



Expanded NRC Questions and Answers related to the March 11, 2011 Japanese Earthquake and Tsunami (October 19, 2011)

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Japanese Power Plants

Fukushima Daiichi Earthquake

Q: *Did the Japanese underestimate the size of the maximum credible earthquake and tsunami that could affect the plants?*

A: The magnitude of the earthquake was somewhat greater than was expected for that part of the subduction zone. However, the Japanese nuclear plants were recently reassessed using ground motion levels similar to those that are believed to have occurred at the sites. The ground motions against which the Japanese nuclear plants were reviewed were expected to result from earthquakes that were smaller, but were much closer to the sites. Although the NRC does not regularly have access to design information on foreign nuclear power plants, information regarding the maximum tsunami height that was expected at the sites is available at the following links:

<http://nei.org/newsandevents/information-on-the-japanese-earthquake-and-reactors-in-that-region/japan-earthquake-additional-nei-updates/japan-earthquake-nei-updates-for-monday-march-21/>

<http://www.tepco.co.jp/en/press/corp-com/release/11040910-e.html>

Q: *Was the damage to the Japanese nuclear plants mostly from the earthquake or the tsunami?*

A: Because this event happened in Japan, the NRC relies primarily on information made available to it by the Japanese government and several organizations involved in responding, assessing, and mitigating the events at the Japanese nuclear plants. These include the Japanese regulator, the Nuclear and Industrial Safety Agency (NISA), Tokyo Electric Power Company (TEPCO), the operator of the Fukushima Daiichi nuclear plants), and the Japan Atomic Industrial Forum (JAIF). Radiation levels in certain areas of the nuclear plants make it very difficult for the NRC, as well as others, to assess the precise conditions of the facilities. Through TEPCO's continued efforts more specific information about the conditions of the plants is learned with each passing day. Based on the information provided by the Japanese, the NRC has learned that the nuclear plants may have sustained some damage from the ground shaking produced by the earthquake, and that the earthquake also caused the loss of offsite power. However, the tsunami appears to have played a key role in the loss of other power sources at the site producing station blackout, which is a critical factor in the ongoing problems. Additional information regarding the damage to the Japanese nuclear plants may also be obtained from the websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>).

Q: *The NRC Near-Term Task Force Report states that a sequence of events like the Fukushima accident is unlikely to occur in the U.S. and some appropriate mitigation measures have been implemented. What are those appropriate mitigation measures? NEW!*

A: The mitigation measures are what are commonly referred to as the B.5.b actions. These are the actions that were taken following the events of 9/11 in the United States. These measures would deal with the loss of large areas of the plant, including the use of portable equipment to provide some level of core cooling, spent fuel pool cooling and/or maintenance of containment integrity. They provide an additional level of mitigation capability that may be of assistance in the event of a significant accident similar to Fukushima.

Fukushima Daiichi Emergency Preparedness

Q: *What is the basis for the dose analyses attached to the March 16, 2011, NRC press release?*

A: The basis for the dose assessment was the limited and unverifiable information on the plant conditions at the Fukushima facility. The facility was modeled in a computer-based dose assessment code as a hypothetical, four reactor site. The dose assessment results are conservative predictions only and may not be representative of any actual radiation releases. The computer-based dose assessment model also utilized predicted meteorological conditions following the events at the Fukushima facility and, therefore, may not be representative of the actual meteorological conditions that occurred for this area. The NRC press release of March 16, 2011, and the predicted dose estimates are available on the NRC's public website and may be accessed at the following link: <http://www.nrc.gov/reading-rm/doc-collections/news/2011/11-050.pdf>.

The assumptions on plant conditions used as the basis for the analyses were indicative of the uncertain and unstable nature of the conditions on Fukushima Daiichi site at the time the analyses were done, and accounted for uncertainty in the future progression of events. Since that time, actions to mitigate the events at facility and to stabilize the reactors and spent fuel at the plant have continued. The NRC continues to support the protective action recommendations provided in the March 16, 2011, press release because conditions at the plant continue to change. The NRC continues to monitor the situation at the Fukushima facility and may reassess its protective action recommendations as additional detailed and verifiable information about actual conditions becomes available.

Q: *Why did the NRC decide to recommend evacuation out to 50 miles from the Fukushima Daiichi facility for U.S. citizens in Japan?*

A: The decision to expand evacuation of U.S. citizens out to 50 miles from the Fukushima Daiichi facility was a conservative decision that was made out of consideration of several factors including an abundance of caution resulting from limited and unverifiable information concerning event progression at several units at the Fukushima Daiichi facility. The NRC based its assessment on information available at the time regarding the condition of the units conditions at Fukushima Daiichi that included significant damage to Units 1, 2, and 3 that appeared to have been a result of hydrogen explosions. Prior to the earthquake and tsunami, Unit 4 was in a refueling outage and its entire core had been transferred to the spent fuel pool only 3 months earlier so the fuel was quite fresh. Radiation monitors showed significantly elevated readings in some areas of the plant site which would challenge plant crews attempting to stabilize the plant. Based on analysis results, there were indications from some offsite contamination sampling smears that fuel damage had occurred. There was a level of uncertainty about whether or not efforts to stabilize the plant in the very near term were going to be successful. Changing meteorological conditions resulted in the winds shifting rapidly from blowing out to sea to blowing back onto land.

Q: *How did the NRC develop its computer-based projections that supported the evacuation decision?*

A: The NRC uses the RASCAL computer code to perform offsite radiation dose projections. The RASCAL computer program contains information about U.S. nuclear reactor design types, radiation release pathways from the nuclear power plant to the environment, radionuclide source terms and meteorology. However, RASCAL is not capable of evaluating concurrent and multiple nuclear plant failures. So, to approximate the events unfolding at the Fukushima Daiichi facility, the NRC developed a model that aggregated information from the three operating reactors and the spent fuel pool. This aggregate model was then evaluated using the RASCAL computer code. The radiation doses calculated by the RASCAL code were predicted to exceed the protective action guidelines (PAGs) established by the U.S. Environmental Protection Agency (EPA) well beyond the 10-mile exposure pathway EPZ and beyond the 30 kilometer sheltering zone recommended by the Japanese authorities. Subsequent aerial monitoring by the U.S. Department of Energy (DOE) fixed-wing aircraft monitoring showed elevated radiation dose rates that were in excess of the EPA relocation PAGs to a distance beyond 25 miles from the facility.

Q: *A chart titled "NRC Dose Estimates" was posted on March 17, 2011, to the Yahoo Group "Know_Nukes," which plots total dose (Rem) vs. distance (miles) for a one reactor site and a four reactor site. Was this document released by the NRC?*

A: No, this document was not released by the NRC. The chart appears to plot the dose information that was included as attachments to the NRC press release of March 16, 2011. This press release provided NRC protective action recommendations for U.S. citizens residing within 50 miles of the Fukushima reactors. The NRC press release had two attachments that gave the results of dose assessments performed for the Fukushima Daiichi facility.

Fukushima Daiichi Event Progression

Q: *As time goes on, does the chance for a meltdown increase?*

A: Based on analyses and information obtained regarding the conditions of the reactors at the Fukushima Daiichi nuclear facility, TEPCO (owner/operator of the facility) has reported that it estimates that considerable melting of the reactor fuel occurred in Units 1, 2 and 3. A report to IAEA from the government of Japan stated that some material originally in the reactor vessel is, to some degree, now outside of the vessel. They are maintaining adequate supplies of cooling water to the fuel in the vessels, however damage and leakage of the reactor pressure vessels is suspected. The fuel has cooled off to the point where no further damage is expected as long as cooling is maintained. The websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>) provide additional information on a daily basis.

Q: *If Chernobyl was a 7 and Three Mile Island was a 5, when does this event move from the 4 level?*

A: The International Atomic Energy Agency (IAEA) rates nuclear events in accordance with its International Nuclear and Radiological Event Scale (INES). IAEA initially assigned the events in Japan an INES rating of 4, "Accident with Local Consequences." This rating is subject to change as events unfold and additional information becomes available. INES classifies nuclear accidents based on the radiological effects on people and the environment and the status of barriers to the release of radiation. IAEA determinations regarding the INES rating of events are made independently.

Three Mile Island was assigned an INES rating of 5, "Accident with Wider Consequences," due to the severe damage to the reactor core.

On April 12, 2011, the Japanese Nuclear and Industrial Safety Agency (NISA) government raised the rating for the events at the Fukushima Daiichi site on the International Nuclear and Radiological Event Scale (INES) from 5, "Accident with Wider Consequences," to 7, "Major Accident," citing calculations by both NISA and the Nuclear Safety Commission of Japan (NSC) of radioactive materials released from the Fukushima Daiichi reactors. This new provisional rating considers the accidents that occurred at Units 1, 2, and 3 as a single event on INES. NISA notes that while an INES rating of 7 is the same as that of the Chernobyl accident, their current estimated amount of radioactive materials released is approximately 10% of the amount from the Chernobyl accident.

Q: *Compare this incident to the Three Mile Island. What are the similarities?*

A: The events at Three Mile Island (TMI) in 1979 were the result of an equipment malfunction that resulted in the loss of cooling water to the reactor fuel. Subsequent operator actions compounded the malfunction ultimately resulting in the partial core meltdown. The events in Japan appear to be the result of an earthquake and subsequent tsunami that knocked out electrical power to emergency safety systems designed to cool the reactor fuel. TEPCO (owner/operator of the Fukushima Daiichi facility) estimates that considerable melting of the reactor cores occurred in three of the six units at this facility. However, the core material in these units is now being cooled with adequate amounts of water. In comparison, only one of the two units at TMI experienced partial core melting in 1979. In both events the final safety barrier, the containment building, remained largely intact and contained the majority of the radioactivity preventing its release to the environment. There appears to have been more radiation released from the Fukushima facility than from TMI and, as a result, the International Atomic Energy Agency on April 12, 2011, raised its rating of the Fukushima accident on the International Nuclear and Radiological Event Scale (INES) from 5, "Accident with Wider Consequences", which was the TMI rating, to 7, "Major Accident." The conditions at the Fukushima Daiichi nuclear facility in Japan continue to be monitored and assessed and actions to mitigate and prevent further releases of radiation to the environment are being actively employed by TEPCO. The websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>) provide additional information on a daily basis.

Japanese Power Plants

Q: *What's going to happen following the hydrogen explosions everyone's seen from the video footage?*

A: The NRC is aware of the Japanese efforts to stabilize conditions at the affected reactors, and those actions are in line with what would be done in the United States. The Japanese owner/operator of the facility (TEPCO) announced in late April that it established a plan toward restoring control of the Fukushima facility. The major aims of the plan involve two steps: (1) achieving a steady decline in radiation dose at the plant, and (2) bringing radioactive materials under control and significantly holding the radiation dose down. TEPCO categorized specific efforts under three major headings of "cooling," "mitigation" and, "monitoring and decontamination." These were further divided into the following five areas: (1) cooling the reactors, (2) cooling the spent fuel pools, (3) containing, storing, processing and reusing the water contaminated by radioactive materials (accumulated water), (4) mitigating radioactive materials in the atmosphere and soil, and (5) measuring, reducing and announcing the radiation doses in areas where evacuation has already taken place and where it is being planned, as well as areas where preparations are being made for emergency evacuation. The websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>) provide additional information on a daily basis.

The NRC continues to monitor information on the status of the reactor cores, the reactor vessels and the containment structures at the Fukushima Daiichi units – all three areas are important to controlling the situation and protecting the public.

