some Member States, detailed SFP seismic analyses, including the structural assessment at high temperatures (e.g. evaluation of the structural response of the SFP structure to temperatures in excess of the design temperature, including an assessment of the maximum credible leak rate following any predicted structural damage), as well as the seismic analysis of structures above and around SFP, such as cranes, gantries, bridges, SFP canal, etc., were analysed for DECs and BDBAs.

5.1.2. Tsunami and flood modelling

As discussed in Section 4.1.2, there have been many analytical assessments to decide on post-Fukushima action to increase robustness of NPPs against the BDB flooding events, including tsunamis, tides, seiches, storm surges, heavy precipitation, waterspouts, downstream dam forming or upstream dam failures, snow melt, landslides into, water bodies, channel changes and work in the channel.

Existing tsunami models are typically constructed based on data and knowledge obtained from the historical tsunamis in the world, including the travels of tsunami across the oceans. They also reflect the information on local tectonic structure in efforts to evaluate/address plate boundary earthquake, tsunami earthquake and outer-slope faulting and volcano eruption and landslide of seabed [19, 59, 126]. It was understood in the post-analysis (joint inversion analysis using observed crust deformation and observed tsunami waveform) that the pattern of multiple segment failure (propagation path and time lag) significantly affects the local tsunami heights. Consequently, tsunami source model, depending on specifics, may need modelling of multiple segment failure [127].

Before the Fukushima Daiichi accident, the main evaluation topic in external flooding was a prediction of water level change due to a maximum possible flooding caused by tsunami. Thus, the prediction methods of tsunami propagation and flooding in horizontal directions had been developed. There were two key models in the prediction methods; first, the prediction model of the initial water level change due to earthquakes or landslides and second, numerical prediction by modelling propagations of water level changes in horizontal directions from sources to sites, such as the shallow water equation models and the approximation model for waves on a free moving fluid surface. For better estimation of the tsunami height, the effect of the breaking wave on the tsunami height had been investigated and applied to the propagation model.

It was highlighted by the tsunami, that hit the east coast of Japan, including Fukushima Daiichi plant, that the impacts of flooding on SSCs need to be considered and the evaluation needed to include the consideration of numerous events, such as:

— Submersion of important components such as emergency seawater pumps;
— Hydrodynamic force and the buoyancy on structures;
— Debris collisions on structures and components;
— Sedimentation of debris at or in intakes;
— Failure of water intake due to lowering water level.

In order to enable evaluation of those impacts on SSC important to safety, inundation/flooding simulation methods at the nuclear power plant site was developed by using two dimensional shallow water equation models and three-dimensional fluid dynamics models. Furthermore, evaluation models of hydrodynamic force with consideration of flooding flow types, as well as debris impact force have also been developed.

On the other hand, assessment of the consequences of impact on SSC important to safety caused by collision, transport, dispersion, and sedimentation of debris needs further development.

Regarding external flood hazards, a very useful compilation of methods was performed in the USA, by the regulatory body, USNRC, which included existing models, technical guidance, and context [128]. Further in this document, probabilistic, deterministic or hybrid analytical techniques, methods, and available hazard simulation models were presented for nuclear power plant owners' use. As such, Ref. [128], together with Ref. [60], was beneficial for nuclear power plant operating organizations in the reassessments of tsunami, surge or seiche hazard and to compare against the current design basis flood hazards. Similarly, hazard re-evaluation approaches were also provided for dam breaks and other flood mechanisms that may affect a nuclear power plant's flood protection or mitigation strategies in Ref. [128]. The regulatory body and industry developed processes and screening approaches to focus resources and analysis on those nuclear power plant sites with the greatest potential for safety improvements.

However, external flood PRA guidance is woefully underrepresented, the NPPs noted. It was generally acknowledged that further development of probabilistic flood hazard analysis (PFHA) methods and modelling is needed to advance the state of practice and reduce uncertainties to support realistic risk estimation. In comparison, PSHA approaches and methods and seismic PRA technology were viewed with significantly greater technical maturity than probabilistic approaches for external flood hazards and external flood PRA. Therefore, external flood PRA had limited use in the post-Fukushima assessments per the value/impact aspects[17] and the reliability but continue to be developed and used for other safety applications including the various ongoing efforts on a consensus guidance, for example by American Nuclear Society (ANS) and EPRI in the USA.

It was also noted that multiple flood mechanisms that require completely different strategies, procedures and equipment effectively count as another completely independent model. A full PRA model (not a screening level or bounding analysis) gets more complicated and costly by three main factors:

— Complex human actions (owing to not well understood/anticipated strategies that require extensive human actions and organization level responses and very limited experience in development of human failure probabilities).
— Non-standard equipment (i.e. difficulty of consideration and inclusion of non-standard equipment in the models in comparison to those with standard equipment).
— Multiple mechanisms affecting the site (i.e. modelling of multiple mechanisms affect a site, which generally requires different responses (human or equipment) are required to handle those, is effectively like developing separate models, e.g. two hazard curves, two sets of impacts in the fault trees, two human reliability assessments (HRAs), etc.

[17] To provide a perspective for the impact/value assessments, a flood hazard curve development costs, for example, in range from US $50 000 to US $500 000 which reflects the variety of flood mechanisms potentially affecting a nuclear power plant with complexity and extent (for example, if just LIP is considered, then a hazard curve development costs approximately US $50 000, but if storm surge, river flooding, etc. are to be considered, it could cost multiples of US $50 000. If a stochastic model is required to substitute simulations for time (actual recorded data), then such project could be in the US $500 000 range could take more than a year. Similarly, a PRA model development can range from US $250 000 to US $500 000 based on the number of human actions required and the set of equipment required for core damage prevention.

Lastly to note, there were several issues reported by the NPPs in their flood modelling and analyses, including:

— Beyond design basis flooding conditions had to be developed based on the 'most recent' data and criteria, such as PMF which is developed based upon the PMP; LIP; tsunami, hurricane, storm surge, etc., as applicable. It in many cases also necessitated recent flood maps, hazards catalogues, etc., as well as requiring external expert and committee consultations.
— Unclarity for the magnitude of maximum flood (e.g. 500-year flood or 10 000-year flood, etc.) resulted in delay of a determined and agreed criterion between regulatory bodies and licensees for the level of conformity;
— Difficulty in determining design basis and the likelihood/evaluation of reaching 'cliff edge' owing to the lack of updated/most recent information which may result in artificially restrictive/conservative evaluation (or conversely, it may reveal that the original design basis is non-conservative based on recent information);
— Complexity of characterisation of the coastal flooding hazard particularly in combined probability of hazards (for example, some nuclear power plants reported that the detailed review of the coastal flooding licensing basis revealed an oversimplification in the combined probability assessment of still water height and wave height required extensive and time intense determination);
— Determination of 'cliff edge' and the likelihood (and uncertainties) of reaching this cliff edge to justify precautionary actions which may be dependent on the initiating event (as shown, for example, by flooding of electric equipment room in Fukushima Daiichi Units 1–4 where the electric equipment submerged lost their function and flooding created a 'cliff edge'.
— Modelling to assess the effect of external flooding on the access to plant in very extreme flooding scenarios.

5.1.3. Modelling of other external hazards

As discussed in Section 4.1.3, there have been many analytical assessments to decide on post-Fukushima action to increase robustness of NPPs against other BDBEs than seismic and flooding including high wind and tornado events, off-site fires (e.g. forest, brush, chemical fires), extreme temperature (high and low) events, off-site and on-site landslides and avalanches due to extreme precipitation, geomagnetic storms, volcanic activity.

Regarding the treatment of other external hazards than seismic and flooding, there were extensive discussions and more differences on the approach by the regulatory bodies in accepting the applicability of existing methods or the adequacy of 'best estimate' methods for DEC/severe accidents. For example, in the USA, following the review of a variety of domestic and international documents and each nuclear power plant's existing resilience to these external hazards, the USNRC determined that for each nuclear power plant no additional actions were warranted. There, the USNRC used the present day methods and ongoing industry work, including approaches that were updated recently for tornadoes independent of Fukushima Daichi accident activities and the additional safety already achieved from other post-Fukushima actions to improve nuclear power plant capacity to endure a broad array of BDB natural phenomena offered through the FLEX program and other improvements (e.g. BWR severe accident capable and reliable hardened vents).

5.1.3.1. High winds modelling

It was generally acknowledged that further development of high winds probabilistic methods and modelling was needed to advance the state of practice and reduce uncertainties to support realistic risk estimation.

Not very many nuclear power plants have performed a high winds PRA mainly due to cost benefit analysis or the availability of models. For example, in the USA, a tornado PRA is currently over US $1 million and the only approved method is proprietary. Technical support organisation, EPRI, is currently performing research in the area of nuclear power plant high winds risk assessments.

Further noted issues while modelling and analysis included setting maximum wind speed and design wind speed considering standard tornado, implementation of missile analysis, implementation of crash analysis, implementation of tsunami walkdown.

5.1.3.2. Volcano modelling

Precise modelling of volcanic phenomena is essential to assess the diverse progress of potential hazards and profiling those. Recent advances in hazard determination are particularly noted in the modelling techniques in all three aspects:

- Ash dispersion model for forecasting the advection process in the atmosphere and deposition of ashfall;
- Ground deformation model for assessing the magma migration in subsurface crust;
- Comprehensive and conceptual model for predicting large caldera unrest which is based on the geological and geophysical observations at major volcanic systems.

Challenges faced in the analysis for volcanic hazard include:

- In modelling ashfall phenomenon, vertical distribution of volcanic ash above the vent and the height of the volcanic ash cloud still needs to be improved for the hazard assessment of future eruptions.
- In modelling large caldera unrest, which is the most preliminary stage of development among the models, research and developments are still maturing, including:
  - The studies on the sequence of eruption and the property of magmas that are effective for revealing a sign of generation, ascent, and accumulation of large silicic magma bodies;
  - Extensive geophysical monitoring on the seismicity and the ground deformation by Global Navigation Satellite System (GNSS) and/or Interferometric Synthetic Aperture Radar (InSAR) that is used to predict the upward migration of magma;
  - The studies of the seismic reflection and/or Controlled-Source Audio-frequency Magnetotellurics (CSAMT) methods that are used to clarify the subvolcanic structure with magma plumbing system.
- Validation of evaluation by benchmarking and tests.
- Determining an appropriate ash concentration in the air for the evaluation of filter choke.

5.2. MODELLING OF THE PHENOMENA OF THE REACTOR ACCIDENT

There are existing analytical models for severe accident phenomena and source term, such as MELCOR, MAAP, ASTEC, CONTAIN, and risk assessment tools for PRA. So far, post-Fukushima modelling of severe accidents has been heavily relying on the adjustments (tuning) of these existing models. However, while doing so, the shortcomings of these adjusted models have been reported when they are applied to the conditions that are outside the applicability of such models, such as multi-source and multi-unit conditions. On the other hand, one of the lessons learned from the Fukushima Daiichi accident was to analyse severe accident scenarios to ensure that operation and monitoring of facilities is guaranteed in severe accident situation. Therefore, improvement of existing models the severe accident models has been an on-going research and development activity. These developments are reported to be mainly in four areas:

- Enhancements to severe accident models, particularly for the evaluation of containment integrity.
- Source term modelling and atmospheric dispersion, for improvement of environmental impact evaluation methodology (e.g. atmospheric/oceanic dispersion analysis method).
- Multi-unit severe accident models, especially multi-unit PRA methods and plant simulator models. Some learnings reported by the survey response included, for example:
  - One NPP with multi-unit site stated that using severe accident models for multi-unit is not significantly different than those for single unit. For that site, an assessment of containment response modelling for multi-unit events using the MAAP code was completed. From this model, a determination of potential PRA modelling improvements for multi-unit events to segregate multi-unit sequences from single unit sequences were identified as:
    - *Scaled Containment Approach*: This approach scales the key parameters (e.g. containment volume and radioactive release) depending upon the number of units participating in the accident. For example, for a four-unit event, the volume of containment is reduced by a factor of four. This approach is most useful when the individual units progress more or less simultaneously through the accident sequence.
    - *Forcing Function Approach*: This approach couples the MAAP4-CANDU stand-alone containment model with external source flows to containment from each unit participating in the accident. The external source terms are generated from a MAAP4-CANDU simulation for a single unit. This approach is used when the timing of accident progressions and terminal configurations in each unit are not the same.
  - The simulator models with multi units affected may be unreliable and uncomprehensive due to uncertainties and unknown-unknowns. Therefore, drills and training to better deal with a multi-unit event scenario.
- Development of a PRA model to study of site accessibility to ensure operation and monitoring of facilities for severe accident scenarios due to unknown ingress/egress condition and overall infrastructure around the plant.

5.3. MODELLING OF EXTENDED LOSS OF AC POWER RESPONSES

There were several post-Fukushima actions that necessitated new analyses to evaluate SSC response during BDBEs. The operating organisations and regulatory bodies agreed that using best estimate methods (although some of these analyses for DBAs required approved deterministic methods) could be sufficient to demonstrate the functionality of those SSCs under BDB conditions, based on the generic analyses performed by internal or external technical support organisations, including technology owners groups, such as PWROG. Regulatory bodies' concurrence allowed the use of 'best estimate' codes, methodologies, analyses, as well as the nominal or realistic values. These analyses included, for example (see also Sections 4.3, 4.4 and 4.6):

- Battery life and load shed studies:
  - Perform assessment of battery duration;
  - Determination of load shedding equipment and timing.
- Room heat-up studies:
  - Battery room;
  - Control room;
  - Containment (whole and/or compartment);
  - DC equipment room, switchgear room, inverter room;
  - Reactor building (in BWR types);
  - TDAFWP pump room: (1) to confirm that TDAFWP will continue to operate during ELAP when pump room ventilation is lost; (2) to investigate failure analyses of the limiting components in the TDAFWP control cabinet; and (3) to determine operator actions to either prop open the TDAFWP pump room door or establish ventilation.
- Primary system response to determine timing and deployment of actions: Thermohydraulic assessment of accident management strategy to confirm timing and actions for coping scenarios to support the approved accident management strategy implementation to ensure that the timing and deployment of equipment would prevent core damage. It should be noted that to ensure the applicability of thermohydraulic models for such conditions and that the costs were challenging during the technical and financial decision making (the average cost for a 'best estimate' plant specific analysis varied from US $50 000 to US $350 000).

  These challenges were resolved by placing boundaries on the applicability of certain model (e.g. only up until two-phase flow in the primary loop) and limiting the number of analysis (by selecting those with largest value). For one nuclear power plant, for example, only two plant specific analyses, that had large conservatism, were performed to maximize the benefits from 'best estimate' analyses. Further assessment of sequences and timescales for protecting the pressure boundary in the event of total loss of cooling would be very complicated and uncertain fault escalation sequences. In this case, the use of expert panels to make informed judgements about what was likely to happen was beneficial to determine which analyses to perform.

— Spent fuel pool analyses:
  - Time to boil analysis for spent fuel pool;
  - Behaviour of the fuel and water in spent fuel pool in prolonged SBO situation;
  - Evaluation of the structural response of the SFP structure to temperatures in excess of the design temperature;
  - Assessment of the maximum credible leak rate following any predicted structural damage.

The best estimate analysis provided correct and realistic estimation of plant response and time critical actions. For example, one nuclear power plant in a Member State used a plant specific best estimate decay heat model in their ELAP analysis which provided a more realistic analysis. By removing the 2-sigma conservatism from the ANS decay heat curves [129], the nuclear power plant was able to gain extensive time margin for the need to refill the CST. The use of the best estimate/realistic analysis allowed the nuclear power plant to extend the time sensitive action for refilling the CST from approximately 16 hours to over 34 hours. This best estimate analysis was deemed acceptable by the regulatory body.

5.4. PROBABILISTIC RISK ASSESSMENT MODEL CHANGES

As noted in the IAEA's Fukushima Daiichi Accident [1], the weaknesses in the Fukushima Daiichi plant design could have been recognized by a more comprehensive PSA, as recommended by IAEA safety standards. Experts of PRA also recognized, from the existing models and their applicability of the events at Fukushima Daiichi accident, the needs for improvements in PRA methodology in the areas of assessment of extreme external hazards, assessment of multi-hazard, multi-unit, multi-source accident conditions, human reliability in harsh environment, reliability in execution of planned accident management in a complex and devastated environment. It would be safe and appropriate to say that improvements of PRA in these points are ongoing.

5.4.1. Multi-hazard/multi-unit/multi-source evaluation

Fukushima Daiichi accident was caused by multiple hazards (earthquake followed by tsunami) in which plant damage status was worsened by subsequent and/or consequent hazards as the core melt occurred in Unit 1, 2 and 3 by CCFs. Multiple hazard affecting the plants at the site included hydrogen explosion in Unit 3 damaging the equipment that were set up for providing coolant makeup to Unit 2 core just when the process of coolant makeup provision was about the begin (i.e. accident at one unit influencing the course of accident in another unit) and loss of coolant makeup capability to Unit 4 SFP (in a condition of full core discharge for shroud replacement), where leakage was suspected, might have added more source term release to the environment. What is expected from an advanced PRA model capable to assess all these complexities by the analysis of CCF leading to the event (such as a prolonged SBO), accident sequence analysis including interaction between units, system reliability analysis, in order correctly estimate risks as well as off-site release from multiple unit installation.

