APPENDIX A

Frequently Asked Questions: Section II, "Accident Analysis"

NOTE: Unless otherwise indicated, all dates in this appendix are for 2011.

Q.1. What are the associated safety systems of a boiling water reactor (BWR) system, and what do they do?

A.1. The following important safety systems can remove decay heat and/or add water to the reactor pressure vessel (RPV) without the need for alternating-current (AC) power:

- isolation condenser system, which is used in BWR/2s and some BWR/3s, including Fukushima Daiichi Unit 1
- reactor core isolation cooling (RCIC) system, which is used in BWR/4s, BWR/5s, BWR/6s, and advanced boiling water reactors (ABWRs)[1]
- high-pressure coolant injection (HPCI) system, which is used in BWR/3s and BWR/4s.

Summary descriptions of these systems are given below. Other safety systems, such as the core spray, residual heat removal, and containment cooling systems, rely on significant AC power for operation and were not available after the tsunami struck Fukushima Daiichi Units 1, 2, and 3. Portions of these systems were used to add water to the RPV via the fire protection system pumps and fire trucks. More detailed descriptions of the isolation condenser, RCIC, and HPCI systems can be found in Appendix F.

The purpose of the isolation condenser system is to remove decay heat and conserve reactor water inventory when the reactor becomes isolated from the turbine/condenser. The isolation condenser system consists of large heat exchangers located in the reactor building above the RPV, which condense steam produced by decay heat and return it to the reactor when the reactor becomes isolated from the main heat sink, the turbine/condenser. Normally, the steam supply line is open, and the condensate return line is closed. Upon receipt of a high-pressure signal, the condensate return valves are opened, and the system begins operation via natural circulation. The shell sides of the heat exchangers contain sufficient water for ~8 hours and then must be replenished.

The primary purpose of the RCIC system is to provide makeup water to the RPV when the RPV is isolated from the turbine/condenser. The RCIC system uses a steam-driven turbine-pump unit and maintains adequate water level in the RPV. The RCIC system is sized to keep up with decay heat inventory losses from the RPV, 90 to 180 m^3/hour (400 to 800 gallons/minute), depending on the reactor design power level. The primary water source is the condensate storage tank (CST), with backup from the suppression pool located inside the containment. Upon receipt of a low-RPV-water-level signal, the RCIC system is started and has sufficient flow to make up for decay heat boil-off as well as losses from small leaks in the primary system. Reactor pressure is maintained via safety and relief valve (SRV) operation, so in a station blackout the decay heat is transferred to the suppression pool.

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[1] In an ABWR, the RCIC system is also an emergency core cooling system (ECCS).

The HPCI system is part of the ECCS network and provides short-term water to the RPV for small-break loss-of-coolant accidents. It also has a steam-driven turbine-pump unit like the RCIC system but has about seven times the capacity. The HPCI system can also act as a backup for the RCIC system in isolation transients in which the main heat sink is lost. The HPCI system is also automatically started upon receiving a low-RPV-water-level signal and has the CST as its primary water source, backed up by the suppression pool. In this case also, the decay heat is transferred to the suppression pool via SRV operation.

Q.2. What is the current energy and nuclear power position in Japan?

A.2. Japan is one of the most energy-efficient countries in the world, with the lowest use of energy per unit of GDP. It is also the most efficient in terms of carbon as Japan produces more GDP per unit of carbon produced than any other country. Nuclear energy supplied ~30% of Japan's electricity from its 54 reactors. At any one time only about three-fourths of Japan's nuclear reactors are being used as the remaining reactors are closed for safety inspections. Japan's total electricity generation is ~158 GW electric (GWe) for a country of 130 million people. So, a 1-GWe reactor produces enough electricity for a city of roughly one million people. Thus, the six Fukushima Daiichi reactors produced enough electricity for approximately six million people. Fossil-fueled plants (oil, coal, and gas) have also been affected by the earthquake; in total, an additional 10 GWe is currently "under review" or under repair. In the short term, Japan needs more refined products to generate electricity.

The electric transmission grid in the western portion of Japan (Osaka and Nagoya) is quite different from that in the eastern portion of Japan (Tokyo Electric Power Company, Tohoku), due to the physical terrain. Only a maximum of 1 GWe of electrical power can be sent by transmission lines from west to east. Yet, any electrical energy disruption affects all of Japan.

Q.3. What is the design philosophy for natural disasters in the United States and Japan?

A.3. In the United States, the U.S. Nuclear Regulatory Commission (NRC) requires nuclear power plant (NPP) designs to withstand the effects of extreme natural phenomena including earthquakes and floods. The requirements, including General Design Criteria for licensing an NPP, are described in 10 CFR 50 (*Code of Federal Regulations*, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," NRC). These requirements include traditional engineering practices such as "safety margins." For these natural events, the historical extreme event in the vicinity of the NPP site (whether earthquake, flood, tornado, or hurricane) would be identified and become the design-basis event with additional safety margins. Practices such as these add an extra element of safety into design, construction, and operations.