On May 30, 2011, the TEPCO Board of Directors announced that it would decommission Units 1 to 4 at its Fukushima Daiichi Nuclear Power Station.

Q: *What is the worst-case scenario?*

A: In a nuclear emergency, the most important action is to ensure the core is covered with water to provide cooling to remove any heat from the fuel rods. Without adequate cooling, the fuel rods will melt. Recent reports from Japan have indicated that considerable amounts of fuel melted inside the reactor vessels of Units 1, 2, and 3, although the fuel remains inside the reactor vessel and adequate cooling water is being provided to the fuel; however damage and leakage of the reactor pressure vessels is suspected. Should the final containment structure fail, radiation from these melting fuel rods would be released to the atmosphere and additional protective measures may be necessary depending on such meteorological factors such as prevailing wind patterns and rainfall.

Q: *What else can go wrong?*

A: The NRC is continuously monitoring the developments at the nuclear power plants in Japan. Circumstances are constantly evolving and new information regarding the specific conditions at each unit is obtained on a daily basis. At this point, it would be inappropriate to speculate on how this situation might develop over the coming days, weeks and months. The challenges for restoring control of each unit at the Fukushima facility are extraordinary and complex. However, plans have been established and implemented to secure, stabilize and control conditions at the Fukushima nuclear units.

Q: *What is the sequence of events at the Japanese reactors?*

A: The Japanese have provided a sequence of events in a report to International Atomic Energy Agency that can be accessed through the IAEA website: <http://www.iaea.org/newscenter/focus/fukushima/japan-report/>. The NRC is currently reviewing the report. The NRC's initial assessment is that is consistent with the agency's understanding of the events that transpired at Fukushima 1 Daiichi following the March 11 earthquake and tsunami. At this early point in the review, no immediate actions for the NRC were identified beyond the Temporary Instructions and the Bulletin that were issued. The NRC is planning to thoroughly review the report and will identify any lessons learned that may be applicable in the United States.

Q: *Where can the public find the report from Japan to the International Atomic Energy Agency? Has the NRC reviewed this report?*

A: The Japanese have provided a sequence of events in a report to International Atomic Energy Agency that can be accessed through the IAEA website: <http://www.iaea.org/newscenter/focus/fukushima/japan-report/>. The NRC is currently reviewing the report. The NRC's initial assessment is that is consistent with the agency's understanding of the events that transpired at Fukushima 1 Daiichi following the March 11 earthquake and tsunami. At this early point in the review, no immediate actions for the NRC were identified beyond the Temporary Instructions and the Bulletin that were issued. The NRC is planning to thoroughly review the report and will identify any lessons learned that may be applicable in the United States.

Japanese Power Plants

Q: *Could an accident sequence like the one at Japan's Fukushima Daiichi nuclear plants happen in the US?*

A: The NRC task force has issued its near-term review of the event and its impact on U.S. plants ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission). The current regulatory approach, and more importantly, the resultant plant capabilities allow the task force to conclude that a sequence of events like the Fukushima accident is unlikely to occur in the United States and some appropriate mitigation measures have been implemented, reducing the likelihood of core damage and radiological release. Therefore, continued operation and continued licensing activities do not pose an imminent risk to public health and safety. The NRC is planning a longer-term review and will review any new specific information regarding the disaster at the Fukushima plant and its applicability to U.S. reactors, identify lessons learned, and determine if any changes to its regulatory requirements are necessary to continue to ensure the health and safety of the public and the environment.

The NRC relies primarily on information made available to it by the Japanese government and several organizations involved in responding, assessing, and mitigating the events at the Japanese nuclear plants. Those sources have described how Fukushima Daiichi Units 1-3 lost all offsite power and emergency diesel generators. This situation is called "station blackout." US nuclear power plants are designed to cope with a station blackout event that involves a loss of offsite power and onsite emergency power. The NRC's detailed regulations address this scenario. US nuclear plants conducted a "coping" assessment and developed a strategy to demonstrate to the NRC that they could maintain the plant in a safe condition during a station blackout scenario. These assessments, proposed modifications to the plant, and operating procedures were reviewed and approved by the NRC. Several plants added additional AC power sources to comply with this regulation.

In addition, US nuclear plant designs and operating practices since the terrorist events of September 11, 2001, are designed to mitigate severe accident scenarios such as aircraft impact, which include the complete loss of offsite power and all on-site emergency power sources.

US nuclear plant designs include consideration of seismic events and tsunamis. It is important not to extrapolate earthquake and tsunami data from one location of the world to another when evaluating these natural hazards. These catastrophic natural events are very region- and location-specific, based on tectonic and geological fault line locations.

Fukushima Daiichi Hydrogen Explosion

Q: *Could explosions like those that occurred in Japan happen at a U.S facility?*

A: The NRC is aware of the Japanese efforts to stabilize and control the plants. While we've learned a great deal, additional investigations and analyses will be necessary to provide a comprehensive and precise explanation for the explosions. In Units 1, 2, and 3 of Fukushima Daiichi, available evidence suggests the explosions were caused by the buildup of hydrogen gas within primary containment produced during fuel damage in the reactor and subsequent movement of that hydrogen gas from the drywell into secondary containment. Available evidence has yet to provide a compelling cause for the explosion in Unit 4. U.S. facilities of similar design have venting capabilities that would allow operators to release hydrogen or other combustible gases to prevent a concentrated buildup that could exceed the flammability limit. The NRC's near term review ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission) recommends making those venting capabilities a regulatory requirement. The Commission will review the Task Force Report and will provide direction to the staff.

Japanese Power Plants

Q: *What's going to happen following the hydrogen explosions everyone's seen from the video footage?*

A: The NRC is aware of the Japanese efforts to stabilize conditions at the affected reactors, and those actions are in line with what would be done in the United States. The Japanese owner/operator of the facility (TEPCO) announced in late April that it established a plan toward restoring control of the Fukushima facility. The major aims of the plan involve two steps: (1) achieving a steady decline in radiation dose at the plant, and (2) bringing radioactive materials under control and significantly holding the radiation dose down. TEPCO categorized specific efforts under three major headings of "cooling," "mitigation" and, "monitoring and decontamination." These were further divided into the following five areas: (1) cooling the reactors, (2) cooling the spent fuel pools, (3) containing, storing, processing and reusing the water contaminated by radioactive materials (accumulated water), (4) mitigating radioactive materials in the atmosphere and soil, and (5) measuring, reducing and announcing the radiation doses in areas where evacuation has already taken place and where it is being planned, as well as areas where preparations are being made for emergency evacuation. The websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>) provide additional information on a daily basis.

The NRC continues to monitor information on the status of the reactor cores, the reactor vessels and the containment structures at the Fukushima Daiichi units – all three areas are important to controlling the situation and protecting the public.

On May 30, 2011, the TEPCO Board of Directors announced that it would decommission Units 1 to 4 at its Fukushima Daiichi Nuclear Power Station.

Fukushima Daiichi Lessons Learned

Q: *How will the U.S. learn from the failures at the Japanese reactors?*

A: The NRC has established a senior level task force to analyze the events in Japan and develop lessons learned and recommendations to improve plant safety, as appropriate. The task force issued its near-term report ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission) and concluded that continued operation and continued licensing activities do not pose an imminent risk to public health and safety. The task force recommended rulemaking activities, orders, certain staff actions, and actions for long-term evaluation. The Commission will review the report and will provide the staff with direction. A long-term evaluation is planned and will assess whether any additional licensing actions are necessary. These actions may include Orders, information requests in accordance with Section 50.54(f) of Title 10 (10 CFR) of the Code of Federal Regulations, license amendments, rulemaking, etc.

The NRC issued an information notice to inform licensees about the effects of the earthquake on nuclear power plants in Japan. In addition, the NRC's staffs at every reactor site have performed targeted inspections to confirm facility responses to beyond design-basis events. The NRC has also issued Bulletin 2011-01 that requires all licensees to verify under oath and affirmation that their mitigation strategies and capabilities are in compliance with relevant NRC regulations.

Q: *Have any lessons for US nuclear plants been identified?*

A: The NRC is continuing to follow and review the events in Japan in real time. The NRC established a senior level task force to conduct both short- and long-term analysis of the lessons that can be learned from the situation in Japan. The task force has completed its near-term analysis of the events and their impact on U.S. plants ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission). The report included 12 overarching recommendations to clarify the NRC's regulatory framework for events of low probability but high consequence, assure protection against accidents caused by severe natural phenomena, enhance mitigation of accidents should they occur, strengthen emergency preparedness for protection of the public, and improve the effectiveness of NRC's programs. The Commission will review the task force report and will provide further direction to the staff. The task force is also going to perform a longer term study based, in part, on recommendations for further study in the near-term report.

Japanese Power Plants

Q: *Have events in Japan changed our perception of earthquake risk to the nuclear plants in the US?*

A: The NRC continues to determine that US nuclear plants are safe. The Japanese quake does not change the NRC's perception of earthquake hazard (i.e., ground motion levels) at US nuclear plants. Even before the events in Japan, the NRC began reviewing the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting a generic review referred to as GI-199, "Implications of Updated Probabilistic Seismic Estimates in Central and Eastern United States on Existing Plants," to again assess the resistance of US nuclear plants to earthquakes. In addition, the NRC has been reviewing updated seismic information regarding the plants in California for many years. The NRC has established a senior level task force to identify areas of further evaluation as a result of the Japanese events. The task force has issued its near-term report, which recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection for each operating reactor. The Commission will review the report and will provide direction to the staff.

Q: *Where can the public find the report from Japan to the International Atomic Energy Agency? Has the NRC reviewed this report?*

A: The Japanese have provided a sequence of events in a report to International Atomic Energy Agency that can be accessed through the IAEA website: <http://www.iaea.org/newscenter/focus/fukushima/japan-report/>. The NRC is currently reviewing the report. The NRC's initial assessment is that is consistent with the agency's understanding of the events that transpired at Fukushima 1 Daiichi following the March 11 earthquake and tsunami. At this early point in the review, no immediate actions for the NRC were identified beyond the Temporary Instructions and the Bulletin that were issued. The NRC is planning to thoroughly review the report and will identify any lessons learned that may be applicable in the United States.

Q: *The NRC Near-Term Task Force Report states that a sequence of events like the Fukushima accident is unlikely to occur in the U.S. and some appropriate mitigation measures have been implemented. What are those appropriate mitigation measures? NEW!*

A: The mitigation measures are what are commonly referred to as the B.5.b actions. These are the actions that were taken following the events of 9/11 in the United States. These measures would deal with the loss of large areas of the plant, including the use of portable equipment to provide some level of core cooling, spent fuel pool cooling and/or maintenance of containment integrity. They provide an additional level of mitigation capability that may be of assistance in the event of a significant accident similar to Fukushima.

Fukushima Daiichi Radiation

Q: *Is there a danger of radiation making it to the United States?*

A: In response to nuclear emergencies, the NRC works with other U.S. agencies to monitor radioactive releases and predict their path. The NRC continues to monitor information regarding wind patterns near the Japanese nuclear power plants. Nevertheless, given the thousands of miles between the two countries, Hawaii, Alaska, the U.S. Territories and the U.S. West Coast are not expected to experience any harmful levels of radioactivity.

Q: *What is the official agency to report radiation numbers and what is the public contact?*

A: NRC regulations require nuclear power plants to report any radiation doses detected at the plant that could be harmful to the public. This would include doses that are generated by the plant or by an external source. During an event in the U.S., it is the state's responsibility to provide protective action decisions for public health and safety. For this incident, the Japanese are responsible for reporting the public dose; nevertheless, should radiation doses be detected within the U.S., it would still be the state's responsibility to provide protective action decisions for public health and safety.