5.4.2. Human reliability assessment under harsh and/or stressful environment

Improvements of HRA model has been going on even before the Fukushima Daiichi accident and the accident provided a benchmark data for real human actions during a severe accident for further modelling changes. For example:

— French human and organizational reliability analysis in accident management (HORAAM), which is a ‘decision tree model’ which is mainly based on the hypothesis that the failure probability of a human action can be evaluated through factors, i.e. ‘influence factors’, that represent the human action framework. HOORAM was developed by the technical support organization to the French regulatory body, Institut de radioprotection et de sûreté nucléaire (IRSN), had only been based on the observation of nuclear emergency drills in France. IRSN had used HORAAM to generate HRA data for French NPPs, though without comparison to an actual severe accident data. The difficulties encountered during the Fukushima Daiichi accident provided a real accident human action data to compare model by a test study to evaluate key influencing factors to determine human reliability. As discussed in Ref. [130], the study confirmed that the choice and the ranking of the first four influencing factors, namely, decision period, information and measurement means, decision difficulty and difficulty for operators, were appropriate. The study also concluded, however, that other three influencing factors, namely, difficulty of the scenario, difficulties induced by the environmental conditions and degree of involvement of the crisis organization needed improvements. The study also concluded that the scope of HORAAM needs to be expanded to assess multi-unit sites (for site PSAs).

— In the USA, integrated decision tree human event analysis system (IDHEAS) [131] method was proposed before the Fukushima Daiichi accident to improve evaluation of human reliability, which depends on cognitive demands associated with the context and task demands of a particular situation. Fukushima Daiichi accident had shown various complex context factors that may influence HRA such as dark control room, loss of signals important to safety, devastated and complex environment created by multiple unit accident and hydrogen explosion. Development activity is ongoing for modelling cognitive/diagnostic failure and execution/recovery procedure to capture human reliability in accident situation.

Nevertheless, it had been already recognized before the accident that HRA in PRA has difficulties in capturing human behaviour in a severe accident situation with significant complexity. This is clearer now with the complexity of Fukushima Daiichi Accident conditions, i.e. damage to facilities/components by earthquake/tsunami and core damage in multiple units, and human actions. Therefore, the industry is working on changes to models, in light of data from Fukushima Daiichi accident, to build an appropriate and comprehensive HRA method.

5.4.3. Component fragility

In parallel to re-evaluation of GMRS described in Section 4.1.1 and Section 5.1.1, EPRI has conducted high frequency seismic testing program for components so as to avoid the further actions [132−134]. The program has verified the vulnerability of components subjected to vibration which contains high frequency components.

For high seismicity sites improvements of analysis model used for fragility evaluation have been desired as the remaining topics after the Fukushima actions. With respect to fragility evaluation, improvement of response modelling and evaluation of functionality are deemed necessary. The response analysis has to take into account the nonlinearity of structural materials and soil foundation and two and three dimensional effects. The SSI effect also affects the structural and in-structure response results. Thus, development of high fidelity analysis codes and models are essential elements for on-going and future R&Ds. With the capacity, seismic qualification of components has to be done by using shaking table tests and seismic experience data. In addition to the above, uncertainty and correlation are major issues for future research so as to improve the quality of seismic PRA.

5.4.4. Risk informed decision making models

In the beginning, the NPPs have not used risk informed decision making (RIDM) in deciding what action to take in response to lessons learned from the Fukushima Daiichi accident. However, some operating organizations later used a similar approach in the determination of the merits of actions being taken (See Section 8).

RIDM process can be conducted using a graded approach, i.e. the level of detail in any analysis needs to be proportional to the level of risk and complexity of the to be decided. In the decision making, RIDM takes into account not only solely technical fundamentals but also human intuition. Methods can include, but are not limited to, the use of PRA, the treatment of uncertainties, safety margin assessments. The process may consider multiple sources of information not included specifically in the assessment of risk as inputs to the decision process in addition to risk information. RIDM is invoked in many different venues, based on the management processes of the implementing organizational unit. These include safety review boards and panels, risk reviews or risk informed committees, engineering design challenge boards/panels, operational/operations decision forums, configuration management processes, among others.

In the U.S., Regulatory Guide 1.174 [135], describes the five principles of RIDM. These are as follows:

- "The proposed change meets the current regulations unless it is explicitly related to a requested exemption;
- The proposed change is consistent with the DiD philosophy;
- The proposed change maintains sufficient safety margins;
- When proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the regulatory bodies safety goals;
- The impact of the proposed change should be monitored using performance measurement strategies." [135].

As another example, in Japan, in order to further support utilities assessment and decision making on nuclear risks, Japanese utilities, in October 2014, had established the Nuclear Risk Research Center (NRRC) in Central Research Institute for Electric Power Companies (CRIEPI). Its function is to support and strengthen Japanese utilities' capability for risk assessment, risk management and risk communication by a group of experts engaged in development of tools and by collecting data from testing. NRRC has a target of establishing good PRA usable for the assessment, management and communication of risk including challenges in cutting edge areas in PRA methodology such as multi-hazard, multi-unit, multi-source accidents situation. By capitalizing on research facilities owned and operated by CRIEPI, it is trying to establish technical basis for risk assessment and risk management such as in fire, tsunami, flooding and earthquake.

Enabling risk informed decision making requires availability of risk assessment tools, a group of experts for assessment and training system, data collection system (including hazard data, initiating event data, component reliability data), decision making system (such as risk committee and supporting review system), all of which requires defined activity/task and associated funding in the organization, specifically in utility and its nuclear power plant site. Risk assessment activities could be outsourced. However, there is a view, in light of the magnitude of risks arising from accident at nuclear power plant, that assessment of risk arising from the assets owned/operated by an entity has to be an integral part of the entity's management.

6. HUMAN AND ORGANIZATIONAL FACTORS CHANGES

While Section 4 and Section 5 discussed the scientific and technical actions taken in light of the lessons learned from Fukushima Daiichi accident, one of the most important lesson learned was the factors that led to the accident and the response to it was the deficiencies in human and organization aspects. Importance of the interactions between human, organizational and technical factors, that was identified during the Three Mile Island and Chernobyl accident, once again was highlighted by the accident at the Fukushima Daiichi NPP. The post-accident investigations called attention to the need for a systemic approach to safety from the design to decommissioning of an NPP, with the consideration of this interaction [1]. This systematic approach including human, technological and organizational considerations is necessary and should consider all stakeholders, among others, licensees, regulators, political leaders and the public, and for accomplishing this, Ref. [1] further elaborated:

> "A diversity of expertise is needed to cover the human, technical and organizational factors […] in order to avoid undue simplifications in interpretations and to better recognize the full picture" [1].

In this section, the elements of this systematic approach that directly or indirectly involved the licensees (i.e. operating organizations) that prompted them to take actions for improvements in the human and organizational areas. These actions were in response to several suggestions that were pointed out in the aftermath of the accident, including:

- Unceasingly questioning and critique of the prevailing thoughts, assumptions, decisions and actions that affect (and could affect) nuclear safety by individuals and organizations, as a part of strong safety culture;
- Developing and maintaining human and organizational resilience capabilities that are based on modern tools and methods on complex sociotechnical systems for safety when designing and operating nuclear power plants;
- Developing standards, programmes, processes and procedures for operation, in order to strengthen human and organizational resilience capabilities, including those establish guidance and instructions for:
  - Training, exercises and drills taking due account of harsh conditions (both physical and psychological) that would be anticipated during postulated severe accident and unexpected situations, preparing humans and organizations for quick and flexible adaptation to unexpected situations in advance, such as the programmes and procedures for deployment of equipment in the management of severe accident and their simulated implementation;
  - Response to a possible nuclear emergency, particularly considering those that could involve severe damage to fuel, reactor core, confinement and including those involving several units at a multi-unit plant possibly occurring at the same time as a natural disaster with clearly defined roles and responsibilities for the operating organization and for local and national authorities;
  - Organizational functions and duties during emergency that cover designation and assignment for clearly specified duties, including the proper preparation and protection.

6.1. CULTURE FOR SAFETY

One survey responder from an operating organization outside Japan stated that they have undertaken the efforts in post-Fukushima accident:

> *"in a spirit of humility and leadership, to protect the future of nuclear generation in the country."*

This statement exquisitely addresses one of the noted human and organizational factor deficiencies, confidence and complacency as highlighted in Ref [1] and Refs [3–5], as Ref. [1] stated:

> "The operators were so confident that the superior technical features of their plants would make nuclear accidents highly unlikely that they did not prepare sufficiently to mitigate the results of such accidents. […] One part of a healthy safety culture of organizations is the capability to challenge or re-examine the basic assumption for safety." [1].

Section 3.2.2 discussed implications of Fukushima Daiichi accident on organizational culture issue on challenging basic assumptions and referred to the IAEA view expressed in the Ref. [1] that the flooding causing a nuclear accident was outside the basic assumptions, which is an invisible part of culture for safety in the organization. Reference [1] also explained:

> "This is part of sustainable safety culture improvement; the basic assumption about safety is recognized as fundamentally directing safety culture. […] Reflection and dialogue are needed within an organization in order to become aware of possible blind spots in basic assumptions." [1].

Furthermore, among lessons learned described in Ref. [5], after recognizing a series of management actions to strengthen safety culture taken in the aftermath of falsification scandal, INPO wrote:

> "The Fukushima event revealed several aspects of a healthy safety culture that require additional attention" [5],

that otherwise could have benefited the prevention of and protection from the accident. This included, *inter alia*:

— 'Questioning attitude', i.e. additional questioning of the assumption that a large tsunami capable of flooding the plant could not occur, while asking, 'What is the worst that could happen?';

— Avoiding 'group think' in accepting unverified assumptions when making decisions that could affect nuclear safety;

— Learning from practices in the world as a part of being 'learning organization'.

The operating organizations reported in the survey that maintaining and further improving the culture for safety of their organization has been part of their day-to-day commitment regardless of Fukushima Daiichi accident and they will continue this as normal business. They also noted that they will continue to follow industry consensus documents regarding the principles of safety culture, for example, the revised and expanded WANO guideline "Traits of a Healthy Nuclear Safety Culture" in 2013 [136, 137], as updated following the Fukushima Daiichi accident.

Furthermore, one Member State licensee responded the survey with the list of questions and attitudes they anchored in evaluating and taking actions in response to the accident, as follows:

— *"What is worth [doing in the country's context of reducing risks as low as reasonable possible] diverting money and resource from more current operational safety issues?*
— *What is the best way to proceed, considering many stakeholders and many different views?*
— *How can we sustain our actions that are being taken in response?*
— *Decisions not to ignore any aspect of the events in Fukushima Daiichi NPP were made in alignment of both leadership and staff (only a top level decision 'just to do it' would not have been a credible reaction by the plant staff and the public).*
— *Openness to views but ultimately self-determination and bearing of responsibility."*

6.2. PROCEDURES

Nuclear power plant operators and other staff are responsible for conducting safe normal operation of the plant in accordance with the plant OLCs and the design and licensing basis, as well as prevention, protection and recovery from abnormal, emergency and accident conditions and mitigation of severe accidents. The actions to be taken to accomplish these are provided to the operators in forms of instructions in associated operating procedures that are developed to enable plant personnel to perform their duties correctly for different operational states of the plant and in accident conditions (i.e. for normal operation, anticipated operational occurrences and accident conditions). As stated in the IAEA's specific safety requirements for commissioning and operation (Ref. [138], Requirement 26, titled "Operating procedures"), the guidance and instructions provided in these written procedures "shall" be clear and concise and strict adherence to the operating procedures is an essential element of safety policy at the plant. Reference [138] also requires:

> "**7.2.** Procedures shall be developed for normal operation to ensure that the plant is operated within the operational limits and conditions.
>
> **7.3.** Procedures shall be developed and validated for use in the event of anticipated operational occurrences and design basis accidents. Guidelines or procedures shall be developed for the management of accidents more severe than the design basis accidents",

and:

> “**5.8.** An accident management programme shall be established that covers the preparatory measures, procedures and guidelines, and equipment that are necessary for preventing the progression of accidents, including accidents more severe than design basis accidents, and for mitigating their consequences if they do occur.” [138].

In Section 2.4 of Ref. [1], all components of accident management are discussed in respect to written accident management guidance covering DBs and BDBEs, including severe accidents, i.e. abnormal operating procedures (AOPs), emergency operating procedures (EOPs) and SAMGs. The IAEA guidance further suggest that, for the BDB conditions, the EOPs and SAMGs need to be symptom based, with clear interfaces between them.

The discussion in Ref. [1] noted that although the relevant procedures were available in the plant, these procedures and guidance were not adequate to address the complex accident conditions experienced. Some of the reasons for ineffective use of procedures was incorrect underlying assumptions, such as the assumption that AC and DC powers would always be available, either from its own power sources in the unit or that it could be easily re-established by connecting with the neighbouring units. For such reasons, the procedures did not provide contingency actions for situations without power, particularly during a prolonged SBO. The procedures also did not consider the possibility that a severe accident could impact several units and SFPs simultaneously or that the off-site infrastructure, including serious deficiencies in communication, transportation and utilities could be seriously disrupted, making it more difficult to receive support in responding to an event or accident.

Additionally, Ref. [1] described an observation on the lack of procedural instructions for mobile equipment that were used during the accident as follows:

> “Although mobile equipment (such as fire trucks) were available on the site and provisions for connection of these sources to the plant had been made, the use of these sources failed or was delayed either due to the absence of adequate procedures or other obstacles in the implementation of the procedures” [1].

Based on these lessons learned and observations, NPPs reviewed existing operating procedures to identify gaps and improvements. When those gaps and improvements were identifies, the operating procedures were revised by a series of technical and administrative controls and solutions for implementation and performance.

Noting that no changes were identified to normal operating procedures (NOPs), Section 6.2.1 and Section 6.2.2 provide the changes to SAMGs and EOPs, respectively, as reported in the survey responses. Therefore, procedure modifications discussed in this Section are not a complete list of all potential impacts, as they will differ depending on the nuclear power plant operation and maintenance practices, effectiveness and extent of existing programmes etc.

Also, the NPPs, which established strategies based on mobile equipment, wrote new procedures (or revised if one has existed) for deployment, operation and maintenance of mobile equipment. These procedures are typically controlled under the existing plant procedure control programmes and, as reported in the survey, some NPPs decided to include mobile equipment operation procedures in SAMGs and/or EOPs, while some used them as supplement to EOPs and SAMGs as an integrated manner. NPP actions for mobile equipment procedures are discussed in Section 6.2.3.

6.2.1. Changes to severe accident management guidelines

The NPPs which responded the survey questionnaire with actions taken for changes to SAMGs provided the following:

- Updating accident management guidance documentation (i.e. SAMG) to improve clarity and usability;
- Expansion of SAMGs to cover shutdown and low power modes;
- Expansion of SAMGs to include SFP;
- Special administrative controls for exclusion of contaminated articles from the MCR;
- Special administrative controls for zoning downwind areas for personnel ingress/egress;
- Standardizing form and format of the SAMGs (with support and efforts by the technology owner groups (i.e. BWROG, COG, PWROG) and/or IAEA for standardization);
- Integration of EOPs, SAMG, Extensive Damage Mitigation Guidelines (EDMGs) and FLEX Support Guidelines (FSGs) (deadline for completion of this integration was less than within two years after BDBE rulemaking by the regulatory body, USNRC).

Challenges encountered in making these changes to SAMGs were reported as:

- To improve clarity and usability with multi-unit events due to the extent and variation of cross-actions and absence of appropriate models (particularly for prolonged SBO, LHS in two or more units at one site);
- To define strategy for shutdown state to set exit conditions to SAMGs;
- Finding a method of maintaining SAMGs current, based on the latest configuration of the plant and potential scenarios discovered;
- Determining the command and control and ultimate decision making (i.e. shift manager who is licensed, technical support centre (TSC) director who may not be even licensed or familiar with the plant, etc.);
- Developing derivation/estimation methods for parameters by measuring and/or trending alternative parameters, as well as observing instrument response to operator/equipment actions;
- Accrued cost of revisions (average cost of SAMG revisions was reported as US $1 million, including the validation efforts, while EOP revisions typically cost between US $150 000 and US $200 000 and FSG revisions cost from US $400 000 to US $750 000).

6.2.2. Changes to emergency operating procedures

The NPPs which responded the survey questionnaire with actions taken for changes to EOPs provided the following:

- Modify EOPs to provide operators guidance during an ELAP event, particularly on declaring an ELAP event, and then entering and prioritizing the use of the approved coping strategy support guidelines (e.g. FSGs in the USA);
- Updating EOPs to improve clarity and usability with multi-unit events (SBO, LUHS);
- Establishing an appropriate procedure for cooling strategy of SFPs at shutdown state.

6.2.3. Mobile equipment procedures

The main prerequisite for adequate and comprehensive mobile equipment procedures was to establish coping strategies (e.g. FLEX strategies in the U.S.) and mobile equipment deployment tactics (e.g. for emergency mitigating equipment (EME), in Canada, or for DBUE, in the UK). Once these strategies and tactics are identified and defined, the NPPs provided changes/additions of mobile equipment procedures.

In most Member States, mobile equipment procedures were not a regulatory requirement; however, commonly the owner/operating organisations decided to develop a suite of operating instructions for the deployment of mobile equipment which was required so that the delivery of capability and functionality can be achieved as intended. These included, among others:

— Developing and implementing the approved coping strategy's support guidelines (e.g. FSGs in the USA) to provide detailed instructions on how to use the portable equipment during an ELAP event.

— Preparing new procedure for recharging batteries due to limited station battery capacity.

— Preparing new procedures for MCR and battery room cooling using portable fans.

— Establishing standard operating guidelines (SOGs) that provide instructions to EPS staff for the management of various mobile EME deployment and operation, e.g. loading, transport and positioning of portable generators; cable routing; connection to electrical receptacles;, directions on how to clear a designated path, retrieve the mitigating equipment and set up and startup and hand over to operations the equipment to operation staff, etc. (The NPP noted that both branches of the EPS workforce, fire and security departments will be utilized to deploy EMEs, as necessary).

