

General Electric Advanced Technology Manual

Chapter 7.1

Fukushima Dai-ichi Earthquake and Tsunami Event

TABLE OF CONTENTS

7.1	FUKUSHIMA DAI-ICHI EARTHQUAKE AND TSUNAMI EVENT	1
7.1.1	Introduction	1
7.1.1.1	Natural Event Description.....	2
7.1.1.2	NRC Immediate Response.....	2
7.1.2	Fukushima Dai-ichi Site Description and Unit Designs	3
7.1.2.1	Status of the Units Prior to the Seismic Event.....	3
7.1.2.2	Spent Fuel Storage	5
7.1.3	Seismic Event and Site Effects	5
7.1.4	Tsunami Event and Site Effects.....	6
7.1.5	Primary Hydrogen Sources.....	6
7.1.5.1	Normal Operations: Hydrogen Chemistry Control and Radiolysis of Water	6
7.1.5.2	Accident Conditions: Zircaloy-Water Reaction	7
7.1.5.3	The Significance of Hydrogen	8
7.1.6	Event Timeline	8
7.1.7	United States Nuclear Regulatory Commission Domestic Responses	10
7.1.7.1	NRC Domestic Actions for the First 3 Months	10
7.1.7.2	Near Term Task Force Mission and Recommendations	11
7.1.7.3	NRC Actions Taken on NRTF Report.....	13
7.1.8	References	16

LIST OF TABLES

7.1.1	Fukushima Dai-ichi Units 1-6 Base Information.....	4
7.1.2	Event Chronology.....	17

LIST OF FIGURES

- 7.1-1 BWR Evolution
- 7.1-2 Site Before The Event
- 7.1-3 Site Location
- 7.1-4 Site Location
- 7.1-5 Isolation Condenser Simplified Flowpath
- 7.1-5a RCIC Flowpaths
- 7.1-6 Zirc-Water Reaction Hydrogen Production in Containment
- 7.1-7 General Elevations and Inundation Level
- 7.1-8 Tsunami Elevations
- 7.1-9 Fukushima Dai-ichi Event Timeline (Figures 9a through 9j)
- 7.1-10 Fukushima Dai-ichi Containment Venting Flowpath
- 7.1-11 Fukushima Dai-ichi on the INES Scale
- 7.1-12 Zirconium Water Reaction
- 7.1-13 Aerial Monitoring Results
- 7.1-14 NRC EOC Update 5-May-2011
- 7.1-15 Unit 1 Photos
- 7.1-16 Unit 3 Photos

7.1 FUKUSHIMA DAI-ICHI EARTHQUAKE AND TSUNAMI EVENT

Learning Objectives:

1. Recognize the immediate effect and Unit 1-4 plant response to the seismic event and subsequent Tsunami.
2. Compare the systems available to mitigate an extended loss of all AC power at Units 1-3 and what factors can impact their continued operation following the event.
3. Recognize the consequences of an extended loss of all AC power.
4. Recognize the sources of hydrogen during normal and accident conditions.
5. Recognize the plant conditions and pathway that led to hydrogen accumulations in the reactor buildings.
6. Recognize the significance of the 10CFR Code requirements as related to the Fukushima event including:
 - a. Reactor Vessel Head Venting
 - b. ECCS Acceptance Criteria
 - c. Containment Inerting
 - d. Containment Combustible Gas Control
7. Recognize the NRC task force recommendations resulting from the event as related to:
 - a. Seismic and Flood Hazards
 - b. Station Blackout Mitigating Strategies
 - c. Hardened Containment Vent Systems
 - d. Spent Fuel Pool Cooling and Instrumentation
 - e. Emergency Response Organization Staffing

7.1.1 Introduction

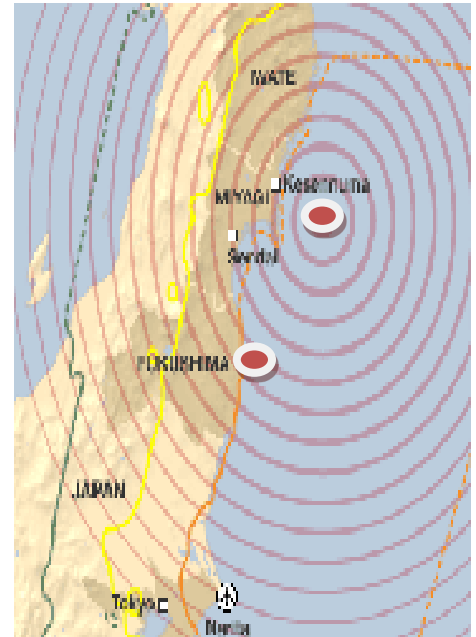
As was the case of the Three Mile Island Accident in March 1979, details of causes, effects and damages at the Fukushima Dai-ichi units will be unveiled over the next many years. The information contained herein is the best and most current information on the topic as of August 2012. Section 7.1.8 is a partial list of the references used to complete this chapter.

7.1.1.1 Natural Event Description.

On March 11, 2011 at 1446 Japan Standard Time (JST), a magnitude 9.0 earthquake occurred off the coast of Japan approximately 112 miles NE of the Fukushima Dai-ichi nuclear power station. Seismic monitors at Fukushima Dai-ichi measured the event at .56g. This is roughly equivalent to magnitude 7.0.

This earthquake triggered a tsunami estimated to be traveling in excess of 500 miles per hour. The initial displaced ocean volume is estimated to be equivalent to 3.1 cubic miles (3.1 miles in each of the x, y and z axis).

Latest data indicates that over 250 miles of the Japan Pacific coast was inundated up to six miles inland by the tsunami estimated to be up to 125 feet high.



The tsunami washed away entire towns and cities as well as devastating much of the areas infrastructure such as roads, railways, power lines, communications, etc. In December 2011 the death toll was estimated to be in excess of 22,000 with thousands still unaccounted for.

Significant to the nuclear industry yet paling in comparison to the nations challenges, was a six unit nuclear power station experiencing an extended station blackout scenario resulting from the tsunami damage.

7.1.1.2 NRC Immediate Response

The event timeline started at 1446 JST (0046 EST) on March 11, 2011. The NRC Headquarters Operations Center (HOC) entered the Monitoring Mode at 0946 EST. Out of concern for American citizens in Japan and the desire to help the Japanese deal with the unfolding nuclear events, the U.S Ambassador in Japan requested NRC representation at the U.S Embassy in Japan. The NRC staff population was canvassed for the availability of a multi-disciplinary team to support the Embassy request. The First team of NRC technical experts was dispatched to Tokyo on March 12th.

The HOC remained in the monitoring mode coordinating data gathering, supporting requests from the Tokyo Team, and providing information to various U.S. Agencies.

The Tokyo team:

- Integrated into periodic (government and other agency) meetings to gather information related to support needs.

- Maintained communications with the NRC HOC and briefed the Commission on Japanese response efforts.
- Provided assistance to the U.S. Ambassador, the Japanese, and other U.S. Agencies regarding the design and condition of the units.
- Coordinated developing, gathering and delivering support needs such as:
 - Environmental monitoring and impact assessment.
 - Initial delivery of military jet pumps for fuel cooling.
 - Conceptual description of a more permanent pumping system and placing it into use.
 - Communicating radiological consumable needs to the U.S. industry.

The NRC maintained a presence at the U.S Embassy in Tokyo throughout 2011. After the initial response, later responders:

- Supported evaluations of core conditions and adverse impacts of contaminant introduction.
- Obtained an understanding of the potential radiological impacts on people and the environment.
- Supported liquid radioactive waste containment issues.
- Continued to provide dose assessment and environmental impact assessments to the Japanese.
- Collected information to support the NRC domestic response and evaluation.

7.1.2 Fukushima Dai-ichi Site Description and Unit Designs

The Fukushima Dai-ichi (II) site (Figure 7.1-2 and 7.1-3) is a six unit General Electric based Boiling Water Reactor (BWR) generating station located approximately 180 miles north of Tokyo in the Fukushima Prefecture. Just a few miles south is the Fukushima Dai-ichi (I) four unit BWR site. Onagawa is a three unit BWR site located about 100 miles to the north. It should be noted that even though only Dai-ichi experienced fuel damage, all three sites experienced significant damage from the seismic and tsunami events.

7.1.2.1 Status of the Units Prior to the Seismic Event

As shown on Table 7.1-1 below, Units 1-3 were operating at full power. There were no significant conditions that departed from normal full power operation. Units 4-6 were in refueling outages. Unit 4 had a full core off-load to the spent fuel pool with the pool to cavity gates installed to facilitate core shroud inspections. Units 5 and 6 had partial core off-loads to the spent fuel pool with the pool to cavity gates removed. The following descriptions are supported by Table 7.1-1 and Figures 7.1-1, 7.1.2 and 7.1-10:

Table 7.1.1 Fukushima Dai-ichi Units 1-6 Base Information

BASE INFORMATION								
	GENERAL ELECTRIC VINTAGE	COMMERCIAL OPERATION	MWth	MWe	FUEL BUNDLES IN CORE	IC or RCIC	STATUS UPON EVENT	FUEL BUNDLES IN SFP
UNIT 1	BWR-3	1971	1380	460	400	IC	FULL POWER	292
UNIT 2	BWR-4	1974	2381	784	548	RCIC	FULL POWER	587
UNIT 3	BWR-4	1976	2381	784	548	RCIC	FULL POWER	514
UNIT 4	BWR-4	1978	2381	784	548	RCIC	OUTAGE	1201 to 1331
UNIT 5	BWR-4	1978	2381	784	548	RCIC	OUTAGE	950
UNIT 6	BWR-5	1979	3293	1100	764	RCIC	OUTAGE	876

Note:
 •A common spent fuel pool separate from the units contains 6000 bundles
 •The site also contains a dry cask storage facility

UNIT 1

1380 MWth (460 MWe) BWR-3 with a Mark I containment. This is similar to 6 units currently operating in the US. This is the only unit at the Dai-ichi site with an Isolation Condenser (Figure 7.1-5). The Isolation Condenser accommodates decay heat post shutdown by condensing steam from the reactor vessel to an atmosphere vented heat sink. By original design, the heat sink is functional for 3 hours without makeup to the tank. The condensate returns to the reactor vessel via gravity drain to the recirculation system. This system is an efficient passive means of exhausting decay heat load but does not provide for inventory makeup to the reactor vessel if needed. If the heat sink is exhausted, subsequent decay heat must be accommodated using Main Turbine Bypass Valves, Safety Relief Valves (SRV's) and / or the High Pressure Coolant Injection (HPCI) system. HPCI (similar to Figure 7.1-5a) is a steam driven turbine system that provides backup for the Isolation Condenser and provides inventory makeup.

UNITS 2-5

2381 MWth (784 MWe) BWR-4 with a Mark 1 containment. This is similar to 19 units currently operating in the US (5 of which have Mark II containments). Rather than an

Isolation Condenser, these units use a steam driven (Figure 7.1-5a) Reactor Core Isolation Cooling (RCIC) System that is backed up by the steam driven High Pressure Coolant Injection (HPCI) System. In this case both systems are capable of decay heat removal as well as reactor vessel inventory makeup. RCIC is significantly smaller than HPCI and is not very efficient at decay heat removal. With only RCIC in operation, excess decay heat must be accommodated by the Turbine Bypass Valves or the Safety Relief Valves.

UNIT 6

3293 MWth (1100 MWe) BWR-5 with a Mark II containment. This is similar to 4 units currently operating in the US. This unit also uses the RCIC (Figure 7.1-5) and High pressure Core Spray combination.

7.1.2.2 Spent Fuel Storage

As shown on Table 7.1-1, the spent fuel pools (SFPs) for units 1 - 4 contained different amounts of spent fuel at the time of the event, generating different heat loads. The Unit 4 SFP had the greatest heat load because the entire core had been offloaded into the SFP to support ongoing outage work and the spent fuel pool to reactor cavity gates were installed reducing the volume available to accept this heat load.

Approximately 60 percent of the spent fuel on site is stored in a separate building in a common spent fuel pool. This pool contained 6,375 fuel assemblies (about 80 percent of pool capacity), but the heat load was very low because the assemblies were stored in their respective unit SFPs for 19 months or longer before being set in the common pool. Calculations determined that cooling can be lost to this pool for 30 days before it becomes a concern. The common spent fuel pool uses fans and air for cooling, so cooling is maintained if the seawater ultimate heat sink is lost; however, AC power is required to power the fan motors and circulating pumps.

Dry cask storage is also used for spent fuel. At the time of the event, the station had nine casks containing 408 spent fuel assemblies.

Although building damage occurred, there was no damage to the common spent fuel pool or dry casks.

7.1.3 Seismic Event and Site Effects

The plants were designed for .45g horizontal and .42g vertical acceleration. As sensed on site, horizontal acceleration was .56g and vertical was .31g. The horizontal acceleration sensed was 25% beyond the design basis of the plant. The effect on the site was:

- Seismic setpoint automatic scram of Units 1-3
- Seismic damage to off-site electrical distribution resulting in a loss of offsite power to all

6 units.

- Emergency diesel generators automatically powered safety busses
- Non-safety related fuel pool cooling was lost to all 6 units.
- Plant conditions were stabilized and controlled
- Plant elevation in relation to sea level dropped approximately 3 feet.

7.1.4 Tsunami Event and Site Effects

The plants were designed for 5.7 meters (18.7 feet) at the shoreline (see figures 7.7 and 7.8). A breakwall at this elevation provided protection from the design basis tsunami. The seismic event caused the plant elevation at the shoreline to drop approximately 3 feet reducing the breakwall elevation protection to approximately 16 feet. The latest estimate is that the tsunami elevation at the site was 46 feet which is 250% of the design basis.

The tsunami effect on the site was:

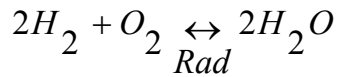
- Flooding of the EDG and Emergency Switchgear Rooms resulting in a loss of all AC and DC power for units 1, 2 and 4 and a loss of all AC power (Station Blackout) for unit 3. Unit 6 retained DC power and one EDG and was able to use the EDG to support fuel pool cooling for both units 5 and 6.
- Unit 1 Isolation condenser operating to control decay heat yet the station blackout removed IC shell makeup capability. Upon IC shell exhaustion, SRV's were used for decay heat removal without suppression pool cooling.
- Units 2-3 were using RCIC/HPCI for vessel makeup and SRV's for decay heat without suppression pool cooling.
- SRV operations in Units 1-3 resulted in significant suppression pool heatup.
- The suppression pools eventually reach saturation temperature resulting in containment pressurization. Elevated suppression pool temperatures challenge the continued operation of HPCI / RCIC.
- As station battery power depletes the remaining injection capabilities (HPCI/RCIC) are lost and a loss of adequate core cooling results.
- The zirc-water reaction discussed in Section 7.1.5.2 results in hydrogen accumulation in the reactor vessel. SRV operation transports this hydrogen to the primary containment. Primary containment leakage / failure transports the hydrogen to the secondary containment where an explosive concentration accumulates and eventually ignites.

7.1.5 Primary Hydrogen Sources

7.1.5.1 Normal Operations: Hydrogen Chemistry Control and Radiolysis of Water

All nuclear power plants that use water as a cooling mechanism in close proximity to the fission process will have hydrogen generation that results from radiolysis. In short, water in the presence of a radiation flux will disassociate into its elemental parts of hydrogen and oxygen. In addition, hydrogen and oxygen in the presence of a radiation flux will also

re-associate to form water.