The NRC specifically requires that safety-significant structures, systems, and components be designed to take into account the following:

- the most severe natural phenomena historically reported for the site and surrounding area, plus a margin for error to account for the limited accuracy of historical data
- appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena
- the importance of the safety functions to be performed.

For example, the U.S. Geological Survey Web site provides general information about earthquakes (http://earthquake.usgs.gov/learning/index.php). It should be noted that the design basis for extreme natural phenomena is under review now by Institute of Nuclear Power Operations requests and NRC orders.

The approach in Japan is supposed to be the same as in the United States. For reasons that are not completely clear, the earthquake design basis for the Fukushima Daiichi nuclear power station and the tsunami hazard were disconnected, and the tsunami hazard was not based on the historical maximum.

Q.4. How did the earthquake and tsunami compare to the design basis?

A.4. At the Fukushima Daiichi nuclear power station (NPS), horizontal accelerations during the earthquake exceeded the design basis at Units 2, 3, and 5 by up to 25%. There is no evidence that the earthquake caused any failures of safety equipment. Many on-site inspections are yet to be completed to verify this for all of the Fukushima Daiichi units; however, the pressure data measured for the primary containment vessels have indicated that the earthquake may have caused leak paths in at least two units. The Fukushima Daini NPS and the other eastern NPSs are also being inspected. The tsunami was less severe than the design basis, except at Fukushima Daiichi, where a height of 14 to 15 m was estimated versus a tsunami design-basis height of 5.7 m ["Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety—The Accident at TEPCO's Fukushima Nuclear Power Stations," Government of Japan (June 2011)]. More detailed information is available in Appendix E.

Q.5. What is meant by the term "meltdown," and have there been meltdowns before?

A.5. The term "meltdown" refers to a partial or complete melting of the nuclear fuel, which is normally a solid ceramic. About 98% of all fission products (i.e., radioactive materials) is retained in the fuel pellets (see Fig. 1) unless the fuel melts; the other 2% is retained as gas by the metal cladding in which the fuel pellets themselves are embedded. Therefore, it is important to keep the fuel "cool" under all modes of normal operation (power, shutdown, and refueling) as well as accidents. Since the fuel continues to generate "decay heat" long after the reactor is shut down, cooling must be provided essentially at all times, or meltdown may occur.

Figure 1. Unirradiated UO_2 fuel pellets before being inserted into the fuel cladding.

The most widely known reactor meltdown prior to that at the Fukushima Daiichi nuclear power station occurred during the Three Mile Island accident in the United States, when the reactor operators did not maintain a stable supply of water coolant to the nuclear fuel, and a partial meltdown occurred. This meltdown was contained in the reactor pressure vessel itself, and the containment remained intact. However, unlike the Fukushima Daiichi accident, the radioactivity releases outside of the nuclear power plant from that accident were negligible.

At Chernobyl, there was a complete meltdown of the fuel, which spread over several rooms and into the basemat of the reactor building. The reactor suffered a graphite fire, and an explosion occurred in the reactor core due to the large energy generation in the reactivity-induced accident (positive reactivity feedback). The reactor core was destroyed, and the radioactivity in the core was released along with steam, which destroyed the confinement building on top of the reactor. Radioactive steam entered the atmosphere as a plume, rising to a high elevation, traveling far and wide in Europe and depositing fission products in many countries.

Q.6. What is the regulatory structure in Japan, and how is it different from other nations?

A.6. The current nuclear regulatory agency, the Nuclear and Industrial Safety Agency (NISA), reports to the Ministry of Economy, Trade and Industry (METI), which has been seen as giving NISA an insufficient level of independence to properly regulate the industry. There may be a potential conflict of interest for METI as both promoter and regulator of nuclear energy. A complete reorganization of nuclear safety regulations and practices is seen as the essential foundation on which to rebuild public trust.

The NISA regulatory body is currently overseen by two other bodies: the Nuclear Safety Commission (NSC), which sets overall safety policy, and the Atomic Energy Commission, which sets nuclear power and research policy. Both of these are part of the Cabinet Office. Neither of these bodies has direct authority over nuclear power plant regulation but are advisory. Japanese officials have announced that a new regulatory structure for its nuclear power industry will be developed with necessary centralized competence exercised by the Ministry of the Environment. The new arrangement will combine NISA with NSC to form a centralized safety agency with the needed functionality and independence. See Fig. 2.