Q: *My family has planned a vacation to Hawaii/Alaska/Seattle next week – is it safe to go, or should we cancel our plans?*

A: The NRC does not expect that residents of the United States or its territories are at any risk of exposure to harmful levels of radiation resulting from the events in Japan. Any changes to travel are a personal decision. The NRC is unaware of any travel restrictions within the United States or its territories.

Japanese Power Plants

Q: *What are the risks to my children?*

A: The NRC continues to believe that protective measures are unnecessary in the United States. No U.S. states or territories have detected harmful levels of radioactivity. In the unlikely event that circumstances change, U.S. residents should listen to the protective action decisions of their states and counties. These protective action decisions could include actions such as sheltering, evacuation, or taking potassium iodide. The NRC will provide technical assistance to the states should they request it. United States citizens in Japan are encouraged to follow the protective measures recommended by the Japanese government. These measures appear to be consistent with steps the United States would take.

Q: *Are there other protective measures I should be taking?*

A: The NRC continues to believe that protective measures are unnecessary in the United States. No U.S. states or territories have detected harmful levels of radioactivity. In the unlikely event that circumstances change, U.S. residents should listen to the protective action decisions of their states and counties. These protective action decisions could include actions such as sheltering, evacuation, or taking potassium iodide. The NRC will provide technical assistance to the states should they request it. United States citizens in Japan are encouraged to follow the protective measures recommended by the Japanese government. These measures appear to be consistent with steps the United States would take.

Q: *I live in the Western United States – should I be taking potassium iodide (KI)?*

A: The NRC continues to believe that protective measures are unnecessary in the United States. No U.S. states or territories have detected harmful levels of radioactivity. In the unlikely event that circumstances change, U.S. residents should listen to the protective action decisions of their states and counties. These protective action decisions could include actions such as sheltering, evacuation, or taking potassium iodide. The NRC will provide technical assistance to the states should they request it.

Q: *The radiation “plume” seems to be going out to sea -- what is the danger of it reaching Alaska? Hawaii? The west coast?*

A: In response to nuclear emergencies, the NRC works with other U.S. agencies to monitor radioactive releases and predict their path. The NRC continues to monitor information regarding wind patterns near the Japanese nuclear power plants. Nevertheless, given the thousands of miles between the two countries, Hawaii, Alaska, the U.S. Territories and the U.S. West Coast are not expected to experience any harmful levels of radioactivity.

Q: *Is the U.S. government tracking the radiation released from the Japanese plants?*

A: Yes, a number of U.S. agencies are involved in monitoring and assessing radiation including EPA, DOE, and NRC. The best source of additional information is the Environmental Protection Agency.

Q: *What should be done to protect people in Alaska, Hawaii and the West Coast from radioactive fallout?*

A: The NRC believes that the actions of the Japanese to control, stabilize and mitigate radioactive releases from the reactors at Fukushima have prevented harmful levels of radiation from reaching U.S. territory. The NRC continues to believe that protective measures are unnecessary in the United States. No U.S. states or territories have detected harmful levels of radioactivity. In the unlikely event that circumstances change, U.S. residents should listen to the protective action decisions of their states and counties. These protective action decisions could include actions such as sheltering, evacuation, or taking potassium iodide. The NRC will provide technical assistance to the states should they request it.

Japanese Power Plants

Q: *What is the magnitude of the radiation release from the Japanese facility at Fukushima?*

A: The NRC sent staff to assist and advise officials in Japan regarding the response and mitigation of the current reactor and spent fuel pool events. The NRC is continuing to work through the U.S. ambassador to Japan regarding these activities. The U.S. Department of Energy (DOE) is continuing to provide support to the Japanese for aerial monitoring and assisting in developing dose predictions for areas surrounding the Fukushima facility. Efforts to mitigate and monitor the radiation releases from the facility continue while activities to control and stabilize conditions at the facility progress. Radiation monitoring by the Japanese and supporting organizations has involved airborne sampling and monitoring, offsite sampling of ground and groundwater contamination in the areas surrounding the facility, and sampling of contaminated seawater and sea beds offshore from the facility. As the situation at the Fukushima continues to evolve, the estimates of radiation releases are in flux. The most up to date radiation release information for the Fukushima event may be found in documents and status updates provided at the websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>).

If an event occurred at a nuclear plant in the U.S., the NRC would model radiation releases using a sophisticated computer program called RASCAL (Radiological Assessment System for Consequence Analysis). Individual states are responsible for deciding when their citizens might need to evacuate or take shelter in response to such an event. There are two Emergency Planning Zones (EPZs) around each nuclear plant; a 10-mile EPZ for plume exposure and a 50-mile EPZ for food exposure. The 10-mile EPZ is the area established as a basis for planning because, at that distance, the projected doses from most accidents would not exceed the Environmental Protective Agency's protective action dose guidelines (1-5 rem). However, the 10-mile EPZ was always considered a basis for emergency planning that could be expanded if the situation warranted. The situation in Japan, with three reactors and two fuel pools experiencing exceptional difficulties simultaneously, along with uncertainty regarding conditions at the plant, led to the decision to recommend evacuation of U.S. citizens out to 50 miles from Fukushima.

Q: *A chart titled "NRC Dose Estimates" was posted on March 17, 2011, to the Yahoo Group "Know_Nukes," which plots total dose (Rem) vs. distance (miles) for a one reactor site and a four reactor site. Was this document released by the NRC?*

A: No, this document was not released by the NRC. The chart appears to plot the dose information that was included as attachments to the NRC press release of March 16, 2011. This press release provided NRC protective action recommendations for U.S. citizens residing within 50 miles of the Fukushima reactors. The NRC press release had two attachments that gave the results of dose assessments performed for the Fukushima Daiichi facility.

Q: *Has the government set up radiation monitoring stations to track the release?*

A: The NRC understands that EPA is utilizing its existing nationwide radiation monitoring system, RadNet, to continuously monitor the nation's air and regularly monitors drinking water, milk and precipitation for environmental radiation. EPA has publicly stated its agreement with the NRC's assessment that we do not expect to see radiation at harmful levels reaching the U.S. from damaged Japanese nuclear power plants. Nevertheless, EPA has stated that it plans to work with its federal partners to deploy additional monitoring capabilities to parts of the western U.S. and U.S. territories.

Q: *What are the short-term and long-term effects of exposure to radiation?*

A: The NRC does not expect that residents of the United States or its territories are at any risk of exposure to harmful levels of radiation resulting from the events in Japan.

On a daily basis, people are exposed to naturally occurring sources of radiation, such as from the sun or medical X-rays. The resulting effects are dependent on the strength and type of radiation as well as the duration of exposure.

Japanese Power Plants

Q: *What is the basis for the dose analyses attached to the March 16, 2011, NRC press release?*

A: The basis for the dose assessment was the limited and unverifiable information on the plant conditions at the Fukushima facility. The facility was modeled in a computer-based dose assessment code as a hypothetical, four reactor site. The dose assessment results are conservative predictions only and may not be representative of any actual radiation releases. The computer-based dose assessment model also utilized predicted meteorological conditions following the events at the Fukushima facility and, therefore, may not be representative of the actual meteorological conditions that occurred for this area. The NRC press release of March 16, 2011, and the predicted dose estimates are available on the NRC's public website and may be accessed at the following link: <http://www.nrc.gov/reading-rm/doc-collections/news/2011/11-050.pdf>.

The assumptions on plant conditions used as the basis for the analyses were indicative of the uncertain and unstable nature of the conditions on Fukushima Daiichi site at the time the analyses were done, and accounted for uncertainty in the future progression of events. Since that time, actions to mitigate the events at facility and to stabilize the reactors and spent fuel at the plant have continued. The NRC continues to support the protective action recommendations provided in the March 16, 2011, press release because conditions at the plant continue to change. The NRC continues to monitor the situation at the Fukushima facility and may reassess its protective action recommendations as additional detailed and verifiable information about actual conditions becomes available.

Fukushima Daiichi Reactor Design

Q: *How do the Japanese reactor designs compare to the US reactor designs of similar vintage?*

A: The NRC is not aware of all of the differences that may exist between the Japanese reactors that are of similar design and vintage as those operated in the U.S. Many improvements have been made to U.S. boiling water reactors (BWRs). For example, NRC Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," conveyed the importance of having a robust pathway for venting primary containment, which contains the suppression pool, in certain severe accident scenarios. In response, all BWRs with Mark I containments that didn't have an existing strengthened or "hardened" pathway for venting directly from primary containment to the outside, made modifications to the plant consistent with the intent of the Generic Letter. This design feature permits a controlled depressurization of primary containment as well as a controlled release of radioactive materials and combustible hydrogen generated by damaged fuel, as may occur during severe accidents. Additional enhancements include:

- Emergency diesel generator (EDG) fuel oil tanks required by NRC regulations are sheltered in safety-related structures or underground in order to withstand an earthquake as well as flooding events. These tanks provide a reliable fuel supply to safety related AC and DC power systems for several days.

- The regulations in 10 CFR 50.63 require all U.S. nuclear power plants to cope with a loss of all AC power (i.e., station blackout) in the event of a loss of station on-site and normal off-site power sources. In addition, nuclear plants are required to have alternate AC sources from separate grid systems separate from the normal off-site power supply.

- A portable emergency diesel-driven water pump for emergency fuel pool cooling is available at all US nuclear sites.

- Emergency operating procedures as well as severe accident management guidelines ensure that the containment structure integrity takes priority in an accident situation. Therefore, in a beyond-design-basis event, such as the one at Fukushima Dai'ichi, U.S. BWR operators are trained to reduce the buildup of explosive concentrations of hydrogen and to preserve primary and secondary containment by venting.

- In parallel with the above, a U.S. facility's emergency operating procedures would prioritize the restoration of offsite power in order to restore vital power needs following a severe event.

Japanese Power Plants

Q: *Can significant damage to a nuclear plant like we see in Japan happen in the US due to an earthquake? Are the Japanese nuclear plants similar to US nuclear plants?*

A: All US nuclear plants are built to withstand environmental hazards, including earthquakes and tsunamis. Even those nuclear plants that are located within areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. The NRC requires that safety-significant structures, systems, and components be designed to take into account even rare and extreme seismic and tsunami events. In addition to the design of the plants, significant effort goes into emergency response planning and accident management. This approach is called defense-in-depth.

The Japanese facilities are similar in design to some US facilities. However, the NRC has required modifications to the plants since they were built, including design changes to control hydrogen and pressure in the containment. The NRC has also required plants to have additional equipment and measures to mitigate damage stemming from large fires and explosions from a beyond-design-basis event. The measures include providing core and spent fuel pool cooling and an additional means to power other equipment on site.

Japanese Power Plants

Q: How many U.S. plants have designs similar to the affected Japanese reactors (and which ones)?

A: Thirty-five of the 104 operating nuclear power plants in the U.S. are boiling water reactors (BWRs), as are the reactors at Fukushima. Twenty-three of the U.S. BWRs have the same Mark I containment as the Fukushima reactors.

Two of the U.S. BWRs with a Mark I containment have an early nuclear steam supply system (NSSS) design designated as BWR-2. Six of the U.S. BWRs with Mark I containments have another early design, designated BWR-3, which are similar to Fukushima Unit 1. The remaining fifteen of the Mark I BWRs have the BWR-4 NSSS, similar to Fukushima Units 2, 3, and 4. The following table lists the operating BWRs in the United States.

