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Review Article

The Fukushima Event: The Outline and the Technological Background

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The paper deals with the evaluation of the Fukushima-Daiichi Nuclear Power Plant (NPP) accident in Units 1 to 4: an attempt is made to discuss the scenario within a technological framework, considering precursory documented regulations and predictable system performance. An outline is given at first of the NPP layout and of the sequence of major events. Then, plausible time evolutions of relevant quantities in the different Units, is inferred based on results from the application of numerical codes. Scenarios happening in the primary circuit and containment (three Units involved) are distinguished from scenarios in spent fuel pool (four Units involved). Radiological releases to the environment and doses are approximately estimated. The event is originated by a natural catastrophe with almost simultaneous occurrence of earthquake and tsunami. These caused heavy destruction in a region in Japan much wider than the land around the NPP which was affected by the nuclear contamination. Key outcome from the work is the demonstration of strength for nuclear technology; looking at the past, misleading Probabilistic Safety Assessment (PSA) data and inadequacy in licensing processes have been found. Looking into the future keywords are Emergency Rescue Team (ERT), Enhanced Human Performance (EHP), and Robotics in Nuclear Safety and Security (RNSS).

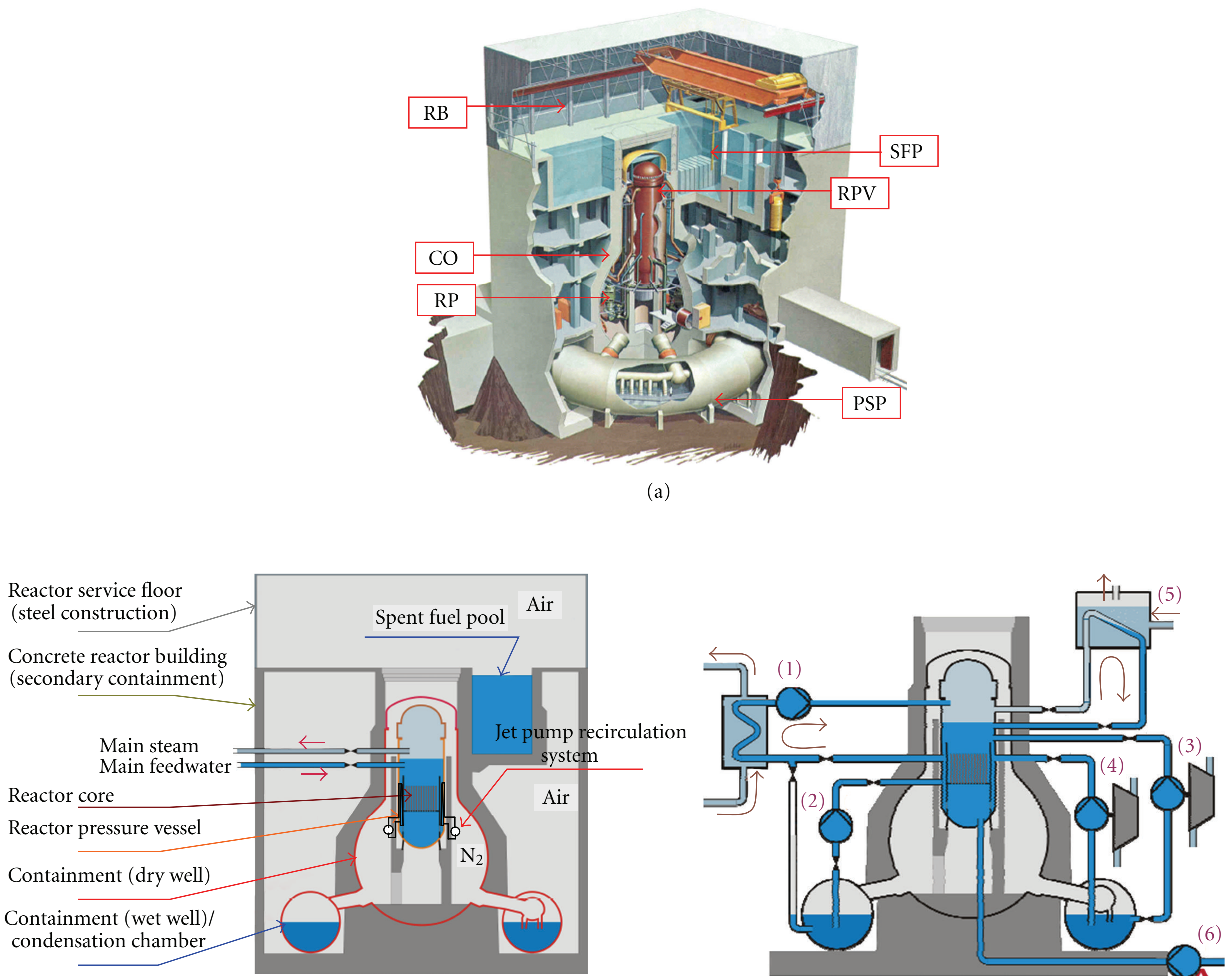
1. Introduction

On March 11, 2011 a magnitude 9.0 (Richter scale) destructive earthquake occurred in the north-east coast of Japan, close to the city of Sendai. The earthquake was followed by an equally devastating tsunami with waves as high as six meters entering a few miles into the land. Thousands of square kilometers of territory were affected, a few percent of the Japan islands surface. Whatever man-made installation like roads, factories, buildings, and dams was destroyed or severely damaged. The death toll accounted for around 30 thousands, that is, more than 2/10000 of the Japanese population.

Several Nuclear Power Plants (NPPs) for the electricity production were (and are) in the affected area, including Onagawa (3 Units), Tokai-2 (1 Unit), Fukushima Daiichi (6 Units), and Fukushima Daini (4 Units). At a larger distance, there are the Kashiwazaki Kariwa (7 Units) and the Tohoku (1 Unit) NPPs. All Nuclear Power Plants safely responded to the earthquake and to the tsunami (where concerned) solicitations, including the NPP closest to the epicenter that is, Onagawa, with the noticeable exception of four over six Units of the Fukushima Daiichi installation. Namely, the nuclear accidents in the Units 1 to 4 of Fukushima Daiichi constitute the reason and the subject for this paper.

At the time of the present writing, less than two months passed from the March 11 earthquake and already thousands of technical and nontechnical documents have been issued and can be found in the web. The title of the paper is ambitious because detailed reconstruction of scenarios has not been attempted yet: it is also too early to provide a technologically comprehensive and systematic set of lessons learned before knowing the specific failures and the reasons for the human reactions on the field. Months or even years are necessary for a sound evaluation of the Fukushima Daiichi nuclear "severe accident", but journalist type and "semitechnical" evaluations are already spread by information channels, for example, [1].

Then the key objectives for the paper are to find what went wrong and to streamline future evaluations of the accident making reference to established facts and to



(a)

(b)

(1) Residual heat removal system
(2) Low-pressure core spray (for LOCA)
(3) High-pressure core injection (for LOCA)
(4) Reactor core isolation cooling (unit 2, 3 [BWR4])
(5) Isolation condenser (unit 1 [BWR3])
(6) Borating system

(c)

Figure 1: (a) Three-dimensional sketch of the Fukushima Units 1 to 4: Reactor Pressure Vessel (RPV), Containment (CO) with toroidal Pressure Suppression Pool (PSP), and Reactor Building (RB) with Spent Fuel Pool (SFP). (b) Sketch of the Fukushima Units 1 to 4. (c) Sketch of the main ECCS of Fukushima Units 1 to 4.

qualified results from previous studies and researches, for example, [2–6]. To this aim, important pieces of information have been collected from different recognized institutions: these are acknowledged in the paper. The bases of the nuclear technology and of the nuclear reactor safety are considered.

Established relevant information connected with the Fukushima Daiichi NPP and the related accident is provided in Sections 2 and 3, with the noticeable exception of the data in Table 2 and in Figure 3. Precursory studies and research findings are discussed in Section 4 and connected with the Fukushima accident and a comparison is made with previous severe accidents in TMI-2 and in Chernobyl-4. Subjective evaluations are reported in Section 5 before conclusions.

The attention is focused on Fukushima Daiichi Units 1 to 4 and no consideration is given to the simultaneous evolutions of transient scenarios in Units 5 and 6 as well as in the "common" spent fuel on the same site.

2. The Fukushima Reactors

The Fukushima Daiichi Units 1 to 4 are Boiling Water Reactors (BWRs) originally designed by General Electrics (GE). The Mark-I containment configuration is adopted in each Unit. The typical sketch is given in Figure 1 (namely, Figures 1(a)–1(c)), and details related to power, age, main constructor, and so forth can be found in Table 1. The Owner

Table 1: Key features of the Fukushima Units 1 to 4.

Unit No	Type*	Power MWe/MWth	Construction Company	Start Operation (year)	No of FA (−)	CO design p (MPa)	No of DG (−)
1	BWR-3 MARK-I	460/1380	GE	1971	400	0.43	
2			GE	1974			2
3	BWR-4 MARK-I	784/ 2381	Toshiba	1976	548	0.38	
4			Hitachi CO.	1978			

*Reactor (above) and Containment (below) type.

(or the Operator) for the NPP is the Tokyo Electric Power Company (TEPCO).

2.1. NPP Design Features Relevant to the Event. An insight into the Fukushima NPP overall configuration and into systems which had some role during the accident of March 11 2011, is presented here, distinguishing between the overall BWR System, Primary Loop, Containment, Emergency Systems, and the Reactor Building.

The Overall BWR System. The BWR was conceived to use the steam produced in the core directly in the turbine, that is, avoiding the intermediate thermal power transfer step that implies the presence of steam generators. The connection between the Reactor Building and the Turbine Building constitutes a potential weak point in case of earthquake because of different foundation platforms, without posing safety concerns. During the Fukushima type of accident and with reference to the loss of electricity, the hydraulic accessibility to the vessel (and then to the core cooling) was possible outside the containment: this may be seen as an advantage related to NPP operated with PWR.

Primary Loop. The primary loop includes the steam lines and the feed-water lines till the respective isolation valves other than the Reactor Pressure Vessel (RPV) and the Recirculation Lines. In case of earthquake, the RPV constitutes a "solid" structure and the recirculation loop constitutes a vulnerable point.

Containment. The "Pressure Suppression Pool" is adopted for the containment of the Fukushima Daiichi Units 1 to 4 NPP as well as in all BWR. The PSP allows a reduction of pressure rise in the containment in case of discharge of thermal power from the primary loop when the same containment volume is taken. Then a reduction of containment volume is the reason for the PSP. The water pool, toroidal shape in Figure 1(a), is located at the bottom and is not at the origin of a "special" concern in case of earthquake. However, the designer improved the containment layout from Mark-I to Mark-III (i.e., passing through Mark-II): those layout changes improved also the seismic resistance. Definitely, the Mark-I containments survived the earthquake in the case of Fukushima.

The containment design pressure is around 0.4 MPa (this set-point value is connected with the containment venting time during the Fukushima accident) and the rupture pressure (this might have had a role during the Fukushima accident) can be estimated at a value roughly 1.8–2 times the design pressure.

Emergency Systems. Standard or nearly standard emergency systems are part of the Fukushima Daiichi Units 1 to 4. The following shall be noted.

(a) Unit 1 is equipped with Isolation Condenser (IC) designed to operate in the case of Station Blackout (SBO) and Loss of On-Site and Off-site Power LOOP; see the following. Related valves are battery (or DC: Direct Current) operated. The cooling pool for the IC heat exchanger is outside the containment into the reactor building and constitutes a potential for containment bypass in case of rupture.

(b) Units 2 and 3 are equipped with Reactor Core Isolation Cooling (RCIC) designed to operate in case of SBO and LOOP. The RCIC system is energized by a turbine driven by steam produced in the vessel at high pressure. The heat sink is constituted by the PSP liquid; that is, the system is operable till the saturation value. The units are also equipped with High-Pressure Coolant Injection system (HPCI).

(c) Steam Relief Valves (SRVs) are installed in the steam lines and discharge steam into the PSP; these can be used to depressurize the system.

Reactor Building. The Reactor Building (RB) surrounds the containment and, noticeably, encases the PSP. The RB is not designed to withstand any pressure greater than the ambient pressure, and so failure of the RB is expectable after venting from the containment or in case of H2 explosion.

Additional Information. At the time of the earthquake, the fuel bundles constituting the core of the Unit 4 were in the SFP due to maintenance.

Each Unit is connected with separate 275 KV external electrical grids.

The elevation of the bottom of the various buildings part of the NPP is about 10 meters above the sea level.

Table 2: (a) Cornerstone events during Fukushima accident, Reactor Core, Vessel and Containment, Units 1 to 3. (b) Cornerstone events during Fukushima accident, Spent Fuel Pool in Units 1 to 4.