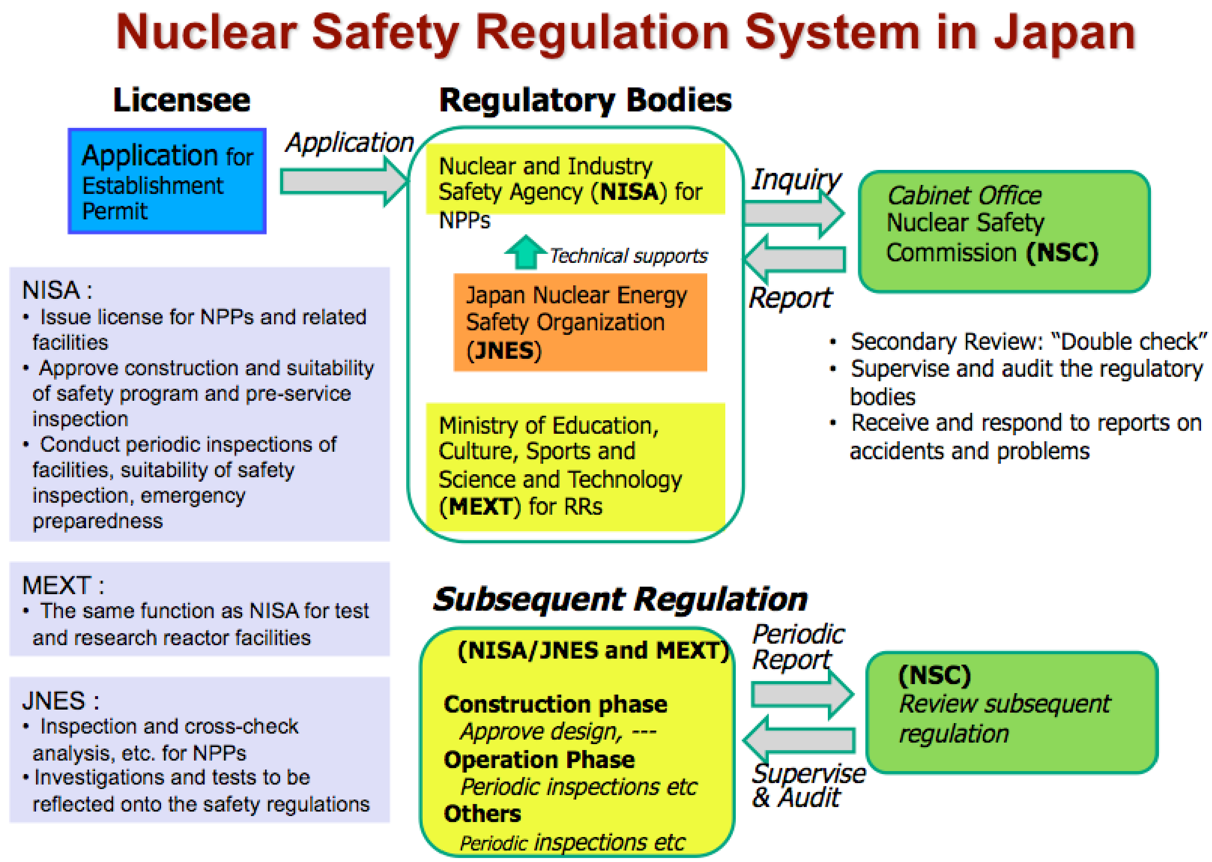

Figure 2. Japan's nuclear safety regulation system.

Q.7. Are nuclear power plants (NPPs) in Japan required to have simulators?

A.7. Nuclear power plants in Japan are not required to have plant-specific (replica) simulators. However, compact simulators (not full scale) are installed on each site. Typically, NPP operators are trained at the BWR Operator Training Center (BTC) (http://www.btc.co.jp/index_e.html) and/or at the Nuclear Power Training Center Ltd (NTC), PWR Power Plants (http://www.jntc.co.jp/en/plants/index.html).

In the opinion of the Committee, the issue is not to train on replica-style simulators but rather to have simulators that can predict the overall behavior of the essential NPP subsystems (i.e., reactor pressure vessel, suppression pool, and containment) for beyond-design-basis conditions, especially before substantial core damage occurs, so that core damage can actually be prevented. High fidelity is not required here; rough estimates for time to fuel exposure, time to start Zircaloy oxidation, time to melt pool formation in the core region, time to reach suppression pool saturation temperature, time to reach containment design pressure, etc., would be very valuable information as the operators and the technical support center staff identify and develop the most effective strategy to manage a prolonged station blackout or other beyond-design-basis sequence. The simulators would need to be able to model the effects of the success or failure of the backup direct-current, backup pressurized nitrogen bottles to operate safety and relief valves, external systems (e.g., fire pump), and containment venting, among other systems.

Q.8. Can it be confirmed that Japan does not have a resident inspector program and, as such, that Japan's regulator must rely exclusively on the licensee to obtain critical information during an accident?

A.8. Japan established a nuclear safety inspector program in 2000, under the authority of the Nuclear and Industrial Safety Agency (NISA). The inspectors are stationed at 21 offices located near various nuclear facilities throughout Japan. Fukushima Daiichi has one such office, ~5 km west of the nuclear power station (NPS). The inspectors perform full safety inspections every quarter, as well as "walkdowns" more frequently. Upon receiving notification from a licensee that an accident is in progress, the Nuclear Safety Inspector Office notifies NISA, surveys the site, and starts an investigation on the causes of the accident. These offices also host the director of nuclear disaster prevention. The director of nuclear disaster prevention advises the licensee on off-site emergency response matters and verifies installation, maintenance, and readiness of equipment and procedures for off-site emergency response. The director of nuclear disaster prevention also advises local governments on disaster prevention plans and, during accidents, disaster prevention implementation.