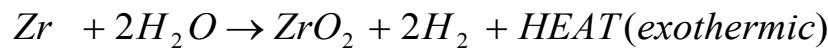
In equilibrium conditions, there will be water, hydrogen and oxygen in stable amounts. Since the water in water-cooled reactors is always passing through the core, some amount of hydrogen and oxygen is always in transport. The flowpaths in Boiling and Pressurized Water Reactors provide collection points where these gasses concentrate. In a BWR the collection point is naturally at the top of the reactor vessel. Depending upon coolant flow rates, the collection point in a PWR can be the reactor vessel, the pressurizer or both. The hydrogen concentration potential is the primary reason for the TMI Action Plan (NUREG-0737) requirement for continuous reactor vessel head venting.

Some BWR plants use hydrogen water chemistry controls to scavenge oxygen from the feed and condensate systems. Oxygen in solution that contacts the carbon steel piping in these large piping systems produces large amounts of corrosion products. These corrosion products accumulate in the BWR reactor vessel. To control the oxygen concentration in solution, excess hydrogen is injected early into the feed and condensate piping. Any leftover hydrogen will eventually make it to the BWR reactor vessel.

Compared to the hydrogen generation from accident conditions, these two sources are relatively small, provide a benefit and are easily managed.

7.1.5.2 Accident Conditions: Zircaloy-Water Reaction

During accident conditions when the core materials are inadequately cooled, the fuel cladding (zirconium alloy) can overheat which promotes and accelerates a corrosion reaction commonly referred to as the Zirc-Water Reaction.



When the core is no longer submerged in water, cladding surface temperature heats up with the uranium fuel being the heat source. At cladding surface temperatures in excess of 2000 °F the reaction rate is significant. At cladding surface temperatures approaching 2500 °F the resultant heat is enough to maintain a high reaction rate (exothermic) regardless of fuel temperature.

This reaction produces .044 pounds of hydrogen for every pound of zirconium that reacts. Using a typical BWR-4 (2436 MWth), if all the zirconium fuel cladding reacts, the result would be approximately 4400 pounds of hydrogen. From Figure 7.1-6, it can be seen that if all of the zirconium reacts and all of the hydrogen is released to a Mark I or II Primary

Containment, the resultant hydrogen concentration would be near 70%.

7.1.5.3 The Significance of Hydrogen

Hydrogen in air has a flammability range of 4 to 78%. The center of this range has the lowest ignition energy requirement. Less than 18% and greater than 58% are the deflagration ranges. In these ranges, hydrogen detonation results in a slow moving (subsonic) burn wave propagation. The range between 18% and 58% is called the detonation range with a characteristic fast moving (supersonic) wave propagation.

This massive amount of hydrogen generation and its flammability characteristic are the reason for two significant design criteria in the BWR Mark I and II plants. To limit the concentrations of hydrogen in air:

- 10CFR-50.44; Containment Combustible Gas Control, requires that all BWR Mark I and II containments maintain an inerted (oxygen free) environment during power operations.
- 10CFR-50.46; ECCS Acceptance Criteria, limits the amount of zirconium-water reaction therefore limiting the amount of hydrogen produced. In part, limiting peak cladding temperature to less than 2200 °F keeps the reaction rate below the exothermic threshold. Using Figure 7.1-6, limiting to 1% maximum hypothetical hydrogen generation will limit the Mark I and II primary containment concentrations to 2-3% hydrogen.

Now let's consider this hydrogen issue backwards from what was seen during the Fukushima event. It is pretty clear that hydrogen leakage from the primary containment resulted in reactor building concentrations in the detonation range of 18% to 58%. The typical BWR-4 reactor building volume is about 1.5 million cubic feet or, about the size of the Mark III primary containment. From Figure 7.1-6, 18% hydrogen in 1.5 million cubic feet results from a minimum of 50% of the core zirconium fuel cladding experiencing the zirc-water reaction. In other words, at least 50% of the core fuel cladding had undergone the reaction in a matter of a few short days. This conclusion is consistent with the hydrogen generation estimates of Figure 7.1.12.

7.1.6 Event Timeline

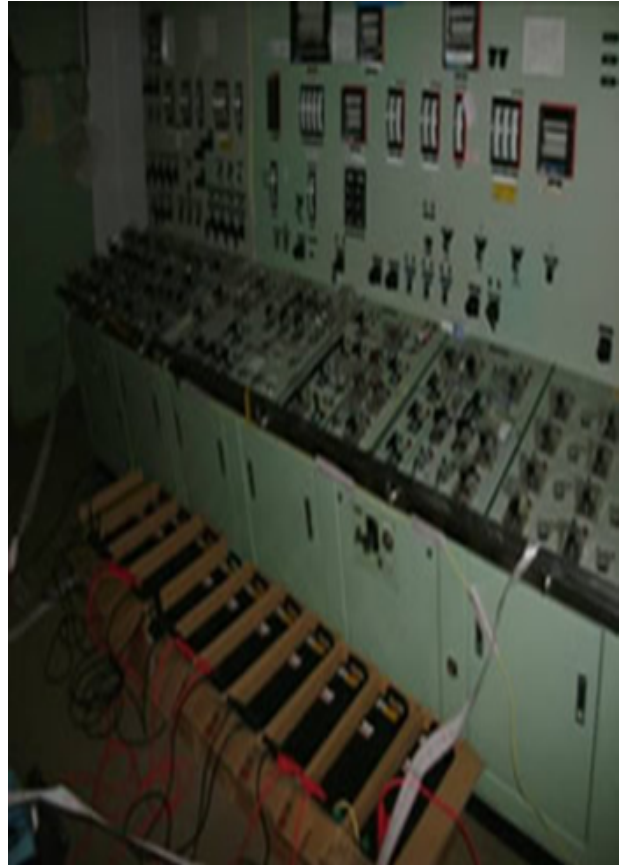
Table 7.1.2 and Figures 7.1.9 a-j describe the event chronology that eventually resulted in the destruction of Units 1-4.

For each of units 1 – 3, the following sequence occurred:

- Earthquake (.56g; 125% design basis) results in:
 - Reactor Scram
 - Loss of Offsite AC Power; Start of Emergency Diesel Generators
 - Operators controlling critical plant parameters with safety related equipment
- Tsunami (46 feet; 250% design basis) results in:
 - Flooding Loss of Emergency AC Power Distribution
 - Flooding Loss of Battery (DC) power distribution (Unit 2 retained limited DC for a while)
 - Operators lose control of critical plant parameters when safety related equipment is lost
 - Overpressure protection operation of safety relief valves depletes reactor coolant
 - Reactor coolant loss results in fuel clad overheating
 - Fuel clad overheating results in accelerated corrosion rates
 - Corrosion results in clad failure (**First Barrier Loss**) and large volumes of hydrogen produced as well as radioactive contents release
 - Vessel overpressure protection valves relieve vessel hydrogen and radioactive contents to suppression pool (**Second Barrier Loss**)
 - Suppression pool reaches boiling temperatures while hydrogen accumulates
 - Primary containment pressurizes due to suppression pool boiling
 - Primary containment is intentionally vented or fails due to over pressurization (**Third Barrier Loss**)
 - Containment contents, including hydrogen, vent to reactor building atmosphere
 - Hydrogen in air at 18 to 58% in air is explosive (detonation range).
 - Reactor Building explodes resulting in direct release pathway of core materials to environment.

For Unit 4, the current hypothesis is that operator venting of the Unit 3 containment resulted in the explosive hydrogen environment of the reactor building via common elevated release pathway piping. Figure 7.1.10 is a simplified diagram of the Fukushima Dai-ichi containment venting system. The systems were installed between 1999 and 2001 to address the need for severe accident venting of pressures that exceed the SGT system design. The design ensures an elevated release point for the vented contents. The valves that align the rupture disk to containment are normally closed and require AC power and/or station air to operate remotely. With a complete loss of AC power, seismic and tsunami debris, darkness and degrading core condition, local operation of these valves was severely challenged.

The picture at the right demonstrates one of the abnormal efforts taken to maintain operation of DC powered equipment such as HPCI, RCIC and SRV's. This picture shows scavenged batteries from various locations, including automobiles in the parking lot, assembled in a makeshift arrangement to provide 125 VDC power at sufficient amperage to operate this DC equipment.



Other similar efforts included:

- Locating portable generators to locally power AC motor operated valves
- Locating portable pneumatic bottles to locally operate air operated valves
- Series, parallel and shuttle operation of site and public fire trucks to transport water of various qualities to the reactor vessels and spent fuel pools
- Hand carrying hundreds of kilometers of high voltage power cable to energize otherwise un-damaged electrical distribution loads centers.

The Fukushima Dai-ichi site event was eventually classified a Category 7, or Major Accident, on the International Nuclear and Radiological Event Scale (INES) (Figure 7.1.11).

7.1.7 United States Nuclear Regulatory Commission Domestic Responses

7.1.7.1 NRC Domestic Actions for the First 3 Months

- On March 18, 2011, IN 2011-05 was issued to inform U.S. Licensees of event details.
- On March 23, 2011, TI 2515/183 was issued to all U.S. Operating License Holders directing inspections to assess similar vulnerabilities at their facilities. The deadline for reporting the results was set at May 13, 2011.
- On April 29, 2011, TI 2515/184 was issued to all U.S. Operating License Holders directing inspections to assess adequacy of the facility Severe Accident Management Guidelines (SAMG's). The deadline for completing the inspection was set at 27 May, 2011.

- On May 11, 2011, BL 2011-01 was issued to all U.S. Operating License Holders requiring a comprehensive verification of facility compliance with 10CFR 50.54(hh)(2), This part requires facility strategies for restoring core cooling, containment and spent fuel cooling after a loss of large area of the plant due to explosion or fire. The deadline for reporting the results of the inspection was set at 11 June, 2011.

7.1.7.2 Near Term Task Force Mission and Recommendations

- On the day of the event, March 11, 2011, data gathering began.
- On March 23, 2011, a NRC Chairman Tasking Memo established the Near Term Task Force responsible for collecting and evaluating information and making recommendations to the Commission.
- On July 13, 2011, the Commission was presented the written report, SECY-11-0093, "Near Term Report and Recommendations for Agency Actions following the Events in Japan," dated July 12, 2011 (ADAMS Accession No. ML11186A950).
- On July 19, 2011, the Task Force presented the recommendations verbally to the Commission.
- On October 3, 2011 the NRC Staff proposes a prioritization plan for implementing the Task Force recommendations, SECY-11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned" (ADAMS Accession No. ML11272A111)
- On December 15, 2011, the Commission approved the prioritization plan for the Task Force Recommendations.

Prioritization Plan

- TIER 1
Those recommendations that should be implemented without unnecessary delay and for which sufficient resources are available
- TIER 2
Those recommendations that cannot be initiated in the near term due to factors that includes the need for further technical assessment and alignment, dependence on Tier 1 issues, or availability of critical skill sets.

- TIER 3
Those recommendations that require further staff study to support a regulatory action, have an associated shorter-term action that needs to be completed to inform the longer-term action, or are dependent on the availability of critical skill sets.

INITIAL TIER 1 RECOMMENDATIONS (Report recommendation number)

- Licensees re-evaluate the seismic and flood hazards against current NRC requirements and upgrade as necessary. (2.1)
- Licensees perform seismic and flood protection walkdowns to identify and address vulnerabilities. (2.3)
- Rulemaking to 10 CFR 50.62 (SBO) for a minimum coping time of 8 hours with on-site resources and 72 hours using off-site resources. (4.1)
- Licensee orders to upgrade the storage and protection of equipment covered under 10CFR50.54(hh)(2). (4.2)
- Order licensees to include reliable hardened vents for Mark I and Mark II containments capable of easy operation during prolonged SBO events. (5.1)
- Order licensees to install sufficient control room safety related Spent Fuel Pool Instrumentation (7.1) (without delay)
- Orders and rulemaking to require SAMG's and EDMG's as a license condition and more realistic hands on staff training to all involved decision makers. (8)
- Order licensees to upgrade the Emergency Plan staffing and strategies for multi-unit and extended SBO events. (9.3)

TIER 2 RECOMMENDATIONS (Report recommendation number)

- Further Spent Fuel Pool Enhancements:
 - Order licensee to provide safety related AC power for SFP makeup. (7.2)
 - Order licensees to revise Technical Specifications for SFP makeup and instrumentation operability during all modes of operation. (7.3)
 - Order licensee to install a seismically qualified SFP portable pump spray system from outside grade level. (7.4)
 - Rulemaking to require all 4 above recommendations. (7.5)
- Emergency preparedness regulatory actions (Recommendation 9.3b)
 - Require licensees to have guidance for multi-unit dose assessment capability
 - Require licensees to hold training and exercises for multi-unit and prolonged SBO scenarios
 - Require that licensees practice the identification and acquisition of offsite resources
- Require that licensees ensure that EP equipment and facilities are sufficient for dealing with multi-unit and prolonged SBO scenarios.
- Other (Related to Recommendation 2.1)
 - Request that licensees perform a new external hazards evaluation using current NRC requirements and guidance.

TIER 3 RECOMMENDATIONS (Report recommendation number)

- Rulemaking to confirm the seismic and flooding hazards analysis and upgrade facilities as necessary every 10 years. (2.2)
- NRC evaluation to determine potential enhancements to the capability to prevent or mitigate seismically induced fires and floods. (3)
- Re-evaluate the need for and issue appropriate Regulatory Action to require reliable hardened vents for other containment designs. (5.2)
- NRC review and identify Fukushima insights about hydrogen control and mitigation inside containment or other buildings as they become available. (6)
- Rulemaking for require Emergency Preparedness upgrades to address multi-unit and extended SBO events (9.1 and 9.2)
- Upgrade/enhance ERDS to maintain capability throughout the events. (9.3)
- Additional EP Topics for Prolonged SBO and Multi-Unit Events (10)
 - Protective equipment requirements for emergency responders
 - Command and control structure and qualification requirements for decision makers
 - ERDS transmission methods independent of hardwire infrastructure
 - ERDS continuous transmission evaluation
 - Evaluate completeness of current ERDS data set with respect to assessment needs.
- EP Topics for Decision Making, Radiation Monitoring, and Public Education (11)
 - Enhancing on-site and off-site emergency response resources and availability with a degraded off-site infrastructure.
 - Evaluate and address further lessons learned from the Fukushima event.
 - Study the possibility of real time radiation monitoring within the EPZ including protected power supplies.
 - Public radiation safety and use of KI training coordinated with other Federal agencies.
- Strengthen the Reactor Oversight Process (ROP) with more focus on the defense-in-depth requirements. (12.1)
- Enhance NRC staff training on Severe Accidents and provide Resident Inspector training on SAMG's. (12.2)

7.1.7.3 NRC Actions Taken on NRTF Report

- The staff has expanded upon the recommendations of NRC's Near Term Task Force and continues to make additions and modifications, as appropriate.
- The staff initiated SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons-Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Tsunami" (ADAMS Accession No. ML12039A103) dated February 17, 2012.
- The Tier 1 Orders and 50.54(f) letter were issued March 12, 2012.