(a)

No.	Event	Unit 1	Unit 2	Unit 3	Notes
1	Earthquake hits the NPP		time "0"		14.46 March 11, 2011.
2	Reactor scram		Within a few seconds after the time $t = 0$		Stop of fission reaction. RP trip. Possible LOCA. Isolation of RPV and of CO.
3	SBO event		Same as above		DBA conditions (expected).
4	Tsunami hits the NPP		55 minutes		Presumed failure of all DG.
5	SBO becomes LOOP		(1–N) hours, where "N" can be between a few and a few dozen.		Following "presumed" attempts to restart DG. SA starts. Mitigation measures needed.
6	Venting to the stack directly from D/W of CO	24 hrs.*	*, **High pressure in RPV***	*, **High pressure in RPV***	To protect containment.*mitigation measures started.**RCIC started.***most of the time.
7	Explosion (**I**) of RB	25 hrs.			15.36 March 12. Possible H2 explosion.
8	Venting to the stack directly from CO.	—		66 hrs.*	11.36 March 12. RCIC lost, HPCI initiated at 12.35 and lost at 02.42 on March 13. 09.20 March 13. CO venting
9	Explosion (**II**) of RB	—		68 hrs.	11.01 March 14. H2 flame.
10	Venting to CO from RPV.	—	80 hrs.	—	13.25 March 14 RCIC lost detected. 16.34 SRV1 opened, depressurization CO from RPV.
11	Explosion (**III**) of CO	—	86 hrs.	—	06.10 March 15. PSP damage.
12	Electricity restored in the Control Room		250–350 hrs.*		March 22–26 period. *CO venting to the stack for Unit 1.
13	Stable conditions reached for core cooling		~300 hrs.		March 25–26 period. Continued water injection into the vessel.
14	End of CO leakage	—	620 hrs.	—	April 6.Unit 2. Breakage not repaired yet.
15	Core status		Damaged*		End-of-the-Accident. *(20 to 80) % core destroyed. **continued steam release. ***less in the case of Unit 2.
16	Containment status	**Undamaged	Repaired	**Undamaged	
17	Reactor Building status		Damaged***		

(b)

No.	Event	Units 1 to 3	Unit 4	Notes
1	Earthquake hits the NPP		time "0"	14.46 March 11, 2011.
2	Occurrence of LOOP		"N" hrs.	See Table 2(a). Mitigation measures.
3	Fire and Explosion (IV) occurred in RB	—	89 hrs.	09.38 March 15. H2 production and radioactivity to environment. SA starts.
4	Injection of water in the SFP	*	200 hrs.	*Started March 31st, 20^{th}, and 17th, for Units 1, 2, and 3, respectively. Various attempts after this time.
5	Electricity restored in the Control Room	*	~ 430 hrs.	March 29. *see Table 2(a). Connected with stable cooling conditions in SFP.
6	SFP status	Cooling recovered with damaged fuel*		End-of-the-Accident. *presumed: majority of fuel bundles affected. **see Table 2(a).
7	Reactor Building status	**	Damaged	

3. Nuclear Reactor Safety and the Fukushima Event

3.1. Background for Nuclear Safety. The safety of NPP is legally quantified by the Final Safety Analysis Report (FSAR). The FSAR constitutes the key element for demonstrating the safety of each individual NPP. A rough overview of the safety can be derived by the following items which shall be seen like a pyramid, where item "(a)" constitutes the top-edge, and the results from the analyses at item (f) constitute the bottom-basis. Those results also prove the safety of any concerned NPP:

(a) fundamental principles, for example, need to protect the environment and the population from the radiations; see, for example, [7];

(b) requirements, or obligations, consequence of fundamental principles, including concepts like Safety Barriers (SB) and Defense in Depth (DiD); see, for example, [8, 9];

(c) identification and consideration of Safety Functions to protect the SB; see, for example, [10];

(d) identification of a Design Envelope (DE) of events or accidents, challenging the Safety Functions (thus the SB) primarily based upon the probability and the consequences (radiological release); see, for example, [11];

(e) establishing Acceptability Criteria (AC) for significant safety parameters connected with the Safety Functions; those criteria may depend upon the frequency of the event and shall be fulfilled during their evolution; see, for example, [10];

(f) performing analyses to demonstrate the compliance between each event part of DE (these are commonly called Design Basis Accidents or DBA) and the AC, for example, [12].

Noticeably, the containment of NPP is designed as a consequence of item (b) and has the double function to protect the environment (and the population) from the NPP and the NPP from the environment. Natural events like flooding, earthquake, and man-provoked events like airplane crash or terrorist attack are considered in the last category.

Furthermore, evaluations or analyses can be requested for events that are outside the DBA envelope. These are called Beyond DBA (BDBA) and include Severe Accident (SA). In this case, the compliance with the DiD shall be demonstrated, and countermeasures, including plans for mitigating of consequences, shall be taken.

The plant safety is under the responsibility of the Operator of the NPP, typically the owner, even though the Regulatory Authority (RA) supervises and eventually endorses whatever, related to safety, is proposed by the Operator. The RA is normally part of a Ministry within a Government (i.e., RA is independent of the industry) and is properly supported by one or more organizations having technological competence in the area of nuclear reactor safety.

3.2. The Fukushima Initiating Events. The DE, mentioned under item (d) in Section 3.1, includes Postulated Initiating Events (PIEs) which can be of internal or external origin.

The earthquake and the tsunami are two among the "external" PIEs part of the DE. Namely, the Design Basis Earthquake (DBE) and the Safe Shutdown Earthquake (SSE) are characterized; see, for example, [2]. A safe response of the NPP must be demonstrated in case of the DBE and the SSE.

In addition to this and somewhat independent from the earthquake, the Station Blackout (SBO), that is, the full loss of electricity from sources exterior to the NPP, constitutes a PIE part of the DE. Suitable sets of Diesel Generators (DGs) must be designed to comply with SBO. Within the framework of the BDBA, an aggravated SBO situation involving the Loss of On-Site and Off-site Power (LOOP) is also studied: in this case DGs are also supposed to fail. Mitigation countermeasures are planned in the case of LOOP, possibly based on proper Severe Accident Management Guidelines (SAMGs) [13]. Then, SBLO, LOOP, and earthquake (and tsunami) are considered within nuclear reactor safety technology.

In order to provide a suitable focus on the Fukushima event, the following items need to be established or defined in advance.

(i) Four NPP Units, at close distance among each other, were severely hit by external events.

(ii) In each unit, except for Unit 4, events occurred affecting the Reactor Core (RC) and the Spent Fuel Pool (SFP).

(iii) Consequential, and somewhat simultaneous, Earthquake, Tsunami, and LOOP occurred.

(iv) The acronym RC hereafter implies that the accident evolution is considered with reference to the pressure vessel, the primary loop till the isolation valves, the ECCS (Emergency Core Cooling Systems), the containment, and the Reactor Building, in this last case, as far as interactions between CO and RB are concerned. The acronym SFP implies that the evolution of the accident is considered inside the spent fuel pool and the reactor building, as far as interactions between SFP and RB are concerned.

Therefore, the analysis of the Fukushima event should consider three Initiating Events (note: not necessarily one "standard" PIE, see what follows) and seven targeted systems. The initiating events are the Earthquake, the Tsunami, and the LOOP including the "early" SBO, consequence of the Earthquake. The targeted systems are RC and SFP in the case of Units 1, 2, and 3 and the SFP in the case of Unit 4.

The Earthquake Event. The Sendai Region or Tohoku-Kanto Earthquake, see Figure 2, can be characterized by the following, see also [14]:

(i) geography: epicenter position: 38° 6′′N and 142° 51′′E (offshore Sanriku coast); epicenter depth 24 km; and distance from Fukushima city and NPP: 177 and 144 km, respectively;

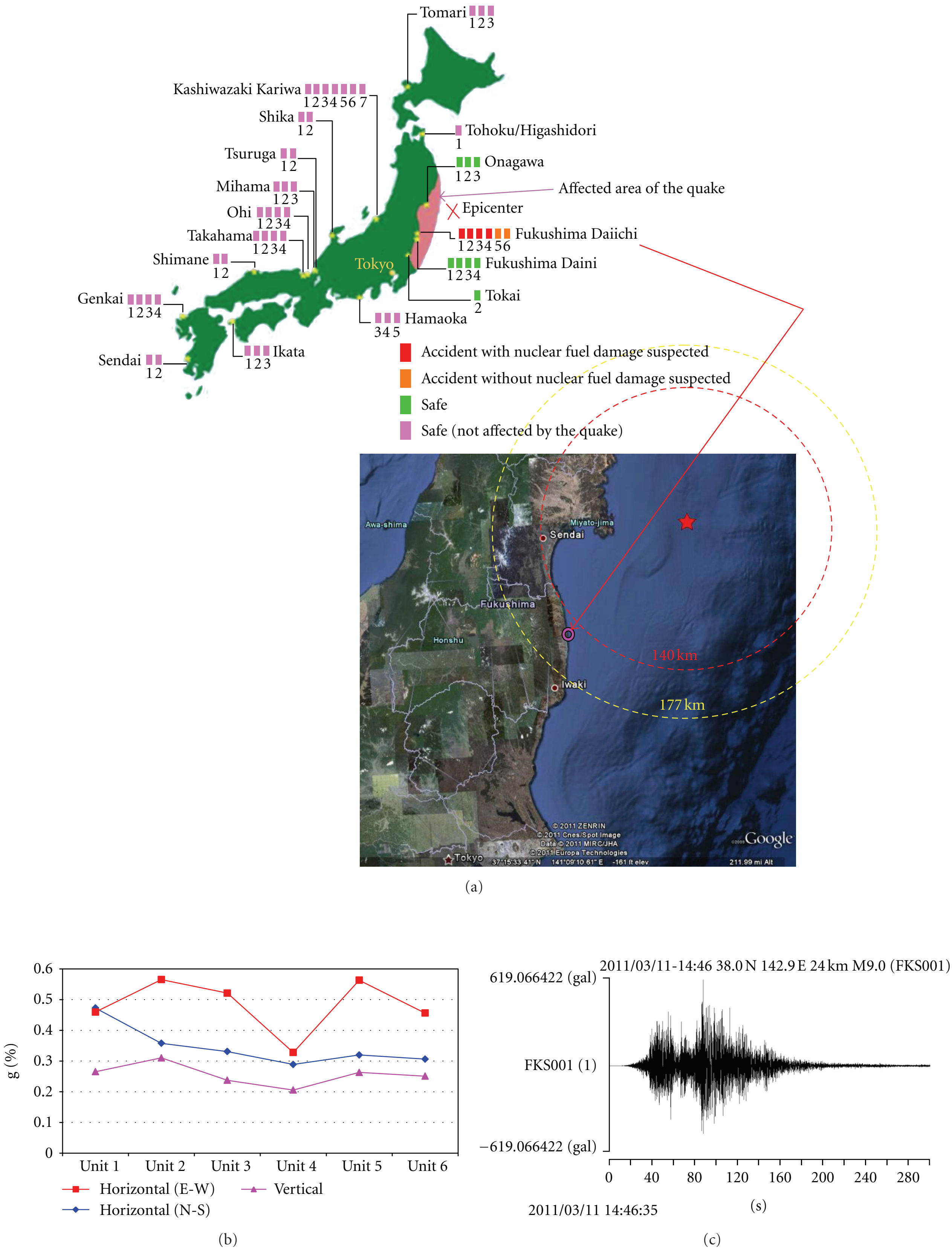


Figure 2: Earthquake: (a) geographical view of the earthquake and tsunami affected area in the North-East region of Japan, including localization of NPP in the Country; (b) maximum acceleration values measured at the Fukushima site; (c) typical measured acceleration time trend at the Fukushima site.

(ii) hypocenter depth: 22 to 32 km; fault "rupture" length 500 km; and displacement at the fault location: about 10 m;

(iii) spectrum (actually acceleration versus time): see Figure 2(c): (1 gal = 1 cm/s^2); note the duration of the oscillatory event around 3 minutes;

(iv) maximum acceleration: at the ground level, at the Fukushima NPP Unit 2 location, Vertical 0.31 g, Horizontal: 0.55 g; note that maximum horizontal ground acceleration measured at the Onagawa NPP site was 2.9 g;

(v) energy released: (0.5–1.9) 10^{11} MJ (equivalent to the thermal power produced by a 1000 MWe NPP in about 20 years).

The earthquake has been classified with Moment-Magni tude = 9.0, [14] and is among the four largest (i.e., in terms of disruptive energy released including maximum ground acceleration) in the world since 1900 and the largest in Japan for 130 years.

The Tsunami Event. The Tsunami has been created by the earthquake as expected. A damage scale for severity of Tsunami as accepted as in the case of the earthquake (e.g., Richter scale) does not exist. However, severity scale can be found in the literature as well as assessment studies of NPP safety against tsunami [15]. The following shall be noted.

(i) A substantial part of the earthquake energy (maybe 50%) has been transferred to the ocean water.

(ii) Waves of different heights and velocities, depending mostly upon the distance from the epicenter and from the under-water earth profile close to the coast, have been created. An idea of the range can be given by the extreme values 3–10 m, and the propagation velocity is of the order of 100 Km/hour (or 30 m/s).

(iii) The sea-water penetrated up to around 10 km into the land.

(iv) At the Fukushima NPP location, according to nonmeasured and uncontrolled information, waves higher than 7 m and up to 14 m hit the Reactor Buildings of Units 1 to 4. "Design tsunami wave height" reported for Fukushima NPP is 6 m.

As in the case of the earthquake, the subsequent Tsunami resulted among the few largest detected ones since ever in the entire globe. In the area of the Fukushima Daiichi NPP the damage to civil buildings from earthquake summed up with the damage from the tsunami. This is also expected to be true for the houses of the TEPCO personnel working at the NPP.