Plant Name	NSSS Type	Containment Design	Location
Browns Ferry 1	BWR-4	Mark I	AL
Browns Ferry 2	BWR-4	Mark I	AL
Browns Ferry 3	BWR-4	Mark I	AL
Brunswick 1	BWR-4	Mark I	NC
Brunswick 2	BWR-4	Mark I	NC
Clinton	BWR-6	Mark III	IL
Columbia Generating Station	BWR-5	Mark II	WA
Cooper	BWR-4	Mark I	NE
Dresden 2	BWR-3	Mark I	IL
Dresden 3	BWR-3	Mark I	IL
Duane Arnold	BWR-4	Mark I	IA
Fermi 2	BWR-4	Mark I	OH
FitzPatrick	BWR-4	Mark I	NY
Grand Gulf 1	BWR-6	Mark III	MS
Hatch 1	BWR-4	Mark I	GA
Hatch 2	BWR-4	Mark I	GA
Hope Creek 1*	BWR-4	Mark I	NJ
La Salle 1	BWR-5	Mark II	IL
La Salle 2	BWR-5	Mark II	IL
Limerick 1	BWR-4	Mark II	PA
Limerick 2	BWR-4	Mark II	PA
Monticello	BWR-3	Mark I	MN
Nine Mile Point 1	BWR-2	Mark I	NY
Nine Mile Point 2	BWR-5	Mark II	NY
Oyster Creek	BWR-2	Mark I	NJ
Peach Bottom 2	BWR-4	Mark I	PA
Peach Bottom 3	BWR-4	Mark I	PA
Perry 1	BWR-6	Mark III	OH
Pilgrim 1	BWR-3	Mark I	MA
Quad Cities 1	BWR-3	Mark I	IL
Quad Cities 2	BWR-3	Mark I	IL
River Bend 1	BWR-6	Mark III	LA
Susquehanna 1	BWR-4	Mark II	PA
Susquehanna 2	BWR-4	Mark II	PA
Vermont Yankee	BWR-4	Mark I	VT

*has concrete secondary containment unlike other BWRs of this type

Fukushima Daiichi Recovery Efforts

Q: *What happens next in Japan? How long will it take to assess the damage to the reactors?*

A: The main focus is ensuring that adequate cooling of the reactor fuel and the spent fuel at each of the affected Japanese reactor facilities is established and maintained. Although much has been learned about conditions in each of the reactor buildings and associated spent fuel pools, the full extent of the damage to these facilities and specific conditions inside each reactor vessel is not yet known. However, efforts to obtain specific information about conditions inside each reactor vessel and containment building continue.

TEPCO (owner/operator of the Fukushima Daiichi facility) announced in late April that it established a plan toward restoring control of the Fukushima facility. The major aims of the plan involve two steps: (1) achieving a steady decline in radiation dose at the plant, and (2) bringing radioactive materials under control and significantly holding the radiation dose down. TEPCO categorized specific efforts under three major headings of “cooling,” “mitigation” and, “monitoring and decontamination.” These were further divided into the following five areas: (1) cooling the reactors, (2) cooling the spent fuel pools, (3) containing, storing, processing and reusing the water contaminated by radioactive materials (accumulated water), (4) mitigating radioactive materials in the atmosphere and soil, and (5) measuring, reducing and announcing the radiation doses in areas where evacuation has already taken place and where it is being planned, as well as areas where preparations are being made for emergency evacuation. The websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>) provide additional information on a daily basis.

Q: *Why did the seawater fail to cool the reactor?*

A: Based on information available to the NRC, it appears that the seawater has been effective at providing some cooling for the reactor. However, based on recent reports, the amount of cooling water provided to cool the reactor fuel, whether seawater or freshwater, was insufficient to prevent considerable melting of fuel in the reactor vessels.

Q: *What should the American public know about the incident in Japan?*

A: The events unfolding in Japan are the result of a catastrophic series of natural disasters. These include the fifth largest earthquake in recorded history and the resulting devastating tsunami. Despite these unique circumstances, the Japanese appear to have taken reasonable actions to mitigate the event and protect the surrounding population. Since the beginning of the event, the NRC has provided support to the Japanese government through the U.S. Ambassador to Japan, sent senior experience staff to Japan to provide technical assistance, and manned its Operations Center in Rockville, MD in order to gather and examine all available information as part of the effort to analyze the event and understand its implications both for Japan and the United States.

TEPCO (owner/operator of the Fukushima Daiichi facility) announced in late April that it established a plan toward restoring control of the Fukushima facility. The major aims of the plan involve two steps: (1) achieving a steady decline in radiation dose at the plant, and (2) bringing radioactive materials under control and significantly holding the radiation dose down. TEPCO categorized specific efforts under three major headings of “cooling,” “mitigation” and, “monitoring and decontamination.” These were further divided into the following five areas: (1) cooling the reactors, (2) cooling the spent fuel pools, (3) containing, storing, processing and reusing the water contaminated by radioactive materials (accumulated water), (4) mitigating radioactive materials in the atmosphere and soil, and (5) measuring, reducing and announcing the radiation doses in areas where evacuation has already taken place and where it is being planned, as well as areas where preparations are being made for emergency evacuation. The websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>) provide additional information on a daily basis.

Japanese Power Plants

Q: *What is the longer term prognosis for keeping the reactors cooled at the Japanese facilities?*

A: In the months since the earthquake and tsunami in Japan resulted in the catastrophe at the Fukushima nuclear facility, all efforts have been focused on cooling the reactors and spent fuel in the cooling pools. Although much has been learned about conditions in each of the reactor buildings and associated spent fuel pools, the full extent of the damage to these facilities and specific conditions inside each reactor vessel is not yet known. However, efforts to obtain specific information about conditions inside each reactor vessel and containment building continue. The NRC continues to coordinate with the Japanese government, private industry, other Federal Agencies, and the military to mitigate the cooling challenges and bring the events to a stable state of operation.

TEPCO (owner/operator of the Fukushima Daiichi facility) announced in late April that it established a plan toward restoring control of the Fukushima facility. The major aims of the plan involve two steps: (1) achieving a steady decline in radiation dose at the plant, and (2) bringing radioactive materials under control and significantly holding the radiation dose down. TEPCO categorized specific efforts under three major headings of “cooling,” “mitigation” and, “monitoring and decontamination.” These were further divided into the following five areas: (1) cooling the reactors, (2) cooling the spent fuel pools, (3) containing, storing, processing and reusing the water contaminated by radioactive materials (accumulated water), (4) mitigating radioactive materials in the atmosphere and soil, and (5) measuring, reducing and announcing the radiation doses in areas where evacuation has already taken place and where it is being planned, as well as areas where preparations are being made for emergency evacuation. The websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>) provide additional information on a daily basis.

Q: *Where can the public find the report from Japan to the International Atomic Energy Agency? Has the NRC reviewed this report?*

A: The Japanese have provided a sequence of events in a report to International Atomic Energy Agency that can be accessed through the IAEA website: <http://www.iaea.org/newscenter/focus/fukushima/japan-report/>. The NRC is currently reviewing the report. The NRC’s initial assessment is that is consistent with the agency’s understanding of the events that transpired at Fukushima 1 Daiichi following the March 11 earthquake and tsunami. At this early point in the review, no immediate actions for the NRC were identified beyond the Temporary Instructions and the Bulletin that were issued. The NRC is planning to thoroughly review the report and will identify any lessons learned that may be applicable in the United States.

Fukushima Daiichi Tsunami

Q: *Did the Japanese underestimate the size of the maximum credible earthquake and tsunami that could affect the plants?*

A: The magnitude of the earthquake was somewhat greater than was expected for that part of the subduction zone. However, the Japanese nuclear plants were recently reassessed using ground motion levels similar to those that are believed to have occurred at the sites. The ground motions against which the Japanese nuclear plants were reviewed were expected to result from earthquakes that were smaller, but were much closer to the sites. Although the NRC does not regularly have access to design information on foreign nuclear power plants, information regarding the maximum tsunami height that was expected at the sites is available at the following links:

<http://nei.org/newsandevents/information-on-the-japanese-earthquake-and-reactors-in-that-region/japan-earthquake-additional-nei-updates/japan-earthquake-nei-updates-for-monday-march-21/>

<http://www.tepco.co.jp/en/press/corp-com/release/11040910-e.html>

Japanese Power Plants

Q: *How high was the tsunami at the Fukushima nuclear plants?*

A: The tsunami modeling team at the National Oceanic and Atmospheric Administration's Pacific Marine Environmental Lab have estimated the wave height just offshore to be approximately 8 meters in height at Fukushima Daiichi and approximately 7 meters in Fukushima Daini. This is based on recordings from NOAA's Deep-ocean Assessment and Reporting of Tsunamis (DART) buoys and a high resolution numerical model developed for the tsunami warning system. The NRC does not normally have access to operating data for foreign nuclear power plants, however, information regarding the tsunami height may be accessed at the following links:

<http://nei.org/newsandevents/information-on-the-japanese-earthquake-and-reactors-in-that-region/japan-earthquake-additional-nei-updates/japan-earthquake-nei-updates-for-monday-march-21/>

<http://www.tepco.co.jp/en/press/corp-com/release/11040910-e.html>

Q: *Was the damage to the Japanese nuclear plants mostly from the earthquake or the tsunami?*

A: Because this event happened in Japan, the NRC relies primarily on information made available to it by the Japanese government and several organizations involved in responding, assessing, and mitigating the events at the Japanese nuclear plants. These include the Japanese regulator, the Nuclear and Industrial Safety Agency (NISA), Tokyo Electric Power Company (TEPCO), the operator of the Fukushima Daiichi nuclear plants, and the Japan Atomic Industrial Forum (JAIF). Radiation levels in certain areas of the nuclear plants make it very difficult for the NRC, as well as others, to assess the precise conditions of the facilities. Through TEPCO's continued efforts more specific information about the conditions of the plants is learned with each passing day. Based on the information provided by the Japanese, the NRC has learned that the nuclear plants may have sustained some damage from the ground shaking produced by the earthquake, and that the earthquake also caused the loss of offsite power. However, the tsunami appears to have played a key role in the loss of other power sources at the site producing station blackout, which is a critical factor in the ongoing problems. Additional information regarding the damage to the Japanese nuclear plants may also be obtained from the websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>).

Q: *The NRC Near-Term Task Force Report states that a sequence of events like the Fukushima accident is unlikely to occur in the U.S. and some appropriate mitigation measures have been implemented. What are those appropriate mitigation measures? NEW!*

A: The mitigation measures are what are commonly referred to as the B.5.b actions. These are the actions that were taken following the events of 9/11 in the United States. These measures would deal with the loss of large areas of the plant, including the use of portable equipment to provide some level of core cooling, spent fuel pool cooling and/or maintenance of containment integrity. They provide an additional level of mitigation capability that may be of assistance in the event of a significant accident similar to Fukushima.

Fukushima Daiichi U.S. Assistance

Q: *What resources are the Japanese asking for?*

A: The Japanese have formally requested equipment needed to cool the reactor fuel. This includes such things as pumps, fire hoses, portable generators, and diesel fuel. The NRC is coordinating with General Electric, which has plant design specifications, to ensure any equipment provided will be capable of meeting the needs of the Japanese.