TIER 1 ORDERS TO LICENSEES

- EA-12-049, Requirements for Mitigation Strategies for Beyond-Design-Basis External Events. (derived from Recommendation 4.2)
 - Requires licensees to develop strategies and procure additional equipment to address extended loss of all AC power.
 - Three phases including initial response using installed equipment, use of FLEX equipment stored on site and use of off-site resources for long term mitigation.
 - All licensees have submitted an implementation plan for NRC review. Strategies must be in place no later than December 30, 2016.
- EA-12-050, Reliable Hardened Containment Vents. This order was later superseded by EA-13-109, Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions containments (derived from Recommendation 5.1).
 - Requires licensees to install a hardened containment vents in Mark I and Mark II containments.
 - Phase 1 is to install a wetwell vent system. Implementation plans are due to the NRC by June 30, 2014 and modifications must be complete no later than June 30, 2018.
 - Phase 2 is to either install a hardened drywell vent system or develop and implement a reliable containment venting strategy that makes it unlikely that a licensee would need to vent from the containment drywell during severe accident conditions. Implementation plans for phase 2 are due to the NRC by December 31, 2015 and the modification or analysis must be completed no later than June 30, 2019.
- EA-12-051, Reliable Spent Fuel Pool Instrumentation (derived from Recommendation 7.1)
 - Requires licensees to install safety related spent fuel pool level instrumentation (or have portable equipment available) that is missile protected, capable of remote read-out and independently powered.
 - All licensees have submitted an implementation plan for NRC review. Modifications must be in place no later than December 30, 2016.

TIER 1 REQUESTS FOR INFORMATION

- Provide information on whether facility configurations, as confirmed by seismic and flooding walkdowns, are in compliance with current facility design bases (derived from Recommendation 2.3)
 - All seismic and flood protection walkdowns are completed.
- Reevaluate the adequacy of facility design bases with respect to seismic and flooding hazards based on the latest information and analysis techniques (derived from Recommendation 2.1).

- Flooding evaluation submittals are broken into three categories based on the potential for the need to complete an integrated assessment. Category 1 plants submitted information by March 12, 2013 while Category 2 and 3 submittals are due March 12, 2014 and March 12, 2015 respectively. The majority of the first set of plants provided the results of their hazard reevaluations by March 12, 2013. Six sites requested and were approved for extensions, primarily to facilitate use of more accurate models. Several sites stated that the results of their reevaluated hazards indicate they will need to take interim actions (e.g., having standby sandbags in place before a permanent barrier can be constructed), and several sites indicated that they will be performing an integrated assessment to determine if permanent changes are needed.
- Seismic evaluation submittals broken into two groups. Eastern and central plants were originally required by September 2013, but the industry proposed an update to the ground motion model and additional screening for some plants that will require an expedited evaluation. These submittals are now due by March 2014. The western plants are required to submit their revised seismic evaluations by March 2015.
- Provide information on current communications system power supplies and their availability during a prolonged SBO event (derived from Recommendation 9.3)
 - The licensees have supplied their responses regarding the assessment of their communications capability for a multiunit prolonged station blackout (SBO) event. The NRC staff has completed its review of the communications assessments and determined that proposed interim actions combined with long-term enhancements will help to ensure that licensees can effectively communicate during a station blackout (SBO) event affecting multiple units.
- Provide information on the required staffing necessary to respond to a multiunit, prolonged SBO event (derived from Recommendation 9.3)
 - The licensees have provided the first part of their staffing assessments regarding the plant personnel needed to respond to a multiunit prolonged SBO. The remaining portions of the staffing and communications request is expected to be impacted by the licensees' mitigation strategies being developed to address Order EA-12-049.

TIER 1 RULEMAKING ACTIVITIES

- Station Blackout Mitigating Strategies (SBOMS) Rulemaking (derived from Recommendation 4.1 and also includes Recommendation 7 for SFP enhancements)
 - Codifies the Mitigating Strategies order to require enhanced capability to mitigate a prolonged SBO.
 - Includes the Tier 2 recommendations for SFP modifications.
 - Advanced Notice of Proposed Rulemaking has been issued.
 - The regulatory basis for the proposed rulemaking has been published in the Federal Register.
 - The final rule is scheduled for completion by December 2016.

- Emergency Procedures Integration Rulemaking (derived from Recommendation 8)
 - Creates a new rule requiring the integration of the emergency procedures and establishes standards that ensure plants can smoothly transition between various emergency procedures, keeping overall strategies coherent and comprehensive. Includes training and qualification requirements for severe accident guidelines.
 - Advanced Notice of Proposed Rulemaking has been issued.
 - The regulatory basis for the proposed rulemaking has been published in the Federal Register.
 - The final rule is scheduled for completion in March 2016.
- External Engineered Containment Filter (Recommendation 5)
 - In SRM-SECY-12-0157, the Commission instructed the staff to initiate rulemaking that considers filtering and confinement strategies for limiting the release of radiological material in the event of a severe accident at BWRs with Mark I and II containments.
 - The technical basis for the rulemaking is scheduled to be presented to the commission by March 2014.

7.1.8 References

- NUREG/CR-2726; Light Water Reactor Hydrogen Manual; Sandia National Labs 1983
- Overview of Events at Fukushima Nuclear Power Plant Sites Following Earthquake and Tsunami on March 11, 2011; N.G. Trikouros, 6 April, 2011
- Tohoku Pacific Earthquake and the Seismic Damage to the NPP's; Nuclear and Industrial Safety Agency Japan, 21 March, 2011
- The Great East Japan Earthquake and Current Status of Nuclear Power Stations; Tokyo Electric Power Company, 18 April, 2011
- Earthquake and Tsunami in Japan on March 11, 2011 and Consequences for Fukushima and other Nuclear Power Plants: VGB Power Tech, 15 April, 2011
- NRC Report by Fukushima Daiichi Near Term Task Force; Recommendations for Enhancing Reactor Safety in the 21st Century
- SECY-11-0137; Prioritization of Recommended Actions to be Taken in response to Fukushima Lessons Learned
- SRM-SECY-12-0035; Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami
- 3/13/12 Orders and 50.54(f) Letter (ML12054A696, ML12054A736 , ML12054A682, & ML12056A046)
- Causes and Countermeasures: The Accident at TEPCO's Fukushima Nuclear Power Stations, Masaya Yasui, Deputy Director General, Nuclear Safety Regulation Reform Ministry of Economy, Trade and Industry (METI) March, 2012
- Fukushima Nuclear Accident (Interim) Report; TEPCO, December 02, 2011
- The Fukushima Nuclear Accident Independent Investigation Commission (NAIIC) Official Report; The National Diet of Japan; June 2012

Table 7.1.2 Event Chronology

JAPAN TIME EST + 14	TIME ZERO PLUS	SUMMARY
3/11/11 1446 (0046 EST)	00:00	Units 1, 2 and 3 operating at full power. Unit 4 in a refueling outage with a full core off load to the spent fuel pool. Units 5 and 6 in a refueling outage with partial core off load to the spent fuel pool.
1446	00:00	Magnitude 9.0 Earthquake offshore approximately 112 miles NE. The magnitude measured at the site was approximately 7.0 (.56g) resulting in: <ul style="list-style-type: none"> • automatic scram on seismic monitoring • damage to power lines resulting in a loss of offsite power • automatic starting of the emergency diesel generators • loss of spent fuel pool cooling for all units • loss of spent fuel pool inventory for all units due to wave movement
1452	00:06	Unit 1 Isolation condenser in service Unit 2 RCIC in service
1503	00:17	Unit 1 operators throttling Isolation Condensers.
1505	00:19	Units 2 and 3 RCIC in service. RCIC subsequently cycled on and off as necessary for RPV level control.
1525	00:39	Unit 1 torus cooling and spray in service Unit 2 torus cooling and spray in service Unit 3 RCIC trips on high RPV level.
1527	00:41	The first of seven tsunamis arrives at site The maximum height tsunami was estimated at 46-49 feet
1528	00:42	Unit 2 RCIC trip on high RPV water level
~1537	00:51	Unit 3 operators have broken condenser vacuum and shed all unnecessary DC loads.
1539	00:53	Unit 2 RCIC manually restarted
1541	00:55	Tsunami flooding of the emergency diesel generators and/or emergency switchgear for units 1-4. This combined with the previous loss of offsite power resulted in a loss of all AC and DC power for units 1, 2 and 4 and a loss of all AC power (station blackout) for unit 3. One (air cooled) of five emergency diesel generators survived for power to unit 6 and later to unit 5 At this time, HPCI's and the Isolation Condensers were all out of service.

Table 7.1.2 Event Chronology (cont.)

JAPAN TIME EST + 14	TIME ZERO PLUS	SUMMARY
1603	01:17	Unit 3 RPV level at 157 inches > TAF Unit 3 RCIC started up for injection.
1810	03:24	Unit 1 RPV level below TAF (TEPCO MAAP analysis)
1850	04:04	Unit 1 start of core damage (TEPCO MAAP analysis)
2007	05:21	Unit 1 reactor pressure 1000 psig (locally). Reactor water level unknown.
2049	06:03	Unit 2 operators estimated core uncover at 2140. Evacuation ordered for 1.9 mi (3 km) around the site.
2050	06:04	Evacuation order for 1.2 mile radius (2 km)
2119	06:33	Unit 1 reactor water level TAF + 8 inches.
2123	06:37	Evacuation order extended to 1.9 mile (3 km) radius. Seek shelter for 6.2 miles (10 km)
2150	07:04	Unit 2 level indication restored and reading 134 inches > TAF.
2325	08:39	Unit 2 RPV pressure 914 psig and Containment pressure 5 psig.
2330	08:44	Unit 1 reactor building entrance dose rate is 120 mrem/hr. Unit 1/2 control room dose rates rising.
2350	09:04	Unit 1 Containment Pressure Instrument (drywell?) re-energized and reading 87 psig. Unit 1 Reactor building access is restricted due to radiation doses.
3/12/11 0230	11:44	Unit 1 drywell pressure at 122 psig. Unit 1 RPV pressure at 116 psig Unit 2 RCIC operation concluded by local indications.
0404	13:18	Unit 1 drywell pressure dropping slowly Site boundary dose rate at .1 mrem/hr (normal is .007)
0514	14:28	Unit 1 drywell pressure 112 psig and slowly dropping. Unit 1/2 control room dose rates rising.
0544	14:58	Evacuation order extended to 6.2 miles (10 km)
0546	15:00	Unit 1 start of freshwater injection using a fire truck shuttle
0903	18:15	Operators attempting to vent Unit 1 Containment in 30 Rem/hr fields. Attempt fails due to exceeding emergency exposure limit of 10 R.
1136	20:48	Unit 3 RCIC lost and could not be restarted (cause unknown)
1205	21:17	Unit 1 RPV level is 59 inches < TAF. Greater than 10,000 gallons of water injected since 0546
1210	21:22	Unit 3 primary containment pressure at 42 psig.

Table 7.1.2 Event Chronology (cont.)

JAPAN TIME EST + 14	TIME ZERO PLUS	SUMMARY
1235	21:47	Unit 3 HPCI auto start at 120 inches > TAF. Operators controlled injection to prevent hi level trip. HPCI causing RPV cooldown.
1410	23:24	Unit 1 drywell pressure 106 psig and still dropping
1430	23:44	Unit 1 containment venting in progress Site boundary dose rates exceed 100 mrem/hr
1453	24:07	Greater than 20,000 gallons of water injected since 0546
1536	24:50	Unit 1 reactor building explodes. The explosion destroyed a temporary generator about to be used for Unit 2 injection and caused a unit 2 blowout panel to open. Also results in significant airborne contamination of Unit 3/4 control room requiring respirators.
1540	24:54	100 mrem/hr detected outside the reactor buildings
1600	25:14	Detectable iodine on site
1700	26:14	Unit 3 RPV pressure at 421 psig and lowering due to HPCI operation.
1730	26:44	Unit 2 containment pressure at 30 psig. RCIC still injecting to Unit 2 from the suppression pool. Preparing to vent the Unit 2 containment.
1825	27:39	Evacuation order extended to 12.4 miles (20 km)
1904	28:18	Unit 1 start of seawater injection
3/13/11 0242	35:56	Unit 3 HPCI tripped and could not be restarted due to battery depletion. RCIC also could not be started due to battery depletion. Local attempts to run RCIC also failed. Unit 3 RPV pressure at 580 psig and rising. Cannot operate SRV's due to battery depletion.
0415	37:29	Unit 3 RPV level at TAF and lowering
0500	38:14	Unit 3 RPV pressure at 1070 psig. Unit 3 RPV level at 79 inches < TAF. Unit 3 containment pressure at 38 psig.
0739	40:53	Unit 3 initiated containment sprays using a fire pumper.
0745	40:59	Unit 3 RPV level is 118 inches < TAF. Unit 3 RPV pressure is 1060 psig Unit 3 primary containment pressure is 52 psig. Unit 3 Torus pressure at 49 psig
0841	41:55	Maximum Unit 3 containment pressure at 78 psig and containment venting commenced.
0855	42:09	Unit 2 containment pressure 39 psig

Table 7.1.2 Event Chronology (cont.)

JAPAN TIME EST + 14	TIME ZERO PLUS	SUMMARY
0856	42:10	Site boundary dose rates at 88 mrem/hr.
0908	42:22	Unit 3 SRV opened on automobile batteries.
0910	42:24	Unit 3 containment pressure is 77 psig. Unit 3 RPV level below TAF (TEPCO MAAP analysis)
0920	42:34	Unit 3 containment venting
0924	42:38	Unit 3 containment pressure is 63 psig.
0925	42:39	Unit 3 RPV pressure is 51 psig. Unit 3 start of freshwater injection
1040	43:54	Unit 3 start of core damage (TEPCO MAAP analysis) Unit 4 hydrogen buildup from unit 3 containment venting with core damage.
1100	44:14	Unit 2 aligned for containment venting. Containment pressure < 62 psig.
1300	46:14	Unit 3 RPV pressure at 28 psig. Unit 3 RPV level at 79 inches < TAF Site boundary dose rates at 156 mrem/hr
1312	46:26	Unit 3/4 control room dose rates up to 1200 mrem/hr Unit 3 start of seawater injection
1431	47:45	Unit 3 RB airlock dose rates from 10 to 30 rem/hr.
1530	48:44	Unit 3 control room at 1.2 rem/hr
3/14/11 0200	59:14	Unit 3 containment pressure 23 psig and trending up. The containment vent valve(s) had failed closed.
0310	60:24	Unit 3 RPV level is 91 inches < TAF
0340	60:54	Unit 3 loss of containment venting. Containment pressure is 33 psig.
0520	62:34	Unit 3 containment pressure 51 psig.
0700	64:14	Unit 3 drywell dose rate at 16,700 rem/hr just prior to commencing venting
1053	68:07	Unit 3 containment pressure at 60 psig and decreasing
1101	68:15	Unit 3 reactor building explodes. Significant damage to ground efforts for electrical power and water injection to Unit 2. Debris outside building up to 1000 rem/hr Unit 2 containment pressure at 52 psig.

Table 7.1.2 Event Chronology (cont.)