The SBO and the LOOP Events. As expected, SBO accident occurred simultaneously with the earthquake. The SBO is part of the DE for accidents in any NPP Unit. Redundant safety features energized by Diesel Generators, DGs (but not only, see what follows), are designed and installed to comply with NPP energy needs following such a postulated initiating event.

SBO conditions and evolutions shall be distinguished for Reactor Core (RC) and Spent Fuel Pools (SFPs). Fukushima Units 1 to 3 are concerned with RC and Fukushima Units 1 to 4 are concerned with SFP.

The following aspects connected with the SBO conditions are relevant for the RC accident evolution.

(i) Other than DG, Fukushima Unit 1 is equipped with a "passive emergency cooling system" specifically designed for SBO situations: this is called Isolation Condenser (IC) and consists of a heat exchanger immersed in a pool that works by natural circulation (Figure 1(c)). Other than DG, Fukushima Units 2 and 3 are equipped with an "active emergency cooling system" specifically designed for SBO situations: this is called Reactor Core Isolation Cooling (RCIC) and consists of a pump driven by a turbine that is energized by the steam produced in the core; the pump suction is connected with the PSP of the containment and injects liquid into the vessel (Figure 1(c)).

(ii) The IC is expected to work, namely, to successfully remove the core decay power, for a time period correspondent to the emptying of the IC pool: the approximate value for this period is 10 hours. The RCIC is characterized by a heat sink (i.e., the PSP) typically ten times larger than the IC pool; however, the turbine and the pump (other than valves) are "active" components having their own characteristic reliability; the approximate duration expected for the time of operation is a few hours.

(iii) DGs are designed to operate for the design-SBO conditions: a reasonable duration for the operation period is 10 hours.

(iv) DG, IC, and RCIC at Fukushima Daiichi NPP have been subjected to an earthquake and a tsunami having characteristics higher than foreseen in the design basis for these systems. Therefore failures of all these systems should have been expected following the actual earthquake and tsunami.

The Fukushima Daiichi NPP accident evolved, one or a few hours after the time when the earthquake hit the site, from SBO to LOOP. Also, the duration itself of the SBO, much higher than 10 hours, is such to bring the accident condition from SBO to LOOP. Noticeably, the LOOP accident scenario is outside the design boundary for accidents: in this case uncontrolled radiation releases are expected and mitigation countermeasures shall be taken.

The following aspects connected with the SBO conditions are relevant for the SFP accident evolution.

(i) Pools are used for cooling spent fuel in Units 1 to 4 of Fukushima NPP. Pools are installed outside the containment and inside the reactor building of each Unit. In the case of absence of cooling, a time period ranging from a few hours to several hours (i.e., "grace period") is needed to produce the lack of cooling conditions that damages the fuel bundles; the duration of the "grace period" depends, other than by the dimension of the pool (fixed and equal in the case of Units 1 to 4), upon the number of fuel bundles installed, their (average-equivalent) burn-up, and the period between the time of beginning-of-stay and the time of lack of cooling.

(ii) Reference "grace period" duration is ten hours (more details can be found in Section 3.3.2); this is a short time compared with the overall-equivalent time of the lack-of or of inadequate cooling (in the case of Fukushima, this is estimated as several days; see the following). The "grace period" could be shorter in case of loss of pool integrity caused by the earthquake.

The LOOP event for the SFP is, as in the case of the RC, outside the design boundaries: furthermore, the pools are not "protected" by the containment, and no countermeasures are designed inside the reactor building to cope with the production of hydrogen originated by the lack of cooling of the pools.

3.3. The Fukushima Accident Scenario. Any nuclear accident scenario shall be characterized by the sequence of main events, by the performance (or accident behavior) of major components and systems, and by the time evolutions of physical parameters, namely, including those which characterize the performance of barriers and safety functions (items "b" and "c" in Section 3.1) and are needed for determining the acceptability conditions (item "e" in Section 3.1).

The analysis of any accident scenario requires a suitable knowledge of initial conditions (e.g., including burn-up of each fuel assembly in the RC or in the SFP, level of core power at the time of the earthquake, etc.) and of boundary conditions (e.g., time of start of DG, working time for the IC and the RCIC, operator actions, etc.). Months or even years will be needed to clarify all those conditions in the case of the Fukushima event. Therefore, the accident analysis here is not rigorous, nor comprehensive as expected for a standard NPP accident analysis. Rather, the purposes of the "rough-preliminary" analysis are as follows:

(a) to perform "bounding calculations" in relation to releases of radioactive products (Section 3.4),

(b) to connect the scenario with established information or understanding in nuclear reactor safety and technology (Section 4),

(c) to streamline the future evaluation of the accident scenario (Sections 4 and 5).

The situation of RC and SFP with the meanings specified in Section 3.2 is distinguished hereafter: the sequences of cornerstone events are given in Tables 2(a) and 2(b), respectively.

3.3.1. Core, Vessel, Containment, and Reactor Building (Excluding SFP) Performance. The earthquake constitutes the external PIE occurring at time "0" (Table 2(a)). The SBO event was the additional expected failure. Notwithstanding the magnitude of the earthquake was larger than the "licensing-accepted" (i.e., by regulatory authority) magnitude for the DBE, the overall NPP, including the civil structures, the major components, and the relevant safety hardware, responded well. The accident remained within the Design Envelope, and no or limited consequences could be expected for the population and the environment, rows 1 to 3 in Table 2(a) whatever is the scenario depicted as follows.

This is true till the time when the tsunami hit the NPP and contributed to (or was at the origin of) the failure of the diesel generators. This occurred about one hour after $t = 0$, row 4 in Table 2(a).

At this point the SBO shall be "upgraded" to LOOP, row 5 in Table 2(a). In this situation the "DC" batteries could be still available, as well as passive devices like the IC in the Unit1 and active systems not requiring external energy to be operated, like the RCIC in Units 2 and 3. This condition is outside the Design Envelope for accidents, but treats of large radioactivity releases to the environment from the core could still be prevented. The last statement does not apply in relation to the spent fuel in the pools (see Section 3.3.2). Additional considerations apply.

(i) The DC batteries have a life of a few hours.

(ii) The IC effectiveness in Unit 1 is connected with the availability of batteries and of water in the related pool: this can be guaranteed for a few hours without refilling. In addition, in the case of Unit 1, the IC related batteries were irreversibly damaged by the tsunami.

(iii) The RCIC in Units 2 and 3 are complex systems, vulnerable by earthquake (e.g., including expected late failure); in any case their effectiveness is connected with the availability of cold water in the pressure suppression pool (PSP). The PSP is definitely larger than the IC pool, but considering water heating up, containment pressurization and reliability of the systems, cooling by RCIC can be guaranteed for a few hours.

(iv) "Classical" or standard duration considered for the loss of AC power in case of SBO within the accident Design Envelope is a few hours.

(v) The design survival life for DG is longer than a few hours, but also because of aging, the actual survival life, even in the absence of tsunami, shall be expected as a few hours.

Because of the aforementioned fourth bullet and of the probability of the PIE, the concerned Fukushima accident should be classified as outside the Design Envelope, that is, BDBA. Actually, the severity of the accident was immediately (a few hours after time "0") recognized by TEPCO. BDBA does not imply yet severe consequences; the actuation of accident mitigation countermeasures can be adopted to prevent core damage and massive radiation releases to the environment.

What (could have been) happened in each of the Fukushima Units 1 to 3 during the time period between

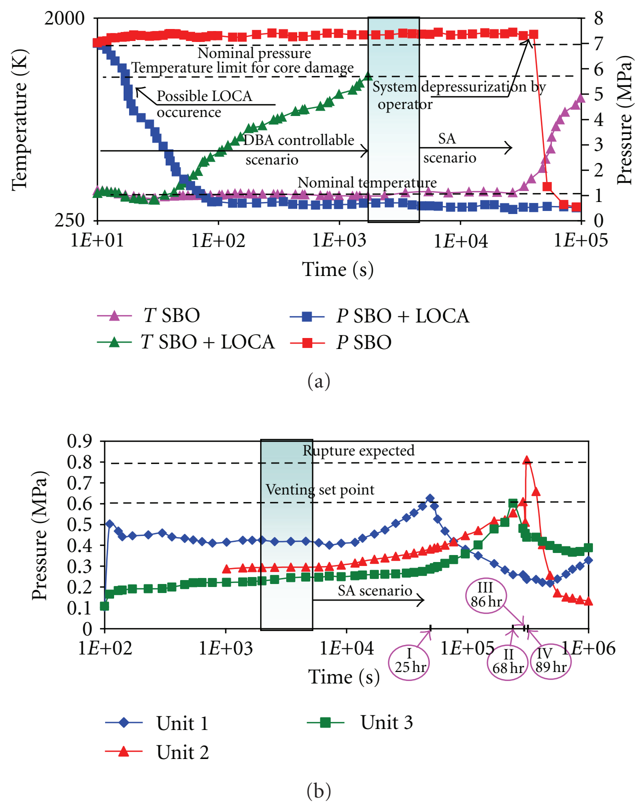


Figure 3: Possible, rough time evolutions of key variables during the Fukushima event in the time period between events: (a) 1 and 5 of Table 2(a), where means T: temperature, P: pressure; (b) 1 and 11 of Table 2(a).

events 1 and 5? This is depicted in Figures 3(a) and 3(b). Note the different time scale in the horizontal axes for the two diagrams. Namely, Figure 3(b) includes the rough consideration of time evolution for the containment pressure up to event 11 in Table 2(a).

The "earthquake-plus-SBO-plus-tsunami" PIE can be associated or less with an early LOCA or an operator-driven system depressurization. This causes differences in the initial part of the transient, Figure 3(a).

(i) In case of early LOCA or of actuation of venting from RPV to the PSP, fast RPV depressurization takes place together with early containment pressurization; furthermore, turbine driven RCIC cannot enter in operation; early dry-out might be observed also depending upon the size of the break; the working capabilities of the IC eventually (in Unit 1) are largely impaired.

(ii) In the situation "no-early LOCA", the IC and the RCIC (in Unit 1 and in Units 2 and 3, resp.) may properly remove decay heat in the absence of AC and DC power; effectiveness of IC and RCIC is connected with the heating of the related pool: the IC pool and the PSP, respectively; heating of pools is expected in a time period (quantity "N" at row 5 in Table 2(a)) between a few hours and a dozen hours.

(iii) Reasonable times for operator-driven depressurization are at the time of "measured" end of operation of IC (in Unit 1) and of RCIC in Units (2 and 3), that is, about 2 hours, 70 hours, and 38 hours after the transient start [14] (therefore N = 2, 70 and 38 for Units 1 to 3, resp.).

Until the time "N" (hours) the core is undamaged or slightly damaged and recovery-prevention procedures can be adopted to halt further damage. Basically the NPP situation is within the DBA (or DE) boundary.

At a time greater than "N" the event scenario exits the DBA boundary, not implying immediate core melt nor loss of substantial geometric integrity of the core. However, without restoration of electricity or of DG system operability, only mitigation measures can be taken (some of these) consistently with the availability of DC power and of suitable operator actions.

Events 6, 8, and 10, also referred hereafter as containment purging, constitute the expected results of implemented EPG (Emergency Power Guidelines) in a BDBA condition aimed at mitigating the consequences of the on-going accident; see, for example, [3]. Namely, the EPGs aim at maintaining the integrity of the containment at the expenses of "minor-early" radioactivity release toward the Reactor Building. Proper openings connect the containment to the RB and are controlled by DC operated valves. The containment pressure rises in each unit because of continued energy deliver from the primary coolant system due to a LOCA (early or delayed) or coming from the design-planned operation of the Steam Relief Valves (SRVs), eventually working under the ADS (Automatic Depressurization System) function.

The SRVs connect the steam lines with the PSP in the wet-well region of the containment. At times identified by events 6, 8, and 10 in Table 2(a), the containment pressure presumably reached the set-point for EPG in Units 1, 3, and 2, respectively. In the case of Units 2 and 3, the loss of RCIC was a precursor for SRV actuation and pressurization of the containment (and, consequently, of CO venting), as already mentioned.

The events 7, 9, and 11 need further explanation.

The events 7 and 9 are expected consequences of the events 6 and 8 in Units 1 and 3. The RB is not designed to withstand pressure associated with containment releases. Furthermore, in the current conditions, H2 burning (either deflagration or detonation) apparently contributed to the damage of the RB. The source of H2 could have been the containment itself due to metal-water reaction in the core during the period before the purging, or, less probably in the case of Units 1 and 3, the metal-water reaction in the SFP (see Section 3.3.2). The failure of the RB caused the release to the atmosphere of the radioactivity:

(i) present in the gas space of the containment,

(ii) (eventually) associated with the SFP.

Furthermore, the disruptive failure of the RB gave a wrong impression to the public in relation to the safety of nuclear installations.

The failure of the containment (CO) in Unit 2, event 11, is associated with root causes different from those related to the failure of the RB. Apparently, the containment pressure release EPG was not effective in the case of Unit 2 where a containment breach, following an explosion, was detected. The containment breach can be a consequence of the containment pressure reaching the rupture or the physical resistance set-point (as depicted in Figure 3(b)), or of a different cliff-edge type of phenomenon, like the vessel rupture, or of any sort of large-energy, pulse-type producing reaction, including chugging loads caused by steam condensation in the PSP. The failure of the containment caused the release to the atmosphere of the radioactivity:

(i) present in the gas and liquid space, noticeably in the PSP, of the containment.