On March 11, there were eight members of the NISA Nuclear Safety Inspector Office performing routine inspections at the Fukushima Daiichi NPS. When the accident started, three inspectors (including the office manager) returned to the off-site office, while five inspectors stayed at the NPS to collect information. The Nuclear Safety Inspector Office established the Local Nuclear Emergency Preparedness Office to direct the off-site emergency response efforts (including evacuation); however, for the first 3 days of the accident, its effectiveness was hampered by the lack of reliable communication and poor people mobility due to the blackout and devastation caused by the earthquake and tsunami. On March 15, the Local Nuclear Emergency Preparedness Office was moved to the Fukushima Prefectural Building.

Q.9. What is the state of practice (roles, responsibilities, and training regimen) of reactor operators as well as operational staff in Japan at an operating nuclear power plant?

A.9. The control room roles and responsibilities in Japan are similar to those in the United States. They have a similar level of staffing and organization. However, the level of operator training with respect to severe accident management in Japan lacks the level seen in U.S. facilities.

Q.10. What is the state of practice (roles, responsibilities, and training) of accident response and crisis management at the (government) institutional level in Japan?

A.10. There are weaknesses due to segmented regulatory existence and responsibilities as well as lack of routine joint emergency drills for such crises. There are also weaknesses in the communication technology and training. There are examples where data were transmitted to the government by the Tokyo Electric Power Company, but because of lack of understanding and timely actions, the off-site radiological information was not communicated in a timely manner to the public.

There was an off-site dose monitoring system, with software codes to predict and monitor the radioactive plume path, that was functioning at the time of the containment venting and rupture events. Responsible government officials were unaware of the existence of the system. People were using the system to track the radioactive plume leaving the site and passing over civilian areas, but the people monitoring the plume path failed to report what they saw because nobody asked them for information. The fact that the responsible government agencies were unaware of the existence of the postaccident radioactive plume monitoring system indicates a significant training/knowledge deficiency issue.

Q.11. What is the accident sequence for each unit at the Fukushima Daiichi and Fukushima Daini nuclear power stations (NPSs)?

A.11. When this question was posed, it was thought to be important to track what had happened at both the Fukushima Daiichi and Fukushima Daini NPSs for lessons learned. Since then, our understanding has improved to where it is the Fukushima Daiichi Units 1, 2, and 3 timelines that are the most important, for the following reasons:

- The fuel located in spent-fuel pools, and in particular for Fukushima Daiichi Unit 4, was not damaged as originally believed.
- Fukushima Daiichi Units 5 and 6 survived without core damage because they were located slightly apart from Units 1 through 4 and at a slightly higher elevation that allowed one (air-cooled) emergency diesel generator to survive and provide electricity to prevent core damage.
- The damage at the Fukushima Daini NPS was limited to temporary loss of the ultimate heat sink, and no core damage occurred.

The timelines for the Fukushima Daiichi NPS Units 1, 2, and 3 are contained in Appendix H.

Q.12. What is the command and control structure in Japan as compared to the United States?

A.12. The control room command and control structure is the same in Japan as in the United States. There are supervisors who direct operation of the controls and in-field work. There is also a shift manager equivalent. There is also a site vice president who, during this accident, was providing direction to vent and add seawater. There was a loss of total communication between the control rooms and site workers with only one line from each control room to the site emergency center. Note that from the time of the accident to 2400 hours (midnight) (Japan time) of March 11, there had been a total of 81 earthquakes of magnitude 5.0 or greater and multiple tsunami warnings that impacted workers.

In the United States, the shift manager is empowered to take actions as necessary to protect the containment by venting without waiting for specific authorization from company senior officials or government officials.

Q.13. What happened to the Fukushima Daiichi Units 1 through 4 spent-fuel pools (SFPs) and the common SFP?

A.13. When off-site power and all emergency diesel generators (EDGs) at the site except for one were lost because of the earthquake and tsunami, normal cooling of the SFPs was lost. Loss of cooling could lead to boiling of the SFP water. The time before the SFP water level drops sufficiently to result in fuel overheating depends on the amount of water in the SFP as well as the heat load of the used fuel. In the absence of a leak in the SFP (or "sloshing" caused by the earthquake), this time could range from several to many days, depending on the decay heat generation in the various SFPs.

The available EDG allowed cooling to be restored to the Units 5 and 6 SFPs before the temperature of these SFPs increased significantly. Power was also restored to the common SFP cooling system before its temperature increased significantly.

On March 12, a hydrogen explosion damaged the upper portion of the structure surrounding the refueling bay on Unit 1. While this explosion may have resulted in material falling into the SFP, there is no evidence that damage to the fuel occurred. On March 14, a hydrogen explosion damaged the structure housing the refueling bay on Unit 3.

Because of the relatively high decay heat associated with the fuel in the Unit 4 SFP [all fuel had been removed from the Unit 4 reactor pressure vessel (RPV) in December 2010], special concern was focused on this SFP. When the refueling floor containment structure was severely damaged because of an apparent hydrogen explosion early in the morning of March 15, this concern was intensified. Initially, since the RPV was defueled, the source of the hydrogen was thought to be the stored spent fuel, implying that SFP inventory had to have been lost early in the scenario. It was determined later that the source of the hydrogen was likely to be from Unit 3 via a pathway to the Unit 4 refueling floor, likely through a shared vent.