JAPAN TIME EST + 14	TIME ZERO PLUS	SUMMARY
1130	68:44	Unit 2 containment pressure is 51 psig
1200	69:14	Unit 3 containment pressure at 38 psig and rising.
1230	69:44	Unit 2 containment pressure is 56 psig and 301 degrees water temperature (saturated).
1325	70:39	Unit 2 RCIC lost due to pump failure. RPV level at 95 inches > TAF. Predicted TAF at 1630. Unit 2 drywell pressure is 52 psig
1600	73:14	Unit 2 RPV level 12 inches > TAF Continuing efforts to vent containment and depressurize the RPV
1620	73:34	Unit 2 RPV level is TAF
1630	73:44	Unit 2 RPV level at BAF (questionable indication) 32 mrem/hr measured at site access gate
1634	73:48	Unit 2 RPV pressure at 1015 psig.
1700	74:14	Unit 2 RPV level below TAF (TEPCO MAAP analysis)
1800	75:14	Car batteries used to open 1 SRV to start RPV depressurization from 1015 psig.
1803	75:17	Unit 2 RPV pressure is 881 psig
1822	75:36	Unit 2 RPV level trended to offscale low (BAF) (core complete uncover)
1903	76:17	Unit 2 RPV pressure 91 psig and lowering slowly through SRV.
1920	76:34	Unit 2 start of core damage (TEPCO MAAP analysis)
1954	77:08	Unit 2 start of seawater injection via fire pumper.
2100	78:14	Unit 2 containment pressure just less than 62 psig (below rupture disk setpoint)
2130	78:44	Unit 2 RPV level 118 inches < TAF
2200	79:14	Unit 2 RPV level at 63 inches below TAF due to fire pump injection. 76 mrem/hr at the site boundary.
2250	80:04	Unit 2 containment pressure is 63 psig. Unit 2 RPV pressure increase from 62 to 264 and RPV level decrease from 27 to 63 inches < TAF since 2240.
2344	80:58	Unit 2 containment pressure at 94 psig and suppression chamber pressure is 44 psig. (questionable)
3/15/11 0002	81:16	Unit 2 containment pressure stable at 95 psig.

Table 7.1.2 Event Chronology (cont.)

JAPAN TIME EST + 14	TIME ZERO PLUS	SUMMARY
0128	82:42	Unit 2 RPV pressure is 91 psig
0300	84:14	Unit 2 drywell at 94 psig and Unit 2 torus at 33 psig (questionable). Attempts to vent containment still unsuccessful.
0600	87:14	Loud noise heard from Unit 2 Torus area. Unit 2 Torus pressure dropped to 0 psig. Drywell pressure remained at 90 psig. Unit 2 RPV level is 110 inches < TAF
0614	87:28	Unit 4 reactor building explodes 40 Rem/hr measured at ground floor of Unit 3 reactor building
0620	87:34	Unit 2 containment pressure at 92 psig and still questionable. Unit 2 torus at <0 psig. (apparently a failed instrument). Containment failure resulting in depressurization and large radioactive release is suspected.
0651	88:05	60 mrem/hr measured at site access gate. Non-essential personnel (650) evacuated leaving 70 on site.
0820	89:34	Site (Main Gate) boundary dose rates 81 mrem/hr
0825	89:39	Steam release through hole in side of Unit 2 reactor building
0938	90:53	Unit 4 reactor building on fire
1125	92:40	Unit 2 containment pressure reduced from 91 to 10 psig after another loud noise.
2305	104:20	Site boundary (Main Gate) radiation dose rate at 455 mrem/hr