Time Period between Events 7, 9, and 11 and Events 12, 13, and 14. A number of operator actions and trials were undertaken during the time period between the events 7, 9, and 11 and the events 12, 13, and 14. These included the injection of water (even sea-water) into the containment and into the vessel as well as attempts to cool from the exterior the containment and venting from containment to the stack.

Those actions were successful for Units 1 and 3 in the sense that the integrity of the containment was kept (eventually by operating the containment purging system) and core degradation did not progress toward worse (catastrophic) conditions. The integrity of the primary circuit and of the vessel was not necessarily preserved, but "large" radioactivity releases to the environment were prevented.

The situation is different for Unit 2. In this unit the operator actions could not prevent releases of radioactive liquid and gas to the environment from the containment. However, the injection of liquid in the containment and in the primary circuit mitigated the radioactivity releases by keeping as low as possible the containment pressure. This condition remained in Unit 2 till the day 25 after the start of the accident (event 14).

Electricity restoration in control room CR and stable conditions reached for core cooling (events 12 and 13 in Table 2(a)) do not imply the "end-of-the-accident" or full control of the NPP: first, restoration of electricity in CR implies "lightning electricity" and not power electricity needed for operating pumps and other heavy components; second, stable conditions for core cooling imply capability to supply needed coolant, but not ensuring that the coolant reaches possible obstructed regions of the core or all possible core molten masses.

However, event 13 for Units 1 and 3 and event 14 for Unit 2, shall be taken as the end of massive releases of radioactivity to the environment.

Summing-Up and Events 15-16. The knowledge of detailed time evolutions of the accident in the primary system of the three concerned Fukushima Daiichi Units will require several months. A few topics are used here to characterize the accident scenarios, consistently with the objective of the paper.

(a) A devastating "earthquake-plus-SBO-plus-tsunami" constituted the initial event for the Fukushima Daiichi NPP-related accident.

(b) Apparently the reactor components and structures relevant to safety, and significantly the containment, reacted to the earthquake forces and survived the tsunami too.

(c) Accident classification: whatever happened during the early hours, after the occurrence at row 5 in Table 2(a), the Fukushima event became a severe accident in each of the three Units. Mitigation measures were needed and were undertaken.

(d) Why Onagawa NPP did not experience the same or a similar accident scenario? The Onagawa NPP is closer to the epicenter than the Fukushima NPP, so the question is relevant. The answers are as follows: (d1) the effect of tsunami upon relevant safety structures was "lower"; (d2) possibly, mitigation measures undertaken were more effective and/or more timely.

(e) Final core and vessel status: important portions of the core are damaged in all three units; primary loop integrity cannot be ensured: possible ruptures may have been originated by the earthquake, or by the high-temperature consequence of core melt, or by any sort of large-energy, pulse-type producing reaction (same phenomena involved as in the case of the containment breach).

(f) End of NPP radiological impact upon the environment: the massive radiological impact from the core regions of the concerned Units is terminated with events 13 and 14 in Table 2(a); in the case of Unit 1 "white smoke" release, possibly indicating that containment steam release was observed for several days after event 14, more details about (relatively minor) long-term radioactivity releases are discussed in Section 3.3.2.

(g) The radiological consequences, that is, the impact upon the environment, of the accident are discussed in Section 3.4.

3.3.2. SFP and Reactor Building Performance. The "background part" of Section 3.3.1 also applies here. Basically the reactor buildings (RBs) and the spent fuel pools (SFPs) in Units 1 to 4 survived the earthquake and the tsunami.

Let us assume no leakage from any of the SFPs (this information is not confirmed from the available data) direct consequence of the earthquake and the tsunami and that LOOP occurred at time "N" (see Table 2(b)). In this case, the performance of the liquid in the pool and of the stored Fuel Assemblies (FAs) only depends upon the amount of liquid in the pool, the FA power, and the distance between the TAF (Top of Active Fuel) and the free surface of the liquid in the pool. The following rough-preliminary data are used:

(i) total liquid water in each SFP (e.g., [2]) in m^3: 1000, Unit 1, and 1400 Units 2 to 4;

(ii) thermal power produced in each SFP (see data in the 1st row of Table 3 and data given in [2] in relation to the average power produced by each FA, assuming that average decay time is 30 days since the last shutdown in each FA) in Mw: 0.29, 0.59, 0.51, and 1.33, Units 1, 2, 3, and 4, respectively;

(iii) time of boiling of each SFP (after the SBO) in hours: 80, 39, 45, and 17, Units 1, 2, 3, and 4, respectively;

(iv) time of fuel uncovery in each SFP (4 m of liquid level above TAF are assumed with total pool depth of 12 m; this is also the time of start of H2 production) in hours: 315, 156, 180, and 69, Units 1, 2, 3, and 4, respectively.

By adding the time values at the 3rd and the 4th items in relation to Unit 4, we get 86 hours that is, roughly, the time of (presumed) H2 smoke detected in Unit 4 SFP (row 3 in Table 2(b)).

A detailed calculation (not performed here) should account of each individual FA including its burn-up and the time of placement in the pool. In this case, the relative position of FA inside the SFP, for example, related to the free liquid surface, shall be considered together with passive structures affecting the thermal balance. Furthermore, no interaction is assumed between the explosions at rows 7, 9, and 11 in Table 2(a) and the conditions of the SFP in the four concerned units.

As in the case of Table 2(a), the severe accident conditions shall be declared at the time of LOOP and consequential mitigation measures should be undertaken. The grace period, that is, the period without fuel damage in the pool and, more important here, the period when the accessibility to the pool was possible without danger for the operators, ends with the event 3 in Table 2(b).

The event 3 in Table 2(b) caused the first (in terms of time) massive release of radioactivity to the environment during the overall course of the Fukushima NPP accident as already mentioned. The site, after this event, was heavily contaminated, and all subsequent mitigation or recovery actions were adversely affected.

Continuous radioactivity releases from the SFP in Unit 4 occurred toward the RB in the time period between event 3 and event 5 in Table 2(b). Similar situations are assumed for the SFP in Units 1, 2, and 3; however, according to the rough calculation reported at the aforementioned items, those units are expected to have a longer grace period assuming no pool damage by the earthquake and no influence upon the SFP integrity of the explosions in the respective reactor buildings.

After event 5 radioactivity releases from the SFPs to the RB in each of SFPs of Units 1, 2, 3, and 4, should be limited to gaseous fission products still exiting the damaged fuel. This does not imply the end of the radiological impact to the environment because radiation can be continuously transported from the RB to the environment till the time when the RB is put in a safe conditions (this may take a few months after the event 5, e.g., in order to get zero radiation release).

Summing-Up and Events 6 and 7. The knowledge of detailed time evolutions of the accident in the SFP of the four concerned Fukushima Daiichi Units will require several months. A few topics are used here to characterize the accident scenarios, consistently with the objective of the paper (some concepts already expressed in relation to reactor core are repeated here for the sake of comprehensiveness).

(a) A devastating "earthquake-plus-SBO-plus-tsunami" constituted the initial event for the Fukushima Daiichi NPP SFP-related accident.

(b) Apparently the SFP structures, significantly the reactor building, reacted to the earthquake forces and survived the tsunami too.

(c) Accident classification: whatever happened during the early hours, after the occurrence at row 3 in Table 2(b), the Fukushima SFP event became a severe accident in Unit 4 and, subsequently, in three other units. Mitigation measures were needed and were undertaken as possible.

(d) Why the SFP in the Onagawa NPP did not experience the same or a similar accident scenario? The Onagawa NPP is closer to the epicenter than the Fukushima NPP, so the question is relevant. The answer is as follows: (d1) the design values for the tsunami are larger than those in Fukushima; (d2) possibly, mitigation measures undertaken were more effective and/or more timely.

(e) Final SFP status: portions of the FA bunches in the SFP are damaged (apparently) in all four units; water level is restored as well as cooling capabilities. However, "minor" radiological releases from the damaged FA to the RB via the liquid surface, namely, fission gases, cannot be avoided.

(f) End of SFP radiological impact upon the environment: the radiological impact from the SFP directly to the environment or from the RB to the environment cannot be easily terminated (a few months are necessary) although "minor" fractions of releases are still possible and expected. Significant releases from SFP from all units are terminated with event 6 in Table 2(b).

(g) The radiological consequences, that is, the impact upon the environment, of the accident are discussed in Section 3.4.

3.4. The Fukushima Accident Consequences. The evaluation of consequences of the Fukushima event will require information and resources not available in the present framework, as already mentioned. The idea here is to set boundaries to the relevant values of quantities which characterize the releases.

At first, the inventory of radiation stored in Units 1 to 4 is established, distinguishing between in-vessel and in-SFP radioactive inventories. Then, suitable upper limits of radiation delivered to the environment are identified (arbitrary assumptions cannot be avoided). Finally, comments

Table 3: Amount (approximate) of radioactivity stored in Fukushima Units 1 to 4 at the time of start of the event.

Item	RC-NPP Unit No.			SFP-NPP Unit No.			
	1	2	3	1	2	3	4
FA no.	**400**	**548**	**548**	**292**	**587**	**514**	**1331**
Irradiated FA no.		—		**192**	**559**	**462**	**1127**
Noble gases (Xe, Kr)	2.49*	3.42*	3.42*	0.06°	0.17°	0.14°	0.34°
Halogens (I, Br)	2.94*	4.04*	4.04*	3.0e10	8.8e10	7.4e10	18.1e10
Alkali metals (Cs, Rb)	0.35*	0.49*	0.49*	1.08°	3.18°	2.64°	6.46°
Tellurium (Te, Sb, Se)	2.45*	3.36*	3.36*	0.05°	0.15°	0.13°	0.31°
Barium-strontium (Ba, Sr)	3.58*	4.91*	4.91*	1.39°	4.07°	3.38°	8.28°
Noble metals (Ru, Rh, Pd, Mo, Tc, Co)	9.63*	13.2*	13.2*	0.83°	2.46°	2.05°	4.99°
Lanthanides (La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am)	10.5*	14.4*	14.4*	1.21°	3.55°	2.94°	7.21°
Cerium (Ce, Pu, Np)	29.2*	40.0*	40.0*	1.45°	4.27°	3.54°	8.67°
Total/RC & SFP Units	61.1*	83.8*	83.8*	6.07°	17.85°	14.82°	36.26°
Total /RC & SFP		**2.29e20**			**0.075e20**		
Total	**2.37e20 Bq** [= 2.37e8 TBq = 6.41e9 Ci]						

* e18; ° e17.

are given in relation to the passage from radioactivity (i.e., Becquerel) to dose (i.e., Sievert/unit time). Broad classes of radioactive isotopes are distinguished.

3.4.1. The NPP as a Radioactivity Repository. The word repository is related to the amount of radioactivity stored in the core and in the SFP at the time when the accident started. The amount is provided in Table 3. The adopted simplified classification of radioactive isotopes is consistent with what is at the basis of the calculation of the "source term" by NRC, that is [16] (NUREG document), for light water reactors. The following can be noted.

(a) Eight categories of products are distinguished (first column of Table 3).

(b) The physical nature of the products is considered: noble gases, for instance, are not stopped by any obstacle; they also do not remain in the human body if inhaled.

(c) The half-life of the products is considered: for instance, this is of the order of days (8) in the case of "I" and of the order of decades (3) in the case of Cs and of the order of hundred-thousand years in the case of Lanthanides and Transuranics (see the following).

(d) Release fraction to be determined: this is typically equal to 1 in the case of Noble gases, Iodine, and Cesium and much less in the case of Lanthanides and Transuranics.

3.4.2. Radioactivity Source Term for the Environment. The word "source term" is related here to the radioactivity released to the environment during the entire course of the Fukushima event. In order to calculate the release to the environment, the core and the primary loop (RC) are distinguished from the spent fuel pool (SFP). Furthermore, in the cases of RC and SFP, releases occur to containment and to reactor building and to reactor building, respectively, before diffusing into the environment, according to Figure 4.

Two situations shall be distinguished:

(1) before the "End-of-the-Accident" (EoA),

(2) after the "End-of-the-Accident".

The EoA is defined, for the present purpose, as the time when the control is taken of the system, that is, when massive radioactivity releases to the environment are terminated (see also items 15 and 6 in Tables 2(a) and 2(b), resp.). Making reference to the sketch in Figure 4, radioactivity releases after the EoA are due to containment leakages and bypass, noticeably, vent valves used for containment depressurization in all units and possible damaged IC line in the case of Unit 1.

The Core Release to Containment. Let us assume as negligible the coolant activity, that is, the radioactivity into the coolant during nominal NPP operation compared with the radioactivity stored and lost from the core during the accident.

Conservatively, full core damage is assumed.

According to [16], the core release can be subdivided into four phases associated to time periods: (a) rod-gap activity release, (b) early-in-vessel release, (c) ex-vessel release, and (d) late-in-vessel release. The corresponding time periods vary between 0.5 hours and 10 hours summing up to about 15 hours. Those time periods have been considered and updated in [6], namely, by increasing substantially the duration of the releases. Average duration values are around 13, 9, 3, and 27 for the four time periods, respectively, based upon the consideration of a dozen transient scenarios and various combinations of equipment failures. Maximum duration time estimated for the late-in-vessel period is up to 6 days (or 144 hours).