Beginning on March 17, various means to add water to the SFPs for Units 1 through 4 were used—helicopter, water cannon, fire trucks, and concrete pumpers—with varying degrees of success; eventually, the normal makeup piping was used. Alternative SFP water cooling systems were installed, and as of September the SFP temperatures in Units 1, 2, and 3 were <35ºC (95ºF) and <40ºC (104ºF) in Unit 4. Typical SFP temperatures for operating boiling water reactors are ~100ºF.

A video recording of the Unit 4 SFP was released on May 9. This video recording showed no evidence of extensive damage. In fact, the fuel racks appeared to be intact with little debris visible in the SFP. A video recording made in the Unit 3 SFP was released on June 16 that showed debris from the containment structure that had fallen into the SFP. It was not possible to confirm the structural integrity of the fuel racks using the video recording, but it is likely that no damage has occurred to the spent fuel.

A more complete discussion of the Fukushima Daiichi SFPs is given in Appendix G.

Q.14. Why did the Fukushima Daini and Onagawa nuclear power plants (NPPs) survive the earthquake and tsunami?

A.14. Basically, because the maximum tsunami height was less than the design basis, emergency diesel generators remained operative. In addition, at these NPPs, one of the off-site power lines was not lost after the earthquake. There was some flooding of pumps used to remove heat, but the motors were repaired, and cold shutdown was achieved within a couple of days. See Appendix E for more detailed information about the earthquake and tsunami.

Q.15. What were the emergency procedure guidelines (EPGs)/severe accident management guidelines (SAMGs) for the Japanese nuclear power plants (NPPs), and how are they different from those in the United States?

The Japanese EPGs and SAMGs are functionally the same as those for U.S. NPPs.

Q.15a. What is the practice for accident sequence analysis when the radiation level is high and thus hinders the operational response?

A.15a. At the Fukushima Daiichi nuclear power station (NPS), operators were authorized to exceed normal dose limits up to the emergency worker limit of 250 mSv. Field operators were dispatched to manually open vent valves and check local indications.

Q.15.b. What is the practice for water-level measurement during normal operations? What is anticipated under transient and accident conditions?

A.15.b. Normally, water level is indicated in the control room. Generally, the same indications are also available under transient and accident conditions. During the accident at the Fukushima Daiichi NPS, field operators were dispatched to check local indications at Units 1 and 2 because the direct-current (DC) power supply to instrumentation was lost, so there was no control room indication.

Q.15.c. What is the practice for valve functionality (open, closed, or percent open) during normal operations? What is the anticipated functionality under transient and accident conditions?

A.15.c. Water delivery systems not normally needed, or only occasionally needed, during normal power operation have the suction side isolation valves open and the discharge side closed. These valves are usually motor operated; some are alternating-current (AC) powered, and some are DC powered. The normal configuration is for the valves outside containment to be DC powered and the valves inside containment to be AC powered. If the NPP conditions require that the system be started in response to a transient or accident, then power is delivered to the appropriate valve(s) to properly position them. If the valve power source is lost, then the valves will fail as-is, but it is possible to manually open or close these valves locally if they are outside of containment.

Examples of the valving arrangements for the isolation condenser, reactor core isolation cooling, and high-pressure coolant injection systems are given in Appendix F. Because these systems have valves inside and outside of containment, they require isolation logic to isolate the system under certain situations such as a system breach. Because of this logic, some systems will isolate upon a loss of power to the system logic. The affected system valves will close if they are open and have valve power to close.

Air (nitrogen) handling systems and depressurization systems typically use air-operated valves that require either the NPP high-pressure air or nitrogen supply or pressurized bottles to supply the motive force plus AC or DC power to operate the controlling solenoid valves. Depending on the valve design, these valves may fail as-is or fail closed upon loss of air (nitrogen) pressure. Opening these valves locally may require portable compressors and portable batteries.

Q.16. Why were there delays in (a) injecting seawater, (b) venting the containments, and (c) reopening the Fukushima Daiichi Unit 1 isolation condenser lines after the tsunami hit?

A.16.a. Seawater injection delay. In order for water to be injected by emergency equipment (on-site diesel-driven fire water pumps or fire truck pumps), the reactor pressure vessel (RPV) pressure has to be decreased to as low as 0.7 MPa (85 psig). At Unit 1, because of complete loss of direct-current (DC) power, including instrumentation power, using safety and relief valves (SRVs) to depressurize the RPV was problematic, and depressurization probably occurred because of leaks in the RPV after core melt, or possibly because of a stuck-open SRV. At Unit 2, attempts to depressurize the RPV via SRVs did not start until after the reactor core isolation cooling system failed. As it took several hours to do this because of equipment problems, core melt occurred before fire pump injection. At Unit 3, the high-pressure coolant injection (HPCI) system had done a good job of reducing RPV pressure before its failure, but the RPV pressure increased again, after the HPCI system failed. As with Unit 2, because of equipment problems it took several hours to reduce RPV pressure via SRVs, and in the meantime, core melt occurred.