— Creating guidelines or instructions for use of EME (e.g. emergency mitigating equipment guidelines (EMEGs)) to support the flexible and diverse mitigation capability. These EMEGs provide instructions to operations staff available, pre-planned strategies for accomplishing specific tasks (in accordance with the developed EME deployment tactics), for example, during a complete loss of electrical power for load shedding and repowering of priority I&C loads.

— Developing programmatic and administrative (e.g. maintenance, test, training etc.) procedures based on the mobile equipment deployment procedures that are based on the coping strategies and tactics.

Some Member States stated that the mobile equipment deployment and operation procedures/guidelines were included in SAMGs, while some chose to use them to supplement (not replace) the existing procedure structure that establish command and control for the event (e.g. AOPs, EOPs, SAMGs). One of those operating organisations, that classified the mobile equipment procedures under the same system with SAMGs, stated that these procedures were written by the authorized staff highly familiar with the plant layout, reactor operation and AOPs, EOPS and SAMGs. Moreover, they were prepared by following the existing writing guide for the station system procedures and using the same syntax and format as AOPs. The NPP also added:

> *"The authors worked closely with reactor safety engineering staff, who were directly involved with the specification of the design requirements and who author the SAMGs, as well as with plant design engineering staff, who prepare the*

electrical design packages, to sequence the tasks for connecting the generators and to prepare the load shedding and supply lists."

NPPs responding to the survey noted some challenges, which were encountered during the development of mobile equipment procedures, and their solutions, as:

- Integrating the mobile equipment guidelines within the existing SAMGs and EOPs framework was reported to be challenging as the implemented deployment and operation strategies should not violate the basis of existing procedures as well as the entry/exit conditions. This necessitated definition of the specification of clear and precise criteria for entry into mobile equipment guidelines to ensure that portable equipment strategies are used only as directed for BDBE conditions and are not used inappropriately in lieu of existing procedures. When a mobile equipment guideline is needed to supplement EOP or SAMG strategies, the EOP or SAMG should govern the entry into, and exit from, the applicable mobile equipment guideline.
- Another challenge was the determination of deployment path to specify in the procedures since some of the site conditions cannot be anticipated. One way to solve this issue was to identify redundant paths in the procedure in case that the primary path becomes unavailable.

6.2.4. Normal operating procedures

No changes to the normal operating procedures have been made, based on the information provided by the owner/operating organisations.

6.3. PROGRAMMES

In general, post-Fukushima action have not necessitated significant changes to the existing plant programmes. The marginal impact of physical, analytical, human and organizational impacts of changes for the DEC/severe accident management on the plant programmes and processes were handled by integrating them primarily in three manners:

- They were annexed to the existing programmes and processes;
- They supplemented, but are governed, by the existing programmes;
- They were incorporated into the existing programmes and processes.

The following sections discusses visible programmes and processes that were impacted: Emergency response programme and emergency plan; training programme; maintenance programme (including test and surveillance and special test programmes); and design control process.

6.3.1. Emergency response

In addition to the physical changes to their plants, as discussed in Section 4.12, based on the lessons learned in human and organizational areas from the accident, the operating organizations made administrative changes to their emergency response organization (ERO) to ensure that it is well prepared to mitigate and respond to both radiological and all hazard emergencies. These included:

— Revision of administrative procedures to ensure clear accountability and ownership of the mobile equipment. For example, in one survey response, the accountability and ownership of portable generators were assigned to station EPS. EPS was elected because they are already highly familiar with the deployment of heavy equipment under emergency conditions.
— Reorganization of ERO with fire and security organizations under a single reporting line such that ERO, Security and Fire Operations crews will be working in a united manner to deliver the service necessary for the deployment of emergency mitigating equipment.
— Installation of forewarning systems for severe weather and sea conditions.
— Improvement of preparedness instructions in common procedures with all stakeholder entities that will be participating in the emergency response.
— Establishment of dedicated off-site emergency teams to provide support activities in case of nuclear emergency by dispatching equipment/staff and tackle on the disaster under harsh conditions (e.g. high radiation), in cooperation with the owner/operator organizations, such as:
  - National and regional (off-site) nuclear rapid action force;
  - National response centres;
  - Regional nuclear emergency assistance centre (NEAC) for support activities in case of nuclear emergency.

6.3.2. Training

The main aims in the preparation and implementation of post-Fukushima Daiichi accident training programme and procedures, that were established by the NPPs, focused on three aspects of the initial and continuous training and qualification exercises/drills such that they:

— Consider human capabilities and resilience for potentially needed very long duration for emergency response tasks;
— Give confidence that arrangements and human resources are fit for purpose for a prolonged severe accident or multi-site events;
— Be sufficient in establishing and sustaining enhanced capabilities.

The initial and continuous training were designed for all the relevant personnel and organizations, i.e. emergency response organization, emergency protection workforce, operations, fire department and security, department etc., that will be carrying tasks in the deployment of equipment. Most of these trainings have qualification requirements for personnel in accordance with new job task descriptions.

Accordingly, initial trainings were established and implemented, for example, for:

— Revised operating procedures (e.g. EOPs, SOPs) and accident management guidelines (e.g. SAMGs, SOGs, EMEGs, deployable backup equipment guidelines (DBUEGs), etc.);
— SAM training at the plant simulator (including the revised SAMG and use of purchased and stored mobile power and water sources);
— Emergency response dealing with a multi-unit event;

— Deployment of mobile equipment based on the deployment strategies and tactics (focusing on complexity of delivery options, methods and timing);
— Debris removal equipment operator training (particularly focusing on operation around sensitive equipment and infrastructure);
— Use (i.e. transportation, connection, operation, control, etc.) of mobile equipment;
— Area preparation and management (both for on-site and off-site, i.e. national and regional training centre personnel);
— Training of off-site personnel at the national and regional centres on deployment of mobile equipment based on the operating organization's deployment strategies and tactics (including delivery options and methods);
— Training of on-site personnel by off-site staff on the operation of special equipment (for example, remote controlled robots);
— Simulator training for with events where multi units are affected (although it should be noted that one lesson learned from these simulator trainings was that simulator training with events where multi units are affected is less effective than drills and classroom and field trainings).

In addition to these initial trainings, training plans for continuing and as-needed topical training to sustain/maintain competency and skills with optimized scope and frequency of the training were prepared.

Also, the classroom and in-field training were supplemented with:

— Required tabletop and full scale exercises and drills (some of which are performed together with off-site emergency response organization and personnel, such as those from national and regional response centres, local, i.e. municipal, fire and emergency response departments, etc.);
— Repeated performance and practice of deployment strategies and tactics by drills and exercises, including the timing trials (in which frequency of drills and exercises increase based on the complexity of event, strategy, organization and time for deployment and the timing of all necessary drills and exercises are country dependent based on national regulatory requirements).

There have been efforts to increase effectiveness and optimization of these training, drill and exercise sessions particularly owing to the 'unknown-unknowns'. For example, IAEA report on the Fukushima Daiichi accident (see Technical Volume 2 of Ref. [1]) stated that:

> "Organizations need to be prepared for the unexpected […] This includes providing appropriate training to all individuals on how to respond to unexpected events" [1].

This unanticipated and unexpected nature of BDBEs has been proven to be problematic and challenging as it necessitated to be prepared for the unprepared. Therefore, some challenges noted in the implementation of training programmes reported as:

— Training for severe accident scenarios is complex as determining the accident scenarios is a challenge and requires extensive engineering analyses and conceptualization;
— Simulator training for SAMGs are complicated owing to the infinite number of severe accident scenarios to exercise on;
— Simulator training with events where multi units are affected is not effective (the solution for this is conducting drills and training which better deals with such an event, where applicable);
— A large population of staff to train, hence potentially long timescales to cover all of them given significant new components of the arrangements (mobile equipment deployment, in particular);
— Establishing processes and plans to sustain and maintain training and exercise programmes into the future since there have been large amount of tacit skills gained by the first set of staff involved in preparation, implementation and practice (this requires a very effective long term planning especially by determining the long term needs);
— When integrated with EOPs and or SAMGs, mobile equipment deployment trainings require special methods to train a large number of personnel from diverse disciplines which may not be efficient to cover all training objectives in accordance with the relevancy of each person/organization in a combined classroom;
— Optimization of maintain and sustaining skills and competencies and expenditures, as initial training costs can run up to US $500 000 to US $1 million range and each drill costs about US $35 000 to US $50 000 (not including the full scale drills with off-site entities).

More importantly, in establishment of training programme for BDBEs, it has been significantly challenging for the operating organizations to find a right balance of the programme between the design basis event and beyond design basis event preparedness needs and requirements. It is understood that there is a need for adequate and enough training and exercises on BDBEs to sustain the enhanced severe accident management means and capabilities. However, there is also a requirement for acquiring and sustaining skills, competencies and capabilities for responding to more likely events, e.g. anticipated operational occurrences (AOOs) and DBAs, such that the need for BDBE response preparation has to be limited to a certain point as not to deflect from the requirement for DBA preparedness capabilities and means. This requires a balance of judgment together with stakeholder consultation.

6.3.3. Maintenance, testing, surveillance and inspections

The maintenance, testing, surveillance and inspection programmes ensure the required facilities and equipment are available and ready operationally to support functionality and an adequate response capability.

In general, post-Fukushima action have not necessitated changes to the existing maintenance programmes except the incorporation of new equipment such that maintenance and testing for equipment can be conducted and performed in accordance with existing plant processes. Also, new maintenance templates and instructions for BDBE response and mitigation equipment were integrated into the existing programmes (e.g. preventive maintenance, surveillance and test programmes). The aim for this integration was that the maintenance programme needs to ensure that equipment reliability is achieved and sustained for BDBA/BDBE equipment.

Industry support organisation EPRI prepared new standard maintenance templates (which includes maintenance performance testing, etc.) for the portable equipment [139]. These were implemented by the member organizations by incorporating into individual NPP's existing programmes and processes to control the maintenance of BDBE mitigation equipment. In the USA, these templates were also used by the national response centre equipment to ensure uniform implementation across the US nuclear power plants.

Also, in the USA, industry guidance [20] provided specific maintenance and testing recommendations for FLEX equipment, particularly for those that directly performs a FLEX mitigation strategy for the core, containment, or SFP should be subject to maintenance and testing (including surveillance and inspection) guidance that is provided in existing processes (e.g. INPO's equipment reliability process that is describe in Ref. [140]) to verify proper function. They also included the guidance for initial testing, periodic testing frequency, preventive maintenance based on the type and expected use of the equipment.

Accordingly, NPPs have created procedures which define the process and frequencies, by which emergency facilities and equipment are periodically inspected, operationally checked, and are tested.

Some NPPs performed equipment survivability and operability evaluations including survivability assessments for equipment and instrumentation for severe accident management.

The main challenge was the lack of operation history for BDBE protection and mitigation equipment for their specific applications. It will take some years to accumulate data and experience which then can be used to update these initial maintenance strategies.

Another challenge that was reported by survey responders was how to deal with the unavailability of portable equipment and applicable connections for which some industry guidance had to be provided. For example, in the USA, Ref. [20] was prepared to provide guidance for determining acceptable availability that directly performs a mitigation strategy for core, containment and SFP, specifically recommending to the US NPPs the following:

- "Equipment may be unavailable for 90 days provided that the site FLEX capability is available.
- Connections to permanent equipment required for FLEX strategies can be unavailable for 90 days provided alternate capabilities remain functional.
- Equipment that is expected to be unavailable for more than 90 days or expected to be unavailable during forecast site-specific external events (e.g. hurricane) should be supplemented with alternate suitable equipment.
- The short duration of equipment unavailability, discussed above, does not constitute a loss of reasonable protection from a diverse storage location protection strategy perspective.
- If equipment becomes unavailable such that the site FLEX capability is not maintained, initiate actions within 24 hours to restore the site FLEX capability and implement compensatory measures (e.g. use of alternate suitable equipment or supplemental personnel) within 72 hours." [20].

6.3.4. Overall design extension condition programme and design control

Once the approved modifications, equipment and coping strategies, are being performed and implemented, a long term programme for preserving, maintaining and sustaining the DEC assessments, analyses, equipment and knowledge, including specific to SAM and emergency response need to be created. This long term programme should include administrative and programmatic controls and ownerships. This have been accomplished in different manners in different NPPs:

- The information and instructions on the deployment and use of equipment in accordance with the DEC and specific SAM strategies and tactics have been incorporated into the plant operating procedures;
- Administrative and functional changes to emergency response have been incorporated in the emergency plans and administrative procedures;
- In most NPPs, the equipment and instrumentation for DEC or specifically for SAM were added to plant maintenance programme, while in some NPPs, some of these were added to the LTO and plant life management (PLiM) programmes;
- The necessary training and drills/exercises were incorporated in the training programmes;
- The development, information and documentation of strategies and design of modifications with underlying DEC 'design basis' were included in the design control process, some of which assigned to quality assurance controls, while some did not. In some NPPs, documented engineering or design basis covered only those SSCs and equipment that are utilized to protect or recover key safety functions (for core, containment and SFP).

Within these numerous challenging aspects (e.g. huge amount of background and implemented work, large distribution of responsibility and ownership), there were, among others, two main challenges for operating organizations:

- Identification and assignment of a long term programme owner organization, i.e. who will own the overall programme for the remaining life of the plant;
- Preservation of information, knowledge, experience and competence gained by the implementation of actions, particularly those that are not subject to the existing quality assurance programme requirements (e.g. design and documentation control) and more importantly, that are tacit.

6.4. ORGANIZATIONAL STRUCTURE

As suggested in Ref. [1], in order to strengthen human and organizational resilience capabilities, organizational functions and duties during emergency need to be assigned to designated organizations. This designation and assignment need to be defined in the plant procedures with clearly and precisely specified roles, responsibilities and tasks and use a systematic approach that includes human, technological and organizational considerations by ensuring:

- Clear roles and responsibilities: It is essential that all staff participating in the activities and tasks concerning accident management clearly understand their role and how their skills and knowledge in specific areas are to be used in conducting these activities and tasks. This requires definition of clear roles and responsibilities. Clear roles and responsibilities the organization need to bet set up such that there are no overlaps of scope of duties, no shared responsibility and no competing authority.

Also, when the duties are coordinated and executed, there is a collegial relationship among all involved personnel and organizations to ensure that the activities are conducted adequately and completely. For doing so smoothly and correctly, cross-organizational channels and protocols also need to be defined to ensure communication with those who are working alongside with them in different organizations.

— Structuring and staffing according to the needs and tasks: There are no 'one-size-fits-all' organizational structure for organization for performing emergency response and accident management functions. However, there are common features that can be accomplish and structure effective organizations, including [141]:

  - Determining what activities will be needed and what the preferred ways and means to perform those;
  - Understanding and knowing the activities and tasks needed for decision(s) and time when they are needed;
  - Identifying needed level of skills and tools;
  - Evaluating available (or potential) means and resources against needed tasks and competencies;
  - Planning, deciding and organizing based on the 'needs' and at-hand and accessible 'means', i.e. human and financial assets;
  - Arranging and sizing the organization such that people and responsibilities are assigned according to the tasks, rather than arranging tasks according to the size, form or personnel in the organization at-hand.

In response to these lessons learned from Fukushima Daiichi accident, NPPs have taken action to structure/restructure their organizations. The following section will discuss some of those.

6.4.1. Internal organizations

For structuring internal organizations, the NPPs reported (in the survey) the following actions:

— In the task determination area:

  - Prepared new (or revised the existing) job task descriptions;

— In the staffing requirements area:

  - Conducted staffing study to determine the minimum staffing necessary to recover the plant during an ELAP, and to establish the timing of actions. In the USA, staffing study was done per the industry guidance (e.g. by the guidance provided in Ref. [122]);

— In the staffing competence area:
  - Established competency mapping;
  - Conducting initial and periodic training and drills/exercises to ensure the staff competency and skills;
  - Developed method for qualification of staff for assigned duties following competence mapping.

— In understanding of roles and responsibilities in decision making:
  - Established or reinforced continuous training and education of leaders and staff on nuclear safety, for 'management for nuclear safety'.

The most significant challenge in structuring the organization and maintaining competency was due to the fact that BDBE scenarios, unlike DBAs, cannot be precisely anticipated or predicted. Because of this, some utilities had to explore a different means to assess the overall response capability and predict the likelihood of success. One utility suggested:

> *"Our planning, execution and review of our set of drills and exercises have convinced us that the best approach is determine the key criteria for the successful management of BDBE and to assess our capability against these criteria as a predictor of future performance. The following eight criteria have been identified as a result of our review:*
> — *Effective design basis minimum compliment in place;*
> — *Exercises and drills demonstrate margin;*
> — *An emergency management organization capable of directing the response to any event including all hazards;*
> — *Emergency mitigating equipment in place including redundancy for key functions;*
> — *Trained and capable response force;*
> — *Effective communications on and off site independent of normal infrastructure;*
> — *Ongoing and effective training program in place;*
> — *Routine drill and exercise program in place."*

One other utility explained an approach for assigning ownership as:

> *"To ensure clear accountability, ownership of the portable generators rests with EPS. EPS was elected because they are already highly familiar with the deployment of heavy equipment under emergency conditions.*
> *After the Fukushima event, the emergency response organization was reorganized with Fire and Security now under a single reporting line so that EPS, Security and Fire Ops crews are working in a united manner to deliver the service necessary for the deployment of emergency mitigating equipment."*

Another utility institutionalized internal oversight and independent review by establishing a safety oversight committee made of experts. Also, an internal risk management committee was founded.

6.4.2. External organizations

In addition to the national/regional (off-site) emergency response action workforce (See Section 4.12.9 and Section 6.3.1), another external organization use was the expertise and experience. Noting that one of the observations that was noted in Ref. [1] was the need for a diversity of expertise. Reference [1] further explained that use of diverse expertise is to avoid undue simplifications in interpretations and to better recognize the full picture during the preparation and execution of tasks.