Making reference to all the four phases, the total maximum releases for each of the eight classes of fission products

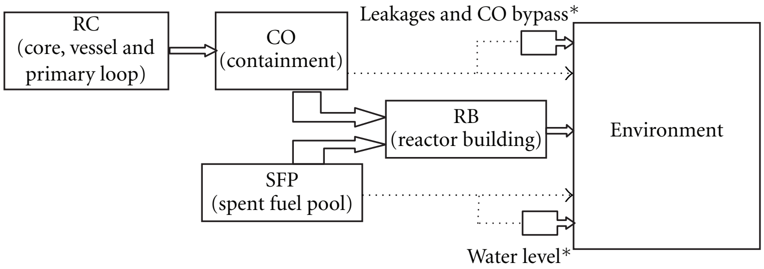


Figure 4: Sketch for radioactivity release *(dotted lines are secondary flow paths for releases; *applies after the end-of-the-accident, rows 15 and 6 in Tables 2(a) and 2(b))*.

Table 4: Fraction of radioactivity release from core to containment in the case of LBU fuel (NUREG 1465, 1995 and SAND 7697, 2007, or [6, 16]).

Fission product category	Fraction LBU***	Fraction HBU***	Notes*
Noble gases	1.0	1.0	
Halogens	0.91	0.55	Lower value (factor 1.5)
Alkali Metals	0.31	0.27	Higher value (factor 2)
Tellurium	0.76	0.80	
Barium-Strontium	0.06	0.07	Higher value (factor 4.5)
Noble Metals**	0.07	0.055	Lower value (factor 6)
Lanthanides**	0.00021	0.0012	Higher value (factor 25)
Cerium**	0.00021	0.0065	

*Calculated in [16], and related to LBU. **There is no gap-release. ***"Decontamination factor".

Table 5: Fraction of radioactivity release from spent fuel pool to reactor building following "zirconium-fire" (NUREG 4982, 1987).

Fission product category	Fraction of initial inventory*	Notes
Noble gases	1.0	
Halogens	1.0	
Alkali Metals	1.0	Conservative value. Large uncertainty.
Tellurium	0.02	1.0 in the case of Sb.
Barium-Strontium	0.002	
Noble Metals	2×10^{-5}	0.10 in the case of Co.
Lanthanides	1×10^{-6}	0.01 in the case of Nb.
Cerium	1×10^{-6}	

*"Decontamination factor".

are given in terms of "fraction of initial core inventory", in Table 4, as derived in [6]. Distinction is made in the table, between Low Burn-Up (LBU) and High Burn-Up (HBU) fuel.

Distinction between LBU and HBU gives a small contribution to the results calculated in the present framework and to the related errors. Therefore, this distinction is not considered any more.

The Spent Fuel Pool Release to Reactor Building. The accident of loss of cooling in a pool is discussed in detail in [2], as already mentioned. Fractions of radioactivity releases for the eight categories of fission products, related to the initial inventory in the pool, are given in Table 5, having the same format as in Table 4. Phenomenology of fire propagation is outlined in the previous referenced document, although emphasizing that considerable uncertainties are associated with the data.

The Overall Release to Environment. The assumptions made hereafter for estimating the radiological releases are not necessarily consistent (however an attempt is made to get such a consistency as far as possible) with the hypotheses introduced before, for example, for calculating the time of boiloff of the SFP.

The results of an intermediate step of the analysis are given in Table 6.

Considering the data in Table 3 (i.e., the overall-reference source term) and the radioactivity propagation (or mitigation) factors from core to containment and from SFP to reactor building in Tables 4 and 5, respectively, one obtains the data in Table 6. The same categories of radioactive

Table 6: Radioactivity inventory released to containment and reactor building of Fukushima Units 1 to 4.

Item	RC–NPP Unit No.			SFP–NPP Unit No.			
	1	2	3	1	2	3	4
Noble Gases	2.49*	3.42*	3.42*	0.06°	0.17°	0.14°	0.34°
Halogens	2.67*	3.67*	3.67*	3.0e10	8.8e10	7.4e10	18.1e10
Alkali Metals	0.11*	0.15*	0.15*	1.08°	3.18°	2.64°	6.46°
Tellurium	1.85*	2.55*	2.55*	0.001°	0.003°	0.003°	0.006°
Barium-Strontium	0.22*	0.29*	0.29*	0.003°	0.008°	0.007°	0.016°
Noble Metals	0.67*	0.92*	0.92*	1.6e12	4.9e12	4.1e12	10.e12
Lanthanides	0.002*	0.003*	0.003*	1.2e11	3.5e11	2.9e11	7.2e11
Cerium	0.006*	0.008*	0.008*	1.4e11	4.3e11	3.5e11	8.7e11
TOTAL/RC & SFP Units	8.01*	11.0*	11.0*	1.14°	3.36°	2.79°	6.82°
TOTAL/RC & SFP		**0.29e20**			**0.014e20**		
TOTAL	**0.30e20 Bq** [= 0.3e8 TBq = 0.81e9 Ci]						

* e18; ° e17.

products are distinguished in Table 6 as in the first column of Table 3. Therefore, "bounding" amounts of radioactivity inside the containments and inside the reactor buildings of each of the Fukushima Units 1 to 4 are reported, bottom rows of Table 6. These values do not constitute the release to the environment (see the following).

Suitable assumptions, other than reliable computer code calculations (outside the scope of the present paper), are needed in order to calculate radioactivity releases to the environment from the data in Table 6. The following cases are distinguished.

(A) *CO Radioactivity Release from Units 1 and 3.* Let us consider the amount of steam produced during the period between March 11 and the EoA equal to 1; then, an approximate steam fraction equal to 10^{-2} is released during venting: the subsequent assumption is that the same fractions of Halogen and Alkali Metals are released from the containment. All other Fission Products (FPs) remain in the containment.

(B) *CO Radioactivity Release from Unit 2.* Let us consider the amount of steam produced during the period between March 11 and the EoA equal to 1 and 1 the total liquid inventory in the containment (i.e., the liquid in the PSP plus the liquid injected into the containment during the same period); then, approximate values for the steam and the liquid fractions released to the environment are 10^{-1} and 10^{-3}: the subsequent assumption here is that the released fraction of Halogen and Alkali Metals is equal to the steam fraction (released) and the released fraction of all other products is equal to the liquid fraction (released).

(C) *RB Radioactivity Release from Units 1 and 3.* Alkali Metals and Noble Gases constitute about the 99% (94% and 5% for Alkali Metals and Noble Gases, resp.) of the radioactivity released from SFP into the RB, Table 6. Then, only Alkali Metals are considered under the present assumption: based on the data in Table 2(b) (noticeably, row 4) and in Section 3.3.2 (noticeably, the estimated time of fuel uncovery), it is conservatively assumed that 50% of the Alkali Metals are released to the environment.

(D) *RB Radioactivity Release from Unit 2.* Introductory remarks as in the previous item apply: based on the data in Table 2(b) (noticeably, row 4) and in Section 3.3.2 (noticeably, the estimated time of fuel uncovery) and considering the final status of the RB, it is assumed that a negligible (related to other SFP) amount of radioactivity is released.

(E) *RB Radioactivity Release from Unit 4.* Introductory remarks as in the previous item apply: based on the data in Table 2(b) (noticeably, row 4) and in Section 3.3.2 (noticeably, the estimated time of fuel uncovery), it is conservatively assumed that 100% of the Alkali Metals are released to the environment.

The containment is undamaged in both Units 1 and 3. For this reason situations at items (A) and (B) are distinguished. The reactor buildings of Units 1, 3, and 4 are severely damaged, compared with the "low" damage of RB in Unit 2. Furthermore, the amount of radioactivity in the SFP of Unit 4 is much higher than that in the other units. This is at the basis of the distinction among situations at items (C), (D), and (E).

A variety of additional assumptions are still needed to pass from the data in Tables 3 and 6 to the data in Table 7. Then, different "conjectural" (or impossible) and "realistic" data for the radiological impact of Fukushima accident are summarized in Table 7.

The first three scenarios in Table 7, that is, I, II, and III, deal with "conjectural" or "hypothetical" releases. The scenario IV is still conservative: the calculation results are based upon the bounding hypotheses indicated at items (A) to (E).

The scenario V shall be seen as a rough attempt to get a realistic result. Related assumptions are as follows.

(1) Mitigation Factors (MFs) recently derived for the TMI-2 accident in [17] are applicable for the releases

Table 7: Conjectural, hypothetical, conservative, and realistic radiological impact on environment of the Fukushima event.

Scenario No.	Type	Release (TBq)			Notes
I	Conjectural		2.37e8		Physical limit—Table 3
II	Conjectural		0.30e8		Theoretical limit—Table 6
III*	Hypothetical		0.21e8		Theoretical limit w/o Noble Gases
—		RC	RB	TOT	—
IV*	Conservative	4.5e5	8.3e5	1.3e6	Assumptions (A) to (E) in text
V*	Realistic	18.0	3.7e4	3.7e4	Mitigation factors (1) to (4) in text

* Without Noble Gases. Total amount of radioactivity associated with Noble Gases in Fukushima NPP, Units 1 to 4, is 9.37e6 TBq (from Tables 3 and 6).

Table 8: The MFs related to the assumptions (1) to (4).

FP Group	MF-Units 1 and 3	MF-Unit 2	MF-SFP in Unit 4
Halogens	1.7e-8	1.7e-6	2.0e-3
Alkali Metals	1.1e-8	1.1e-6	2.7e-3
Tellurium	1.4e-8	1.4e-6	2.4e-3
Barium-Strontium	8.3e-9	8.3e-7	4.0e-2
Noble Metals	1.7e-11	1.7e-9	1.5e-8
Lanthanides	7.6e-10	7.6e-8	1.3e-3
Cerium	8.4e-12	8.4e-10	8.1e-4

from Units 1 and 3: namely, source term data in Table 3 are multiplied by MF data in the second column.

(2) The MFs for Unit 2 are assumed 100 times larger than at the previous item. This derives by the consideration that fluid release caused by the containment break is assumed 100 times larger in Units 2 compared with Units 1 and 3 where (only) controlled releases and (normal) leakages are present.

(3) The MFs for the SFP of Unit 4 are derived from the severe accident analysis related to core performed in [4]. The MFs derived in case of containment failure (without drywell spray, that is, sequence RC4 in Table 4 of [4]) are "arbitrarily" applied to the source term from the SFP reported in Table 3.

(4) It is assumed that (possible) radioactivity releases from SFP of Units 1, 2, and 3 are negligible.

The MFs related to the assumptions (1) to (4) are reported in Table 8. MF is defined as the ratio between the concerned value (e.g., the environment related value) and its source-value (e.g., the amount stored in the core of in the SFP at the time of accident start).

All calculated, or hypothesized, releases are given in the 3rd main column of Table 7. Related to Noble Gases (NGs) it can be emphasized that an MF of around 10^{-4} was used in [17]. In the case this factor is used here, the data in the last two rows of Table 7 shall be increased by about 1000 TBq. Summing up, the following results are obtained.

(i) Bounding limit for overall radiological release from the Fukushima event is 1.3 million TBq (+1000 TBq for NG).

(ii) Realistic value for overall radiological release from the Fukushima event is 37000 TBq. (+1000 TBq for NG).

In the aforementioned table, the word "realistic" shall not be taken as a synonymous for "best-estimate" because the best tools for performing analyses are not used. Furthermore, realistic does not mean "precise": an error of (at least) a factor 10 shall be applied to the results in the last row of Table 7; compensating errors in the estimations are likely to occur.

It shall be stressed again that the EoA does not imply the end of the radioactivity emissions owing to the minor flow-paths sketched in Figure 4 (consistently, at the end of April, JAIF in its website reports a presumed continuing emission for about 20 TBq/day, other than about 3.7 10^5 TBq as total emission till the end of April).

In the case of Chernobyl, 5.2e6 TBq were released (without considering the related error, still close to a factor 10): realistic and conservative values for Fukushima radiological release are 100 and 5 times less, respectively.

The Radioactive Contamination. The radiation contaminates the environment. Namely, land, water (river, lakes, or sea, as well as underground water), atmospheric air (inside or outside the buildings), man-made surfaces (e.g., houses wall or cars), animals (in the air, in the water, or in the soil), and plants (including greens) are contaminated other than humans.

Various radioactive isotopes (e.g., several tens) and a wide variety of chemical species (e.g., several tens) contribute to the radioactive contamination. A key parameter relevant for the contamination is the half-life for each isotope, even though one shall consider that any radioactive isotope may be part of a radioactive chain and, therefore, be formed as a function of time other than decaying. The half-life of selected isotopes is provided hereafter (approximate value, half-life order is used):

(i) Te-129: 70 minutes;

(ii) La-140: 48 hours;

(iii) Mo-99: 66 hours;

(iv) I-131: 8 days;

(v) Cs-136: 13 days;

(vi) Co-56: 77 days;

(vii) Ru-106: 1 year;

(viii) Co-60: 5 years;

(ix) Cs-137: 30 years;

(x) Ag108m: 418 years;

(xi) Cs-134: 2 years;

(xii) Sr-90: 29 years.

Very spread values of half-life (please consider that there are elements having half-life of the order of 10^5 years and, at the same time, infinite half-life implies stable, nonradioactive element) shall be noted including values much lower than the duration of the release: this implies that any value derived from a global evaluation (i.e., the entire duration of the accident), like the one in Table 7 shall be taken with caution. Namely, radioactivity associated with elements like Te-129 of even La-140, eventually emitted on the first day of the accident, may be not present at the end of the accident: thus, characterization of releases independently of time of release is ambiguous (and conservative in the sense that the overall amount of emitted radioactivity is not present at the same time). Rather, the aforementioned values give an idea of the time of permanence on the earth of each species.