Q: *A chart titled "NRC Dose Estimates" was posted on March 17, 2011, to the Yahoo Group "Know_Nukes," which plots total dose (Rem) vs. distance (miles) for a one reactor site and a four reactor site. Was this document released by the NRC?*

A: No, this document was not released by the NRC. The chart appears to plot the dose information that was included as attachments to the NRC press release of March 16, 2011. This press release provided NRC protective action recommendations for U.S. citizens residing within 50 miles of the Fukushima reactors. The NRC press release had two attachments that gave the results of dose assessments performed for the Fukushima Daiichi facility.

Japanese Power Plants

Q: *Did the NRC consult the Department of Energy (DOE) or the Nuclear Energy Institute (NEI) for assistance in developing the protective action recommendation?*

A: Although the DOE assisted in providing radiation dose rate information to support the analysis performed by the NRC, the protective action recommendation was made by the NRC.

Q: *What is the basis for the dose analyses attached to the March 16, 2011, NRC press release?*

A: The basis for the dose assessment was the limited and unverifiable information on the plant conditions at the Fukushima facility. The facility was modeled in a computer-based dose assessment code as a hypothetical, four reactor site. The dose assessment results are conservative predictions only and may not be representative of any actual radiation releases. The computer-based dose assessment model also utilized predicted meteorological conditions following the events at the Fukushima facility and, therefore, may not be representative of the actual meteorological conditions that occurred for this area. The NRC press release of March 16, 2011, and the predicted dose estimates are available on the NRC's public website and may be accessed at the following link: <http://www.nrc.gov/reading-rm/doc-collections/news/2011/11-050.pdf>.

The assumptions on plant conditions used as the basis for the analyses were indicative of the uncertain and unstable nature of the conditions on Fukushima Daiichi site at the time the analyses were done, and accounted for uncertainty in the future progression of events. Since that time, actions to mitigate the events at facility and to stabilize the reactors and spent fuel at the plant have continued. The NRC continues to support the protective action recommendations provided in the March 16, 2011, press release because conditions at the plant continue to change. The NRC continues to monitor the situation at the Fukushima facility and may reassess its protective action recommendations as additional detailed and verifiable information about actual conditions becomes available.

Q: *Why did the NRC decide to recommend evacuation out to 50 miles from the Fukushima Daiichi facility for U.S. citizens in Japan?*

A: The decision to expand evacuation of U.S. citizens out to 50 miles from the Fukushima Daiichi facility was a conservative decision that was made out of consideration of several factors including an abundance of caution resulting from limited and unverifiable information concerning event progression at several units at the Fukushima Daiichi facility. The NRC based its assessment on information available at the time regarding the condition of the units conditions at Fukushima Daiichi that included significant damage to Units 1, 2, and 3 that appeared to have been a result of hydrogen explosions. Prior to the earthquake and tsunami, Unit 4 was in a refueling outage and its entire core had been transferred to the spent fuel pool only 3 months earlier so the fuel was quite fresh. Radiation monitors showed significantly elevated readings in some areas of the plant site which would challenge plant crews attempting to stabilize the plant. Based on analysis results, there were indications from some offsite contamination sampling smears that fuel damage had occurred. There was a level of uncertainty about whether or not efforts to stabilize the plant in the very near term were going to be successful. Changing meteorological conditions resulted in the winds shifting rapidly from blowing out to sea to blowing back onto land.

Q: *The United States has troops in Japan and has sent ships to help the relief effort – are they in danger from the radiation?*

A: The NRC is not the appropriate federal agency to answer this question. DOD is better suited to provide information regarding its personnel.

Q: *What other U.S. agencies are involved, and what are they doing?*

A: The entire federal family is responding to this event. The NRC is closely coordinating its efforts with the White House, DOE, DOD, USAID, and others. The U.S. government is providing whatever support requested by the Japanese government.

Japanese Power Plants

Q: *What is the NRC doing in response to the situation in Japan?*

A: The NRC has taken a number of actions:

- (1) From March 11 until May 16, 2011, the NRC manned its Operations Center in Rockville, MD, in order to gather and examine all available information as part of the effort to analyze the event and understand its implications both for Japan and the United States.
- (2) A team of NRC staff with expertise in boiling water nuclear reactors have deployed to Japan as part of a U.S. International Agency for International Development (USAID) team and continues to provide support.
- (3) The NRC maintains communications with its counterpart agency in Japan, offering assistance and technical expertise as requested.
- (4) The NRC continues to coordinate its actions with other Federal agencies as part of the U.S. government response. In addition, the NRC's senior-level task force conducted a near-term analysis of the lessons that can be learned from the situation in Japan.

The Near Term Task Force has examined all the available information from Japan to understand the event's implications for the United States and performed a systematic and methodical review to see if there should be changes to be made to NRC programs and regulations to ensure protection of public health and safety. Its report was issued on July 12 and is available to the public (ADAMS Accession No. ML111861807). On July 19, 2011, the Task Force presented its findings to the Commission in a meeting open to the public and proposed improvements in areas ranging from loss of power to earthquakes, flooding, spent fuel pools, venting and emergency preparedness. The Task Force discussed its report and recommendations with the public on July 28, 2011, in NRC headquarters in Rockville, Maryland.

The NRC has issued the following documents related to the events in Japan:

- Information Notice 2011-05 provided information to licensees on the effects of the earthquake and resultant tsunami on the Fukushima Daiichi nuclear power station in Japan.
- Temporary Instruction 2515/183 provided instructions for NRC inspectors to perform independent assessments of the adequacy of industry-initiated efforts to respond to the fuel damage events at the Fukushima Daiichi nuclear station. This involves a high-level look at industry's preparedness for events that may exceed the design for a plant.
- Temporary Instruction 2515/184 provided instructions for NRC inspectors to determine: (i) that the severe accident management guidelines (SAMGs) are available and how they are being maintained, and (ii) the nature and extent of licensee implementation of SAMG training and exercises.
- Bulletin 2011-01 required all holders of operating licenses for nuclear power reactors to provide a comprehensive verification of their compliance with the regulatory requirements in 10 CFR 50.54(hh) associated with mitigating strategies for beyond design basis events.

Japanese Power Plants

Q: *What is the NRC doing to ensure this (Japan event) doesn't happen at US plants?*

A: The NRC continues to conclude that US nuclear plants are safe. Since the beginning of the event, the NRC has provided support to the Japanese government through the U.S. Ambassador to Japan, sent senior experience staff to Japan to provide technical assistance, and manned its Operations Center in Rockville, MD in order to gather and examine all available information as part of the effort to analyze the event and understand its implications both for Japan and the United States.

The NRC has established a senior level task force to conduct both short- and long-term analysis of the lessons that can be learned from the situation in Japan. The task force is examining all the available information from Japan to understand the event's implications for the United States. They are performing a systematic and methodical review to see if there are changes that should be made to NRC programs and regulations to ensure protection of public health and safety. This will undoubtedly lead to the identification of issues that warrant further study in the longer term. The task force is scheduled to provide a report to the Commission in July 2011 identifying the results of its review and providing recommendations for short-term action, if necessary, and longer-term study.

The NRC has also issued the following documents related to the events in Japan:

- Information Notice 2011-05 provided information to licensees on the effects of the earthquake and resultant tsunami on the Fukushima Daiichi nuclear power station in Japan.

- Temporary Instruction 2515/183 provided instructions for NRC inspectors to perform independent assessments of the adequacy of industry-initiated efforts to respond to the fuel damage events at the Fukushima Daiichi nuclear station. This involves a high-level look at industry's preparedness for events that may exceed the design for a plant.

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- Bulletin 2011-01 required all holders of operating licenses for nuclear power reactors to provide a comprehensive verification of their compliance with the regulatory requirements in 10 CFR 50.54(hh) associated with mitigating strategies for beyond design basis events.

Q: *How did the NRC develop its computer-based projections that supported the evacuation decision?*

A: The NRC uses the RASCAL computer code to perform offsite radiation dose projections. The RASCAL computer program contains information about U.S. nuclear reactor design types, radiation release pathways from the nuclear power plant to the environment, radionuclide source terms and meteorology. However, RASCAL is not capable of evaluating concurrent and multiple nuclear plant failures. So, to approximate the events unfolding at the Fukushima Daiichi facility, the NRC developed a model that aggregated information from the three operating reactors and the spent fuel pool. This aggregate model was then evaluated using the RASCAL computer code. The radiation doses calculated by the RASCAL code were predicted to exceed the protective action guidelines (PAGs) established by the U.S. Environmental Protection Agency (EPA) well beyond the 10-mile exposure pathway EPZ and beyond the 30 kilometer sheltering zone recommended by the Japanese authorities. Subsequent aerial monitoring by the U.S. Department of Energy (DOE) fixed-wing aircraft monitoring showed elevated radiation dose rates that were in excess of the EPA relocation PAGs to a distance beyond 25 miles from the facility.

Q: *What is the NRC doing about the emergencies at the nuclear power plants in Japan? Are you sending staff over there?*

A: We are closely following events in Japan, working with other agencies of the federal government, and have been in direct contact with our counterparts in that country. The NRC has sent several staff to Tokyo to assist with this emergency by working through the U.S. Ambassador in response to the Japanese government's request for assistance. The NRC continues to support the Japanese efforts with various staff members both in Japan and at the NRC headquarters in Rockville, MD.

Japanese Power Plants

Fukushima Daini Earthquake

Q: *Did the Japanese underestimate the size of the maximum credible earthquake and tsunami that could affect the plants?*

A: The magnitude of the earthquake was somewhat greater than was expected for that part of the subduction zone. However, the Japanese nuclear plants were recently reassessed using ground motion levels similar to those that are believed to have occurred at the sites. The ground motions against which the Japanese nuclear plants were reviewed were expected to result from earthquakes that were smaller, but were much closer to the sites. Although the NRC does not regularly have access to design information on foreign nuclear power plants, information regarding the maximum tsunami height that was expected at the sites is available at the following links:

<http://nei.org/newsandevents/information-on-the-japanese-earthquake-and-reactors-in-that-region/japan-earthquake-additional-nei-updates/japan-earthquake-nei-updates-for-monday-march-21/>

<http://www.tepco.co.jp/en/press/corp-com/release/11040910-e.html>

Q: *Was the damage to the Japanese nuclear plants mostly from the earthquake or the tsunami?*

A: Because this event happened in Japan, the NRC relies primarily on information made available to it by the Japanese government and several organizations involved in responding, assessing, and mitigating the events at the Japanese nuclear plants. These include the Japanese regulator, the Nuclear and Industrial Safety Agency (NISA), Tokyo Electric Power Company (TEPCO), the operator of the Fukushima Daiichi nuclear plants), and the Japan Atomic Industrial Forum (JAIF). Radiation levels in certain areas of the nuclear plants make it very difficult for the NRC, as well as others, to assess the precise conditions of the facilities. Through TEPCO's continued efforts more specific information about the conditions of the plants is learned with each passing day. Based on the information provided by the Japanese, the NRC has learned that the nuclear plants may have sustained some damage from the ground shaking produced by the earthquake, and that the earthquake also caused the loss of offsite power. However, the tsunami appears to have played a key role in the loss of other power sources at the site producing station blackout, which is a critical factor in the ongoing problems. Additional information regarding the damage to the Japanese nuclear plants may also be obtained from the websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>).

Fukushima Daini Event Progression

Q: *Where can the public find the report from Japan to the International Atomic Energy Agency? Has the NRC reviewed this report?*

A: The Japanese have provided a sequence of events in a report to International Atomic Energy Agency that can be accessed through the IAEA website: <http://www.iaea.org/newscenter/focus/fukushima/japan-report/>. The NRC is currently reviewing the report. The NRC's initial assessment is that is consistent with the agency's understanding of the events that transpired at Fukushima 1 Daiichi following the March 11 earthquake and tsunami. At this early point in the review, no immediate actions for the NRC were identified beyond the Temporary Instructions and the Bulletin that were issued. The NRC is planning to thoroughly review the report and will identify any lessons learned that may be applicable in the United States.