In response to these lessons learned from Fukushima Daiichi accident, NPPs have taken action to use external expert support organizations. These external organizations included external expert and committees, some of which were institutionalized by the operating organizations, safety review boards and panels, risk reviews or risk informed committees, engineering design challenge boards/panels, operational/operations decision forums. Some examples of utilizing such external special purpose organizations included:

- In the application of realistic models and methods and the performance of 'best estimate' analyses, the complication and uncertain fault escalation sequences of assessment of sequences and timescales for protecting the pressure boundary in the event of total loss of cooling. In this case, expert panels were consulted to make informed judgements about the possible phenomena and their likelihood to determine which analyses to perform.
- For SSHAC[18] process experts and expert panels were consulted.
- Development of beyond design basis flooding conditions necessitated external expert and committee consultations.

The most visible challenge in the utilization of external expert support organizations was finding experts for the panel(s) and finding available time of the experts in those panels for scheduling the panel meetings in timely manner. It was also noted by the NPPs that the time that it took to establish advisory group/committee, particularly as parts of the regulatory and industry initiatives, imposed a challenge for the post-Fukushima action implementation schedule.

6.5. MUTUAL TRUST AMONG STAKEHOLDERS

Reference [1] highlighted that licensees need to conduct a transparent and informed dialogue with the public on continuous basis, when assessing the Fukushima Daiichi accident regarding the basic assumptions of the main stakeholders regarding nuclear safety. It noted:

[18] SSHAC is the entity that developed the SSHAC Guidelines for which the process to be used in seismic hazards analysis, such as PSHA models and studies.

> "In addition to meeting the technological challenge of making nuclear power plants as safe as reasonably possible, it was essential to make sure that the public would consider nuclear energy production as safe" [1].

This necessitates the explanation of their activities and risks in the use of nuclear electricity generation and how safety is maintained.

Following the Fukushima Daiichi accident, reactions from the public has been strongly influenced by the way the actions in response to lessons learned from the accident were carried out by NPPs. Therefore, NPPs have been in communication with the public on their activities addressing the improvements of the safety of their plants in light of Fukushima Daiichi accident lessons learned. The NPPs noted that, with the understanding of the need for building a constructive dialogue with the public, mutual trust and respect are the main cornerstones for the roles both stakeholders play. As a part of this open and constructive dialogue, the NPPs have taken actions openly and competently inform the public about the actions taken by the NPPs. These actions included, but not limited to:

- In order to encourage public's stakeholder involvement, new visitor centres were opened (one survey responder said that this was done at all their power station sites, in their fleet).
- Taking the actions in a spirit of humility and leadership to protect the future of nuclear generation in the country and demonstrating this attitude to the public.
- Valuing the views of public in proceeding with the best way that is agreed by the stakeholders, including public.
- In one Member State, the public was made a direct input and concurrence provider in the decision making.

7. VERIFICATION AND VALIDATION OF EFFECTIVENESS

7.1. FUNCTIONALITY

Verification of equipment functionality, i.e. the ability to perform its intended function when needed during an event, of the post-Fukushima safety enhancements has typically been performed using several methods, including:

— An analytical basis is established using engineering calculations and/or computer codes to determine the functional requirements of equipment (in some Member States survivability assessments for SAM equipment and instrumentation, were also used);
— The equipment functional verification is then performed either at the factory where the equipment is built, or on the plant site but not connected to the plant (for mobile equipment);
— Once the equipment is purchased or installed, it will be subject to:
  - Periodic confirmation via maintenance and testing programmes to ensure that functionality is still maintained by preventive maintenance testing and surveillance (See Section 6.3.3);
  - Integrated systems testing (in some Member States this was deemed to be not necessary; however, it is a method of verification, particularly for being able to credit in the DBA or licensing basis space).

For human actions, verification is done either using a tabletop exercise, via simulator training, performing plant walkdowns, or classroom training. This verification is performed to ensure that the human can in fact perform the necessary task in the timeframe allowed. To ensure long term functionality of human actions, periodic drills and exercises are also to be conducted.

In most of the NPPs, the verification process has been incorporated into the existing plant programmes and processes, such as the design change process, procurement process or procedure/guideline development process.

As an example of verification and validation, in the USA, validation of FLEX strategy actions is performed using the industry guidance, that is contained in the Appendix E of Ref. [20] which outlined a process to reasonably assure that required tasks, manual actions and decisions for FLEX strategies are feasible and may be executed within the defined time sensitive actions. There, a graded approach to the validation of FLEX strategy actions is used in order to apply a higher level of detail and rigor to validations critical aspects (for example, time sensitive actions that have limited available margin and would be necessary when personnel resources may be at minimum administrative staffing levels).

Here, it should be noted that Ref. [20] differentiates between ‘validation’ and ‘verification’, as follows:

— Validation [of FLEX strategies] process is used to validate the feasibility of individual strategies and the includes actions that are time constraints required, using an integrated review of the strategies. The purpose of this integrated review is: *“to ensure that adequate resources (personnel, equipment, materials) are available to implement the individual strategies to achieve the intended results.”* [20] The validation is a ‘one-time effort’ unless the strategies change.

— On the other hand, verification process is performed before the validation that is to verify the capabilities and performance of equipment, connections, tools, physical plant modifications and revised procedures/guidelines.

7.2. CONFORMITY

Verification and validation of conformity is to ensure that the actions taken are appropriate and conform with the regulatory and industry requirements/guidance and the expectations of operating organizations. The conformity is verified and validated by inspection and review of the aspects of post-Fukushima actions and physical, analytical and organizational modifications implemented by those actions.

The conformity verification and validation inspections and reviews have been (and will be) conducted by regulatory bodies for the conformance with regulatory conformance and guidance. In Europe, for example, a checklist for verification and validation is provided by WENRA (to guide operating organizations and the regulatory bodies). This checklist consists of a set of questions [8, 9], as follows:

— "Has the licensee a sufficiently rigorous process to identify shortfalls in preventing and mitigating radioactive releases?

— Is the process shown to be adequate? (e.g. identifies the modern safety standards, encompasses all of the faults and hazards that could lead to a release, all modes of operation, includes DEC.)

— Has the licensee considered what could be done to remove or reduce the shortfalls? (this should cover all levels in defence in depth that could contribute to prevention or mitigation of radioactive releases, and not be restricted to the specific technology that a new reactor uses to meet the modern safety standard)

— Has the licensee taken due account of national and international practices?

— Of the reasonably practicable options available to reduce a shortfall, is the one selected that gives the largest safety benefit?

— Where an option is considered not reasonably practicable has the licensee provided an adequate justification that the measure is disproportionate taking account of the nature and scale of the shortfall?

— Has the licensee considered alternative measures to address the shortfall?

— Has the licensee taken account of the time for implementation in the selection process?

— Do the licensee's processes embrace continuous improvement as well as PSR led improvement?" [9].

Additionally, industry self-safety organizations (e.g. INPO, WANO) reviews for the conformance with industry requirements and self- or peer-review of NPPs for conformance with industry guidance and expectations are being conducted, as follows:

— Regulatory inspections and reviews may include a combination of:
  - Review of proposed actions;
  - Pre-compliance audits;
  - Readiness assessments;
  - Review of integrated procedures and guidelines;
  - Baseline inspection upon completion of actions;
  - Initial drill evaluations;
  - Follow up inspection and assessments, such as periodic facility inspections, drill evaluations and reviews of changes to approved configuration and strategies.

— Self, peer and/or industry plant assessments may include:
  - Various periodic assessment (self or peer) in accordance with quality assurance and operational safety programmes;
  - Plant evaluation;
  - Corporate reviews.

Regardless of who is performing the verification and validation of conformity, the inspections and reviews typically involve field walkdowns, examinations of selected procedures and records, observations of activities and interviews with plant management and personnel. These inspections and reviews [reported to] have included the verification and validation of:

— Adequate determination and implementation reliable mitigation strategies for overall BDBEs and specific to severe accidents;

— Correct and adequate selection and installation of equipment (SSCs, portable mean, etc.) for the mitigation of BDBEs (which could be separated from those for specifically severe accidents);

— Correct and comprehensive development and implementation of emergency preparedness, DEC event response and specific SAM enhancements, including, but not limited to:
  - Development and issuance of operational procedures (AOPs, EOPs, etc.) and accident management guidelines (e.g. SAMG, EMEGs, DBUEGs, etc.), including the interface/transition between, and entry to and exit from, existing operating procedures and guidelines with the newly developed ones;
  - Provision of means to protect BDBE response and SAM equipment and instrumentation from site specific internal and external hazards;
  - Development and implementation of adequate testing and maintenance of BDBE response and SAM equipment to ensure their availability and capability;
  - Sufficiency and training of staff to assure number of personnel with skills, competency and proficiency in BDBE response and SAM;
  - Development of procedures and binding agreements to ensure that the necessary off-site equipment and human support will be available from off-site locations;
  - Establishment and adequacy of preventive maintenance, emergency response, training programmes for the BDBE response and SAM equipment;

- Development and implementation of plans and/or procedures to ensure periodic equipment inventories were in place and being conducted;
- Additionally, the team examined the introductory and planned periodic/refresher training provided to the staff most likely to be tasked with deployment and operation of the BDBE response and SAM equipment;
- Development and implementation of the introductory and planned periodic training for the emergency response organization and supporting personnel;
- Selection and deployment of systems and equipment to ensure that the communications can be maintained during DEC/severe accident conditions.

7.3. LONG TERM SUSTAINABILITY

In addition to completing all post-Fukushima action, it is essential to maintain and preserve those in the long term. It has already been noted in Section 6.3.4 as one of the challenges is to preserve information, knowledge, experience and competence gained by the implementation of actions.

Primary reason for this challenge is the innovative nature of post-Fukushima actions in their design and strategies for implementation, n, including concepts and ideas with long term and progressive thinking. Consequently, there have been many underlying analyses, strategies and tactics for the changes that were made (or being made). More importantly, in the performance of these analyses, various concepts and innovative thoughts were considered, discussed and realized, and unfortunately, some were done in ad hoc basis.

Furthermore, during the evaluation, decision making and implementation, vast amount of tacit knowledge and experience were collected by the core group of people managers and frontline experts. These people have understood the concepts, bases, history, operational experience and importance of the physical modifications, analytical, programmatic, procedural and organizational changes and training needs at their plant, as they have been working on these activities for the last nine years as part of the their NPPs and industry's response to Fukushima lessons learned. It is also very important to note that, unfortunately, some of these people are leaving the field[19].

Therefore, turnover/handover and long term preservation of information, knowledge and experience is the key factor for long term sustainability and effectiveness of post-Fukushima actions taken by NPPs. However, for many NPPs, if not all, this remains a big challenge and a significant concern. This is also an angst of the regulatory bodies, as one of the USNRC senior managers who participated in a senior leadership visit to Japan in 2014 reflected in the essay of his personal reflection, published in NUREG/KM-0008 [142]:

> "Following the TMI accident, the NRC published a listing of TMI action items in NUREG-0737, "Clarification of TMI Action Plan Requirements," issued November 1980, to ensure that all the actions required were captured for knowledge management. About 7 or 8 years later, a member of Congress asked the agency to provide a status update of the TMI action items. The staff had not kept adequate records and had to scramble to recreate an accounting. Once the dust settled, we found that although many

[19] Even during the preparation of this publications, five contributors to the publication have retired, three contributors have been assigned to different organizations or move to other field of work.

> licensees had fully implemented the TMI lessons learned, some had not. We are resolved that we do not repeat that experience with the Fukushima lessons learned" [142].

There have been several approaches for ensuring long term sustainability of actions taken in light of lessons learned from the Fukushima Daiichi accident, including:

- Proper, comprehensive and complete integration of the post-Fukushima actions into the existing plant programmes and processes to assure the continued sustainability, reliability and effectiveness, such as:
  - Maintenance, testing and surveillance programme(s);
  - Configuration control programme(s);
  - Procurement process;
  - Work control process;
  - Quality assurance programme;
  - Design change and design basis control process;
  - Training programme;
  - Procedure development and control process;
  - Document control processes;
  - Emergency plan;
  - Staffing;
- Periodic inventory accounting;
- Continuing regulatory approval and update. For example, in the U.S., the approach for long term sustainability relied on use of a multi-step process covering:
  - Pre-compliance readiness review audits:
    - Using technical guidance approved by the regulatory body, each nuclear power plant operating organization provided proposed action plans for plant BDBE strategies. The regulatory body conducted reviews of each written plan and technical basis, on-site plant audits and plant walkdowns. Finally, prior to declaration of compliance for each nuclear power plant, the RB issued a detailed plant specific safety evaluation report.
  - Post-compliance baseline inspections:
    - After a nuclear power plant owner declares compliance with requirements: Once the NPP operating organization declares, to the regulatory body, compliance with requirements, the regulatory body inspects the NPP to verify all actions are complete (as well as verify appropriate corrective action process is applied when issues are discovered).
  - Close out:
    - A close out letter is sent to each plant operating organization that provides a comprehensive inventory of key documents related their plant specific actions. This letter provides a 'roadmap' of documents and decisions for future reference to assist individuals involved with maintaining and/or modifying Fukushima related, plant specific actions.

- Long term control of plant changes to equipment, procedures or FLEX strategies:
    - Long term periodic regulatory body inspections of each nuclear power plant that is governed by a change control process that is provided in the approved industry guidance. Also, periodic verification of nuclear power plant compliance with pending new requirements for (1) nuclear power plant periodic drills, and (2) integration of each plant's EOPs, EDMGs, FSGs will be verified through the inspection program.

8. MERITS OF ACTIONS TAKEN

During the decision making for post-Fukushima actions for BDB space, particularly for those actions that added/modified SSCs and for the procurement and integration of mobile equipment, connections, instrumentation, etc., one factor was the possibility of use of those for enhancing reliability and resilience within the design basis space. Therefore, evaluations have been performed (pre- and post-implementation of actions) to assess how the safety enhancement modifications and portable equipment could be utilized — within the bounds of the member state regulations/requirements — and may provide tangible additional benefits to plant operations and safety.

The scoping studies have shown that these side benefits of the equipment BDBE response and SAM for operational safety, safety risk and performance can be gained via reductions in the site outage risk profile, outage duration, online risk profile, etc. For example, any critical path limitations in a plant's current outage configurations can be reviewed to determine if the safety enhancements/modifications and strategies could be used to address such limitations, and ultimately reduce safety risk and outage time. It is recommended that this review be performed by an individual or a group who was independent of the safety enhancement process and who has a background in plant operations and outage management. The subject review can be performed by classifying the potential improvement mechanisms into three categories:

- Procedure/strategy changes;
- Physical plant changes;
- Scheduling optimization.

This can be done to emphasize that in many cases, there is no need to modify the plant but to simply take credit for what portable equipment is already available, e.g. pre-deploy a piece of portable equipment for use within the allowed out of service time. By taking credit for what is already available in the safety enhancement strategies and portable equipment, many benefits can be realized without additional expenditure of funds.

In some Member States, it is an industry initiative to study the use of such equipment since the NPPs have purchased them and developed associated, programmes, processes and procedures towards enhancement and robustness against BDBEs with reliability and resilience. For example, in the US, several organizations are exploring the use of FLEX equipment in PRA, HRA and other areas where crediting and utilization of such equipment can be optimized for NPP operations and maintenance, risk reduction and operational safety [143–145].

Therefore, NPPs are also investigating potential trade-off, multi-purpose utilization and credit of actions taken (or are to be taken) considering the diversity, redundancy and abundancy that is provided by the SSCs and mobile equipment purchase/built/installed for BDBE response and SAM.

8.1. CREDITING BEYOND DESIGN BASIS ACCIDENT/EVENT EQUIPMENT IN PROBABILISTIC SAFETY AND RISK ASSESSMENTS

In some Member States, where failure rates and reliability data for the BDBA/BDBE equipment meet PRA acceptability standards (e.g. those of Ref. [125]), some NPPs have done some scoping work and/or modification to reduce risk from other hazards than extreme external events. Furthermore, in the USA, for example, the industry issued guidance to support expanded uses of portable equipment to improve site operations and safety [146, 147] for the development of risk reduction strategies and to establish regulatory or operational risk reduction credit. In the guidance, it is also described what evaluation need to be performed when crediting portable equipment in a PRA or DiD application. It was reported by one of the contributors of this publication, in a regulatory body and industry meeting (the public meeting, NEI/Industry Task Force on FLEX in RIDM had with the USNRC, that took place on 15 November 2019) application of RIDM was discussed, as the contributor stated:

— *“The staff noted that implementation of FLEX in a variety of different applications does add to overall plant safety, and as such, utilities should be encouraged to use FLEX equipment to reduce risk.*
— *The staff would like to see more consistency in FLEX credit in oversight (e.g. SDP evaluations).*
— *Currently the NRC licensing reviewers are using the May 30th letter (which was the staff response to receiving NEI 16-06) when FLEX is used in a risk informed application. The staff is developing an internal desktop guide to support future reviews and the staff does not intend to create any interim staff guidance (ISG) or changes to the SRP.*
— *NRC/RES staff requested that EPRI provide the NRC with the raw failure data. EPRI responded that they cannot share the plant specific data with the NRC without the consent and approval from each utility and without ensuring the data is good.*
— *The staff acknowledged that the majority (about 80%) of the HRA related modelling for FLEX is already covered under currently approved methods. The other 20% involves command and control decision making and will require further research and development.”*

There have already been several examples of utilizing portable equipment for PRA and DID evaluations and applications. For example:

— In one NPP, it was shown that the utilization of a portable SG makeup pump during certain fire scenarios may help reduce CDF.
— Another utility, in exploration of risk reduction, was able to reduce contribution of the internal flood to the CDF by utilizing BDBA/BDBE equipment in risk informed operation concept. As discussed in Ref. [148], this was performed as a part of the evaluation of the utilization of a small modification made to the plant as BDBE (installing an uninterrupted power supply to fire protection remote monitoring panel) and the use of BDBE mobile generator. This evaluation showed that the contribution to CDF from design basis internal flooding was reduced from 47 per cent to 17 per cent

(or from being the largest contributor to the second largest, after SBO) in the risk profile.