BWR EVOLUTION

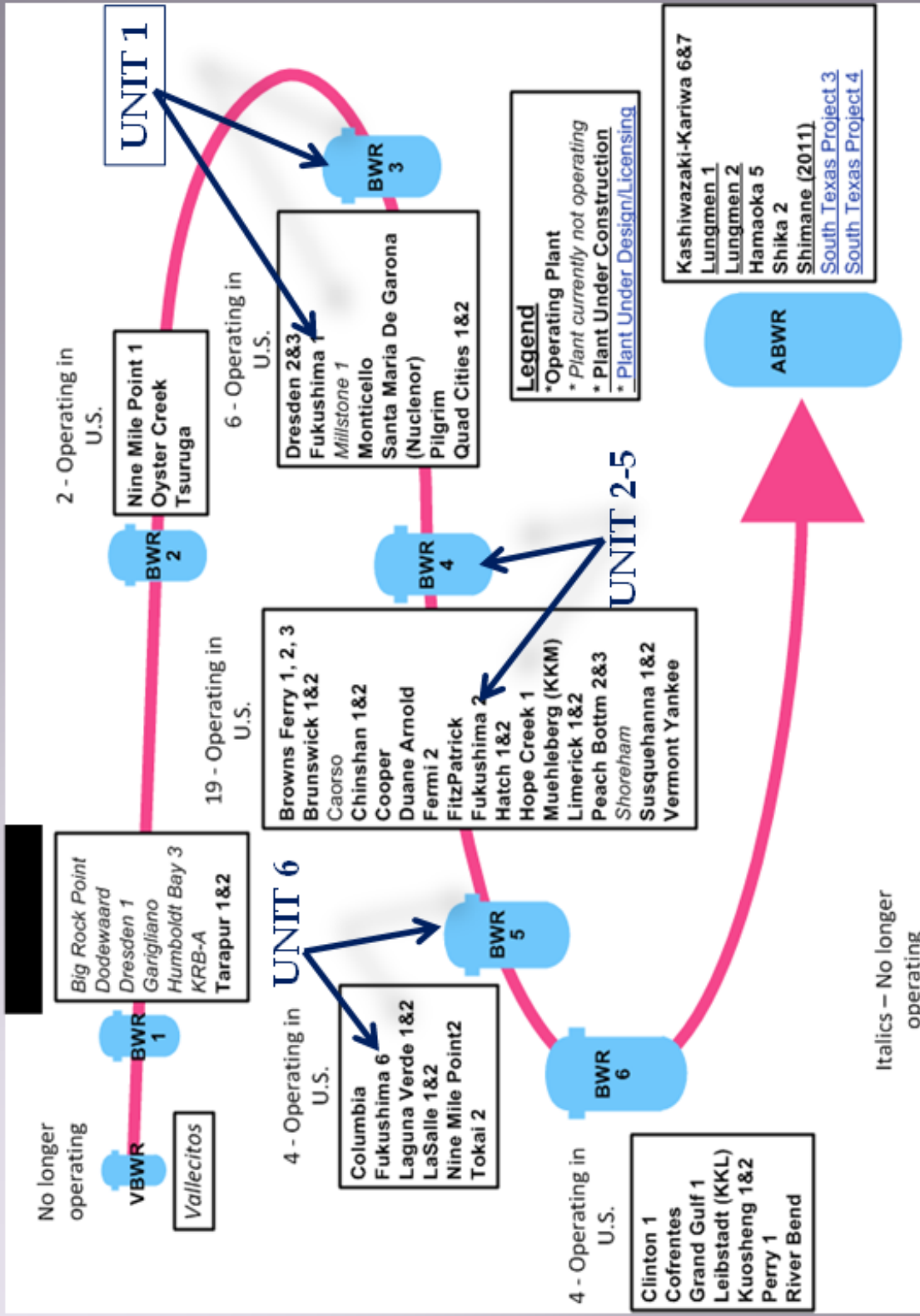


Figure 7.1-1 BWR Evolution

FUKUSHIMA DAI-ICHI SITE BEFORE EVENTS

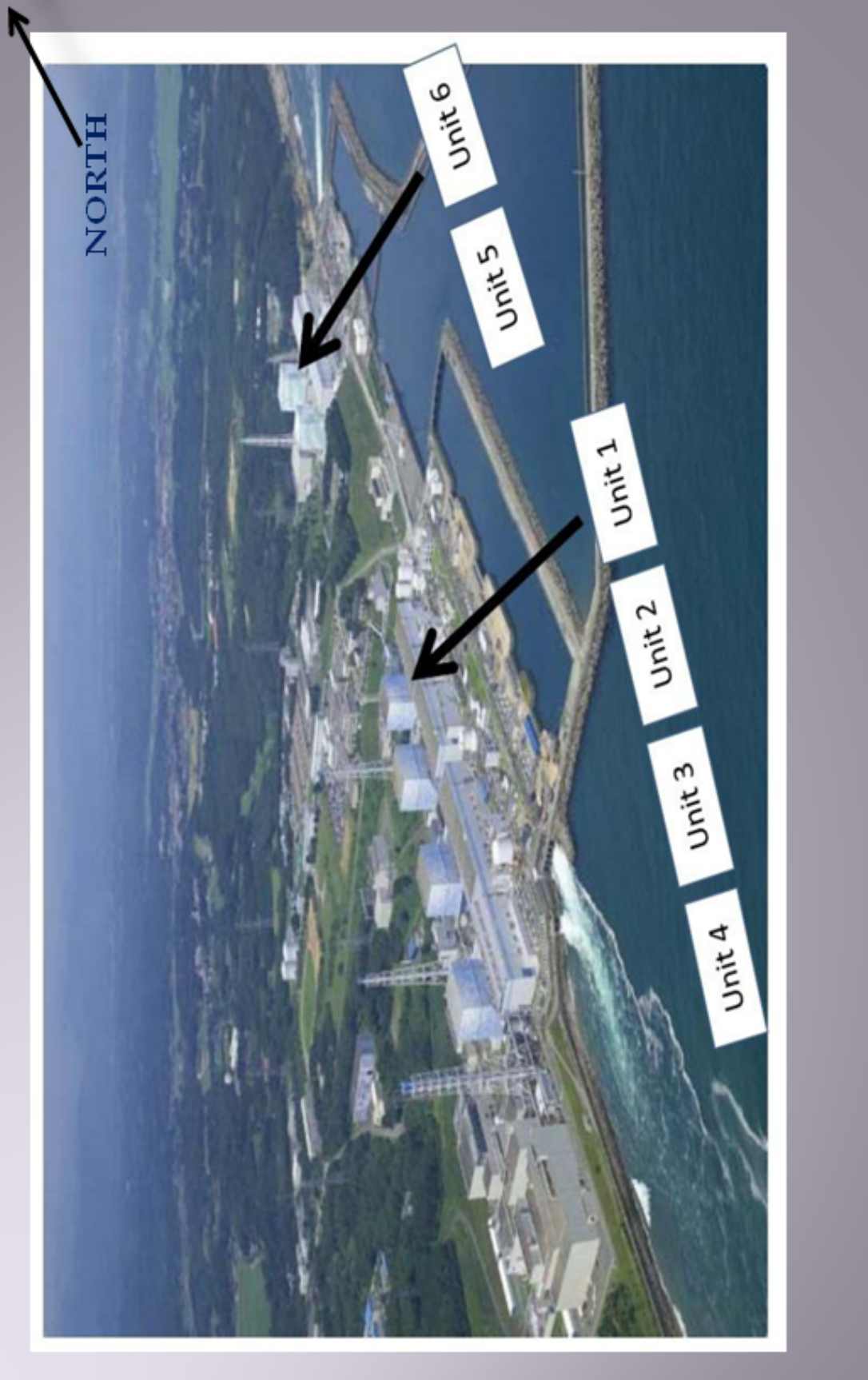


Figure 7.1-3 Site Location

SITE LOCATION



Figure 7.1-4 Site Location

TYPICAL BWR 2/3 DECAY HEAT REMOVAL / CORE COOLING SYSTEMS

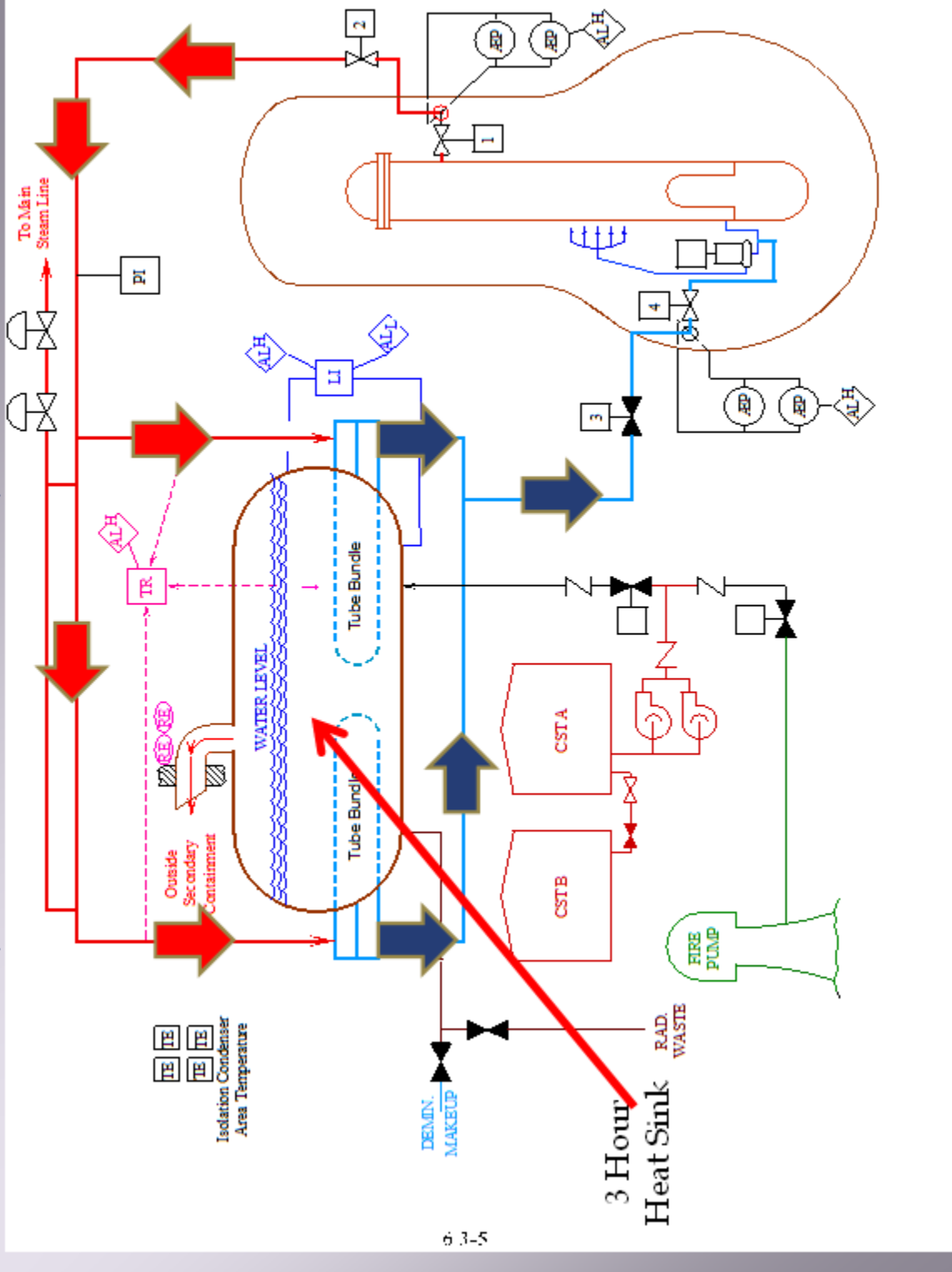


Figure 7.1-5 BWR 3 Isolation Condenser

TYPICAL BWR 4 DECAY HEAT REMOVAL / CORE COOLING SYSTEMS

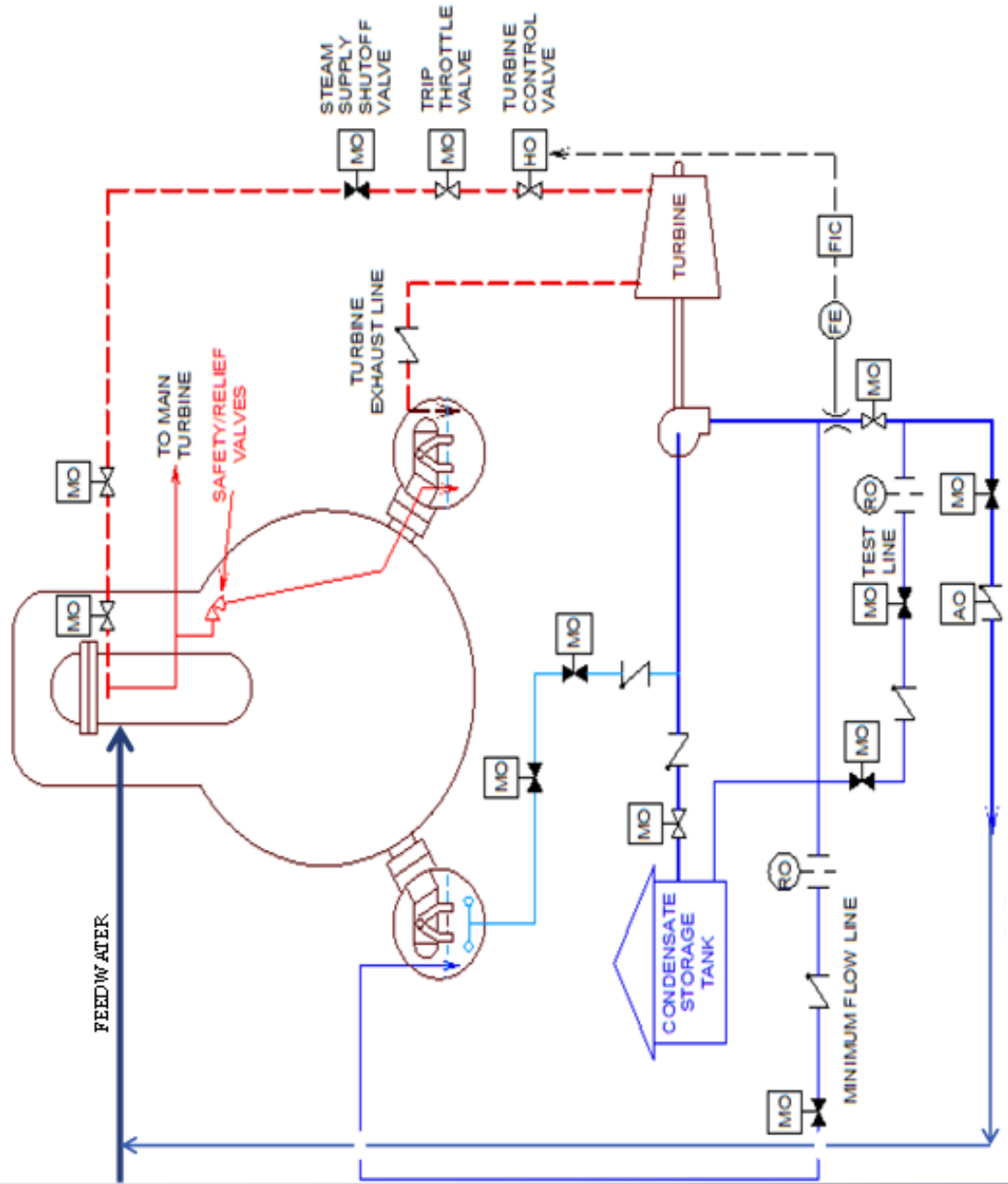


Figure 7.1-5a BWR 4 RCIC

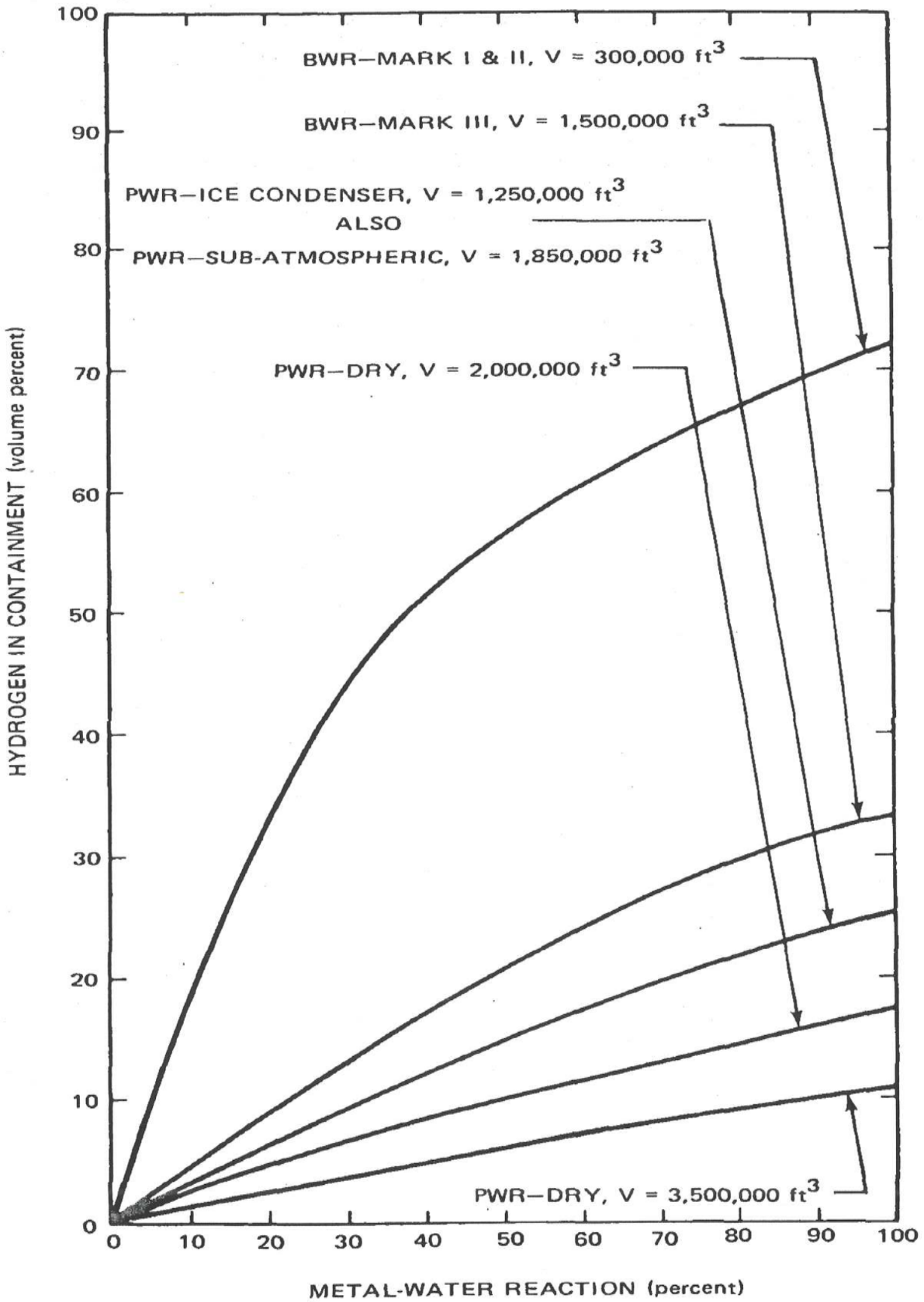


Figure 7.1-6 Zirc-Water Reaction Hydrogen Production in Containment

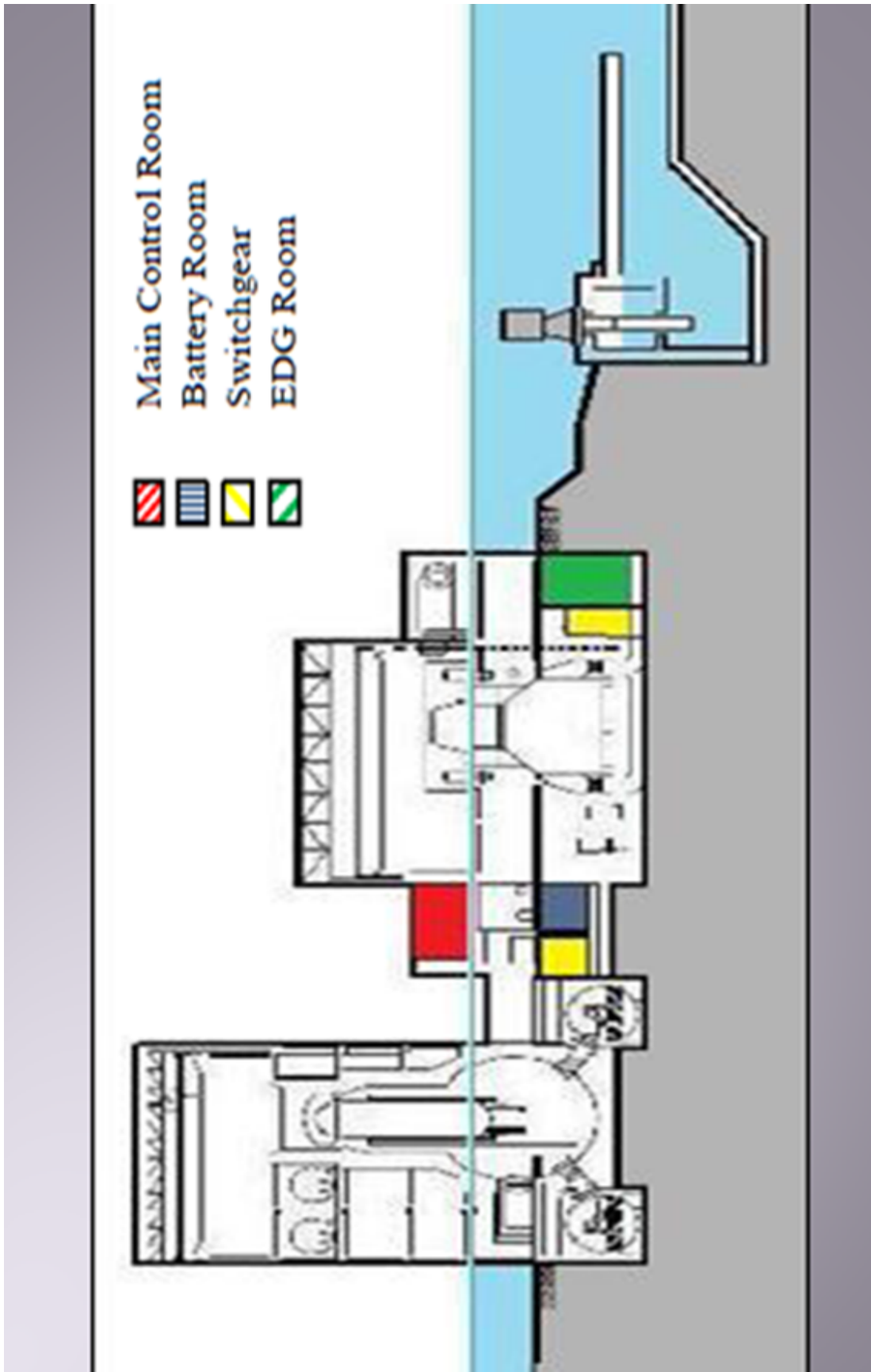
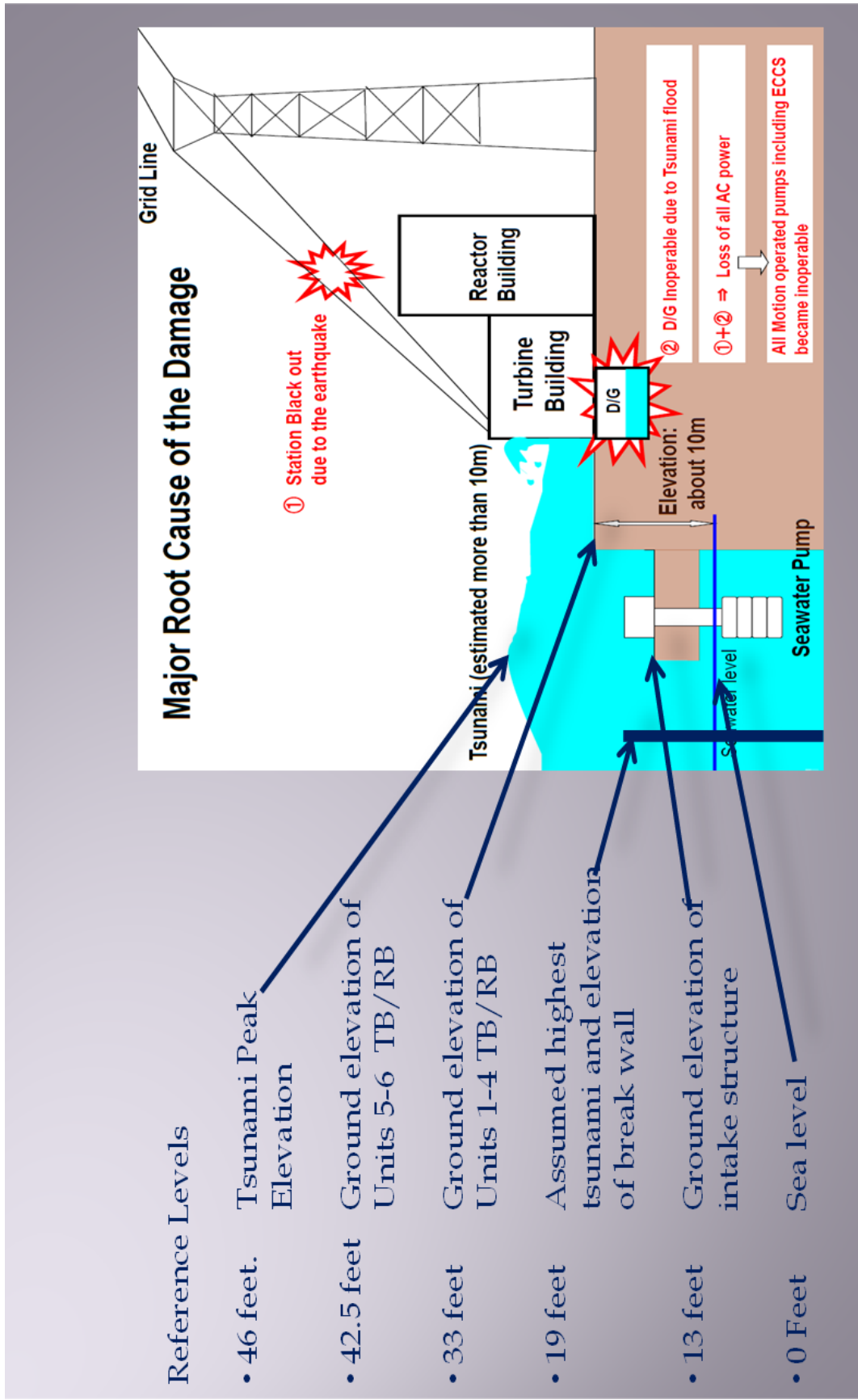