Fukushima Daini Lessons Learned

Q: *Have any lessons for US nuclear plants been identified?*

A: The NRC is continuing to follow and review the events in Japan in real time. The NRC established a senior level task force to conduct both short- and long-term analysis of the lessons that can be learned from the situation in Japan. The task force has completed its near-term analysis of the events and their impact on U.S. plants ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission). The report included 12 overarching recommendations to clarify the NRC's regulatory framework for events of low probability but high consequence, assure protection against accidents caused by severe natural phenomena, enhance mitigation of accidents should they occur, strengthen emergency preparedness for protection of the public, and improve the effectiveness of NRC's programs. The Commission will review the task force report and will provide further direction to the staff. The task force is also going to perform a longer term study based, in part, on recommendations for further study in the near-term report.

Japanese Power Plants

Q: *Have events in Japan changed our perception of earthquake risk to the nuclear plants in the US?*

A: The NRC continues to determine that US nuclear plants are safe. The Japanese quake does not change the NRC's perception of earthquake hazard (i.e., ground motion levels) at US nuclear plants. Even before the events in Japan, the NRC began reviewing the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting a generic review referred to as GI-199, "Implications of Updated Probabilistic Seismic Estimates in Central and Eastern United States on Existing Plants," to again assess the resistance of US nuclear plants to earthquakes. In addition, the NRC has been reviewing updated seismic information regarding the plants in California for many years. The NRC has established a senior level task force to identify areas of further evaluation as a result of the Japanese events. The task force has issued its near-term report, which recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection for each operating reactor. The Commission will review the report and will provide direction to the staff.

Fukushima Daini Reactor Design

Q: *Can significant damage to a nuclear plant like we see in Japan happen in the US due to an earthquake? Are the Japanese nuclear plants similar to US nuclear plants?*

A: All US nuclear plants are built to withstand environmental hazards, including earthquakes and tsunamis. Even those nuclear plants that are located within areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. The NRC requires that safety-significant structures, systems, and components be designed to take into account even rare and extreme seismic and tsunami events. In addition to the design of the plants, significant effort goes into emergency response planning and accident management. This approach is called defense-in-depth.

The Japanese facilities are similar in design to some US facilities. However, the NRC has required modifications to the plants since they were built, including design changes to control hydrogen and pressure in the containment. The NRC has also required plants to have additional equipment and measures to mitigate damage stemming from large fires and explosions from a beyond-design-basis event. The measures include providing core and spent fuel pool cooling and an additional means to power other equipment on site.

Fukushima Daini Recovery Efforts

Q: *Where can the public find the report from Japan to the International Atomic Energy Agency? Has the NRC reviewed this report?*

A: The Japanese have provided a sequence of events in a report to International Atomic Energy Agency that can be accessed through the IAEA website: <http://www.iaea.org/newscenter/focus/fukushima/japan-report/>. The NRC is currently reviewing the report. The NRC's initial assessment is that is consistent with the agency's understanding of the events that transpired at Fukushima 1 Daiichi following the March 11 earthquake and tsunami. At this early point in the review, no immediate actions for the NRC were identified beyond the Temporary Instructions and the Bulletin that were issued. The NRC is planning to thoroughly review the report and will identify any lessons learned that may be applicable in the United States.

Fukushima Daini Tsunami

Q: *Did the Japanese underestimate the size of the maximum credible earthquake and tsunami that could affect the plants?*

A: The magnitude of the earthquake was somewhat greater than was expected for that part of the subduction zone. However, the Japanese nuclear plants were recently reassessed using ground motion levels similar to those that are believed to have occurred at the sites. The ground motions against which the Japanese nuclear plants were reviewed were expected to result from earthquakes that were smaller, but were much closer to the sites. Although the NRC does not regularly have access to design information on foreign nuclear power plants, information regarding the maximum tsunami height that was expected at the sites is available at the following links:

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<http://www.tepco.co.jp/en/press/corp-com/release/11040910-e.html>

Japanese Power Plants

Q: *How high was the tsunami at the Fukushima nuclear plants?*

A: The tsunami modeling team at the National Oceanic and Atmospheric Administration's Pacific Marine Environmental Lab have estimated the wave height just offshore to be approximately 8 meters in height at Fukushima Daiichi and approximately 7 meters in Fukushima Daini. This is based on recordings from NOAA's Deep-ocean Assessment and Reporting of Tsunamis (DART) buoys and a high resolution numerical model developed for the tsunami warning system. The NRC does not normally have access to operating data for foreign nuclear power plants, however, information regarding the tsunami height may be accessed at the following links:

<http://nei.org/newsandevents/information-on-the-japanese-earthquake-and-reactors-in-that-region/japan-earthquake-additional-nei-updates/japan-earthquake-nei-updates-for-monday-march-21/>

<http://www.tepco.co.jp/en/press/corp-com/release/11040910-e.html>

Q: *Was the damage to the Japanese nuclear plants mostly from the earthquake or the tsunami?*

A: Because this event happened in Japan, the NRC relies primarily on information made available to it by the Japanese government and several organizations involved in responding, assessing, and mitigating the events at the Japanese nuclear plants. These include the Japanese regulator, the Nuclear and Industrial Safety Agency (NISA), Tokyo Electric Power Company (TEPCO), the operator of the Fukushima Daiichi nuclear plants, and the Japan Atomic Industrial Forum (JAIF). Radiation levels in certain areas of the nuclear plants make it very difficult for the NRC, as well as others, to assess the precise conditions of the facilities. Through TEPCO's continued efforts more specific information about the conditions of the plants is learned with each passing day. Based on the information provided by the Japanese, the NRC has learned that the nuclear plants may have sustained some damage from the ground shaking produced by the earthquake, and that the earthquake also caused the loss of offsite power. However, the tsunami appears to have played a key role in the loss of other power sources at the site producing station blackout, which is a critical factor in the ongoing problems. Additional information regarding the damage to the Japanese nuclear plants may also be obtained from the websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>).

American Centrifuge Plant

Q: *Are special procedures employed at the ACP for other natural phenomena?*

A: Yes. The ACP's Emergency Plan describes emergency measures to be taken in response to emergencies such as accidents or natural phenomena events (i.e., earthquake, flood, high winds/tornadoes, lightning strikes, and snow load hazards). It describes the protective actions to be implemented on-site and off-site to ensure exposures of personnel and members of the public are limited in case of an accidental release of licensed material to the environment.

On-site protective actions include evacuation, shelter in place, accountability, search and rescue, and monitoring and decontamination.

For off-site protective actions the ACP's Incident Commander (IC) is responsible for providing protective action recommendations to local officials. These recommendations are based on assessment actions and an understanding of the actual or potential conditions. Recommendations include sheltering in place, evacuation or advisories that no action is needed. Appropriate off-site authorities are responsible for alerting and notifying the public on the recommended off-site protective actions.

Q: *Are the ACP emergency power diesel generators built to withstand the effects of an earthquake? If not what happens when power to the facility is lost?*

A: The ACP maintains emergency power diesel generators that have been designed for key operational areas/buildings. The ACP maintains, inspects, and tests the emergency power diesel generators to ensure they are capable of activation in the event power is lost to the facility for they are supporting. The emergency power diesel generators are designed to support the critical items within a facility until such time that primary power can be restored. The existing buildings utilized by the ACP are designed and constructed to withstand a 1,000-year seismic event. All newly constructed buildings, including extensions to existing buildings, are designed to withstand a 1,000-year seismic event. In some cases, uninterruptable power supplies and batteries are in place. Although the emergency power diesels are not specifically designed for a seismic event, the facilities for they are in, or supporting, are designed to withstand a 1,000-year seismic event.

Key facilities within ACP, or supporting ACP, with emergency power diesels include: X-104 Guard Headquarters, X-112 Computer Support Facility, X-300 Plant Control Facility, X-1007 Fire Station, X-1020 Emergency Operations Center, X-3001 Process Building, and X-3012 Process Support Building. The X-3002, X-3346, and X-7725 will have emergency power diesels as well.

The X-640-1 and X-6644 Fire Water Pump Houses are equipped with emergency diesel fire water pumps.

Q: *What is the design basis earthquake for the ACP?*

A: Seismic specifications for the ACP design are based on the risks and potential consequences from seismic events involving the primary facilities. This approach results in two criteria being applied depending upon whether or not the normal operations therein involve liquid UF₆. Facilities where liquid UF₆ operations occur are required to withstand the forces resulting from a 10,000-year return period seismic event (Peak Ground Acceleration - PGA 0.48 G). All other primary facilities are required to withstand the forces resulting from a 1,000-year return period seismic event (PGA 0.15 G) because UF₆ operations therein involve UF₆ in either gas or solid form.

The existing buildings utilized by the ACP are designed and constructed to withstand a 1,000-year seismic event. All newly constructed buildings, including extensions to existing buildings, are designed to withstand a 1,000-year seismic event. The Customer Services Building will be the only building within the complex that will handle liquid UF₆ and is designed to withstand a 10,000-year seismic event. Process-related equipment in the ACP buildings will also be seismically qualified to meet the 1000-year or 10,000-year return earthquake, as applicable. Therefore, while the unprevented frequency of a seismic event that would be expected to impact these buildings is estimated to be "Not Unlikely," no release of hazardous material is expected during a design basis seismic event.

U.S. Fuel Cycle Facilities - NEW!

Q: *What are the potential impacts of an earthquake that exceeded the design of the ACP facility?*

A: While a seismic event of a magnitude beyond the design of the facility could result in damage to the buildings and equipment, a review of design calculations for these buildings show significant reserve with respect to the design basis seismic capacity. The models and assumptions used in these calculations indicate that the capacity of the structural components would exceed the basic requirements with significant margin. If a seismic event were to occur such that there was a breach of process systems and equipment containing UF6 in these facilities, it is unlikely that a release of licensed material would produce any significant impact to the off-site public because the systems and equipment operate below atmospheric conditions, contain low inventories of licensed material, and the structures and equipment are not expected to fail catastrophically. Previous experience with similar equipment (at the GDP) has shown that a breach in equipment operating below atmospheric pressure has resulted in minimal release to the environment (the ACP operates at much lower pressures than the GDP). Besides, some of the ACP equipment (i.e., autoclaves with roll or tilt capability) are designed to withstand a 100,000-year return period seismic event.

The Customer Service building would be expected to survive the beyond design basis earthquake without experiencing a release of hazardous material that would produce a significant impact to the off-site public. However, since UF6 in liquid state could be located within the building during a seismic event, the building is considered to be a high seismic risk location and, as such, the margin of safety for the design has been increased appropriately. Liquid UF6 would be in cylinders and piping, all of which in turn would be located in autoclaves. Given this configuration, it is unlikely that the seismic damage suffered would be sufficient to breach a cylinder or process piping. Therefore, no release of hazardous material would result.

Q: *What would happen if an earthquake occurred in the vicinity of the ACP?*

A: There are no major geologic fault structures in the vicinity of the ACP and there have been no historical earthquake epicenters within 25 miles from the site. However, there have been eight earthquake epicenters within 50 miles. The maximum event had an epicenter intensity of over IV on the Modified Mercalli (MM) scale. These events had intensities between I and IV. The maximum peak ground acceleration (PGA) of a MM level IV event roughly corresponds to 0.02 G. Historically, the maximum earthquake-induced PGA experienced at the reservation was in 1955 and had a value of only 0.005 G.