— One of the safety enhancements/ modifications being considered at some plants is to install diverse methods of making up to the SFP by providing two pathways to transfer water from the CST through a portable pump. By pre-deploying the portable pump during an outage and making the connections, this adds redundancy to the SFP makeup system (which is a critical system during a refuelling outage), improves the SFP inventory safety functions success paths and provides additional DiD.

It is also discussed in Refs [143−145] that safety and economic impact of plant occurrence, for example, component failure can be reduced by developing and applying a PRA model incorporating portable BDBE and SAM equipment for which the efforts are being conducted for a US utility plant PRA model.

8.2. CREDITING EQUIPMENT IN PLANT EVOLUTIONS

In some cases, utilization of the safety enhancement modifications and the BDBE response and SAM equipment — that are unutilized and idle for most of the time in NPP lifecycle — can result in additional operational benefits, such as avoiding plant shutdown when a component that is in the plant OLCs fails, performing maintenance with reduced risks and expenditures. These benefits gained from utilization of such equipment during normal plant operation, which can only be done/applied within the bounds of the regulatory requirements, may reduce O&M costs without adversely impacting safety. Furthermore, crediting portable equipment to avoid an imminent unit shutdown, for a failed component, could improve the quality of maintenance activities allowing a longer time to complete without time pressure as a precursor of human error.

In addition to the examples provided in Section 8, there have also been several examples of utilizing portable equipment for plant maintenance and outage evolutions. For example:

— In one U.S. NPP, using FLEX equipment, workers at a nuclear power plant developed a plan to replace an emergency service water pump to improve its reliability (although the pump was operable and this would be plant performance improvement) while the plant was online. Consequently, the operating organization requested a license amendment from the regulatory body. After reviewing the request, the regulatory body approved plan amending the license. As a result, the NPP was able to avoid an outage and remained online during the repair, saving more than US $3 million in replacement power costs alone [149].

— In another nuclear power plant unit in the USA, the 'FLEX Plus' (2 MW/4160 VAC) portable DGs and a SG makeup pump success paths contributed to a successful license amendment request in response to the event when a catastrophic failure of an EDG occurred. Without the utilization of the temporary equipment installed by the post-Fukushima actions, the unit would have had to be shut down after 10 days (i.e. the 'time-to-shutdown' requirement in the plant's OLC) until the EDGs are fixed and put back in service which would be costly considering that the average replacement power cost is US $1 million per day (note that this varies between winter and summer with the summer being higher).

The contribution from the utilization of aforementioned post-Fukushima equipment to the risk input presented in the license amendment request, which was reviewed and approved by the regulatory body, ultimately allowed that plant to remain online during EDG repair effort. The license amendment request extended the 10-day OLC 'time-to-shutdown' time by 52 days, allowing a 62-day repair (which was completed in 57 days).

In this unit's case, the replacement power costs would be nearly US $57 million during the repair that took 57 days while the plant remained online. In comparison, the cost of the portable two 2 MW/4160 kV AC portable DGs that were utilized to accomplish this evolution was approximately US $1.8 million (US $900 000 each). It should be noted that the DGs were one time purchase to be stored at the site for a need during a BDBE, and there were other insignificant costs associated with the installation of breakers (see Section 4.2.2).

It should be also noted that there were key input, assumptions and factors submitted to the regulatory body for review and approval in granting 52 days extension including:

- The plant has to have a PRA compliance with the regulatory requirements (i.e. Regulatory Guide 1.200 [125]);
- The plant has to have PRA models for: Internal flood, internal fire and seismic events;
- All other events have to be screened out;
- The plant configuration has to match the assumptions in the PRA.

Also, the following risk management actions became license commitments in the license amendments requests for this unit:

- Demonstrate proficiency in the time based use of the temporary installed portable equipment;
- Determine and implement processes and procedure changes for control of activities in the applicable unit to reduce risk (e.g. fire watches, stringent control of transient combustibles, all activities through the outage control centre (OCC));
- Minimize or eliminate all discretionary work at the applicable unit;
- Implement a 'protected equipment scheme' including switchyard, SBO generators, portable equipment, etc.;
- Assign a 'regulatory risk manager' (in OCC) and a dedicated temporary equipment operator;
- Additional 'time to shutdown criterion' (six hours) based on unavailability of the protected portable equipment.

Repeating the example provided in Section 8.1, the provision of additional DiD was without adding any additional cost.

9. ANALYSIS OF COSTS AND BENEFITS OF POST-FUKUSHIMA PLANT IMPROVEMENTS

In decision making, the operating organizations naturally considered the associated costs and added benefits, as discussed in Section 3.4.3. In many Member States, costs of a specific action, such as those required by the regulatory bodies, were not a factor in deciding whether the action is to be taken or not, but they were considered in deciding on alternative options, if available, or in the prioritization, planning and scheduling of actions. In one Member State, to be noted, the regulatory body collected the estimated costs of proposed/considered enhancements at the beginning of the process and referred to standard methodologies for value/impact assessment[20], such as the one provided in Ref. [40], in deciding on the requirements on actions to be taken and their implementation schedules.

This section surveys and studies the costs and benefits associated with the post-Fukushima actions, including, to some degree, the costs and benefits for the other stakeholders than the operating organizations.

The discussions in this section is not to justify or criticize the cost/benefit considerations, which would be impossible to do since each utility has different approaches, alternatives, corporate strategies, long term plans, regulatory framework, electricity market structures, reactor types, locations, and so forth. Those differences were discussed in Section 3.4, where the decision making approaches by the NPPs were presented. Rather, the following subsection analyse the costs and benefits with an overview of what have been experienced by the NPPs (and the Member States, at large).

9.1. COST ANALYSIS OF THE IMPROVEMENT SYSTEMS

In the immediate aftermath of the accident, various institutions or national regulatory bodies have tried to estimate the cost of actions to be implemented. These initial ‘ballpark’ figures had wide range of uncertainties and expected variations at the plant and utility level. These uncertainties and variations mainly due to the ever growing extent of lessons learned from the accident which drove the preliminary scope and extent of the actions and unestablished requirements for their implementation.

The study herein adopts two approaches to collect and derive costs of post-Fukushima enhancements. The first approach is based on the costs from public statements made by national authorities, companies or other bodies about the overall magnitude of the costs (see Section 9.1.1.1). This can be characterized as a *top down approach*, a compiled estimate of the total cost per reactor in each country.

The second approach is a more detailed *bottom up approach*, where it is necessary to establish a breakdown of the work structure to cost all the subtasks individually to a unit level, like tonnes of concrete and hours of labour and then sum the cost (see Section 9.1.1.2).

Each method has its own advantages and disadvantages, but they should lead to similar results, assuming that the published costs are correct and the subtasks are correctly identified and the costs for each are reasonable.

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[20] Here, value means reduced risks and impact means costs incurred.

There are, however, some important issues that need to be addressed in deriving cost estimates (see also Section 3.4.3):

— It is a complex task to define what is an appropriate level of enhancements, e.g. achievement goals, ‘reasonable’ assurance of safety, DiD, ‘acceptable’ risk or consequences, etc. Expenditure on enhancement can be very high if every conceivable safety and safety related addition is made to an operating plant and to its surrounding infrastructure (although to be noted, IAEA Safety Standards, in particular safety fundamentals and requirements, provide an objective, transparent and technologically neutral basis for what constitutes a high level of safety, or ENSREG in Europe, for example, can reach consensus on the definition of ‘reasonable’.

— Nevertheless, regulating safety is a national responsibility so there were no internationally standard requirements; operators need to follow their national safety authorities’ requirements. National regulatory frameworks differ somewhat in their approaches to dealing with the licensees. Some are highly prescriptive, making lists of requirements which have to be followed exactly, while others place more responsibility on the operators proposing and implementing their own safety improvements.

— It is arguable that cost was a consideration but not a major one in the post-Fukushima environment, where it has been taken as given that many safety enhancements had to be carried out (even regardless of the cost) to potentially mitigate the probability of a similar accident happening.

— With the existing stock of reactors already, on average, more than 30 years old, it can be difficult to separate out costs specifically related to Fukushima Daiichi accident from those related to equipment enhancements that are necessary specifically to extend operating licence for 10 or 20 more years[21]. As such:

  - Some reactors may already have enhancements in their original design
  - Previous regulations and compliance with them have already been implemented;
  - Periodic safety reviews and associated upgrades have already been implemented;
  - The same level of safety necessitates different enhancements in older or newer plants.

— Even within the same country which subject to same regulations, the costs vary, depending on:

  - Completed upgrades that were required prior to Fukushima Daiichi accident;
  - Upgrades serving multiple goals (e.g. CST hardening could be under Fukushima upgrade or under fire protection compliance, or a SPRA model could be under capital project of all PRA modelling);
  - Meeting the minimum requirements versus implementation for long term strategies, including trade-off, multi-purpose utilization and crediting some actions, etc., during normal operations, i.e. gaining safety, design and operational margin for performance and production.

— For nuclear power plant operators who operate in competitive power markets, costs data are confidential and the information in the public domain is limited.

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[21] For example, in the case of the USA, a majority of the reactors have already achieved 20-year extensions, extending each reactor’s operating license to 60 years. In the case of France, a large fleet of reactors is now approaching 40 years of operation and plant life extension is under consideration, that would require substantial expenditure.

— A high level assessment of the available data shows substantial differences in the costs between different countries. To some extent, these may reflect differences in reactor types, the age of the operating units, different regulatory approaches and the local supply chain cost of key components and systems.

9.1.1. Summary of costs

Two approaches have been utilized for developing estimates on the basis of breaking down the required work into subtasks and costing them individually: a literature search (estimated costs) and the IAEA survey requested from the owner operating organizations (actual costs), as presented in the following sections.

Here, it should be also noted that, in the USA, the regulatory body, USNRC initially collected estimated cost data from all US plants to be used in the value/impact assessment in the rulemaking. This data was very comprehensive and, at the end of the implementation, estimated cost listed in Ref. [37], which consists of a set of appendices, App. C through App. M) of Ref. [84], were proven to be very close to actual costs that were reported by the US plants in response to IAEA survey and discussed in Section 4 through Section 6.

9.1.1.1. Literature search

Firstly, a search of the available literature was made and some figures procured, in an attempt to obtain some initial 'ballpark' numbers. Table 1 presents the initial estimates of various SSCs that were being planned to be purchased and installed/stores. To synthesize this, a general estimate of the costs can be calculated (in current USD) per reactor. The numbers in Table 1 add up to total cost estimation of around US \$150–160 million per reactor, only for safety upgrades. This does not take into account the equipment needed for off-site facilities, for example the 'nuclear rapid response force' which is estimated to be around US \$225 million by EDF [150]. Also, the extra cost of decision making, such as the stress tests, is not taken into account. Therefore, it is a rough estimate of the most commonly recommended safety enhancements to be implemented on reactors without taking into account any of the surrounding needs (e.g. additional emergency and preparedness training, heavy equipment like bulldozers, fire trucks).

TABLE 1. EARLY COST ESTIMATION OF SAFETY UPGRADES

Cost type	Cost per reactor (million US$)	Description
Hazard protections	20	For all kinds of hazard protection seems reasonable, even if it is a very site specific charge.
AC/DC emergency support	100	An overall estimation of can be used because the detailed requirement for a number of chargers, batteries, portable diesel generators, and the possible need for an extra diesel generator is not clear. It is clear, however, that the requirements are very plant and modification specific, and the estimate here would not represent a typical cost for a typical modification.
Filtered containment venting systems (FCV)	15–30	With all the examples, a rough estimate per reactor can be assumed.
Passive autocatalytic recombiners (PAR)	3.5	For a PWR in the USA
Passive autocatalytic recombiners (PAR)	2	In Europe - The estimated price in Europe was lower because the USA imports them from Europe.
Pumps	1	The extra number of pumps will not be standardized from one reactor to another but can be estimated as US $5 million per reactor.
Total cost	**150–160**	This was proven to be very off (See Section 3.4.3)

9.1.1.2. Survey results

Secondly, a survey was distributed to nuclear power plant operators requesting information on the costs. Results of this survey are also included in Section 4 through Section 6.

One can summarize the information on costs from the survey results under key headings, trying to provide rough averages per reactor, as illustrated in Table 2.

TABLE 2. ACTUAL COSTS REPORTED BY OWNER/OPERATING ORGANIZATIONS

Cost Type	**Cost Per reactor (million US$)**
Seismic walkdowns, hazard assessment and analytical work	5
Flooding walkdowns, assessment and analytical work	2
Low voltage AC power supplies	1
Medium voltage AC power supplies	2
DC power supply	1
Core cooling systems	3
Spent fuel protection	2
Hardened containment vent	10
Modifying ultimate heat sink	1
Emergency management systems	2
Post-accident management systems	2
Construction of protected storage	5
Total	**36**

The total of all these items is US $36 million per reactor. It is clear, however, that there are substantial differences dependent on the individual circumstances, such as the nature of the reactor site, the type of reactor and the particular countries concerned.

9.1.1.3. Reconciling the two approaches

Particularly where there are multiple units on one site and/or the operator has several nuclear stations, it should be possible to cut costs per reactor by economies of scale such as any action taken in the site or fleet level. Industry association of technologies with similar technologies, e.g. common structures, analyses, assessment.

9.1.2. Comparing the cost figures

Comparison of costs can be concluded and summarized as follows:

— There is a considerable variation in the cost of necessary post-Fukushima safety enhancements.

— Excepting Japan, where expenses are far higher, the costs lie in the range of US $20 million to US $200 million per reactor. High end of the costs was mainly driven by permanent and safety grade, external hazard proof structures, equipment, achievement goals.

— By comparison with other capital expenditure to allow reactors to operate, the post Fukushima costs are not particularly high; however, many units have already done those major plant upgrades, and therefore, they are sunk costs for those units. (then it should be counted as accumulated).

— More money spent on enhancements in regulated market than deregulated market owing to the possibility of passing the cost to the consumers.

9.2. BENEFIT ANALYSIS OF POST-FUKUSHIMA IMPROVEMENTS

The benefits of post-Fukushima safety improvements depend on the perspective and goals of each stakeholder:

- For the plant owner, the relevant decision is whether to implement the safety improvements required by the government post-Fukushima. The direct benefit to the plant owner of implementing post-Fukushima safety improvements is that the reactor can continue operation, sell electricity, and generate revenues. The owner can avoid costs that may incurred when accident happens such as liability, loss of asset, needed power replacement cost, etc.
- For electricity consumers, any direct benefits relate to the electricity price. Thus, they depend on whether plant owners choose to (or are forced to) pass on any net benefits they get to consumers in the form of lower electricity prices. In that sense, regulated market plants are in better position to spend more than minimum required.
- For the society, as a whole, direct benefits are the reduced risks of/by accident in terms of adverse consequences (and probability) that include both health (radiological, psychological, and/or degraded quality of life, caused by evacuations, etc.) and socioeconomic (e.g. contaminated land and its recovery cost, evacuation/relocation of residents, etc.) impacts. The society can also benefit by avoiding/minimizing the burdens in overall impact in daily and future living, such as greenhouse gas (GHG) emission, energy and power supply security, jobs and local economy near nuclear power plants.

The important decision takes place earlier, when the society, through its government, decides to require new, post-Fukushima safety improvements. The principal benefits of requiring such improvements are the avoided costs associated with what might happen if those requirements were not imposed.

Such costs include, first, the costs of the possibility of another Fukushima accident. Second, because the requirements probably increase safety beyond reducing the risk of another Fukushima accident, the costs avoided by the new regulations are probably greater than just the costs of the possibility of another Fukushima accident. Third, without required post-Fukushima safety improvements, the risk of public opinion forcing the shutdown of reactors would be higher, and shutdowns would eliminate external socioeconomic benefits associated with nuclear power. Finally, there are indirect economic benefits to society as a whole if plant owners choose to make required post-Fukushima safety improvements and continue operating rather than shutting down. This is because it is cheaper to make required improvements and continue operating than it is to switch to the next least cost alternative. Cheaper electricity is good for the economy, and thus creates positive economic benefits for society as a whole.

9.2.1. Benefits for plant owners

9.2.1.1. Direct benefits

For the nuclear power plant owner, the direct benefit of implementing required post-Fukushima safety improvements is that the reactor does not get shut down. It can continue to operate, sell electricity, and generate profits.

The value of that benefit is equal to the costs that the owner would incur if the reactor were shut down, hereafter referred to as *avoided costs*.

The principal such cost would be the cost of replacement power which varies among reactors, and the country at large, particularly in the short term. Regardless of the cost, even if it is expensive as in case of Japan ($30 billion per year for 2011–2015), the ability to build new lower cost power plants may also be limited. Where new power plants (whether gas, coal or renewable) would be lower cost options than continued power purchases, the costs of the least-cost alternative will vary from site to site — from possibly less expensive gas fuelled plants using fracked natural gas in the USA to more expensive coal fuelled plants using imported coal in countries like Japan and Republic of Korea. In the short term this would be the cost of purchasing replacement power. This might be the only replacement power cost if the reactor were nearing the end of its planned lifetime and due to soon shut down permanently. But if the reactor were to still have most of its planned life ahead of it, and if the least-cost long term alternative for providing the lost power is a new power plant (whether gas, coal, or renewable), then the long term cost of replacement power would be the cost of the least-cost alternative plant.