Figure 7.1-7 General Elevations and Inundation Level



Reference Levels

- 46 feet. Tsunami Peak Elevation
- 42.5 feet Ground elevation of Units 5-6 TB/RB
- 33 feet Ground elevation of Units 1-4 TB/RB
- 19 feet Assumed highest tsunami and elevation of break wall
- 13 feet Ground elevation of intake structure
- 0 Feet Sea level

Figure 7.1-8 Tsunami Elevations

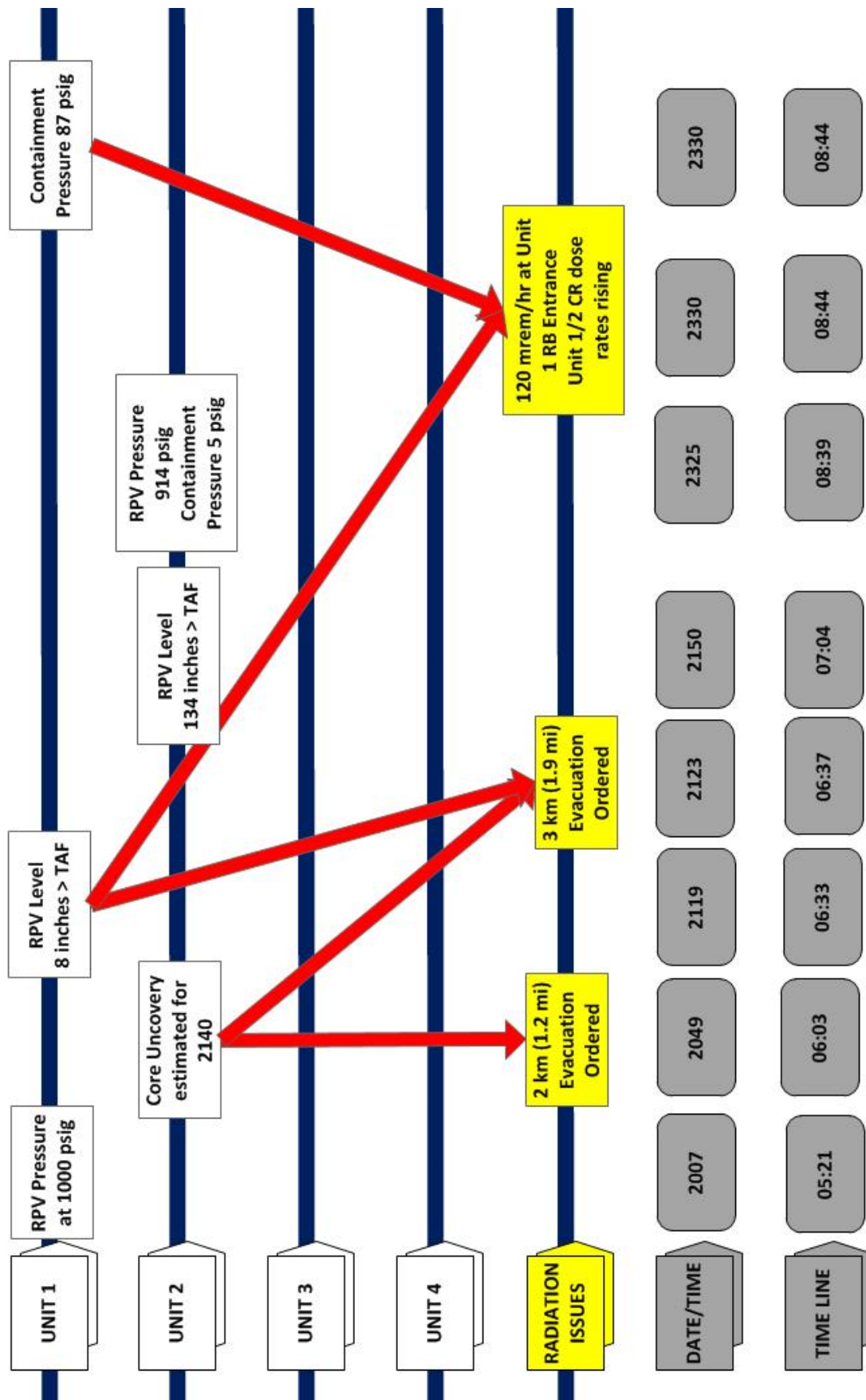


Figure 7.1-9b Fukushima Dai-ichi Event Timeline

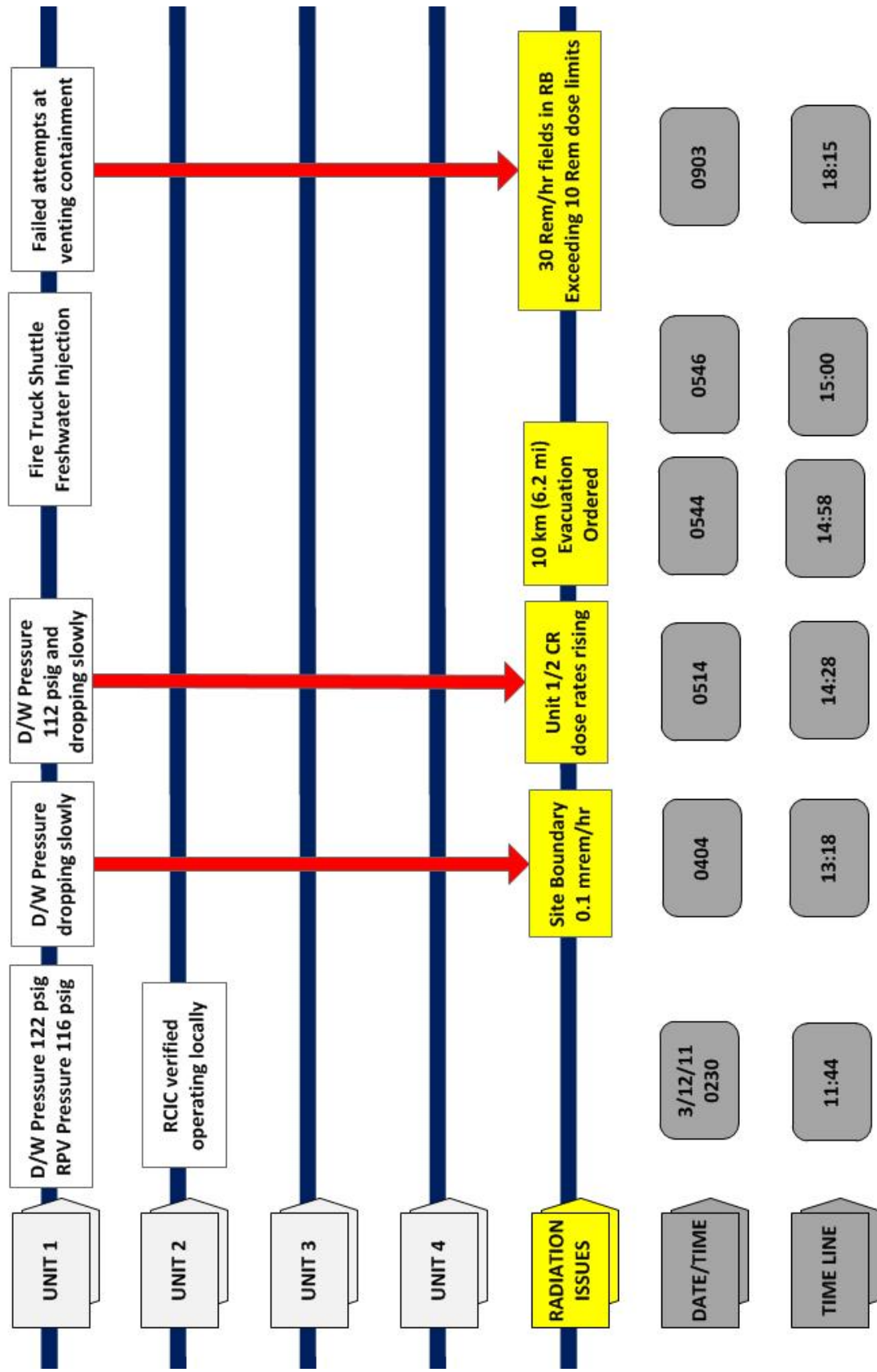


Figure 7.1-9c Fukushima Dai-ichi Event Timeline (cont.)

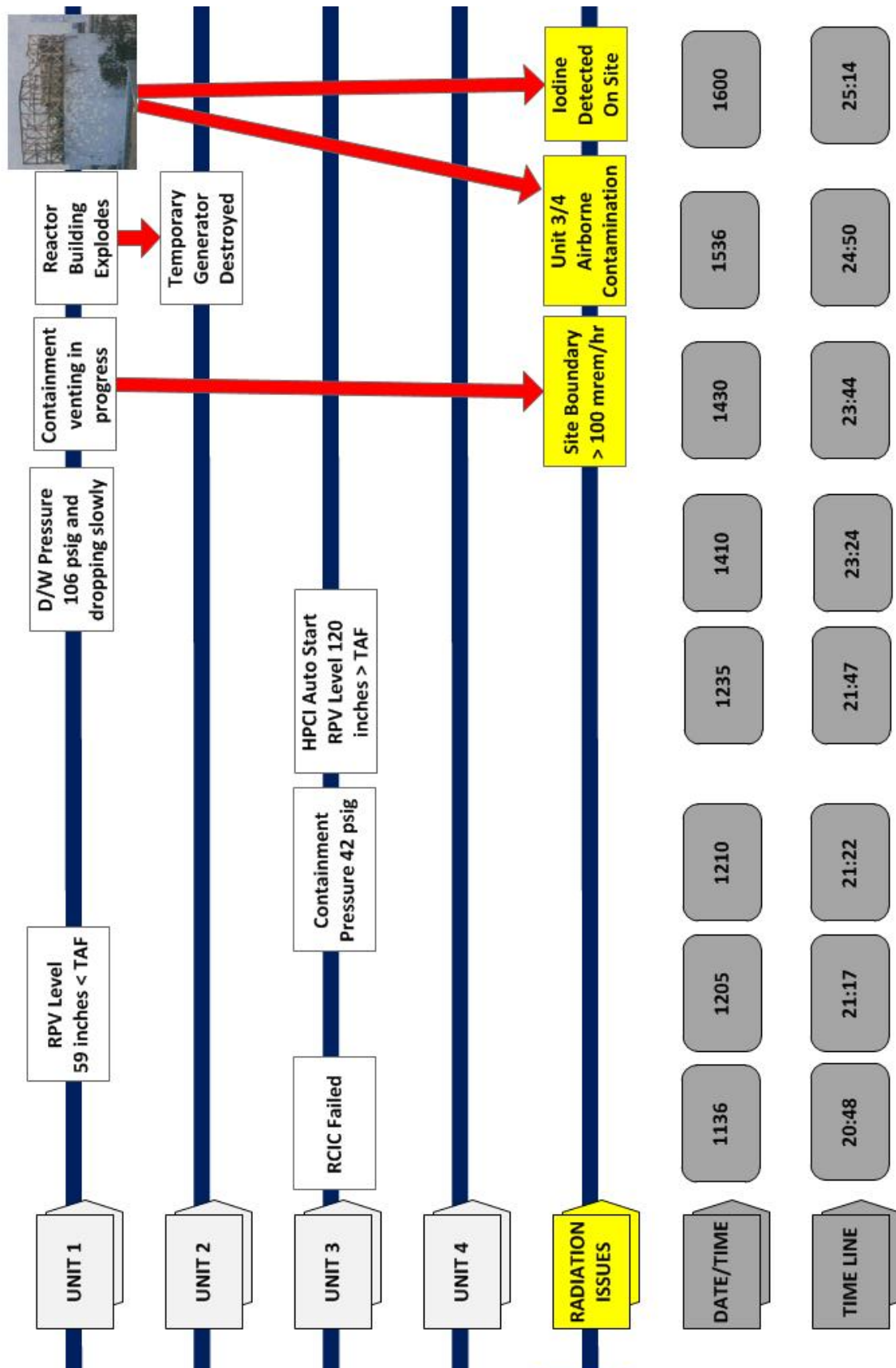


Figure 7.1-9d Fukushima Dai-ichi Event Timeline (cont.)

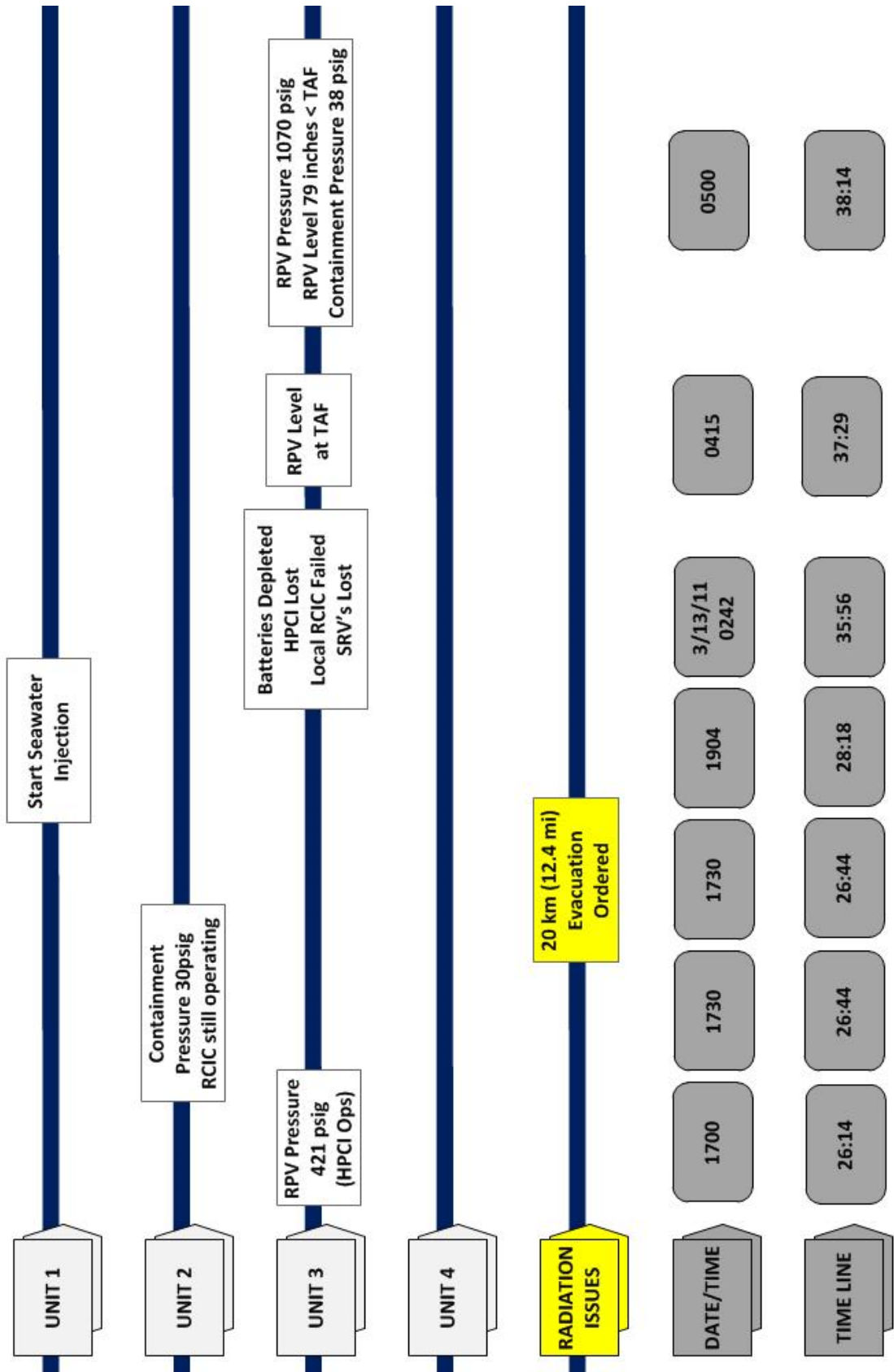


Figure 7.1-9e Fukushima Dai-ichi Event Timeline (cont.)