A.16.b. Containment venting delay. By emergency procedure, the local population has to be notified, and evacuation orders have to be given and accomplished. It took some 18 hours after the start of the event for this to be accomplished, and the first venting started at Unit 1. Venting at Units 1 and 3 was not started until 2 days into the event because containment pressures were lower and no core damage had yet occurred. The manual operation of the valves, in the dark and in a high-radiation environment, was quite difficult.

A.16.c. Reopening Unit 1 isolation condenser lines delay. The loss of DC power required manual opening of the isolation valves. This took a couple of hours and was only partially successful. See also **A.18**.

Q.17. Have all three containments (Unit 1, Unit 2, and Unit 3) failed at the Fukushima Daiichi nuclear power station? If so, what were the failure mechanisms?

A.17. The containment at Unit 1 reached a maximum pressure of 0.84 MPa (107 psig), which is well in excess of the design pressure of 0.53 MPa (62 psig), and stayed there for 7 to 8 hours, meaning there had to be a leak, at least at this pressure. There is also a suspicion that at least some of the hydrogen that built up in the refueling floor area of the reactor building came from leaking gaskets in the drywell head cover. In postaccident recovery operations, the Tokyo Electric Power Company found it not possible to flood containment above the downcomer joint with the drywell, which is another indication of a breach in containment integrity. The failure mechanism is probably overpressure.

At Unit 2 it took >36 hours for containment pressure to reach 0.315 MPa (31 psig), but if there were no leaks, the containment should have reached the design pressure of 0.53 MPa (62 psig) in ~16 hours. Further confirmation of a breach in the containment boundary comes from the postulated hydrogen explosion (certainly a loud sound) that occurred in the torus room. The failure mechanism is probably the earthquake; it is interesting to note that Unit 2 recorded seismic accelerations significantly above the design basis.

As with Unit 2, the Unit 3 containment pressures in the range of 0.4 to 0.5 MPa were not reached until ~40 hours into the event, implying some containment leakage was happening. The failure mechanism is probably the earthquake; like Unit 2, Unit 3 recorded seismic accelerations significantly above the design basis. For further information, refer to Appendix E.

Q.18. Why were workers not able to refill the tanks of the isolation condenser system at Fukushima Daiichi Unit 1?

A.18. There are two isolation condenser trains at Unit 1, and the total stored water on the shell side is at least 8 hours of decay heat boil-off. According to the "Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety—The Accident at TEPCO's Fukushima Nuclear Power Stations," Government of Japan (June 2011), both trains were deliberately shut down in the time period between the earthquake and the arrival of the tsunami because with both isolation condenser trains working, the reactor pressure vessel temperature decrease was exceeding the prescribed technical specification cooldown rate of 55°C/hour (100°F/hour). Train A was restarted and stopped several times but was stopped at the time the tsunami hit. Train A was attempted to be started by manually opening the isolation valves, but this was apparently only partially successful. Water was supplied to the shell side of train A, but this did not happen until after core melt.

Q.19. Why did the reactor core isolation cooling (RCIC) system fail at the Fukushima Daiichi nuclear power station?

A.19. Unit 1 does not have an RCIC system, but Units 2 and 3 do. At Unit 2 the RCIC system actually did run for 70 hours. This might be attributed to good planning and load shedding by the nuclear power plant (NPP) operators. The initial water source was the condensate storage tank, but it was switched to the suppression pool after ~14 hours because of high water level in the suppression pool. There are several reasons why the RCIC system might have failed, including high water temperature (net positive suction head and seal cooling limits), high room temperature, and loss of battery power. Most probabilistic risk assessments conclude that the RCIC system will probably fail in ~8 hours because of one of these causes. This one ran much longer than the NPP had a right to expect. Operators had installed, in parallel with RCIC system operation, an alternative water injection line that was destroyed by the explosion in Unit 3.

At Unit 3 the RCIC system ran for 20 hours before failure, but then the high-pressure coolant injection system (also steam driven) ran for an additional 15 hours before stopping also.

Q.20. Did hydrogen leak into the buildings at the Fukushima Daiichi nuclear power station (NPS) through a gap in the containment vessel head (due to high containment pressure) or through weak vent ductwork?

A.20. The hard vent piping and instrumentation diagrams described in the "Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety—The Accident at TEPCO's Fukushima Nuclear Power Stations," Government of Japan (June 2011) show at least one closed direct-current motor operated valve and one air (nitrogen) operated valve (AOV) normally closed in series with a rupture disk that cannot be bypassed. Since the containment cannot be vented until the pressure is in excess of the rupture disk setpoint, a problem with the configuration is indicated, which will be the subject of regulatory scrutiny in both Japan and the United States. Current U.S. configurations have been approved via the 10 CFR 50.59 [*Code of Federal Regulations*, Title 10, "Energy," Part 50, "Domestic Licensing of Production and Utilization Facilities," Sec. 59, "Changes, Tests, and Experiments," U.S. Nuclear Regulatory Commission (NRC)] process. Some NPSs have installed dedicated pathways; some use existing pathways, such as the off-gas system. In some, the rupture disk can be broken at any pressure. Most use AOVs, which could be subject to the same difficulties as faced by the operators at the Fukushima Daiichi NPS. The advanced boiling water reactor (ABWR) (NRC, certified but not yet constructed in the United States) has AOVs only (plus a rupture disk), normally open; this configuration has been approved by the NRC.