The radioactivity associated with each isotope may have different nature including "α", "β", and "γ" particles or rays as well as combination of those. Energy of each particle or ray can be very different (typical ranges from "ev" to "Mev") and, consequently, the impact upon the matter. Radioactive elements, as any chemical species, follow specific diffusion paths on the earth (different paths in air, water, and soil) and their concentration unavoidably decreases with the distance from the emission area.

How much is the local contamination already constitutes a complex questions. Three parameters or boundary conditions, for example, time since the emission to be connected with the half-life, space (or distance from the emission point), and atmospheric conditions (winds, rain, etc.), affect the contamination (damage consequence of contamination is discussed in next section).

Acceptable limits are fixed in any country in relation to most of the radioactive isotopes: these limits are connected with the damages of human bodies, better to say: acceptable limits for radiation are set to prevent damage of human bodies. Examples for Japan are the following (average values are given; there can be different thresholds for different isotopes of the same element):

(i) iodine in tap water <300 Bq/kg or <100 Bq/kg in the case of children (J),

(ii) cesium in tap water <200 Bq/kg (J).

Typical measurements of radioactivity on the earth are pointwise or obtained by local detectors: measured point values are extrapolated to average regions. Before considering local measurements in the Fukushima region, let us start from the data in Table 7 to obtain reference average values.

If the radioactivity calculated for the case V in Table 7 is spread in a radius of 20 km around the Fukushima Daiichi NPP, including the ocean, the surface contamination would result in 3000 Bq/cm^2 (this value can be transformed in 300 Bq/cm^3 or in 150 kBq/kg of land-soil, assuming that 10 cm of soil is contaminated and that the soil density is 2000 kg/m^3). In case all the radioactivity is spread in the ocean water in half of the above area and assuming an average depth of the ocean of 100 m, the ocean water contamination would result in 0.6 Bq/cm^3. In case all the radioactivity contaminates the air only, assuming 20 km as the diameter of the contaminated region and 10 km as the concerned height of the atmosphere, the average air contamination would result in 0.003 Bq/cm^3 (or 3000 Bq/m^3). Local measured values can be much higher than the reported values, but extrapolations to average values larger than the aforementioned values are questionable.

Selected values, here taken as reference to give an idea, reported in relation to contamination measurements are as follows (distances are from the NPP; source of information is given in parentheses):

(a) 40 km, air: 15 Bq/m^3 (NISA-JNES) (this value is in the N-W direction related to the Fukushima Daiichi NPP, that is, the region where the highest radioactivity concentrations are expected because of the meteorological conditions);

(b) 40 km, soil: 12–1300 kBq/kg ("JAEA-NSTC" joint institution) (this value is in the N-W direction related to the Fukushima Daiichi NPP, that is, the region where the highest radioactivity concentrations are expected because of the meteorological conditions);

(c) 100 m, sea-water: 0.02–200 Bq/cm^3 (NISA-JNES);

(d) 20 km, sea-water: < 0.08 Bq/cm^3 (NISA-JNES).

Namely, all the measured values are consistent with the bold-underlined average values obtained. However, in all cases extrapolations of local values to average values are questionable.

In Europe, maximum I-131 concentrations detected in France, Iceland, and Germany were 1.78, 3.0, and 3.7 mBq/m^3, respectively, all of these measured around the end of March or at the beginning of April. The same concentration values in the concerned measurement stations of France and Iceland decreased down to 0.6 and 0.14 mBq/m^3, respectively, toward the middle of April. In France a maximum of 3 Bq/l was measured in sump-collected rain-water.

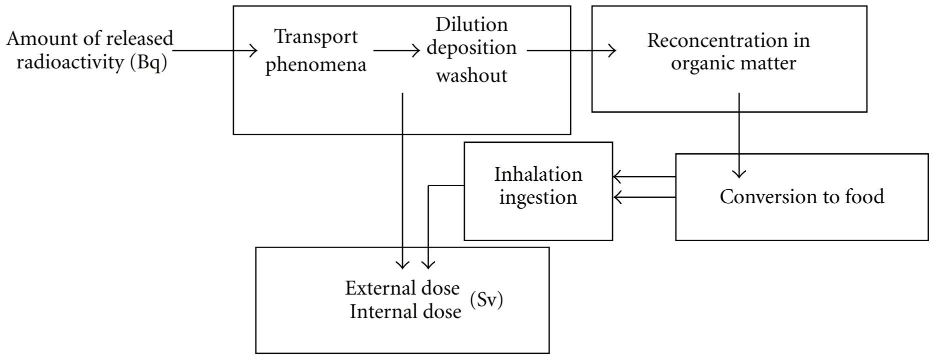


Figure 5: Simplified flow-path: from radiation (Bq) to doses (Sv).

Radioactivity to Dose. The transformation of contamination to dose implies the presence of humans at the location of the contamination: no humans = no dose (previous units for the measurement of radiological dose were "rem = roentgen equivalent man": the need of the "presence" of man was "embedded" into the unit-definition; in the case of Sievert [Sv], the straightforward connection between "man" and "dose" is lost, even though 1 Sv = 100 rem).

In addition to the parameters previously indicated, the type of particles and their energy is needed in order to characterize the contamination and to estimate the radiological damage or the dose. A simplified sketch to derive the dose in Sievert units from contamination in Becquerel units can be seen in Figure 5.

The effects of radiations upon the human body are very many and very different, as well established, going from "diffused" beneficial effects (typically, if any, solar radiation) to "local" therapeutic effects (nuclear medicine) to local damages (e.g., from a substance which fixes in the bones) to deathly effects. Before discussing about Fukushima doses, it seems relevant to report the following data from UNSCEAR, for example, [18] (all values are per-person):

(i) worldwide average annual dose: 2.4 mSv (range is 1–10 mSv); = 0.27 μSv/hr;

(ii) dose to selected air-travel crew: 3 mSv;

(iii) dose to radon exposed workers: 4.8 mSv;

(iv) natural yearly dose at Guarapari (Brazil): 10 mSv;

(v) dose originated by Chernobyl: from 0.04 mSv (in 1986) to 0.002 mSv (in 2000);

(vi) deadly dose range (whole body): 3–5 Sv (delayed, 50% mortality) to 10 Sv (prompt, 99% mortality).

Furthermore, acceptability values for doses exist and are derived assuming that one person remains in one location during the due time and is exposed to a variety of established radiations. Typical values are as follows:

(i) 50 mSv/year: permitted for professional radiation workers (internationally accepted value);

(ii) 100 mSv: maximum permitted for staff working in emergency cases (extended to 250 in the case of Fukushima);

(iii) 20 mSv/year: planned evacuation area in Fukushima.

Fukushima doses have been documented (measured and reported) by different institutions in different locations around the NPP. Typical reported values are as follows (distances are related to Fukushima Daiichi NPP; in the majority of cases, assumptions to pass from Becquerel to Sievert are not available):

(i) "site boundary of Fukushima": 1000 down to 30 μSv/hr, period March 15–April 20, [19]; up to 12 mSv/hr in the air (NISA-JNES, reporting TEPCO data);

(ii) inside reactor buildings of Units 1 and 3 at the end of April: around 50 mSv/hr (TEPCO data);

(iii) "area around Unit 3" on March 15 at 10.00 local time: 400 mSv/hr [19];

(iv) various locations within 30 km radius: 50 (March 12) down to 5 μSv/hr (JAIF); at 30 km distance in N-W direction values up to 170 μSv/hr are reported by "JAEA and NSTC" (joint institution);

(v) various locations in Ibaraki prefecture, more than 100 km distance, 5 (March 20) down to 0.3 μSv/hr (JAIF);

(vi) dedicated flight around, around 1000 m altitude, April 1st, Fukushima region during two hours: 0.04 down to 0.015 μSv/hr, that is, well within natural conditions, reported by "JAEA and NSTC" (joint institution).

Correspondence between contamination and dose is given by "JAEA and NSTC" (joint institution) in relation to soil measurement also reported previously, without providing the procedure for passing from one unit to another:

(i) 12 Bq/m^3 of I-131, and 2.4 Bq/m^3 of Cs-137 = 4 μSv/hr,

(ii) 5.8 Bq/m^3 of I-131, and 1.5 Bq/m^3 of Cs-137 = 10 μSv/hr.

No person (namely worker at the NPP) is reported to have absorbed more than 300 mSv whole-body, and two workers are reported to have absorbed to the legs 2–3 Sv, that did not require treatment (NISA-JNES information). Evacuation of residents was ordered on March 12 (i.e., before massive radiological releases) within the 20 km radius from Fukushima Daiichi NPP.

The overall set of data shall give an idea to the reader of the radiological impact of the Fukushima Daiichi upon the environment and upon the population. A more comprehensive and more reliable estimate of the impact of radiation upon the environment and the population will be concluded in the next months or even years. Controversial results, namely, in terms of population doses are expected: hopefully, the aforementioned information is suitable to catch the root reasons for different results, for example, number of deaths, number of cancers, no of genetic diseases, and so forth.

INES Classification. Temporary INES classifications were issued four times from NISA related to Fukushima Daiichi NPP event: on March 12 (ten hours after the earthquake occurrence), level 3 was declared, based on the strength of the earthquake and tsunami; on March 12 (late in the day), level 4 was declared because of radiation measured above the normal level on the site; on March 18 reevaluation was carried out assuming as realistic the possibility of core damage, and level 5 was declared; finally, level 7 was declared taking into account the estimated radiological release. Level 7 was declared by NISA after evaluating a radiological release as large as 3.7 10^5 TBq (already mentioned) emphasizing that such an amount is about 10 times less than the Chernobyl release.

Some inadequacy of the INES classification transpires from the depicted situation: the direct connection between radioactivity release (in Becquerel) and the level in the scale seems misleading. Relatively small amounts of radioactivity may give rise to lethal doses to humans and huge amounts of radioactivity released in a desert region may not create any dose. A proposal-recommendation for modification of INES scale is discussed in Section 5.

4. The "Precursors": Reactor Design, Research, Regulations, and Events

The objective of this section is to discuss the predictability of the Fukushima Daiichi NPP accident and to identify possible deficiencies in established nuclear technology and reactor safety understandings. Without having the ambition to summarize the main lessons learned, let us start from the previous nuclear disasters: these had an important impact on the improvement of the nuclear technology.

(i) The TMI-2 accident in 1979 brought the attention on the inadequate consideration of SBLOCA in safety analyses, including incomplete understanding of thermal-hydraulic phenomena, for example, [17].

(ii) The Chernobyl-4 accident in 1986 allowed the characterization of a number of (more or less "minor") design deficiencies for the concerned Soviet-type reactor and demonstrated the lack of connection between nuclear safety cultures in the "Eastern Countries" and in the "Western Countries" (time period of cold-war and of Soviet Union), for example, [20].

(iii) Both TMI-2 and Chernobyl-4 accidents were largely affected by operator errors and showed the importance of qualification and responsibility for the operators: without severe human failures, both accidents would have never progressed to the respective damage levels.

The Fukushima Daiichi accident has a few or no aspects in common with the two major predecessor accidents, excluding the core damage and the radioactivity release: the Fukushima accident was originated by a natural event; the Fukushima reactors are well known by technologists as well as the occurring phenomena; the safety culture is well established in the Country; human factors did not play a key role or did not play any role at all in relation to the initiating event; early fatalities or nuclear-related injuries did not occur.

In order to better understand the Fukushima accident, keeping in mind TMI-2 and Chernobyl-4 accidents, eight topics are discussed and failures, if any, are associated with each topic: (1) the extent of the external PIE, (2) the SBO and the LOOP evolutions and the DSA support studies, (3) the LOOP mitigation effort in Units 1 to 3, (4) the SFP liquid draining phenomena; (5) the PSA support studies for PIE and DG reliability, (6) the PSA support studies for core damage and containment failure, (7) the PSA support studies for SFP cooling loss, and (8) the licensing attention to BWR4 equipped with Mark-I containment.

The Extent of the External PIE. In the case of TMI-2 and Chernobyl-4, human failures were at the origin of the concerned accident (errors in maintaining of circuitry in the secondary side and attempt to recover the reactor core fission power in the presence of large amounts of xenon). Nothing similar in the case of Fukushima happened: here natural events constituted the external PIE. The magnitude of both the events, see Section 3.2, was outside the reactor design boundaries. Human failures cannot be attributed to designers nor to operators of the NPP; rather, human failures may be associated with the process of fixing, typically by designers, and accepting, typically by regulators, the maximum amplitudes for earthquake and tsunami.

The statistical bases for fixing the design limits of earthquake and tsunami for Fukushima Units 1 to 4 were those available till the end of 1960s in the past century. Wider and more precise database was available at the end of the century. New evidence included earthquake and tsunami having amplitudes comparable or bigger than the concerned PIE. Thus, *the human failure here is not having considered the need to update the design parameters in view of the latest evidence.*

The SBO and the LOOP Evolutions and the DSA Support Studies. The SBO is intensively studied by deterministic codes (DSA framework) assuming a spectrum of originating events wider than the earthquake or the tsunami. SBO duration, for example, the time from the start of the accident till DG working (i.e., around 1 hour), is well understood; see, for example, [3]. At present, the comparison between Fukushima individual Units data (namely time-evolutions of relevant quantities, sequence of events) and results of thermal-hydraulic system code calculation are not available yet: however, no deficiency of analytical capabilities, including humans who designed the codes or performed the calculations, is expected. Moreover, no help could have been expected to make the transient from any sophisticate DSA study milder (e.g., including evaluation of three-dimensional effects).