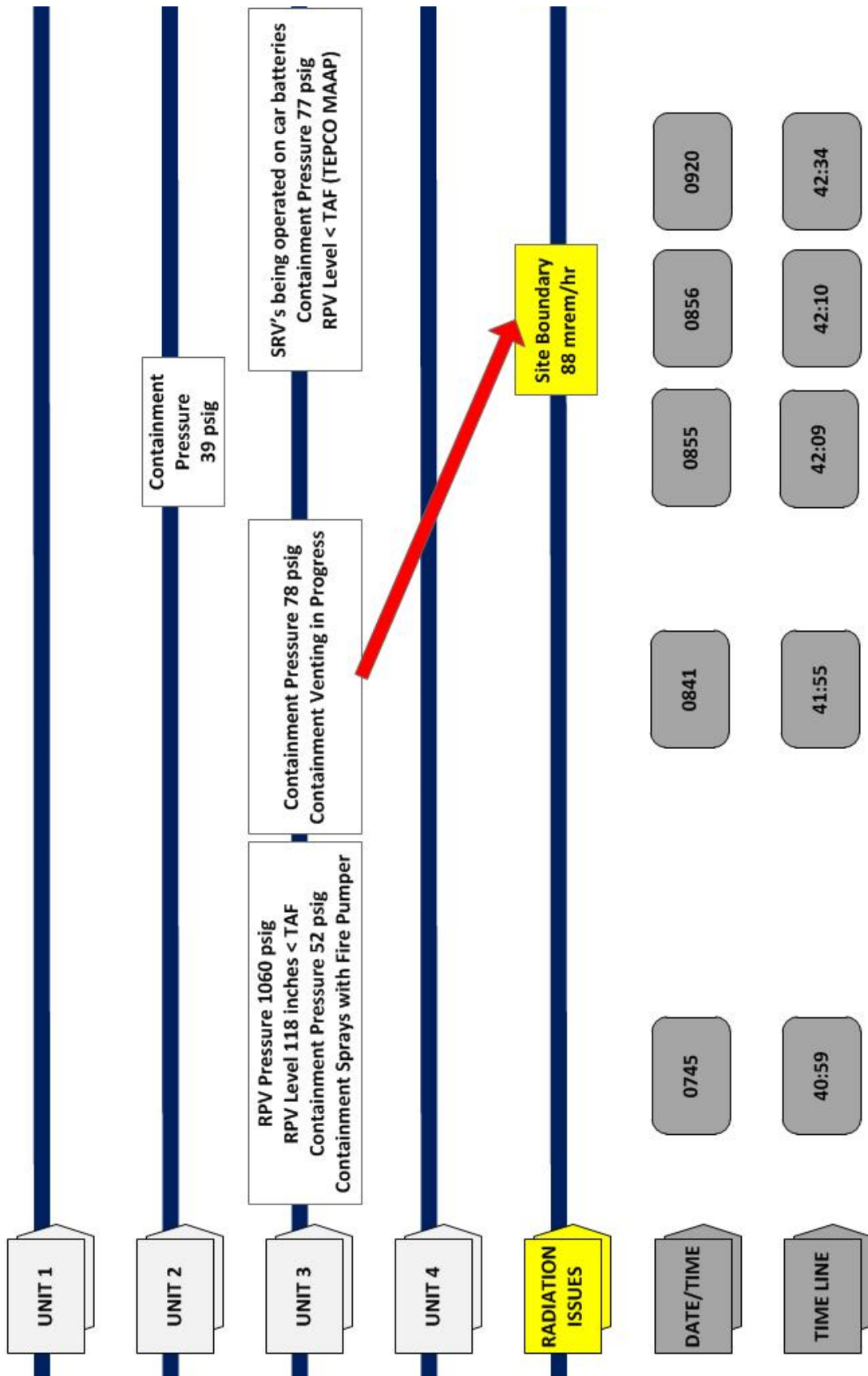


Figure 7.1-9f Fukushima Dai-ichi Event Timeline (cont.)

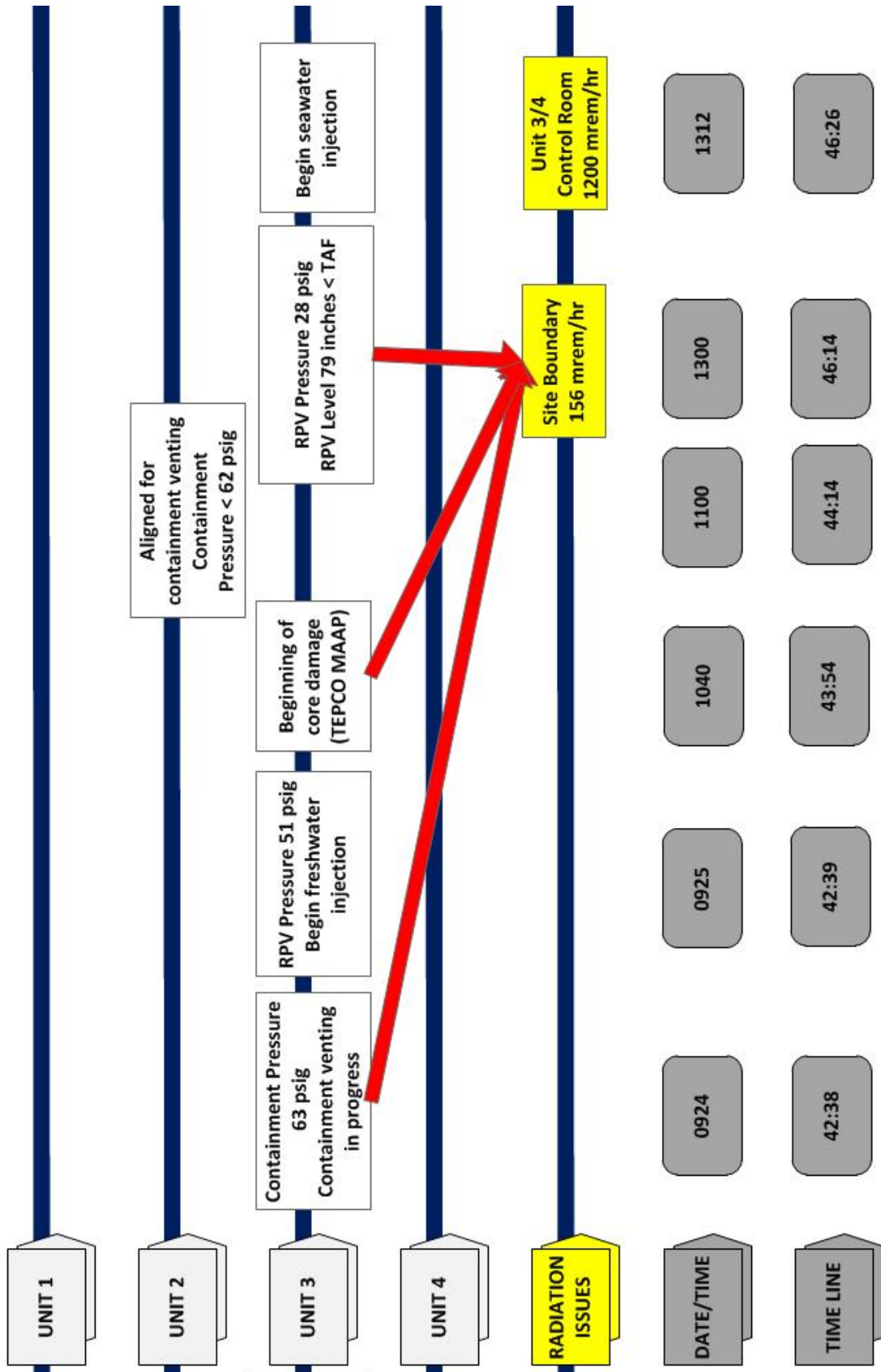


Figure 7.1-9g Fukushima Dai-ichi Event Timeline (cont.)

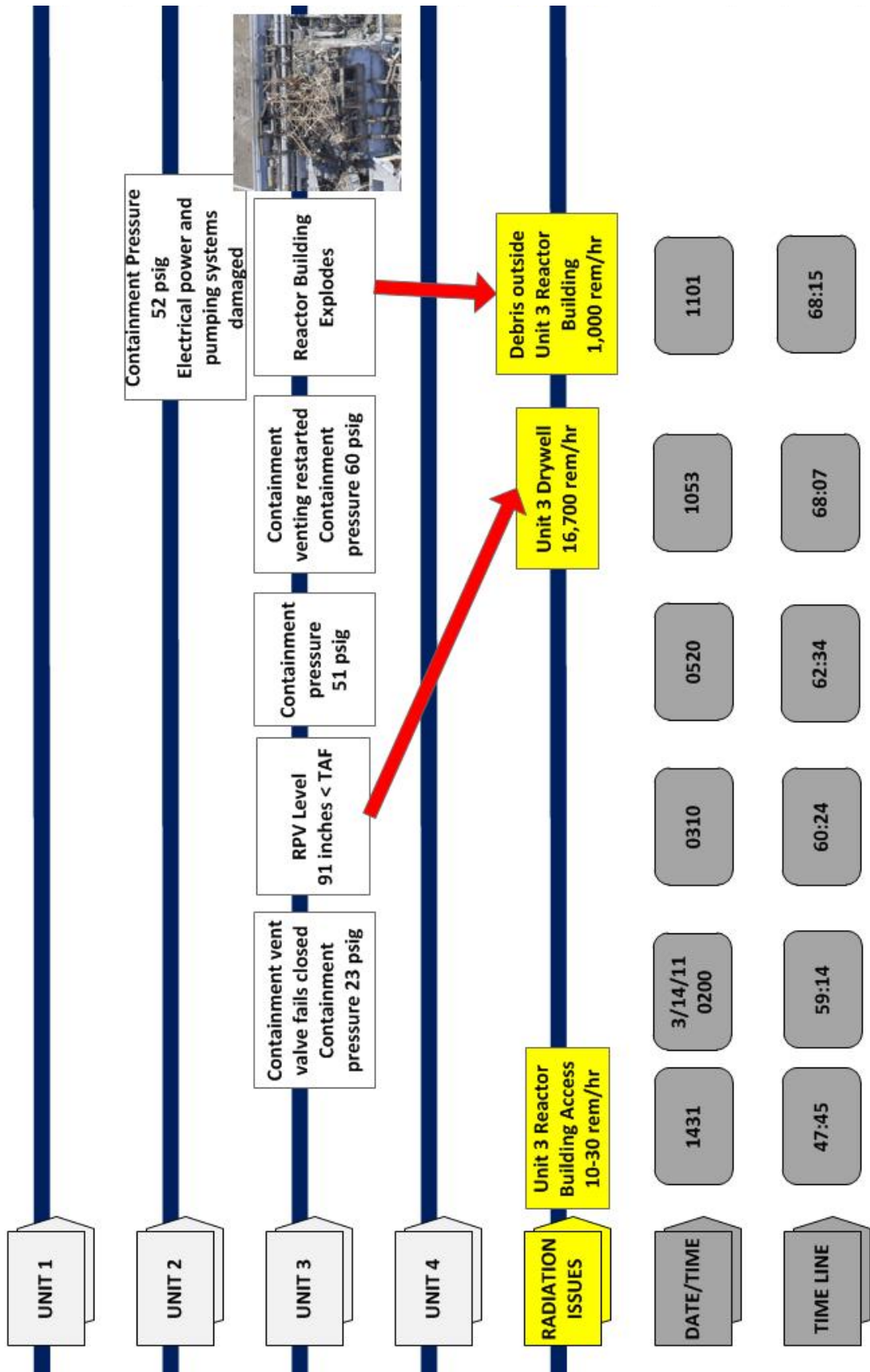


Figure 7.1-9h Fukushima Dai-ichi Event Timeline (cont.)

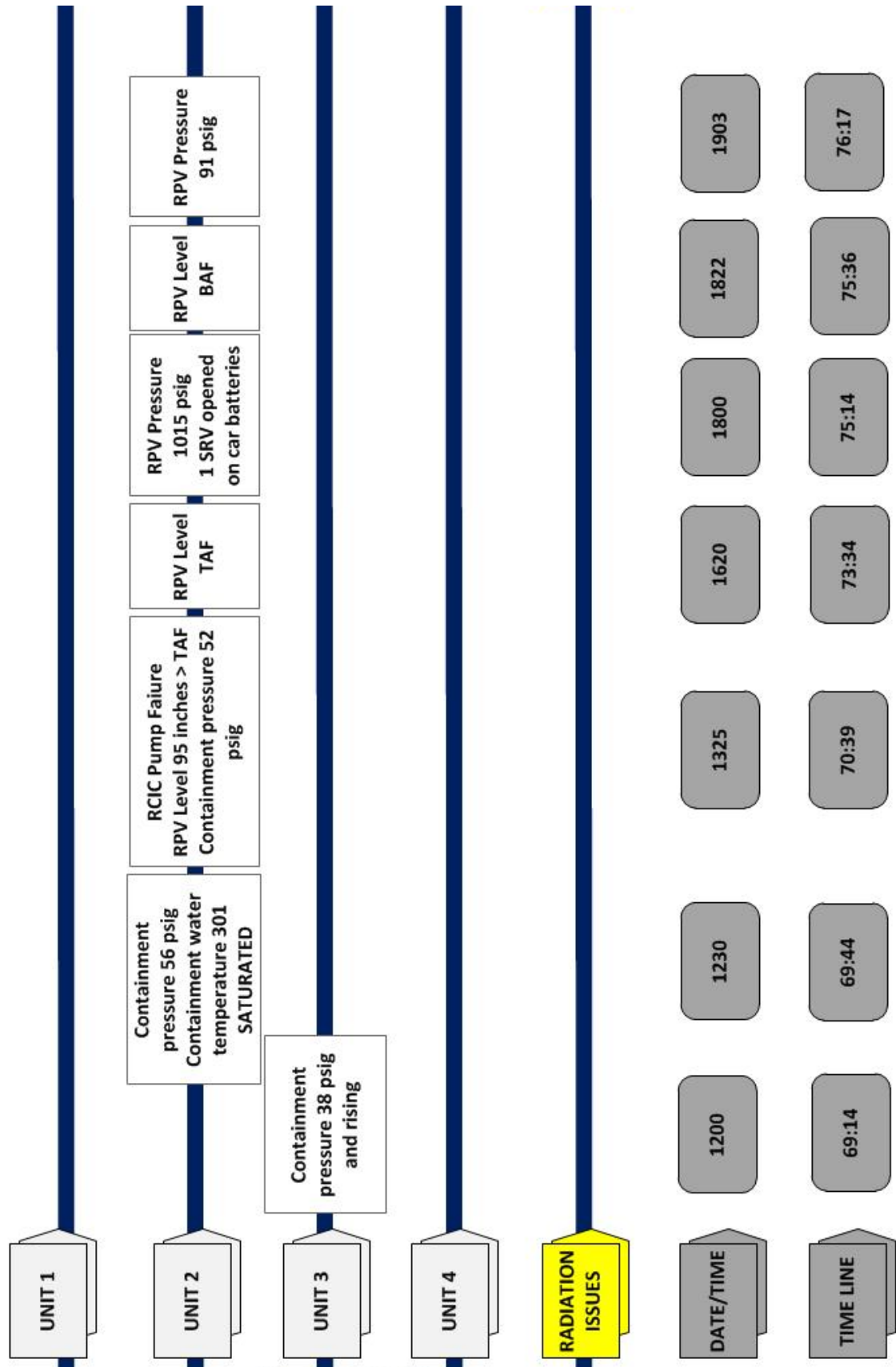


Figure 7.1-9i Fukushima Dai-ichi Event Timeline (cont.)