Since it takes a hydrogen concentration of >5% to have a flammable mixture and >10% to form a detonable mixture with air, it is clear that a significant fraction of the generated hydrogen during core melt succeeded in escaping the containment. In the case of Units 1 and 3, at least some of the hydrogen that exploded in the reactor building came from containment leaks, although it has been speculated that some of the hydrogen may have also come from backflow through the standby gas treatment system that also terminates at the same plant stack as the hard-piped vent.

At Unit 4, the source of hydrogen was not from fuel melting in the spent-fuel pool as first feared but was due to backflow from the common ductwork that Units 3 and 4 share at the plant stack.

Q.21. We know the earthquake did not cause massive damage to the Fukushima Daiichi nuclear power plant (NPP) structures, but little cracks here and there may have had major safety consequences. For example, were the containment vent pipes damaged by the earthquake? Were the reactor core isolation cooling (RCIC) system steam turbine–driven pumps and pump supports damaged by the earthquake?

A.21. It is unlikely that the earthquake itself caused safety-significant damage to the reactor pressure vessels (RPVs) and safety systems. Note the following:

- The data show no abnormal pressure drop in the RPVs after the earthquake; thus, a major leak in the reactor system should be excluded. Small leaks, such as a recirculation pump seal leak, cannot be ruled out, though they would be readily made up for by the RCIC system, at least at Units 2 and 3.
- The Tokyo Electric Power Company and the "Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety—The Accident at TEPCO's Fukushima Nuclear Power Stations," Government of Japan (June 2011) indicate that all safety systems at the NPP operated normally prior to the tsunami.
- It was initially speculated that the isolation condenser lines at Unit 1 were closed prior to the tsunami for fear of a leak in the isolation condenser lines caused by the earthquake. However, later reports indicate that the isolation condenser lines were manually closed, according to procedures, because the cooldown rate was high.

There is indirect evidence of mechanical damage to the containments:

- About 13 hours into the event, the containment pressure at Unit 1 started to drop, and high radiation levels were detected at the site. That containment venting was not attempted until 18 hours into the event suggests that leaks through the containment boundary had occurred, at least at high containment pressure.
- At Unit 2 the containment pressure grew much more slowly than expected, i.e., 0.315 MPa (30 psig) in 36 hours versus the ~0.5 MPa (62 psig) design pressure expected in ~16 hours,

suggesting a leak in the containment boundary. Further confirmation of a breach in the containment boundary at Unit 2 comes from the loud sound that occurred in the torus room, originally postulated to be due to a hydrogen explosion, but now thought to be a structural failure.

- Unit 3 containment pressures in the range of 0.4 to 0.5 MPa (45 to 60 psig) were not reached until ~40 hours into the event, which again suggests that some containment leakage had occurred.

At the present time it is not possible to determine if the aforementioned damage to the containments was caused by the earthquake (note that some ground accelerations measured at Units 2 and 3 did exceed the design-basis values) or by high containment pressures.

Q.22. Is there any evidence of water leaks at the spent-fuel pools (SFPs)?

A.22. As indicated in **A.13**, the available evidence does not yet permit the SFP scenarios to be completely described. While no hard evidence has been reported that leakage of SFP inventory occurred, there is also no hard evidence presented that leakage did not occur. "Additional Report of the Japanese Government to the IAEA—The Accident at TEPCO's Fukushima Nuclear Power Stations (Second Report)," Government of Japan, Nuclear Emergency Response Headquarters (September 2011) concludes that fuel damage was not likely to have occurred in the units because of low SFP inventory levels, which implies that any leakage was not sufficient to be a factor in damaging the fuel.

As also indicated in **A.13**, early comments were made that a damage mechanism in addition to SFP boiling had occurred in the Unit 4 SFP. However, these comments might have represented an attempt to explain the source of hydrogen in the Unit 4 refueling bay. At the time of the comments, the source of this hydrogen had not yet been argued to be Unit 3.

There was concern as to the structural strength of the support for the Unit 4 SFP and that concern was sufficient to lead TEPCO to place steel columns to provide additional support. Reports indicate that the base concern was that an additional earthquake might cause damage to the weakened SFP support. The Committee has not found reference to any observed leaks.

Q.23. What was the cause of the "explosion" in or near the containment in Fukushima Daiichi Unit 2? If it was within the containment, it cannot be due to hydrogen since the containment was filled with steam and nitrogen at that time. Was it a steam explosion due to corium contact with water under the reactor pressure vessel (RPV)?