The LOOP scenario, starting when DG failed, has a much longer duration, that is, several days, till EoA, or at least till the time when stable cooling was established for the damaged core. In this situation the precision of severe accident codes is challenged, but, again, no major deficiency is expected in relation to analytical capabilities as also discussed in [21]; see also [22]. Eventually, modeling capabilities for severe accident codes can be improved from an accurate study of the damaged cores of Units 1 to 3, but no limitation is expected that has any interaction with the occurrence or the evolution of the Fukushima accident.

The LOOP Mitigation Effort in Units 1 to 3. Mitigation countermeasures in case of LOOP were successfully actuated and prevented either the high-pressure failure of the vessel or the failure of the containment with the noticeable exception of Unit 2 where loss of integrity of the containment was experienced. It was (so far) common understanding that loss of on-site and off-site power during time duration larger than 10 hours may be in the noncredible domain. This is true notwithstanding the requirement for "new" generation reactor of 72 hours grace period; see, for example, [23].

No human failure shall be associated with the mitigation measures implemented in Fukushima. The following comments apply.

(i) The destruction in the Fukushima area caused by the earthquake-tsunami presumably affected the ownerships and the families of operators.

(ii) The same destruction could have revealed to responsible technologists or NPP managers that off-site electricity restoration was not ensured in the forthcoming dozen hours.

Mitigation measures outside the "NPP-procedures" were actually taken (use of fire-pumps, injection of sea-water, etc.) and were partially successful, as already mentioned. However, an eventual failure for managers can be identified as follows: *no success to bring to the Fukushima site (e.g., by helicopters) additional DG and pumps, namely, during the early hours of the accident*.

Minor inadequacy is connected with the lack of emphasis given to the possibility of H2 explosion after containment venting: direct containment venting to the atmosphere would have prevented the destruction of reactor building and the connected very negative impact on the public.

The SFP Liquid Draining Phenomena. Simple energy and mass balance equations can be used to predict the amount of cooling needed in any SFP as well as the time when fuel uncovery occurs; see, for example, Section 3.3.2.

Furthermore, the accident in Paks NPP (2003) constitutes a noticeable precursor of what happened in the SFP of Fukushima, for example, [24]. Nuclear fuel in a pool went dry causing harmful radioactivity releases to the environment, for example, [25]. More recently experiments were performed to characterize the nuclear fuel physical performance after the loss of cooling, for example, [26].

Then the technological understanding of phenomena consequence of loss of cooling in spent fuel pools shall be considered as suitable as well the capabilities of related computational tools. Possibly, what went wrong in connection with SFP? One may answer nothing, noting that a few kg/s of ambient temperature and pressure water would have been sufficient to avoid any radioactivity release from SFP. Namely, *one or a few responsible operators could have reacted soon (i.e., within a few hours after the earthquake) with "light pumping devices"* (even by carrying water to the pools). Obviously, after fuel uncovery and in the presence of radioactivity, the same operation (this was done!) did become troublesome.

The PSA Support Studies for PIE and the DG Reliability. Let us consider first the earthquake and the tsunami. The established occurrence and monitoring in the last 100 years of half-a-dozen earthquakes, having the energy and the destruction power potential as the current one, should bring to the result that probability for such an earthquake is 10^{-1}/year, worldwide. What in Japan for NPP design purposes? Japan is one of the most active seismic regions in the globe: consistently, the same probability 10^{-1}/year should be selected for NPP design in Japan. A large number of NPP can only increase the probability value, specifically in cases when different (and independent among each other) faults exist in the same geographical region.

Probabilities of SBO and of LOOP are recently discussed in [27]. In case of a prototype reactor located in India they found probabilities for SBO ranging from values larger than 1/year, when duration of electricity unavailability is a few minutes, down to 10^{-2}/year when duration is of the order of 100 hours. In case of LOOP the same authors achieve internationally recognized data which range between 10^{-2}/year (short duration) and, approximately, 10^{-5}/year when the duration varies from a few minutes to about 20 hours. For more than 20 hours, that is, the case of Fukushima, negligible probability shall be assumed, that is, 10^{-6}/year or lower.

Reliability data for Fukushima NPP DG should take into account the age, even though the DGs were never called in operation during their life. However, the DG failure probability with the condition of a few hundred operation hours (this time period was needed in the case of the Fukushima event) can be very close to unity. As a consequence, one might conclude that the influence of the tsunami, considered as responsible for the DG failure, might be negligible as far as the overall evolution of the accident is concerned. In other terms, the Tsunami only accelerated the evolution of the Fukushima accident.

Now, combining probabilities of a destructive "earthquake + tsunami" and considering the SBO event as certain after such a PIE and LOOP as having a reasonable (not available in the literature) consequential probability, the overall probability of the "earthquake + tsunami + LOOP" having long duration (e.g., greater than 20 hours) may result greater than 10^{-4}/year. This value would imply consideration of the event into the design envelope (DE) for NPP. Thus, *the human failure here is not having considered the need to update the design parameters (in this case the probability of the initiating event) in view of the latest evidence*.

The PSA Support Studies for Core Damage and Containment Failure. Independently upon the PIE, global core damage frequency, involving or not the containment failure, is widely studied. The comprehensive analysis made by several authors for US NRC, for example, NUREG-1150, [28], is well established and accepted: accepted range for probability of core melt is between 10^{-4} and 10^{-6}/reactor-year, where the upper limit applies to the "old", the first and the second generations of nuclear reactors like Fukushima Units 1 to 3.

The previous severe accident was Chernobyl in 1986. The operating experience between 1986 and 2011 implies more than 10^4 reactor/years. Thus a core melt event (common cause failure can be considered for the three units in Fukushima) should not be surprising in 2011.

The PSA Support Studies for SFP Cooling Loss. Similarly as in the case of the previous paragraph, SFP cooling failure has been studied in the past, for example, [2]. Although emphasizing the lack of earthquake-probability data (in 1987), the authors conclude that probabilities of the order of 10^{-4}/spent-fuel-pool-year apply when estimating the loss of SFP cooling capability and the H2 production.

Let us exclude from the present consideration the noticeable event of the Paks NPP already discussed in relation to phenomenological aspects, because this was not originated by SFP fuel. Let us consider that there are more SFP than NPP Units. Assuming as valid the above value for SFP severe accident probability, then an SFP melt event should not be surprising in 2011.

The Licensing Attention to BWR-4 Equipped with Mark-I Containment. Licensing attention has been put toward all NPP Units in the world since ever. It seems worthwhile to report here concerns raised in 1989 by US NRC in relation to severe accidents in BWR equipped with Mark-I containment; see [29]. The following is taken from this document.

> The NRC staff has identified certain containment performance improvements that would likely reduce the vulnerability of the Mark I containment to severe accident challenges.... The Commission expects that licensees of Mark I plants will seriously consider these improvements during their Individual Plant Examinations. It should be noted that these improvements should be considered in addition to improvements that stem from the evaluation and implementation of the hardened vent.
>
> (a) Alternate Water Supply for Drywell Spray/ Vessel Injection. An important improvement would be to employ a backup or alternate supply of water and a pumping capability that is independent of normal and emergency AC power. By connecting this source to the low pressure residual heat removal system (RHR) system as well as to the existing drywell sprays, water could be delivered either into the reactor vessel or to the drywell, by use of an appropriate valve arrangement. An alternate source of water injection into the reactor vessel would greatly reduce the likelihood of core melt due to station blackout or loss of long-term decay heat removal, as well as provide significant accident management capability.
>
> Water for the drywell sprays would also provide significant mitigation capability to cool core debris, to cool the containment steel shell to delay or prevent its failure, and scrub airborne particulate fission products from the atmosphere.
>
> A review of some BWR Mark I facilities indicates that most plants have one or more diesel driven pumps which could be used to provide an alternate water supply. The flow rate using this backup water system may be significantly less than the design flow rate for drywell sprays. The potential benefits of modifying the spray headers to assure a spray were compared to having water run out of the spray nozzles. Fission product removal in the small crowded volume in which the sprays would be effective was judged to be small compared with the benefit of having a water pool on top of the core debris.
>
> (b) Enhanced Reactor Pressure Vessel (RPV) Depressurization System Reliability. The Automatic Depressurization System (ADS) consists of relief valves which can be manually operated to depressurize the reactor coolant system. Actuation of the ADS valves requires DC power and pneumatic supply. In an extended station blackout after station batteries have been depleted, the ADS would not be available and the reactor would be re-pressurized. With enhanced RPV depressurization system reliability, depressurization of the reactor coolant system would have a greater degree of assurance. Together with

Table 9: Summary of background for the Fukushima Daiichi accident.

No.	Topic*	EHP	Notes
1	External PIE features	YES	Feedback from monitoring of natural events.
2	DSA: SBO and LOOP analyses	—	Notes at rows 6 and 7 apply.
3	LOOP mitigation effort	YES	Promptly bringing additional DG to the site**.
4	SFP loss of cooling	YES	Light pumping device sufficient**.
5	PSA: PIE and DG reliability	YES	Failure in estimating DG reliability under tsunami (and aging).
6	PSA: core and containment failure	—	Core melt “expected”.
7	PSA: SFP failure	—	SFP fuel failure “expected”.
8	The licensing	YES	Aging and integrated individual NPP to be better considered.

*Topics (1) to (8) are listed in a synthetic way. **Operators or managers from the Fukushima Utility.

> a low pressure alternate source of water injection into the reactor vessel, the major benefit of enhanced RPV depressurization reliability would be to provide an additional source of core cooling which could significantly reduce the likelihood of high pressure severe accidents, such as from the short-term station blackout.
>
> Another important benefit is in the area of accident mitigation. Reduced reactor pressure would greatly reduce the possibility of core debris being expelled under high pressure, given a core melt and failure of the reactor pressure vessel. Enhanced RPV depressurization system reliability would also delay containment failure and reduce the quantity and type of fission products ultimately released to the environment. In order to increase reliability of the RPV depressurization system, assurance of electrical power beyond the requirements of existing regulations may be necessary. Performance of the cables needs to be reviewed for temperature capability during severe accidents as well as the capacity of the pneumatic supply.
>
> (c) Emergency Procedures and Training. …Revision 4 to the BWR Owners Group EPG is a significant improvement over earlier versions in that they continue to be based on symptoms, they have been simplified, and all open items from previous versions have been resolved. The BWR EPG extend well beyond the design bases and include many actions appropriate for severe accident management. The improvement to EPG is only as good as the plant-specific EOP implementation and the training that operators receive on use of the improved procedures.…

Definitely, suitable licensing attention was put on individual NPP. So failure at licensing level can hardly be identified. However, the following shall be considered.

(i) *Implication of NPP aging should be more carefully evaluated; for example, very old DG may not react as brand new devices.*

(ii) *Expertise and attention is needed from the side of licensing authorities in relation to individual NPP (integrated knowledge of site-position, earthquake faults, aging, etc.).*

4.1. Evaluation and Summary Remarks. The eight topics discussed in this section are listed in the second column of Table 9. In the same table, the 3rd column deals with possible Enhanced Human Performance (EHP) and addresses the question whether EHP could have been successful in mitigating the Fukushima Daiichi accident. The word “Human” (inside the acronym EHP) is related to NPP Designer, Operators, Regulators, and Scientists involved with nuclear technology.

The performed analysis brings to synthesize the connection between the Fukushima accident and the existing nuclear technology, by distinguishing five areas.

(a) *Previous Severe Accidents*. No direct connection exists between TMI-2 (1979) and Chernobyl-4 (1986) accidents and Fukushima1-4 accident. Rather the Paks fuel accident (2003) can be considered as a precursor in phenomenological terms of the SFP accident in Fukushima.

(b) *Human Performance*. From the operator side, guidelines were followed and no operator misconduct is identifiable; however, “enhanced” reactions in the areas of LOOP mitigation and cooling of SFP (rows 3 and 4 in Table 9) could have substantially reduced the severity of the Fukushima accident. In addition, technologists (or designer, regulators, and scientists) having various connections with the nuclear technology failed (rows 1, 5, and 8 in Table 9) in the following:

(i) updating, within the NPP design, the characteristics of the earthquake and of the tsunami based on recent monitoring (row 1 in Table 9);

(ii) estimating the reliability of DG, RCIC, and IC systems (including all system structures and components) in the presence of earthquake and tsunami like those hitting Fukushima Daiichi NPP system (row 5 in Table 9);

(iii) imposing, that is, from the side of regulator, a careful consideration of both plant aging and plant modernization actions in view of the recent technological understanding or knowledge (row 8 in Table 9).

(c) *DSA*. Possible failures under DSA can be classified as human failures in general. No related deficiency is reported under item (b). Then no deficiency related to computational tools features, to their applications, or to code-user can be associated with the Fukushima accident. However, analysts performing DSA shall include scientists performing the activity under the first item of item (b).

(d) *PSA*. Possible failures under PSA can be classified as human failures in general. So, what reported under the item of item (b) applies (see also row 5 in Table 9). Obviously, such failures have consequences in the evaluations concerned with rows 6 and 7 of Table 9.