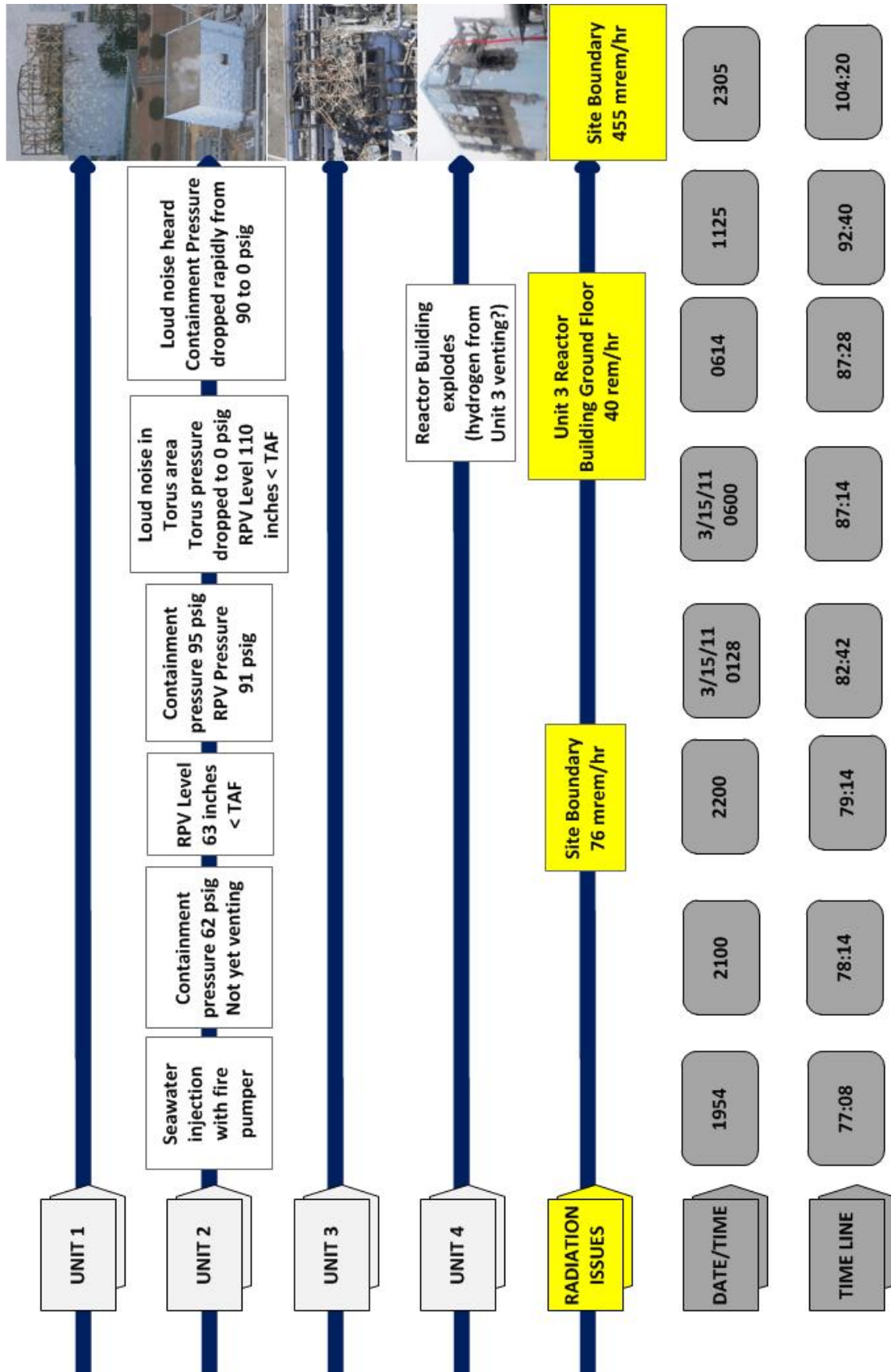


Figure 7.1-9j Fukushima Dai-ichi Event Timeline (cont.)

AOV'S

- Normally Closed
- Remote pneumatic operation from the Control Room
- No local control handwheel
- Fail Closed on loss of power or pneumatics

MOV

- Normally Closed
- Remote electric operation from the Control Room
- Local handwheel control
- Fail as is on loss of power

RUPTURE DISC

- Ruptures at 62 psig (Containment ASME Design Pressure)

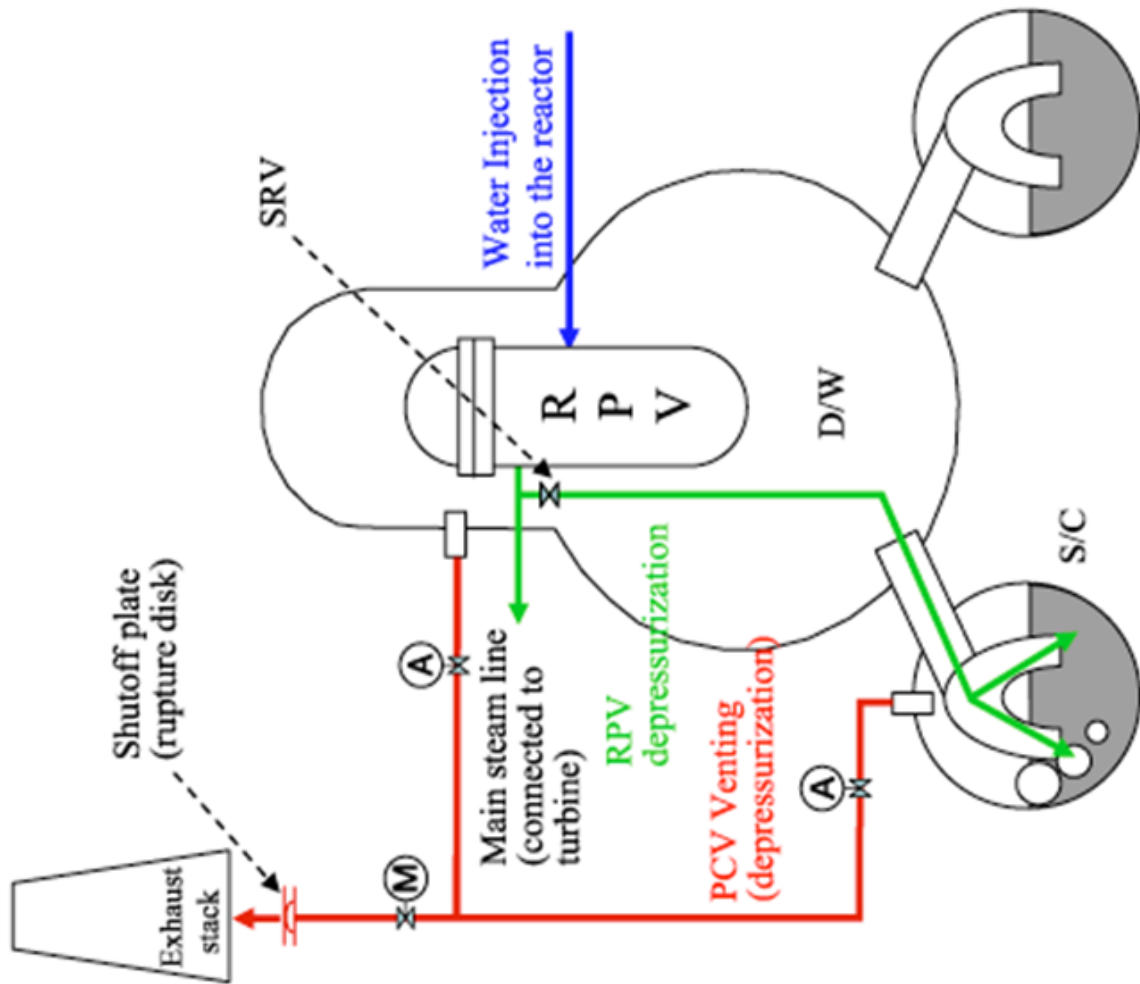


Figure 7.1-10 Fukushima Dai-ichi Containment Venting Flowpath

INES

THE INTERNATIONAL NUCLEAR AND RADIOLOGICAL EVENT SCALE

ACCIDENT INCIDENT

FUKUSHIMA DAI-ICHI, 2011
CHERNOBYL 1986

KYSHTYM, 1957

THREE MILE ISLAND, 1979

7 MAJOR ACCIDENT

6 SERIOUS ACCIDENT

5 ACCIDENT WITH WIDER CONSEQUENCES

4 ACCIDENT WITH LOCAL CONSEQUENCES

3 SERIOUS INCIDENT

2 INCIDENT

1 ANOMALY

**Below Scale / Level 0
NO SAFETY SIGNIFICANCE**

Figure 7.1-11 Fukushima Dai-ichi On The INES Scale

FUKUSHIMA DAI-ICHI HYDROGEN GENERATION

- Cladding exceeds ~2500°F
- $Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 + \text{heat}$
 - Exothermic reaction further heats the core
 - Generation of hydrogen
- Unit 1: 300-600kg^{**} **660-1322 pounds**
- Unit 2/3: 300-1000kg^{**} **660-2200 pounds**

Figure 7.1-12 Zirconium Water Reaction

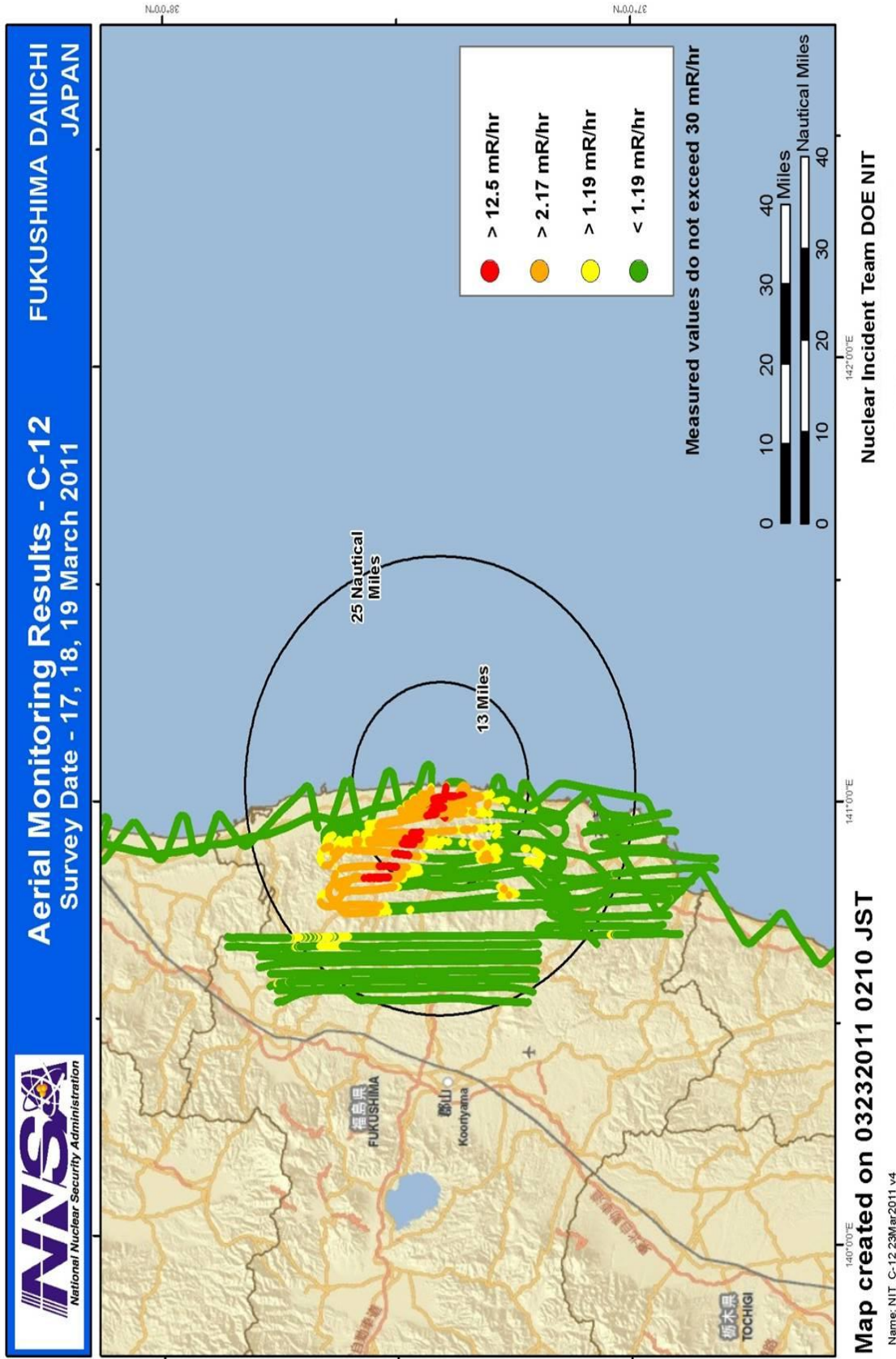


Figure 7.1-13 Aerial Monitoring Results

NRC EOC Update 5 May, 2011 1200 EDT

100 R/hr debris outside Unit 3 Reactor Building

UNIT	CORE DAMAGE ESTIMATE	CORE COOLING	PRIMARY CONTAINMENT	SECONDARY CONTAINMENT	DRYWELL RADIATION LEVELS	TORUS RADIATION LEVELS
UNIT 1	55% and uncovered	26 gpm fresh water via feedwater line	Damaged Leak rate est. at 3 cubic meters per hour	Severely damaged by hydrogen explosion	Unknown	114 R/hr 4 mR/hr at west gate
UNIT 2	35% and uncovered	30 gpm fresh water	Damage suspected	Damaged	2070 R/hr	40 R/hr 4 mR/hr at west gate
UNIT 3	30% and uncovered	31 gpm fresh water	Damage suspected	Damaged	1260 R/hr	50 R/hr
UNIT 4	In spent fuel pool	N/A	Relaxed for outage	Severely damaged by hydrogen explosion	N/A	N/A
UNIT 5	None	N/A	N/A	N/A	N/A	N/A
UNIT 6	None	N/A	N/A	N/A	N/A	N/A

Figure 7.1-14 NRC EOC Update 5-May-2011

UNIT 1



Figure 7.1-15 Unit 1 Photos

UNIT 3



Figure 7.1-16 Unit 3 Photos