A second cost of shutting down is the cost of bringing the reactor's decommissioning fund up to the level needed to decommission the shutdown reactor. This will also vary among reactors. For old reactors it could be small. For new reactors it could be substantial. Another aspect may be not being able to deliver ancillary services like grid stability.

Currently, a harmonized approach as to what needs to be in value, particularly image and power replacement costs do not exist. Since the extent to which nuclear power plants (i.e. reactors of similar design or all the reactors) are forced to shut down, whether they are shut down until licensed to restart (e.g. in case of Japan) or if phased out forever (e.g. German case), or else, would depend on societal and political environment of the country where accident occurred, decision on inclusion of power replacement costs would depend on country specifics which may be a significant factor preventing harmonization among the Member State nuclear power plants.

As noted in Section 9.1.2, the costs of post-Fukushima improvements are not high compared to costs that plant owners normally incur to extend a reactor's lifetime or replace major equipment (e.g. SGs). It is therefore not surprising that, the benefits of making the required improvements and continuing operation are significant, i.e. the costs of those improvements are much less than the avoided costs of shutting down the reactor and acquiring replacement power.

9.2.1.2. Indirect benefits

Implementing post-Fukushima safety improvements allows a plant owner to keep the reactor running. That generates the direct benefits discussed in the previous section. However, the new post-Fukushima safety equipment, safety procedures or safety margins may also allow the plant owner to resolve certain operational problems more cheaply than would otherwise have been the case. We refer to cost savings from these situations as indirect benefits to the plant owner. A specific example, the 2016 catastrophic failure of an EDG that was presented in Section 8 is the best way to make the point. For this reactor, the usual cost of replacement power during a shutdown was US $1 million per day. Thus, this incident alone generated a US $57 million indirect benefit to the plant owner, more than the average cost of US post-Fukushima safety improvements and much more than the cost of the equipment involved, US $1.8 million.

Calculating illustrative averages for indirect benefits is much more difficult than calculating illustrative averages for direct benefits. Still, indirect benefits are already much greater than zero, as the above example shows, and they are likely to increase as operators learn how to cost-effectively use their post-Fukushima improvements in situations other than just a Fukushima-like natural catastrophe.

9.2.2. Consumer benefits

Section 9.2.1 defines the direct benefit to a plant owner of implementing post-Fukushima safety improvements as equal to the avoided costs associated with shutting down. Similarly, the direct benefit to the consumer is the avoided price increase that would have been associated with shutting down. This depends on what cost share the plant owner would have passed on to the consumer (which would be dependent on market structure and rules) in the event of shutting down, and what share of the improvement costs get passed on to the consumer.

For the purposes of this report, however, little would be gained by analysing how such costs might be divided between different plant owners and their consumers. It is sufficient to discuss the total benefits as equal to the avoided costs of shutting down, plus any indirect benefits. How they are divided between plant owners and consumers does not change the important conclusion that the benefits of the improvements outweigh their costs.

9.2.3. Wider socioeconomic benefits

9.2.3.1. From the perspective of the plant owner/operating organization

To what extent should a plant owner look beyond the direct and indirect benefits in Section 9.2.1 and consider broader socioeconomic factors? For example, keeping the reactor running might avoid external costs, such as, increased GHG emissions from non-nuclear power replacement (in case of corporate strategy being a low carbon emitting company), loss generation or impact on local socioeconomics and loss of corporate reputation.

The answer is that these 'external costs' should be considered by the plant owner to the extent that they have been internalized by government rules. For example, in many jurisdiction electricity generators are required to hold permits for carbon emissions associated with their electricity generation. If a plant owner has to buy replacement power from a generator using coal, for example, the cost of emission permits for the coal plant will be included in the cost of replacement power. It will therefore be included in the plant owner's calculation of the costs he will avoid by keeping his reactor running.

To the extent that the items listed above have not been internalized into the plant owner's cost calculations by government rules, they do not feature in the plant owner's decision. They remain true external costs. The government may take account of them in its decisions, but if it has not created rules that internalize them into the plant owner's costs, the society has, through the government, decided that these costs are not the plant owner's responsibility.

9.2.3.2. From the perspective of the government

The government's decision to impose new post-Fukushima requirements is different from the plant owner's decision about whether to implement improvements or shut down. The government's decision is whether the benefits of imposing the new post-Fukushima requirements are worth the incremental costs to the plant owners and possibly others (such as, at least, government regulators themselves).

The principal benefits of requiring such improvements are the avoided costs associated with what might happen if those requirements are not imposed. Such costs include, first, the costs of the possibility of another Fukushima accident.

Second, because the requirements probably increase safety beyond reducing the risk of another Fukushima accident, the costs avoided by the new regulations are probably greater than just the costs of the possibility of another Fukushima accident.

Third, without required post-Fukushima safety improvements, the risk of public opinion forcing the shutdown of reactors would be higher. Shutdowns would eliminate the external socioeconomic benefits of nuclear power, such as low GHG emissions and greater energy supply security.

Also, there are other categories of wider socioeconomic benefits affecting the government's decision to require post-Fukushima safety improvements, including:

- Lost jobs if the reactor shuts down (mainly local but can be expanded to the supply chain);
- A lost contribution to the local economy, education;
- Possible negative gross domestic product (GDP) impacts from shutting down.

Further benefits to the plant owner and/or the government of requiring post-Fukushima safety improvements includes avoiding, with at least a higher probability than before, these accident related costs. Some such lesser accidents might only involve unplanned temporary shutdowns and thus impose only the socioeconomic costs of GHG emissions from replacement power or a small reduction in energy supply security. Bigger accidents avoided because of the post-Fukushima improvements might have socioeconomic costs in all of the categories above, even if they would not match the Fukushima accident's costs. However, any effort to quantify the avoided costs of such lesser accidents would require comparisons of probabilistic risk assessments done before and after post-Fukushima safety improvements[22].

Once the government introduces such requirements, there are also wider socioeconomic benefits if plant owners choose to meet the requirements rather than shut down. This is because it is cheaper to make required improvements than it is to switch to the next best alternative.

[22] It is not publicly available information whether any such comparisons have been made at any plant, sector or economy level. Furthermore, it is not included in the scope of this publication.

10. FURTHER NEEDS FOR ACTION

The forensic analyses of plant equipment and instrumentation failures has been an on-going effort and they will be continuing in the long term. Not only the potential future finding, but also the existing information and knowledge will be a part of necessary actions in the future to strengthen the nuclear safety. In this section, the scientific and engineering actions that may create more lessons learned and more associated post-Fukushima actions by and for operating organizations are discussed, based on the knowledge thus far.

10.1. RESEARCH AND DEVELOPMENT NEEDS HIGHLIGHTED BY THE FUKUSHIMA DAIICHI ACCIDENT

10.1.1. BWR water level measurement

In BWR type of reactors, several SAM measures rely on the reactor vessel water level indications. In Fukushima Daiichi accident, owing to the loss of DC power and the harsh or extreme conditions, the level indicators were either not available or provided false indications, some of which are proven to be non-conservative. These problems with the water level indications were also deemed to be associated with instrument calibration since reference columns and electronics for level and pressure readings were affected or potentially affected by the harsh conditions. One of the reasons for unpredictable and abnormal measurements or instrument readings for the same parameter, such as one instrument reading offset from another or divergent readings, could be possibly that.

Nevertheless, the evaluations so far are not conclusive as to provide an action to take for NPPs, although some have been exploring and installing potential fixes. For example, a Finnish NPP explained in Ref. [151] for the diversification of reactor water level measurement:

> "Diversification of reactor water level measurements: The reactor water level measurement system consists of four parallel subsystems, two of which are sufficient for implementing the protection function (from high and low level). The subsystems are based on differential pressure measurement. TVO has studied possibilities to supplement the currently used low level measurement system with another system based on a different measuring principle. TVO's plans to implement the modification have been delayed. Design work is progressing and the current schedule is to install the new devices for test use in annual outages of 2020 and 2021. However, at the same time TVO also studies whether similar safety benefits could be achieved by other methods and will send an application to STUK during 2019" [151].

Therefore, it has been suggested to conduct additional research on the failures of Fukushima Daiichi' reactor vessel water level instrumentation to determine root and contributing causes to enhance the instrumentation and indications in BWRs in general.

10.1.2. Accident tolerant fuel

The current design of most of the commercially used nuclear fuel consists of uranium oxide pellets clad in a zirconium alloy tube. While nuclear fuel designs have been optimized over the years for economic reasons, the zirconium cladding has remained relatively unchanged. While zirconium provides an excellent material for strength and resistance to neutron absorption, it does react under high temperature conditions with steam to oxidize and create hydrogen gas. As demonstrated at Fukushima Daiichi, the hydrogen gas can worsen the plant conditions during a severe accident.

Redesigning nuclear fuel remains a very costly and lengthy process due to the amount of testing required to confirm the safety of any new fuel design, including lead test rods, lead test assemblies, and full fuel changeout which can take at least three operation cycles to complete. While concepts for an accident tolerant fuel that would not worsen an accident existed, there was no real economic driver for such fuel development to be completed in the near term.

However, the consequences of the Fukushima Daiichi accident renewed interest and funding commitment from several sources to develop new accident tolerant fuel (ATF) designs, as discussed in the IAEA publication TECDOC-1797 on ATF meeting [152] as:

> "With the renewed interest in ATF, the nuclear fuel suppliers all have begun Research and Development programs looking a different Uranium pellets and cladding systems. Owning to the significant benefits to both safety and economics of the new fuel design, nuclear plant operators have been willing to offer their reactors to install lead test rods and assemblies. Current schedules list lead test rods ready by 2018 and lead test assemblies by 2022, this aggressive schedule has been made possible by the funding by governments, utilities and fuel vendors owning to the enhanced safety and economics that ATF can provide" [152].

10.1.3. In-vessel retention

In-vessel retention of the molten core has been hypothesized as an accident mitigation strategy to keep the core within the vessel. This strategy consists of flooding up the reactor cavity and putting water around the outside of the vessel during the accident. When the core melts and relocates to the lower plenum of the vessel, heat transfer would occur through the vessel wall to the water which would remove the heat through steam generation.

Due to the uncertainties associated with an accident progression and exactly how the core would relocate, and the difficulty in actually performing full scale testing that accurately simulates the core melt progression, in vessel retention remain a topic of debate by scientists and academia. Some nuclear plant designs rely on the in-vessel retention (IVR) phenomenon to avoid doing other severe accident mitigation strategies such as installing containment filtered vents.

The IVR remains an area where additional research has been proposed, although justifying the cost of the research remains difficult.

10.1.4. Passive cooling

Passive cooling is the process where operator actions are not required to remove decay heat either from the reactor core or spent fuel pool during an event. While there have been some postulated ideas such as thermosiphons, there has not been much success in backfitting passive cooling systems into existing nuclear reactor designs. The passive cooling systems tend to be expensive, especially when they are required to be hardened to address beyond design basis external events. With the safety enhancements that have been made via equipment and human actions, the need for passive cooling systems has been reduced even further.

It should be noted that the new reactor designs do have passive cooling systems fully integrated to the designs.

10.1.5. Containment integrity

Research and development, as well as benchmark on the effectiveness of containment vent systems and utilization of new and better technologies to filter gaseous iodine are on-going by several Member States.

10.2. IMPROVEMENT OF ANALYTICAL MODELS

10.2.1. Advanced severe accident analytical methods

Tool development for severe accident simulations (i.e. code and models), particularly for multi-unit/hazard/source is needed to better predict severe accident progression.

10.2.2. Component fragility models

For high seismicity sites improvements of analysis model used for fragility evaluation have been desired as the remaining topics after the Fukushima actions. With respect to fragility evaluation, improvement of response modelling and evaluation of functionality are deemed necessary. The response analysis has to take into account the nonlinearity of structural materials and soil foundation and two and three dimensional effects. The SSI effect also affects the structural and in-structure response results.

Thus, development of high fidelity analysis codes and models are essential elements for on-going and future R&D. With the capacity, seismic qualification of components has to be done by using shaking table tests and seismic experience data. In addition to the above, uncertainty and correlation are major issues for future research so as to improve the quality of seismic PRA.

10.2.3. Risk assessment models

Risk informed decision making continues to be a growing area of opportunity for nuclear plant operators that are supported by regulators with this mindset. By using risk informed decision making, plant safety can be improved while unnecessary burdens are reduced.

The continued development of risk informed decision making provides reasonable assurance the safety enhancements that are made have the highest impact on the safety of the plant. The short term actions implemented in response to Fukushima were not necessarily risk informed, but once risk informed insights were incorporated the enhancements provided safety benefits for more than postulated beyond design basis Fukushima type event. For instance, low leakage RCP seals not only provide benefit for coping times for a BDBE/BDBA, they also can be used to provide benefit for fire risk event.

10.2.4. Risk informed decision making models

Another set of models that were highlighted and has been investigated and improved is the RIDM methods and tools.

There is a view, in light of the magnitude of risks arising from accident at nuclear power plant, that assessment of risk arising from the assets owned/operated by an entity needs to be an integral part of the entity's management.

10.2.5. Human reliability assessment under harsh and/or stressful environment

It had been already recognized before the accident that HRA in PRA has difficulties in capturing human behaviour in a severe accident situation with significant complexity. This is clearer now with the complexity of Fukushima Daiichi Accident conditions, i.e. damage to facilities/components by earthquake/tsunami and core damage in multiple units, and human actions. Therefore, there is a need for changes to models, in light of data from Fukushima Daiichi accident, to build an appropriate and comprehensive HRA method.

11. MAIN OBSERVATIONS FROM THE IMPLEMENTATION OF POST-FUKUSHIMA ACTIONS AT THE NUCLEAR POWER PLANTS

This publication collected and discussed challenges and needs of nuclear power plant owners and operating organizations which implemented (are implementing) and maintaining post-Fukushima efforts in the Member States. In other words, this publication provides a report of the *lessons learned from the lessons learned*, i.e. what lessons have been learned by the industry from the response to lessons learned from the Fukushima Daiichi accident.

Post-Fukushima actions by, for and of the nuclear power plants demonstrated that the implementation of actions in order to respond to the lesson learned from the accident resulted in:

(1) increased innovation (concepts, designs, ideas);

(2) progression (thinking and strategizing long term).

More importantly, the accident, and the response to it, took nearly all the operating organizations out of complacency that was based on a belief in superior technology and on a very high comfort on human and organizational capabilities, as one operating organization outside Japan which responded to survey stated that the actions have been taken:

> *"in a spirit of humility and leadership, to protect the future of nuclear generation in the country"*.

The response to the IAEA survey and the presentations and discussions among the experts from operating organizations in the IAEA Member States during periodic IAEA technical meetings, conferences and expert meetings showed a continued commitment from the experts to prevent and mitigate severe accident and willingness to learn from global practices.

They also showed that in order to increase public trust and confidence in operating organizations' commitment to prevent and be ready for events similar to the Fukushima Daiichi accident.

However, the operating organizations recognize several challenges both in the implementation of actions in response to Fukushima Daiichi accident as well as the future progress and sustenance. The challenges in post-Fukushima actions were brought up by the operating organizations included:

- Preservation of information, knowledge, experience and competence gained by the implementation of actions, particularly those that are not subject to the existing quality assurance programme requirements, such as design and documentation control;
- Development of analytical methodologies such as realistic damage descriptor for fragility assessment of SSCs, multi-unit risk assessment, evaluation of human reliability in harsh environments including multi-hazard, multi-unit accident environment;
- Development of methodology for complementary use of probabilistic and deterministic approach and advanced plant analysis tools, particularly for severe accidents that is necessary to better predict accident progression as well as model the uncertainties and assumptions, in order to prepare guidance, training and simulators;

- Method and programme development for qualification, testing, maintenance and protection of post-Fukushima SSCs from credible hazards, particularly for seismic considerations and harmonization of regulations in terms of qualification standards and on redundancy requirement;
- Need for definition and application of design extent conditions, BDBE/BDBAs and associated level of DiD (i.e. Level 3 or Level 4 and 5), particularly involving and provoking requirements of redundancy, independency, physical separation, etc.;
- Diversity in the hazard frequency for different hazard to consider for actions, for example ranging which ranged from 10^{-4} per year for flooding to 10^{-7} per year for hurricane within the same country;
- Delays in technology developments and the long duration that it takes to deploy them, e.g. accident tolerant fuel that would help to avoid non-condensable gas production during the core melt and relief the operating organizations from some of the actions that are implemented and to be sustained;
- Acceptable and adequate methods and programmes to ensure availability, operability and sustainability of severe accident equipment.

This report overall tried to explore answers to two key questions for the benefit of owner/ operator organizations as to how their peers thought, decided and implemented:

i) If we had known what we know now on 12 March 2011, what would we have done differently?

ii) What would we wish to have/do during the implementation of the actions?

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APPENDIX I.
SURVEY QUESTIONNAIRE ON THE CHARACTERISTICS OF THE PLANT

The first part of the survey questionnaire distributed to the nuclear power plant operating organizations was about the technical and administrative characteristics as their location, design, configuration, size, age (including the vintage of technology), corporate structure and strategies, effectiveness and extent of existing programmes, etc., as shown in Table I.1. The questions included the status of the units about their:

— Design and technology type (i.e. light water, heavy water or gas cooled);
— Age (years of operation);
— Number of units at the site;
— Cooling source (i.e. lake, river, sea, ocean);
— Life extension (obtained/planned and how many years beyond the originally licensed life);
— Management/corporate structure (i.e. single unit utility or fleet).

The information requested from the plants about their characteristics was used to determine commonality and differences in actions/schedules (i.e. what made that unit to take this unique action or what made a unit not to take an action that is common for the units, say for a BWR unit versus another similar BWR unit). Naturally, the information was also used to identify different initiating events, for example, extent of issue and associated actions for flooding from a tsunami in an ocean or lake for the units located at such water bodies versus flooding from heavy precipitation for a unit on a river.