(e) *Licensing*. Possible failures under licensing can be classified as human failures in general. So what reported under the third item of item (b) applies (see also row 8 in Table 9). Definitely, the concepts of SSE and DBE need proper additional consideration and application.

5. Streamlining the Evaluation of the Event

The analysis of the Fukushima Daiichi accident has been completed by discussing "facts-as-available" in Section 3 and "connection-with-nuclear-technology" in Section 4. Here, an attempt is made to put the basis for developing the lessons-learned from the accident. It is too early to identify a proper-valuable set of lessons learned and, to this aim, a wide-range assessment is needed, including evaluations by different experts in different disciplines. So the objective here is to provide a help to derive lessons learned.

Repetition of concepts from previous sections is unavoidable, even though an effort is made to minimize such a repetition. The topics are not ordered according to their relevance, but an attempt is made to have more important topics at the top of the list. At the end of each topic a Streamlined Recommendation and Lesson learned (SRL) is proposed.

(i) The GE company modified (i.e., substantial layout changes) the containment from Mark-I to Mark-III, without introducing substantial changes in primary system layout (excluding ABWR equipped with a different containment type). The overall resistance to earthquake was a reason. Consistently with this issue, the following question should be addressed: why Onagawa NPP Units (BWR equipped with Mark-II containment) survived a stronger earthquake? SRL: *careful evaluation by regulators of important design changes and possible counter-action request for upgrading the safety of Mark-I equipped Units*.

(ii) The earthquake hitting Fukushima Daiichi NPP had characteristics outside the design basis, but its probability of occurrence is (much) larger than other events already part of the design basis envelope. SRL: *the accident design envelope for nuclear power reactors should be changed, including the severity of PIE, earthquake and tsunami, and their probability*.

(iii) Radioactivity and loss of radioactive material is not a synonymous of environmental impact, nor of injury or death for humans: infinite doses may cause zero impact if released in the desert. This was well clear before the Fukushima accident. SRL: *continuing informing the public that radiation impact is not proportional to radioactivity*.

(iv) In the case of Fukushima accident, failure of several NPP structures and components like DG and DC batteries was due to the tsunami. However, due to the long duration of the Station Blackout (a few dozen hours), the acceptability of both DG and DC batteries is not guaranteed. So-called stress tests, already defined by WENRA as "targeted reassessment of safety margins of NPP in the light of the events which occurred at Fukushima: extreme natural events challenging the plant safety function and leading to a severe accident", [30] are recommended. SRL: *to reevaluate the resistance of important safety components as a function of aging and considering extreme external events*.

(v) Estimation of reliability of systems and components success under serious conditions of flooding in the presence of common cause failure, radiation environment, and aging seems incorrect. This is valid for DG, RCIC, and IC systems, batteries, and so forth, SRL: *reevaluate PSA level 1 analysis and results based on the new failure evidences*.

(vi) Building-up nuclear units close to each other has been considered so far as an example of good engineering practice to save land and to provide a back-up to a possible fault unit by other working units. SRL-1: *having units close together has the potential to trigger a domino effect in case of failure of one unit and dispersion of radiation in the territory*. SRL-2: *a crisis center may concentrate its effort more easily toward a single unit while simultaneous failures on the same site may dilute the capability of action*.

(vii) It is difficult to find formal mistakes in the implementation and in the adoption of emergency procedures during the Fukushima accident. However, there are a number and a variety of actions that could have been performed during the first 20 hours (roughly) of the accident preventing substantial radiation releases. SRL-1 *working-organization of operators, supervisors, and heading personnel could be better organized under the concept of Enhanced Human Performance (EHP)*. SRL-2: *an Emergency Rescue Team should be created having own devices to cool the core (e.g., DG, pumps piping, and access points in each NPP Unit) and ready to act in a few hours in each nuclear site of the Country (e.g., intervening with dedicated helicopters and robots)*.

(viii) The absence of "robots" during the initial days of the Fukushima accident is remarkable in a Country which is leading in the technology. Namely, managing of radioactive fuel in the SFP, damaged or not, seems a job for robots. SRL: *to create a new nuclear technology branch like "robotics in nuclear accidents"*.

(ix) Political or public-emotion driven issues in nuclear technology, specifically in the research area, noticeably the nuclear waste, the nuclear proliferation, the decommissioning, the detailed studies of very low probability events, and undue severe regulations (if any), could have shifted resources and attention from more important technology problems. SRL: *sound engineering and actual technological needs should orient research in nuclear technology*.

(x) The failure of cooling in the SFP of Fukushima should be related to human failures. SRL-1: *create team of operators with suitable responsibility-capability-resources to* *operate fuel pools*. SRL-2: *SFP should be inside the containment (well-established concept for new NPP); H2 burners and cooling systems with suitable reliability (redundancy) should be installed in reactor building hosting SFP*.

(xi) Venting from containment to reactor building is questionable and created harmful situations during the Fukushima accident, not only due to the H2 flame. SRL: *venting from containment to the atmosphere should be considered as an accident management procedure, possibly though devoted proper chimney-loop*.

(xii) The operators of Fukushima Daiichi, at some time during the accident, decided to inject sea water into containment and into the vessel apparently. Other than an irreversible damage to structures, components, and systems, this action equals to "property-lost" and could have had safety relevant consequences including core channel blockage. SRL: *each NPP should have available, maybe underground-protected, a reservoir of a fresh water for a few-thousand cubic meters*.

(xiii) Probabilities of core melt, of containment failure, of SBO and/or LOOP occurrence, of SFP failures, and so forth, are (and were) well established in literature and are (unavoidably) consistent with the occurrence in Fukushima. Nuclear technologists are aware of the related risks and (more or less implicitly or consciously) accept those risks. SRL: *no surprise for the Fukushima accident and continuing teaching the public about the features of NPP*.

(xiv) The early evacuation ordered in the Fukushima area, well before radiation started to contaminate the environment, prevented early fatalities and most of the direct-connectable late fatalities. SRL: *a (very) severe nuclear accident may end up without radiation-connected fatalities*.

(xv) The Fukushima nuclear operators, maybe a few hundred men working at the NPP at the time of the earthquake and Tsunami, survived those events. Presumably and unfortunately, this is not true for their houses and their relatives. Their reactions on the working place were necessarily affected. SRL: *selected safety-related NPP operators should have living places not connectable with events which may occur on the site*.

(xvi) The INES value of the Fukushima event has been updated a couple of times by NISA, as already mentioned. The current INES is (satisfactorily) based on "People and the Environment", "Radiological Barriers and Control", and "Defense-in-Depth": two accidents involving the same damage of barriers and the same radioactivity release to the environment are classified at the same level. SRL: *there should be at least two indices to characterize an event: (a) one, varying in the range between 1 and 7 and dealing with radioactivity release and barrier failures, including also property loss; (b) another one, for example, varying between "a" and "g" (again seven levels) dealing with impacts on the population and (to a lower extent) on the environment. In this case, Chernobyl-4 may (remain) 7 g and Fukushima may be classified as 7a or 7b*.

6. Conclusions

A severe accident affected the Fukushima Daiichi NPP. Heavy environment, land and ocean water, and radiological contamination occurred. No early fatalities were detected due to nuclear radiations and the impact of radiations on humans was much less than expectable based on the size of the release. The "technologically satisfactory" performance was obtained in an area destroyed by earthquake and tsunami. The root causes for the technological performance are the following:

(i) the resistance of major components and structures of the individual NPP Units, although they we're not designed to withstand such strong natural events,

(ii) the early evacuation order, given well before massive releases of radiation occurred,

(iii) the effectiveness of the overall mitigation measures undertaken by the operators.

In this connection, the Fukushima accident shall not be compared with the Chernobyl-4 accident where late evacuation did not prevent large doses to the population and with TMI-2 where evacuation revealed unnecessary.

Rather, major nuclear industrial accidents in the past were initiated and were strictly affected by human failure: TMI-2 (1979), Chernobyl-4 (1986), Tokai-Mura (1999), and Paks (2003) have direct connection with operator errors. In the case of the Fukushima accident, the human intervention was a consequence of a nature originated disaster. No important failure could be attributed to operators of the Fukushima Units, in the sense that human performance is consistent with plant manuals or guidelines. However, it seems clear that Enhanced Human Performance (EHP), also involving managerial procedures, had the potential of largely impacting the severity of the accident (further investigation needed in this context).

A severe accident like the Fukushima one, independently upon the initiating event, shall be considered as "technologically expected" owing to its probability and to the number of reactors in operation: therefore, there should be no surprise (by technologists), nor emotional feedback actions, but consideration of lesson learned.

The first (in terms of time) massive radiological release during the overall Fukushima NPP accident shall be associated with the lack of cooling in the spent fuel pool of Unit 4, that is, to a system which receives "a secondary" attention in safety analyses. The resulting nuclear contamination on the site affected subsequent recovery and mitigation actions related to all other units. Paradoxically, the most easy-to-protect system, that is, the spent fuel pool requiring a few liters/second of fresh water at atmospheric pressure to be protected, constituted a significant root-cause of one of the most severe nuclear accidents in the history.

Although suitable lessons learned will need a more comprehensive, systematic, and multidisciplinary study than the present one, a series of Streamlined Recommendation and Lesson learned has been given.

Relevant findings and observations are as follows.

(i) All responsible scientists involved with nuclear technology should search for inadequacies from their professional life sharing possible connections with the Fukushima Daiichi accident.

(ii) The probability of occurrence of earthquake, tsunami and consequent Station Blackout and Loss of On-Site and Off-Site Power, could be much larger than currently estimated.

(iii) The reliability values adopted for components and structures like diesel generators, batteries, passive systems, and so forth should be revised in the presence of a situation like the Fukushima scenario and considering aging.

(iv) License renewals processes including permissions for life prolongation have been severely challenged by the Fukushima accident. Suitable "stress tests" should be considered.

(v) Differences between radioactivity releases and doses, involving detected and predictable human fatalities or injuries, should be considered when characterizing a nuclear accident. The current INES could be improved.

Recommendations for the future are as follows.

(i) First is to constitute a national (or regional) Emergency Rescue Team (ERT) capable of physically intervening in a failed NPP Unit having own devices and access locations in each unit: this might be seen as a new (active) barrier part of the defense-in-depth and summing up with the current (mostly passive) standard barriers.

(ii) There is room to exploit the capabilities of workers on the nuclear site with main reference to operators and technology managers as synthesized by the words Enhanced Human Performance (EHP).

(iii) A new nuclear technology branch, that is, Robotics in Nuclear Safety and Security (RNSS, with the last topic, that is, "security" not discussed in the present framework), shall be created or advanced.

Key outcome from the work is definitely the demonstration of strength for nuclear technology; looking at the past, misleading PSA data and inadequacy in licensing processes have been found. Looking into the future keywords are ERT, EHP, and RNSS.

Abbreviations

ABWR: Advanced Boiling Water Reactor
AC: Acceptability Criteria or Alternate Current
ADS: Automatic Depressurization System
ANS: American Nuclear Society
BDBA: Beyond DBA
BWR: Boiling Water Reactor
CO: Containment
CR: Control Room
DBA: Design Basis Accident
DBE: Design Basis Earthquake
DC: Direct Current
DE: Design Envelope
DG: Diesel Generator
DiD: Defense in Depth
DSA: Deterministic Safety Analysis
EHP: Enhanced Human Performance
EoA: End of Accident
EOP: Emergency Operating Procedures
EPG: Emergency Procedure Guidelines
ERT: Emergency Rescue Team
FA: Fuel Assembly
FP: Fission Products
FSAR: Final Safety Analysis Report
GE: General Electrics
HBU: High Burn-Up
IAEA: International Atomic Energy Agency
IC: Isolation Condenser
INES: International Nuclear Event Scale
IRSN: Institut de Radioprotection et Sureté Nucléaire
JAEA: Japan Atomic Energy Agency
JAIF: Japan Atomic Industrial Forum
JNES: Japan Nuclear Energy Safety Organization
LBU: Low Burn-Up
LOCA: Loss Of Coolant Accident
LOOP: Loss of On-site and Off-site Power
MF: Mitigation Factor
NISA: Nuclear and Industrial Safety Agency
NSTC: Nuclear Safety Technology Center (in Japan)
NRC: Nuclear Regulatory Commission
PIE: Postulated Initiating Event
PSA: Probabilistic Safety Assessment
PSP: Pressure Suppression Pool
PWR: Pressurized Water Reactor
RA: Regulatory Authority
RB: Reactor Building
RC: Reactor Core
RCIC: Reactor Core Isolation Cooling
RHR: Residual Heat Removal
RNSS: Robotics in Nuclear Safety and Security
RP: Recirculation Pump
RPV: Reactor Pressure Vessel
SA: Severe Accident
SBLOCA: Small Break Loss of Coolant Accident
SBO: Station Blackout (in the present paper, Loss of Off-site Power, or LOP)
SFP: Spent Fuel Pool
SRL: Streamlined Recommendation and Lesson learned
SRV: Steam Relief Valves
SSE: Safe Shutdown Earthquake
TAF: Top of Active Fuel
TEPCO: Tokyo Electric Power Company
TMI: Three Mile Island
UNSCEAR: United Nations Scientific Committee on the Effects of Atomic Radiation
WENRA: Western European Nuclear Regulators Association.

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