Independent calculations and a review of the seismic hazard analyses for the reservation were performed by the USGS and the results were documented. For a return period of 500 years, the PGA was determined to have a value of 0.10 G. For a return period of 250 years, the PGA was determined to have a value of 0.05 G. Earthquakes with large ruptures are highly unlikely to occur near the reservation because of low values of maximum magnitude.

Q: *How would the NRC and USEC respond to an earthquake at the ACP facility?*

A: NRC and the ACP would respond consistent with standard procedures described in its Emergency Plan.

Q: *What happens if the ACP were to lose offsite power as a result of an earthquake?*

A: Should an earthquake occur that causes the loss of offsite power, emergency backup power is or will be available to support an orderly shutdown of the enrichment processes. The enrichment processes are based on fail-safe operation so there are no active safety systems that are required to support safe shutdown. However, critical business operations do indicate a need for cost effective shutdown of enrichment equipment.

The ACP essential electrical system designs shall provide for continued operation of essential utility services in accordance with 10 CFR 70.64(a)(7). The ACP's essential electrical systems (EES) will be provided reliable electrical power as recommended by Articles 700 and 702 of ANSI C1/NFPA 70-2005. Each EES will be connected to a power source derived from the X-5000 Substation via a local single or double-ended 480V substation and another power source consisting of a standby diesel-generator set. This configuration will ensure reliable power for the essential electrical systems that support various systems that are necessary to protect life safety, maintain critical communications systems, and protect valuable process equipment.

Installation of cables, cable trays, and conduit systems will comply with ANSI C1/NFPA 70-7005. The cables will be suitable for their environment (hot, cold, wet, dry, and/or corrosive) and comply with applicable flame retardancy requirements. Physical supports for conduits, trays, panels, and cabinets will equal or exceed ANSI C1/NFPA 70-2005 requirements.

Areva Richland (NP)

Q: *What are the most significant hazards at the Areva Richland site?*

A: A criticality accident represents the potential for a lethal radiation dose to a worker within 10 to 50 feet of the criticality, and lesser but significant doses out to 100 feet or more. The total radiation to an individual at the site boundary would exceed regulatory limits, but would not result in the potential for radiation doses large enough to cause injury. There are no criticality scenarios that would be initiated by an earthquake or loss of offsite power event.

Q: *What is the design basis earthquake for the Areva Richland fuel fabrication facility?*

A: There is no design basis earthquake per se. Buildings were constructed in accordance with the local commercial building code in effect at the time of construction. Buildings constructed recently were designed to meet the seismic load resistance specified in the code for UBC Zone IIB. The code design loads are based on an earthquake with a 2,500 year return period and a peak ground acceleration of 0.20 g. The peak ground acceleration for a ground motion with an annual exceedance probability of 10⁻⁴ is 0.398g.

The staff determined that it would not be likely that the structures at the facility would suffer severe damage from a seismic event with a peak ground acceleration of 0.398g, because of sufficient safety margin associated with the design.

Q: *What are the potential impacts of an earthquake that exceeded the design of the Areva Richland facility?*

A: Building codes incorporate occupant safety margins such as maintaining structural integrity long enough to allow occupants to leave the building. Maintaining structural integrity would reduce the damage that could lead to fires and releases of hazardous materials. If an earthquake exceeded the design of the facility, structural failure would be more likely increasing the risk that occupants would be injured and unable to escape.

The likelihood of fires and releases would increase. Although offsite impacts may increase, the potential for radiation doses large enough to cause an acute fatality or early injury to a member of the public is not considered plausible.

Q: *What would happen if an earthquake occurred in the vicinity of the Areva Richland facility?*

A: A significant earthquake could cause damage resulting in fires or the release of hazardous materials. Earthquakes are considered an unlikely cause of a criticality accident because such accidents require enriched uranium must be accumulated in an unsafe, critical mass. The contents of broken pipes and containers tend to be dispersed, not accumulated.

Q: *How would the NRC and the licensee respond to an earthquake at the Areva Richland facility?*

A: The NRC has a resident inspector at the nearby Columbia generating Station facility. NRC Headquarters and Region IV Emergency Response staff from the Fuel Cycle Safety Team and Protective Measures Team would monitor the situation.

Q: *What preparations are currently in place to respond to such an emergency at Areva Richland?*

A: The site Emergency Plan contains instructions and Protective Action Recommendations (PARs) for the postulated emergencies. The facility has an onsite Plant Emergency Response Team with medical, incipient fire fighting, and incipient hazardous material response capabilities. Agreements are in place with the City of Richland fire department, police, and hospital. For these types of events Notifications are made to the Richland Emergency Dispatch Center, which in turn notifies both Benton and Franklin Counties Emergency Management personnel and the City of Richland. Additional assistance is available from the DOE Hanford fire department, approximately ½ mile away. The DOE Richland Operations Office is notified. The Washington State Emergency Management Division is notified.

Q: *What happens if the Areva Richland facility were to lose offsite power as a result of an earthquake?*

A: There are no safety significant scenarios which would result from a loss of offsite power. Back-up power batteries are installed to allow an orderly shutdown of plant operations.

U.S. Fuel Cycle Facilities - NEW!

Q: *Are the Areva Richland emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: The emergency diesels are not seismically qualified. There are no safety significant scenarios which would result from a loss of offsite power. Risk-significant safety controls are passive or fail to a safe position without relying upon electric power.

Q: *What would be the impact of a tsunami or flood at the Areva Richland site?*

A: The site is located approximately 120 miles from the Pacific Ocean. The Columbia River is approximately 1 1/2 miles east of the site. The facility is located approximately 25 feet above the Columbia River and 370 feet above sea level.

The combination of the Probable Maximum Flood (PMF) and failure of any of the dams on the Columbia River upstream of the Hanford area has been calculated to be 1 E -6. It is also noted that following an upstream event, some time would be available for site personnel to place the facility in a safe shutdown mode.

Q: *How many people live near the Areva Richland site?*

A: The closest resident to the facility is approximately 1 1/2 miles to the southwest in the City of Richland.

Offsite population totals are as follows:

1 mile: 0
2 miles: 376
3 miles: 5020
4 miles: 15,560

Q: *Does the Areva Richland site have a spent fuel storage area?*

A: No.

Babcock and Wilcox Nuclear Operations Group (B&W NOG)

Q: *Does the B&W NOG site have a spent fuel storage area?*

A: The Lynchburg Technology Center has a cask unloading pool. Note, however, that unlike a power reactor site, the LTC does not possess large quantities of fuel elements immediately following refueling, and is limited to a maximum of 4 irradiated fuel assemblies.

It is possible that a shielded cask could fall into the pool, causing a fuel element to rupture. The worst case would be the sudden and complete release of fission gases in the transfer canal. Normally, these gases are filtered by HEPA filters and flow up the stack. A significant earthquake which caused damage to the filters, exhaust system, or stack would not result in the potential for radiation doses large enough to cause an acute fatality or early injury to a member of the public.

Q: *What is the design basis earthquake for the B&W NOG facility?*

A: The B&W NOG site is located near the western limit of the Piedmont physiographic province. Seismic activity in the Central Virginia region is classified as moderate. The site falls within the western part of the Central Virginia cluster region which is classified as Seismic Zone 2, indicating that moderate damage could occur as the result of earthquakes. Since 1774, there have been 18 earthquakes reported as having an intensity VI or higher on the Modified Mercalli Scale, defined as "felt by all, many frightened and run indoors, falling plaster and chimney bricks, damage small." It is comparable to 4.5 on the Richter scale.

The building structures are designed to meet the requirements of the IBC (International Building Code) National Building Code as adopted by the Commonwealth of Virginia. The IBC requires a design based on maximum earthquake spectral response acceleration for periods of 0.2 and 1 second. The peak ground acceleration is 0.26g and 0.09g for 0.2 and 1 second periods, respectively. The peak ground acceleration may be equated to an earthquake with a return period of 2500 years.

U.S. Fuel Cycle Facilities - NEW!

Q: *What are the most significant hazards at the B&W NOG site?*

A: A criticality accident represents the potential for a lethal radiation dose to a worker within 10 to 50 feet of the criticality, and lesser but significant doses out to 100 feet or more. The total radiation to an individual at the site boundary would exceed regulatory limits, but would not result in the potential for radiation doses large enough to cause injury. There are no criticality scenarios that would be initiated by an earthquake or loss of offsite power event.

Q: *What preparations are currently in place to respond to such an emergency at the B&W NOG?*

A: The site Emergency Plan contains instructions and Protective Action Recommendations (PARs) for the postulated emergencies. The facility has an onsite Emergency Team with medical, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, and hospital. For these types of events B&W NOG notifies the Virginia Department of Emergency Services, and Campbell, Amherst, and Appomattox counties. Recommendations are based on the release quantity/chemical/physical state and atmospheric conditions and are discussed with the agencies listed above. Campbell County officials are the decision-makers for evacuations.

Q: *How many people live near the B&W NOG site?*

A: The closest resident to the facility is approximately 4,500 feet directly west. The nearest potential off-site worker would be an occupational worker at the AREVA facility, approximately 3,000 feet to the northeast.

Offsite population totals are as follows:

1 mile: 302
2 miles: 1100
3 miles: 2424
4 miles: 4557
5 miles: 9070

Q: *What would be the impact of a tsunami, hurricane, or flood at the B&W NOG site?*

A: B&W NOG is located approximately 170 miles from the Atlantic Ocean and at an elevation of 820 feet above sea level where any impact from a hurricane or tsunami is very unlikely. The James River borders three sides of the site. Flooding of the James River occurs infrequently. There have been 11 significant floods since 1771, which averaged 28 feet above the normal river elevation. The main manufacturing facility is located approximately 110 feet above the river, and 75 feet above the 100-year flood level.

Q: *Are the B&W NOG emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: The emergency diesels are not seismically qualified. There are no safety significant consequences that would result from a loss of offsite power. Risk-significant safety controls are passive or fail to a safe position without relying upon electric power.

Q: *What happens if the B&W NOG facility were to lose offsite power as a result of an earthquake?*

A: There are no safety significant consequences that would result from a loss of offsite power. Back-up power supplies are installed to allow an orderly shutdown of plant operations. There are no plant processes that require continuous electrical power after shutdown.

Q: *What are the potential impacts of an earthquake that exceeded the design of the B&W NOG facility?*

A: Building codes incorporate occupant safety margins such as maintaining structural integrity long enough to allow occupants to leave the building. Maintaining structural integrity would reduce the damage that could lead to fires and releases of hazardous materials. If an earthquake exceeded the design of the facility, structural failure would be more likely increasing the risk that occupants would be injured and unable to escape.

The likelihood of a high consequence event would increase. Although offsite impacts may increase, the potential for radiation doses large enough to cause an acute fatality or early injury to a member of the public is not considered plausible.

U.S. Fuel Cycle Facilities - NEW!

Q: *What would happen if an earthquake occurred in the vicinity of the B&W NOG facility?*

A: A significant earthquake could cause damage resulting in fires or the release of hazardous materials. Earthquakes are considered an unlikely cause of a criticality accident because such accidents require enriched uranium must be accumulated in an unsafe, critical mass. The contents of broken pipes and containers tend to be dispersed, not accumulated.

Q: *How would the NRC and the licensee respond to an earthquake at the B&W NOG facility?*

A: The NRC has a resident inspector at the facility. NRC Headquarters and Region II Emergency Response staff from the Fuel Cycle Safety Team and Protective Measures Team would monitor the situation. If warranted by events, NRC will dispatch an emergency site team to monitor licensee activities and serve in an advisory role to state and local officials who may be considering protective actions to further ensure the protection on the public.