The differences in plant characteristics have illustrated some different decision making on actions and schedules, as well as the cost of implementation. Individual plant or the utility specific conditions, such as the nature of the reactor site, the type and age of reactor and the corporate strategies of owners and/or operating organizations in decision making for actions played an important role. Thus, there have been large variations in the costs and schedules of changes made (or planned), as well as sustaining the level of assets and margins gained by these implementations from one member state to another, one fleet to another, one site to another, or even, from one unit to another in the same utility or site.

TABLE I.1. SURVEY QUESTIONNAIRE ON THE UNIT AND SITE CHARACTERISTICS

Unit Technology		**Unit Age (years)**		**Site**		**Cooling Source**		**Life extension (obtained/planned)**		**Management/Corporate structure**	
BWR	☐	< 10	☐	Single Unit	☐	Lake/River	☐	+10 years	☐	Fleet	☐
PWR/VVER	☐	10-20	☐	Multiple Unit	☐	Ocean	☐	+15 years	☐	Stand alone	☐
Heavy Water	☐	20-30	☐	No. of units	☐	Other	☐	+20 years	☐		
Gas-cooled	☐	30-40s	☐	NOTES:							

APPENDIX II.
SURVEY QUESTIONNAIRE ON THE ACTIONS AND THEIR TYPE IN ACCORDANCE WITH THE SECTIONS OF THIS PUBLICATION

The main goal of the survey was to collect information and experiences in decision making, planning and implementing of actions, particularly regarding:

— Good practices and lessons learned from the implementation of actions;
— Encountered challenges and their resolutions during the implementation of actions;
— Effectiveness, sustainability and advancement of these actions in terms of impact/value (cost/benefit) in the long term horizon.

Accordingly, the survey requested a first-hand information, experience and knowledge from the nuclear power plant operating organizations regarding their actions taken (or are to be taken) as to their:

— Nature/title and aspect, i.e. physical (modifications to SSCs), analytical (calculation, evaluation assessment, modelling, walkdowns, etc.) or human and organizational (programmatic, administrative, procedural, man-machine interface, organization structure, etc.);
— Corresponding sections in the publication (table of contents was provided with synopsis);
— Drivers of actions, i.e. regulations, industry initiative, plant/fleet owner decision/expectation;
— Type of modification, e.g. permanent or temporary modification, mobile or stationary equipment, etc.
— Schedule for implementation in terms of the time after the accident as: Immediate (within six months), short term (six months to one year), mid-term (1–5 years), long term (5–8 years) or future/planned (more than eight years);
— Status of action, i.e. completed, in progress or planned (not started);
— Basis/cause/justification for implementing (or not implementing), as well as prioritization (or deferral) of actions;
— Approximate cost of implementation (rounded to the nearest thousand, hundred thousand or million, etc.), i.e. per NNP unit/block or per unit of component, etc. (for example, if each NPP unit/block gets two pumps, each costing US $8 500 it could be reported as ‘procurement of pumps US $17 000 per unit/block’ or ‘procurement of two pumps, with a price of US $8 500 per pump for each unit’;
— Challenges encountered and resolution of these issues;
— Proposed/implemented method of verifying and sustaining benefit and effectiveness, e.g. analytically, by testing, by inspection, etc.;
— Any other information or lessons learned from the decision making and implementation, such as costs/benefit (impact/value) of the action, considered options, comparison of differences in cost of options, and reasons for the differences in options considered, trade-off, multipurpose utilization and crediting of some actions, etc.

A blank survey that was sent to the operating organizations is shown in Table II.1.

TABLE II.1. THE SURVEY STRUCTURE AND CONTENTS IN FORM OF THE BLANK SURVEY THAT WAS DISTRIBUTED

REPORT SECTION	ACTION	ASPECT	DRIVERS	TYPE	SCHEDULE	STATUS	CHALLENGES	RESOLUTIONS	COSTS	METHODS FOR VERIFICATION OF EFFECTIVENESS	NOTES AND OTHER INFO
EXAMPLES											
Example: Physical Change 4.X.X	*Purchased portable, low voltage (480V) diesel generator (DG)s*	*P*	***Regulatory Requirement*** *Required by the regulatory body's order REF: [xxx]: "Each NPP shall make available a portable low voltage diesel generator that will connect into the primary or backup connections to the safety related bus and repowers the batteries."*	*M*	*M*	*C*	• *Storage of diesel generators (have to be protected)* • • *Qualification and classification;*	• *Stored in a building on-site protected from external events.* •• *Non-Q class is selected based on the approval by the regulatory body*	*The cost for a typical 800kW 480-volt Portable DG is ~ US $330,000.00 per DG*	*The regulatory body (RB) will conduct an audit prior to the preparation of the safety case. The audit will review the strategies, the analytical basis for the strategies and the implementation, as well, and issue a safety report. The NPP will conduct periodic testing as indicated in the test programme and implement a maintenance work order, in order to ensure functionality and operability*	*The 480-volt generators can repower the busses for a positive displacement charging pump, not necessarily a high pressure RCS injection pump depending on a specific plant design.* *For the 480-volt generator, either adequate isolation existed (spare breaker) or adequate isolation was installed.*

REPORT SECTION	ACTION	ASPECT	DRIVERS	TYPE	SCHEDULE	STATUS	CHALLENGES	RESOLUTIONS	COSTS	METHODS FOR VERIFICATION OF EFFECTIVENESS	NOTES AND OTHER INFO
EXAMPLES											
Example: Analytical Change 5.X.X	*Revised flooding safety analysis*	*A*	***Required by regulation**: Regulation REF[XXX] requires reanalysis of flood safety assessment with consideration of combined effects of natural hazards.*	*P*	*M*	*C*	*• Finding and hiring qualified technical support •• Significant debate about how to interpret the result and what to do if flooding hazards for extremely rare events exceeded plant design basis*	*• Resource sharing with other NPPs •• Agreed with RB that equipment, if protected and deployable, could be used to mitigate extreme flooding*	*100 000 Euro*	*n/a*	*n/a*
Example: Human / Organization Change 6.X.X	*Developed procedure for cooling spent fuel pools*	*H*	***Industry initiative** and **NPP decision***	*P*	*M*	*O*	*Developing a generic strategy for shutdown state to cover all potential scenarios*	*Utilities have volunteered to update and maintain a standard procedure to account for the lessons learned from Fukushima*	*25 000 (in local currency) per member of the utility group*	*The procedure will be tried intermittently in plant simulator during operator training*	*May need more engineering analyses to establish credible scenarios*

ACTIONS TAKEN (OR ARE TO BE TAKEN) AT YOUR NPP
(please fill as shown in examples, providing as much information as possible)

APPENDIX III.
NUCLEAR POWER PLANTS RESPONDING TO THE SURVEY

It is difficult to provide a breakdown of individual plants since some responses were provided per unit while some provided the fleet/site information with specific notes on individual units or provided two sets information on groups of units with different technologies in one site or fleet. Furthermore, since the survey is anonymous, this publication could not provide some unit characteristic that would make it very easy to guess (know) which units (unless the responders permitted or the information was publicly available). However, Table III-1 provides a breakdown in generic manner based on the units falling in the same or similar categories. Overall, 267 plants from six countries responded the survey (not including the information that was provided by 11 countries representing 67 plants during the technical committee and consultancy meetings, as well as the plants of which the information is available publicly and included in this publication).

TABLE III.1. BREAKDOWN OF NUCLEAR POWER PLANTS RESPONDING TO THE SURVEY

PLANT/ FLEET	TECHNOLOGY	AGE	SITE	LTO (+)	FLEET?	SIZE
I	PWR	>20	Multi-Unit	20	No	>1000
II	PWR	>20	Multi-Unit	None	Yes	<1000
III	PWR	>20	Single-Unit	None	Yes	<1000
IV	PWR	>20	Single-Unit	None	Yes	<1000
V	PWR	>20	Multi-Unit	None	Yes	>1000
VI	PWR	>20	Single-Unit	None	Yes	>1000
VII	PWR	>20	Single-Unit	None	Yes	>1000
VIII	HWR	>30	Multi-Unit	15	No	<1000
IX	BWR	>30	Single-Unit	15	Yes	not indicated
X	PWR	>30	Multi-Unit	15	Yes	not indicated
XI	BWR	not indicated	Single-Unit	not indicated	not indicated	not indicated
XII	PWR	not indicated	Multi-Unit	not indicated	not indicated	not indicated
XIII	PWR	>30	Single-Unit	20	No	not indicated
XIV	AGR	>30	Multi-unit	not indicated	Yes	not indicated
XV	PWR	>20	Multi-unit	10	Yes	not indicated
XVI	AGR	>30	Single-Unit	N/A	Yes	not indicated
XVII	PWR	>20	Single-Unit	10	Yes	not indicated

ANNEX I.
FUKUSHIMA DAIICHI ACCIDENT SEQUENCE FOR CRITICAL SAFETY FUNCTION

This annex adds a brief presentation of key aspects of the accident, particularly those concerning the fundamental safety functions, which initiated all the subsequent investigations, analyses and evaluations. It could be beneficial to the readers to recall the accident sequence which resulted the identification and implementation of actions based on the lessons learned. Therefore, Fig. I-1 is provided herein to illustrate the accident sequence and plant responses as a reference [I-1].

REFERENCES TO ANNEX I

[I–1] INTERNATIONAL ATOMIC ENERGY AGENCY, The Fukushima Daiichi Accident, IAEA, Vienna (2015).

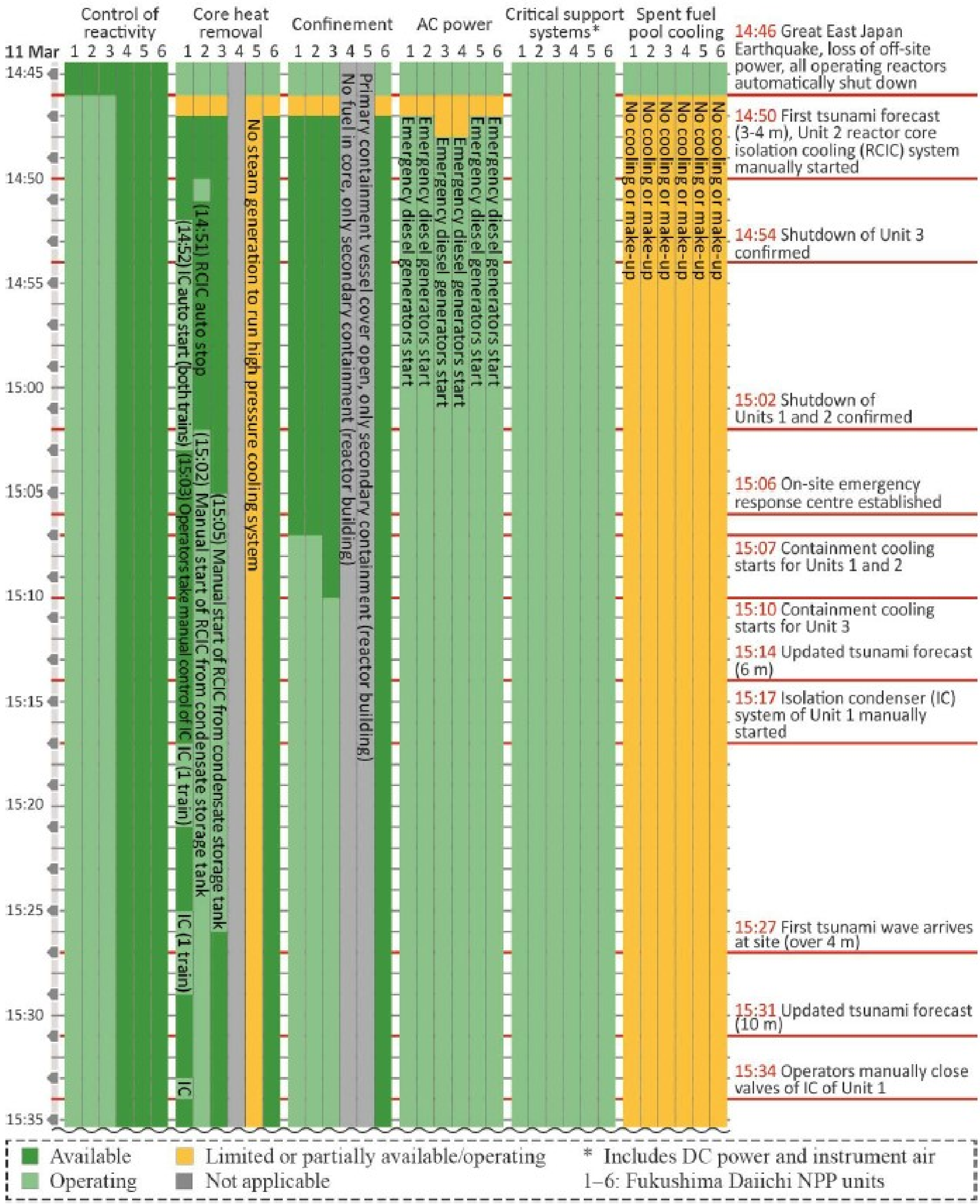

FIG. I-1(a). Sequence of events and conditions of fundamental safety function during the Fukushima Daiichi accident [I-1].

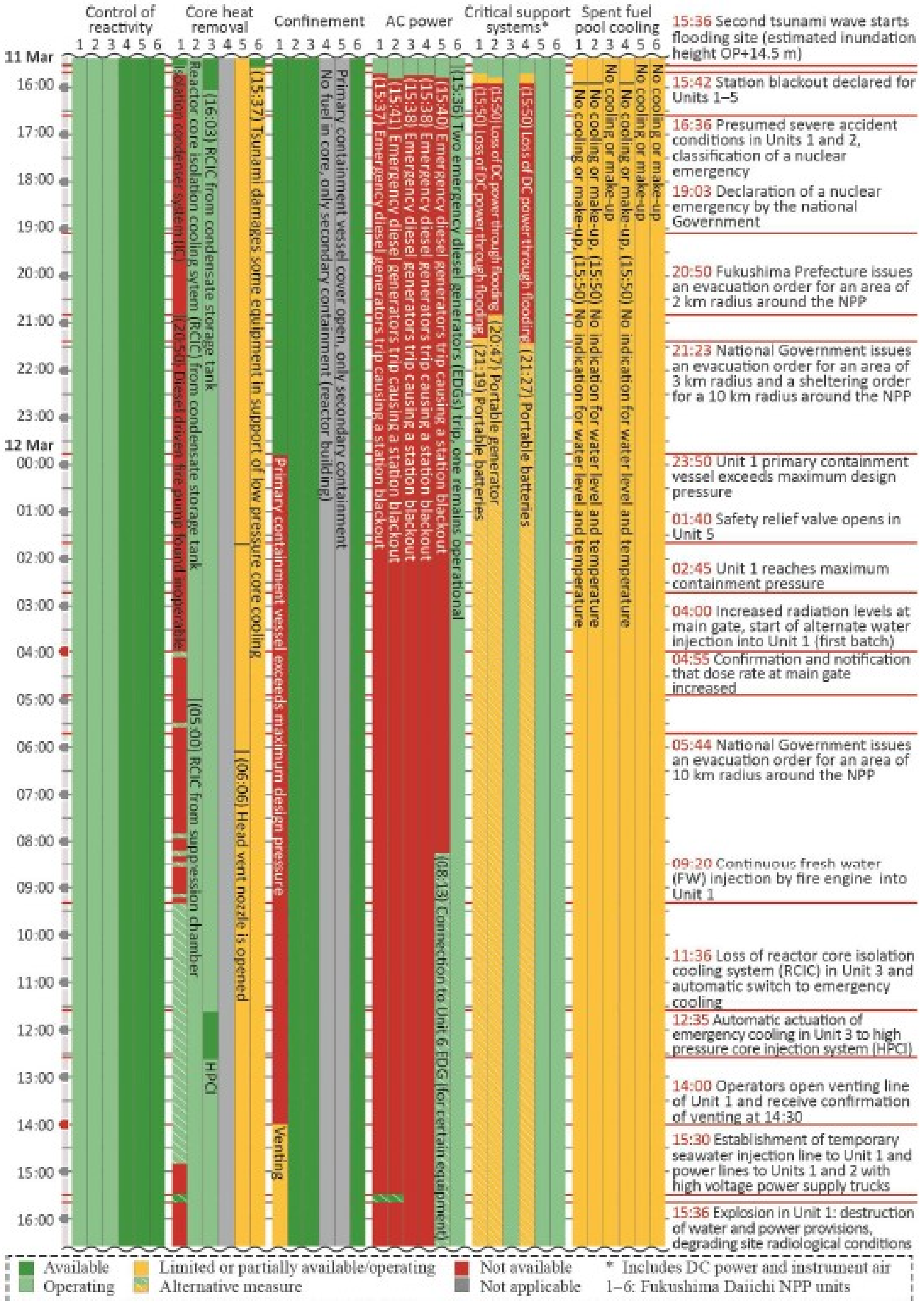

FIG. I-1(b). Sequence of events and conditions of fundamental safety function during the Fukushima Daiichi accident [I-1].