A.23. It is currently thought that at the time of the explosion, the RPV boundary was still intact, so no steam explosion could have occurred. However, if there had been a breach of the RPV lower head by molten core, even if there was water above the corium in the RPV, no water would have been in the lower drywell when the molten core arrived, so a steam explosion could not have been the explanation. Steam explosions are usually associated with loss-of-coolant-accident–type scenarios where sufficient water can end up in the lower drywell before core melt. Hydrogen external to the containment in the torus room was first thought to be the explanation for the sound, but as of December 2011, it is thought the sound came from a sudden break in the containment boundary.

Q.24. Why did the hydrogen venting not go to the stack but rather into the reactor building? On one reactor one could understand that earthquake damage might have severed the duct, but not on the others. What manual actions were required to open the vents?

A.24. Manual opening of air (nitrogen) operated valves (AOVs) and direct-current operated valves *and* containment pressure in excess of rupture disk pressure (0.549 MPa) were required in order for the hard-piped vent pathway to be opened to the plant stack. As there are no accessible valve operators for the AOVs that can be manually operated, portable air compressors were also required to open these valves. In addition, it took many hours to coordinate with local and national officials to obtain permission to vent after the neighboring population had been evacuated or confined. In the meantime, containment leaks provided another pathway for hydrogen to migrate into the reactor buildings, and high local radiation levels complicated the ability to manually open isolation valves. Finally, once the hard-pipe pathway was open, it was also possible for hydrogen to return to the same or shared unit via reverse flow through other ductwork using the same stack.

Q.25. What caused the batteries that provide power to instrumentation and RCIC and HPCI control to die? Why was there no means to charge them via the steam-driven turbines?

A.25. Reactor core isolation cooling (RCIC) and high-pressure coolant injection (HPCI) turbines do not have generators/alternators on the turbine shaft. The only turbine that is coupled to a generator is the main turbine. It takes direct-current power to run the RCIC and HPCI turbine speed control system and the injection valves, and unfortunately, the flow is not controlled to match decay heat. This means that the systems run full-out until high-water-level trip followed by cyclic action to start and stop these systems as the water level cycles between high and low water levels, unless operators intervene to manually control these systems. There is a drain on the batteries every time the valves are opened and closed. In the case of Fukushima Daiichi Unit 3, operators did use HPCI in a clever way before failure by routing most of the HPCI discharge back through the full flow test line to maintain a relatively constant water level in the reactor pressure vessel, avoiding high-level trips and more drain on the batteries.

Q.26. How much diesel fuel was at the Fukushima Daiichi nuclear power station? Was it really stored only in the two tanks swept away by the tsunami? Why were the tanks not protected better?

A.26. There are two 50,000-gallon diesel fuel tanks for the emergency diesel generators (EDGs). One feeds Units 1 and 2, and the other feeds Units 3 and 4. Those tanks are still intact. The two tanks that were washed away were heavy heating oil tanks, ~250,000 gallons each.

The fuel tank for Units 5 and 6 remained intact, as evidenced by the fact that after the tsunami, the air-cooled EDG of Unit 6 continued to operate. The other EDGs at Units 5 and 6 were stopped because of flooding, not because of loss of fuel. The air-cooled EDG was able to supply power to both Units 5 and 6 through some modification of wiring.

Q.27. Where did coupling of services (ventilation, power, etc.) between nuclear power plants at Fukushima Daiichi help or hurt accident response?

A.27. Ventilation coupling allowed for transfer of hydrogen at least between Units 3 and 4. Power is coupled between Units 1 and 2 and between Units 3 and 4, but since there was a total loss of power, it probably did not matter. Coupling saved Unit 5, in which power was supplied by a Unit 6 emergency diesel generator that had not been damaged by the tsunami.

Recovery might have been helped with shared power ability. Fire protection is shared among all units, which allows for faster lineups.

Q.28. When did Fukushima Daiichi nuclear power plant (NPP) operators lose (due to lack of electricity) the information about the pressure and temperature that is usually available to them, and when was the information restored?

A.28. At Units 1 and 2, the direct-current batteries that supply power for instrumentation were flooded when the tsunami struck, so there was no information available in the control room for the NPP operators during the critical time period between ~1535 (~3.35 p.m.) and ~2200 (~10:00 p.m.) on March 11, at which time some functions were restored in the shared Units 1 and 2 control room. By sending operators into the NPP with car batteries to energize transmitters, local measurements were taken a couple of times during the complete blackout period for key parameters such as reactor pressure and water level and containment pressure.

Q.29. Did the Tokyo Electric Power Company (TEPCO) have training of nuclear power plant (NPP) operators on severe accidents? Did that include loss-of-power–type events?

A.29. Japan NPP operators receive training on severe accident management guidance, similarly to operators in the United States. Based on Regulatory Guide 1.155, "Station Blackout," U.S. Nuclear Regulatory Commission (August 1988), TEPCO modified the Emergency Operational Guide to include actions necessary in the case of total loss of alternating-current (AC) power on August 30, 1990. Then, throughout the 1990s, other measures were taken by TEPCO to enhance the capability to cope with total loss of AC power at its NPPs, such as the addition of air-cooled emergency diesel generators (one at the Fukushima Daiichi NPP) and installation of hard-piped containment vents.