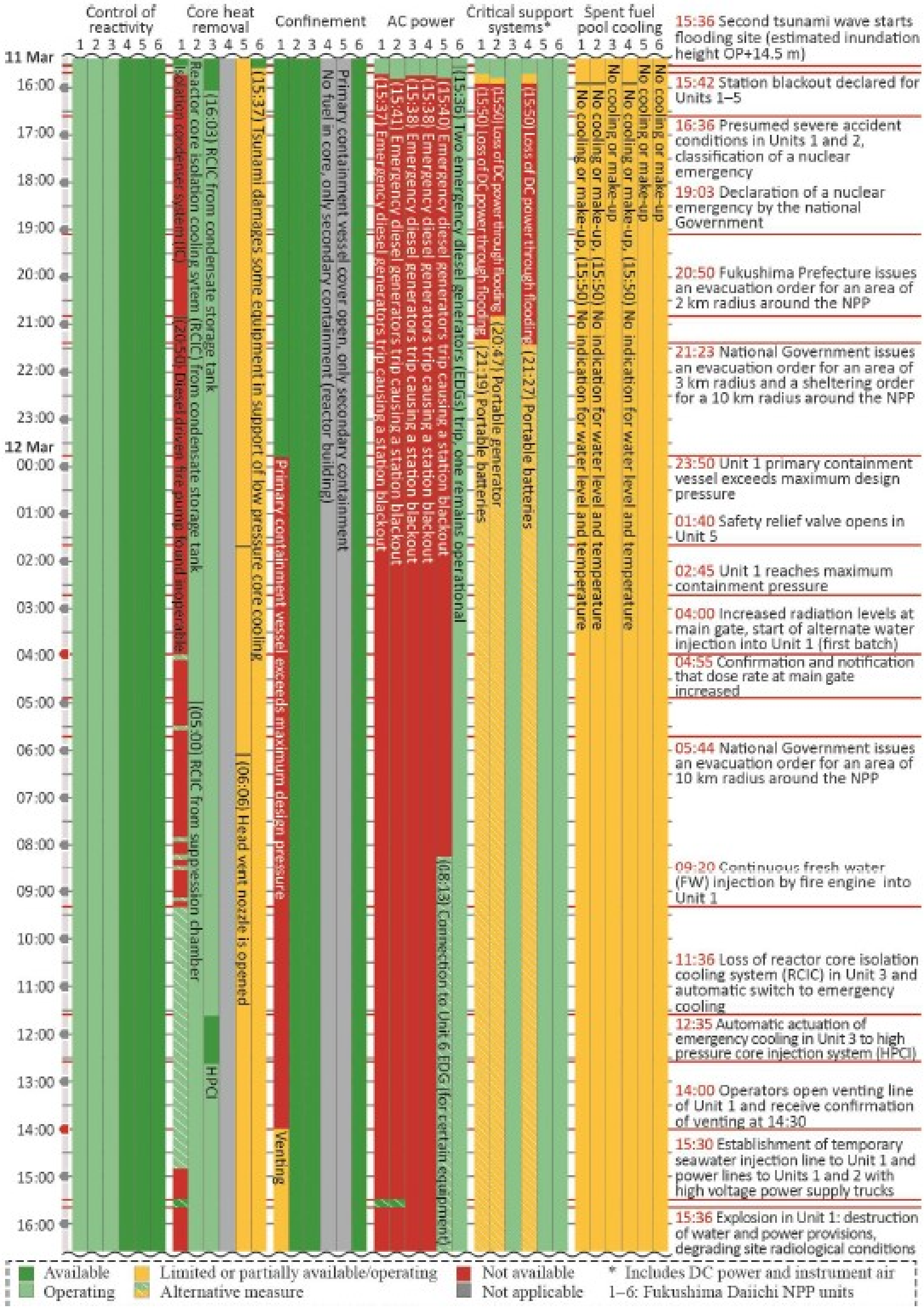

FIG. I-1(c). Sequence of events and conditions of fundamental safety function during the Fukushima Daiichi accident [I-1].

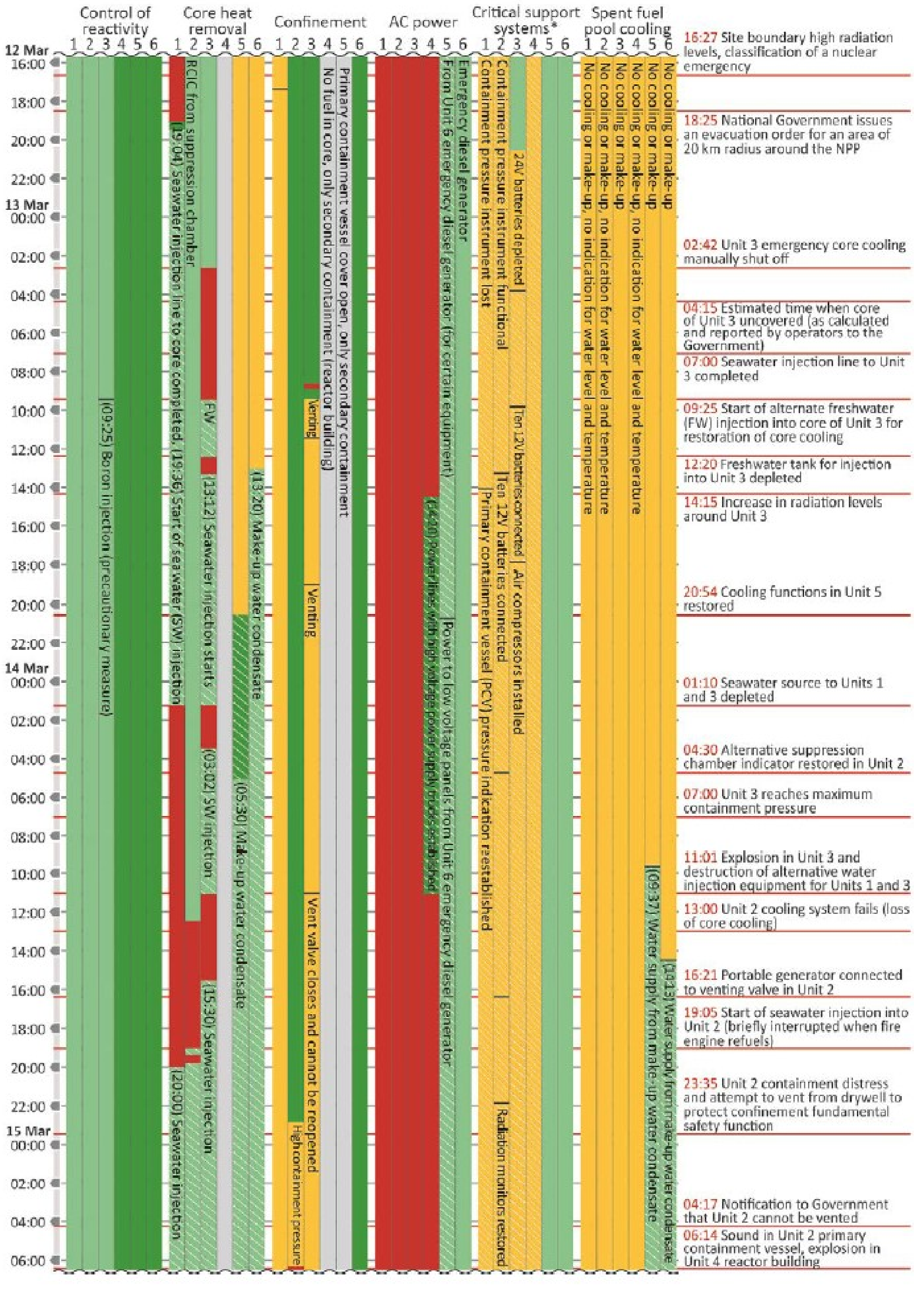

FIG. I-1(d). Sequence of events and conditions of fundamental safety function during the Fukushima Daiichi accident [I-1].

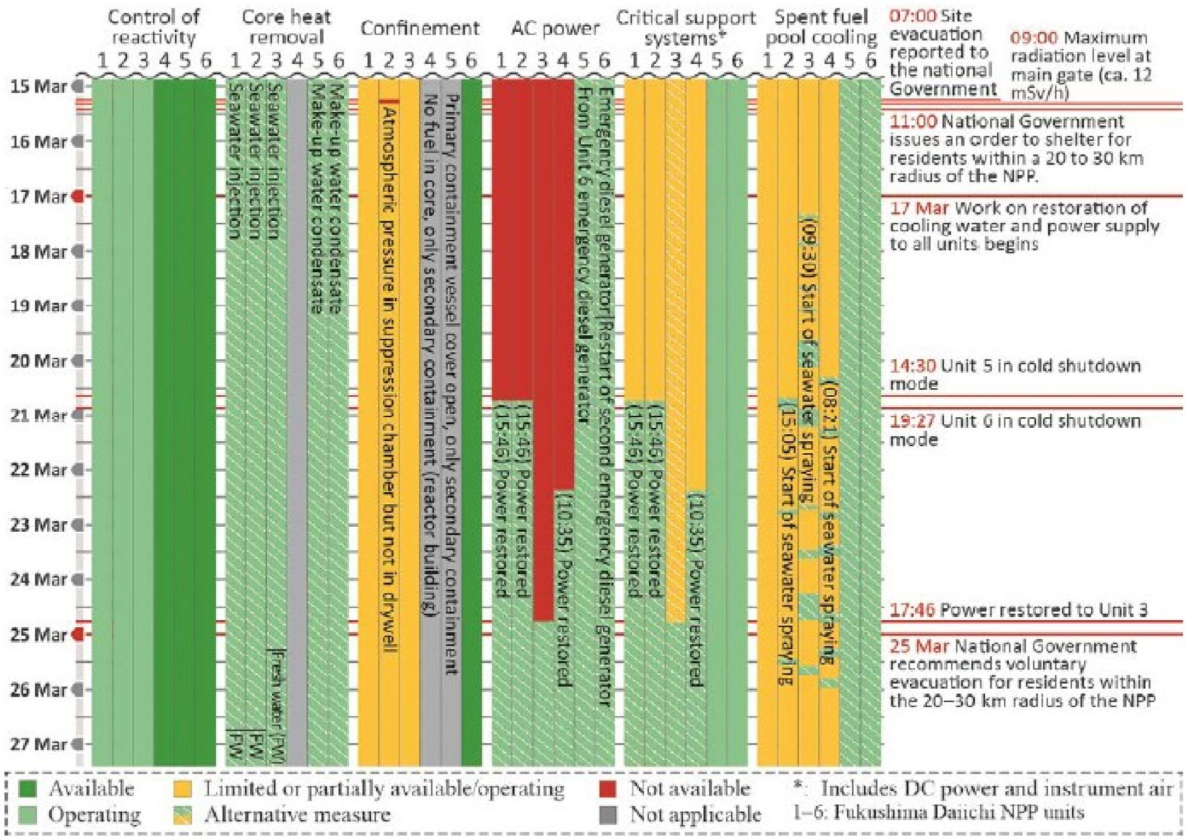

FIG. I-1(e). Sequence of events and conditions of fundamental safety function during the Fukushima Daiichi accident [I-1].

GLOSSARY

beyond design basis accident. Postulated accident with accident conditions more severe than those of a design basis accident. Also referred as beyond design basis event (BDBE).

beyond design basis event. See beyond design basis accident.

design basis external events. The external event(s) or combination(s) of external events considered in the design basis of all or any part of a facility.

design basis accident. A postulated accident leading to accident conditions for which a facility is designed in accordance with established design criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits. Also referred as design basis event (DBE).

design basis event. See design basis accident.

design extension conditions. Postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. For nuclear power plants and research reactors, design extension conditions comprise conditions in events without significant fuel degradation and conditions in events with melting of the reactor core.

ground motion characterization. Describes the excitation and propagation of an earthquake ground motion for all earthquakes that affect a nuclear power plant site (function of earthquake magnitude, distance, and frequency content).

nuclear steam supply system. The reactor and the reactor coolant pumps (and steam generators for a pressurized water reactor) and associated piping in a nuclear power plant used to generate the steam needed to drive the turbine generator unit.

nuclear unit. Comprises a nuclear reactor and all the auxiliary equipment (generator, transformers, motors, pumps, electrical supplies, protection systems, etc.) that are required for operation. A nuclear power plant may have one or more nuclear units.

operational limits and conditions. A set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of an authorized facility. Also referred as Limiting Condition of Operation (LCO).

plant evolution. The development or sequence of changes in the status of a system or equipment that is initiated by plant transients or by deliberate action of the operator.

safety case. A collection of arguments and evidence in support of the safety of a facility or activity. Normally includes the findings of a safety assessment and a statement of confidence in these findings.

seismic source characterization. Describes an earthquake occurrence in area of interest - where and how often they occur; how big they are when they do occur. This results in an estimate of the recurrence rates and intensities of earthquakes at their respective sources.

severe accidents. Accident more severe than a design basis accident and involving significant core degradation.

technical support. An activity (or part of an activity) to assist decision makers with technical and scientific input in decisions on the achievement of design and performance objectives.

technical support organization. Any organization (or individual or group) that provides technical and scientific support to decision makers for decisions on preparation for a nuclear power plant project and afterwards, for the design, licensing, construction, commissioning, operation, maintenance and decommissioning of plant.

Tier 1 screening. Completion of checklists of evaluation statements that identifies potential deficiencies in a building based on performance of similar buildings in past earthquakes.

Tier 2 evaluation. An approach applicable to certain types of buildings and Performance Objectives based on specific evaluation of potential deficiencies to determine if they represent actual deficiencies that may require mitigation. Analysis of the response of the entire building may not be required.

Tier 2 retrofit. The mitigation of deficiencies identified in the Tier 1 screening.

Tier 3 evaluation. An approach to evaluation in which complete analysis of the response of the building to seismic hazards is performed, implicitly or explicitly recognizing nonlinear response.

utility. An entity that owns assets and operates facilities for the generation, transmission or distribution of electricity/energy for commercial sale to the individual and/or industrial consumers.

ABBREVIATIONS

AC	alternating current
AGR	advanced gas-cooled reactor
ANS	American Nuclear Society
AOO	anticipated operational occurrence
AOP	abnormal operating procedure
AOV	air-operated valve
ASCE	American Society of Civil Engineers
ATF	accident tolerant fuel
BDB	beyond design basis
BDBE	beyond design basis event
BDBEE	beyond design basis external event
BUHS	backup ultimate heat sink
BWR	boiling water reactor
BWROG	BWR Owners Group
CANDU	Canadian deuterium uranium reactor
CCF	common cause failure
CDF	core damage frequency
CFVS	containment filtered venting systems
COG	CANDU Owners Group
CRIEPI	Central Research Institute for Electric Power Companies
CST	condensate storage tank
DBA	design basis accident
DBE	design basis earthquake
DBF	design basis flood
DBUE	deployable backup equipment
DBUEG	deployable backup equipment guidelines
DC	direct current
DEC	design extension condition
DG	diesel generator
DiD	defence in depth
ECC	emergency crisis centre
EDF	Electricité de France
EDG	emergency diesel generator
EDMG	extensive damage mitigation guidelines
ELAP	extended loss of AC power

EMC	emergency management centre
EME	emergency mitigating equipment
EMEG	emergency mitigating equipment guidelines
ENSREG	European Nuclear Safety Regulators Group
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
EPS	emergency protective services
ERC	emergency response centre
ERO	emergency response organization
FEPC	Federation of Electric Power Companies
FLEX	Diverse and Flexible Coping Strategies
FSG	FLEX support guidelines
GHG	greenhouse gases
GMC	ground motion characterization
GMRS	ground motion response spectrum
HCVS	hardened containment venting system
HOF	human and organizational factors
HORAAM	human and organizational reliability analysis in accident management
HRA	human reliability analysis
HVAC	heating, ventilation and air conditioning
I&C	instrumentation and control
IEEE	Institute of Electrical and Electronics Engineers
INPO	Institute of Nuclear Power Operations
IRSN	Institut de radioprotection et de sûreté nucléaire
IVR	in-vessel retention
JANSI	Japan Nuclear Safety Institute
LCO	limiting condition of operation (see also OLC)
LERF	large early relief frequency
LIP	local intense precipitation
LTO	long term operation
LUHS	loss of ultimate heat sink
MCR	main control room
MOV	motor operated valve
NEI	Nuclear Energy Institute
NRA	Nuclear Regulatory Authority
NRRC	Nuclear Risk Research Center
NSRC	National SAFER Response Centres
NSSS	nuclear steam supply system
NTTF	Near Term Task Force
OBE	operating basis earthquake

OCC	outage control centre
OLC	operational limits and conditions (see also LCO)
PAR	passive autocatalytic recombiner
PFDHA	probabilistic fault displacement hazard analysis
PHWR	pressurized heavy water reactor
PMF	probable maximum flood
PMP	probable maximum precipitation
PORV	power operated relief valve
PRA	probabilistic risk analysis
PSA	probabilistic safety assessment
PSBO	prolonged station blackout
PSHA	probabilistic seismic hazard analysis
PSR	periodic safety review
PWR	pressurized water reactor
PWROG	PWR Owners Group
RCS	reactor coolant system
RCP	reactor coolant pump
RHR	residual heat removal
RIDM	risk informed decision making
RMWT	reactor makeup water tank
RPVH	reactor pressure vessel head
SAFER	Strategic Alliance for FLEX Emergency Response
SAM	severe accident management
SAMG	severe accident management guidelines
SBO	station blackout
SFP	spent fuel pool
SG	steam generator
SIT	safety injection tank
SOG	standard operating guidelines
SPRA	seismic probability risk assessment
SRV	safety relief valve
SSC	systems, structures and components
SSE	safe shutdown earthquake
SSHAC	Senior Seismic Hazard Analysis Committee
SSI	soil-structure interaction
TDAFWP	turbine driven auxiliary feedwater pump
TEPCO	Tokyo Electric Power Company
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
USNRC	United States Nuclear Regulatory Commission

VSAT	very small aperture terminal
WANO	World Association of Nuclear Operators
WENRA	Western European Nuclear Regulators' Association
WWER	water-water energetic reactor (VVER)

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Vienna, Austria: 30 May–1 June and 7–10 November 2016, 27–30 March and 9–12 October 2017, 11–13 April 2018

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