Eagle Rock Enrichment Facility (EREF)

Q: *What are the potential impacts of an earthquake that exceeds the design of the EREF?*

A: Although the facility is designed to withstand an earthquake with a one in 10,000 chance of occurring, analysis of the design shows that the facility design has sufficient margin to be able to maintain radiological safety even if it is shaken by an earthquake with a one in 100,000 chance of occurrence.

In the case of an earthquake which leads to a breach in the piping for the uranium hexafluoride (UF₆) systems, it is assumed that there would be a release of UF₆. Although the uranium feed material is radioactive, the primary consideration with regard to human health and safety is chemical rather than radiological. Thus, in the event of an unmitigated release, the chemical effects are greater health and environmental concerns than the amount of radiation that might be released. The NRC has modeled the potential consequences from such a release. Based on its modeling, NRC has found that the consequences to workers are potentially high, while consequences to offsite public are low (below the appropriate Acute Exposure Guideline Level). Mitigating measures will further reduce the consequences.

Q: *How would the NRC and EREF respond to an emergency, such as an earthquake, at the facility?*

A: NRC regulations require that the proposed EREF have an emergency plan. The emergency plan contains onsite and offsite Protective Action Recommendations (PARs) for emergencies. The EREF will have an onsite Emergency Response Organization with first aid, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, and hospital. In the event of an alert or site area emergency, as part of the emergency plan, EREF would notify key offsite agencies, namely the Bonneville County Emergency Management Services, Idaho Bureau of Homeland Security, and the NRC Emergency Operations Center. Once notified, NRC staff would monitor the situation.

Q: *What happens if the EREF were to lose offsite power?*

A: Items relied on for safety for the proposed EREF will be designed to maintain their safety functions or to fail into a safe state in the event of a loss of off-site power. Standby diesel generators will be provided for investment protection purposes only.

Q: *How many people live near the EREF?*

A: The distance to the nearest resident to the proposed EREF is approximately 5 miles. The population density around the site and region is generally low. The nearest population center is Idaho Falls which is about 20 miles southeast of the site. Its estimated population is 52,786 (based on 2006 data).

Q: *What would be the impact of a tsunami or flood at the EREF?*

A: The proposed EREF site is located inland, approximately 575 miles from the coast. Additionally, the site is not located near any large body of water that could cause a flood at the site. The nearest large surface waters are the Snake River which is about 20 miles east and Lake Wolcott which is approximately 75 miles southwest of the proposed site.

U.S. Fuel Cycle Facilities - NEW!

Q: *Are special procedures employed at the EREF for other natural phenomena?*

A: As part of the license review, the resistance of the proposed EREF to natural phenomena was evaluated, including consideration of: seismic hazards, volcanic hazards, tornado hazard, high winds, extreme precipitation, flood, snow, and lightning.

Q: *What is the design basis earthquake for the EREF?*

A: The license application was submitted in December 2008 and is currently under review. The applicant submitted and staff reviewed a site-specific probabilistic seismic hazard assessment (PSHA) for the proposed EREF. The design basis earthquake for the EREF has a peak ground acceleration of 0.16 g. The EREF design will allow it to withstand, without serious consequences to the public, the effects of earthquakes that have less than a one in 100,000 chance of occurring.

The proposed EREF site is situated in a seismically inactive region of the Eastern Snake River Plain. The largest historical earthquake to strike the Snake River Plain was the 1905 Shoshone earthquake, with an estimated magnitude of between 5.3 and 5.7.

GE-Hitachi Laser-Based Uranium Enrichment Facility (GLE)

Q: *What happens if the GLE facility were to lose offsite power as a result of an earthquake?*

A: The NRC staff is currently in the process of evaluating GLE's license application.

Q: *What is the design basis earthquake for the GE-Hitachi Laser-Based Uranium Enrichment Facility (GLE)?*

A: The NRC staff is currently in the process of evaluating GLE's license application. The proposed site of the facility is located in Wilmington, North Carolina. This location is considered a low earthquake hazard area. The proposed facility is located inland 10 miles west and 26 miles north of the Atlantic Ocean. In the guidance is NUREG/CR-6966, "Tsunami Hazard Assessment at Nuclear Power Plant Sites in the United States of America," this location is greater than 1 mile inland from the coast and is not considered susceptible to a tsunami.

Q: *Are special procedures employed at GLE for other natural phenomena events?*

A: The NRC staff is currently in the process of evaluating GLE's license application.

Q: *Are the GLE emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: The NRC staff is currently in the process of evaluating GLE's license application.

Q: *What would happen if an earthquake occurred in the vicinity of the GLE facility?*

A: The NRC staff is currently in the process of evaluating GLE's license application.

Q: *What are the potential impacts of an earthquake that exceeded the design of the GLE facility?*

A: The NRC staff is currently in the process of evaluating GLE's license application.

Q: *How would the NRC and GLE respond to an earthquake at the GLE facility?*

A: NRC and GLE would respond consistent with standard procedures for emergency events. GLE procedures are described in its Radiological Contingency and Emergency Plan.

Generic Criticality Safety

Q: *Can a U.S. fuel cycle facility have a criticality accident as the result of an earthquake?*

A: In general, fuel cycle facilities are constructed to the local Uniform Building Code so as to withstand anticipated earthquakes. Facilities' Integrated Safety Analyses (ISAs) also consider natural phenomena hazards, including earthquakes and severe weather, and must demonstrate that chemical and radiological hazards, including criticality, have an acceptable level of risk. For example, criticality accidents must be shown to be highly unlikely.

Criticality, in general, requires the accumulation of a sufficient mass of nuclear material into a compact geometry, such as a sphere. It also requires a certain quantity of moderator, materials that slow down neutrons and enhance their ability to cause fission, the most common of which is water. During an earthquake, nuclear material would tend to be dispersed over a wide area, the opposite of what is needed for criticality. Nuclear facilities are required to be designed so that at least two unlikely, independent, and concurrent changes in process conditions would be needed before criticality is possible. Accumulation, rather than dispersion, of nuclear material, in the right geometric shape and with the right quantity of moderator for criticality to occur, would require the occurrence of several unlikely events and is considered to be very unlikely.

Q: *What would be the consequences of a criticality accident at a fuel cycle facility to a member of the public?*

A: A criticality accident is considered a high consequence event to a worker in the immediate area. Outside approximately fifteen feet, a lethal dose is unlikely. Beyond approximately 200 feet, a significant dose is unlikely. Fuel cycle facilities are required to perform analyses to evaluate the chemical and radiological consequences to members of the public. Most fuel facilities in the U.S. are situated on sites where the distance to the site boundary, and the presence of shielding material, such as concrete walls and steel containers, precludes any significant exposure to members of the public. Criticality is therefore considered a localized industrial hazard with little or no off-site consequences.

Global Nuclear Fuels - Americas (GNFA)

Q: *What would be the impacts of a tsunami or a flood at the GNF-A site? How would the NRC and the licensee respond to these events?*

A: The facility is more than 10 miles from the Atlantic coastline and at an elevation 25–30 feet above sea level. Tidal bores are also highly unlikely given the distance from the coastline, the quick dissipation as bores travel upstream, and the elevation of the facility site.

Q: *What are the most significant hazards at the GNF-A site?*

A: A criticality accident represents the potential for a lethal radiation dose to a worker within 10 to 50 feet of the criticality, and lesser but significant doses out to 100 feet or more. The total radiation to an individual at the site boundary would exceed regulatory limits, but would not result in the potential for radiation doses large enough to cause injury. There are no criticality scenarios that would be initiated by an earthquake or loss of offsite power event.

Q: *Are the GNF-A emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: The emergency diesels are not seismically qualified. There are no safety significant scenarios which would result from a loss of offsite power. Risk-significant safety controls are passive or fail to a safe position without relying upon electric power.

Q: *What happens if the GNF-A facility were to lose offsite power as a result of an earthquake?*

A: Process equipment will fail safe if the electrical service is interrupted. Emergency power is provided for a supervised alarm system and essential equipment. There are four emergency diesel generators onsite. Additionally, the facility has emergency power capabilities for support services, the controlled areas, and the emergency control center. A diesel-operated generator will provide an automatic startup and a switch over to the emergency system in the event of a power failure.

U.S. Fuel Cycle Facilities - NEW!

Q: *What preparations are currently in place to respond to an emergency at GNF-A?*

A: The site Emergency Plan contains instructions and Protective Action Recommendations (PARs) for the postulated emergencies. The facility has an onsite Emergency Team with medical, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, hospitals, New Hanover County and the State of North Carolina to provide support and assistance if there is an emergency.

Q: *What would happen if an earthquake occurred in the vicinity of the GNF-A facility?*

A: Impacts to the facility will depend on the magnitude of the earthquake and the location of its epicenter. The licensee will implement its emergency plan and will coordinate with local and State emergency management agencies and organizations to respond to the event.

Q: *What are the potential impacts of an earthquake that exceeded the design of the GNF-A facility?*

A: Impacts to the facility will depend on the magnitude of the earthquake and the location of its epicenter. However, based on the inventory of material that GNF-A is licensed to possess, it is unlikely that there will be a significant release of radioactive material.

Q: *What is the design basis earthquake for the GNF-A facility?*

A: The GNF-A facility was built to the Uniform Building Code (UBC) in effect at the time of construction. The UBC has identified the facility area as Seismic Zone 1 and considers seismic events of minor magnitude (5.5 to 6.0 in the Richter scale). Since there are no significant geologic features in the Wilmington region that would produce a "major" earthquake, it is highly unlikely that a "major" earthquake could affect the GNF-A facility.

Q: *How would the NRC and GNF-A respond to an earthquake at the facility?*

A: Local (New Hanover County) and the State of North Carolina emergency management agencies are both notified.

The licensee provides recommendations based on the release quantity/chemical / physical state and atmospheric conditions.

Honeywell International Metropolis Works Facility (MTW)

Q: *How would the NRC and the licensee respond to an earthquake at the MTW facility?*

A: The NRC currently has two permanent resident inspectors on site at the Paducah USEC facility which is approximately 15 miles from MTW. These inspectors are available and on call 24/7 in the case of any plant emergency including an earthquake. These inspectors could quickly respond to an emergency at MTW from the USEC site if necessary. If applicable the licensee would notify the NRC Operations Center and additional resources would be mobilized from the R-II office or headquarters depending on the severity of the hazard and the particular area of concerns within the plant. The onsite NRC inspectors dispatched from PGDP would relay pertinent information through the Operations Center to the appropriate headquarters staff to continually reassess the hazards and the agency response. The NRC Operations Center would also assist in coordinating additional government agency responses. It is important to note that the licensee would have the lead in the response and the NRC would monitor the licensee's response providing oversight, assistance and coordination as necessary.

New responses and strategies may be developed as a result of the new assessment underway.

Q: *Can an earthquake as large as Japan also happen at MTW?*

A: The Japan earthquakes experienced a ground motion corresponding to 2.7 g. Under current design bases for MTW the maximum credible ground motion at MTW is between than 0.4 g and 1 g. Thus, an earthquake as large as the Japan event was not plausible.

Under the new USGS information, a ground acceleration of 2.124g is associated with the Maximum Credible Earthquake. However, MTW is in the process of investigating this new information and assessing the potential impacts and possible need for plant and system modifications.

U.S. Fuel Cycle Facilities - NEW!

Q: *What are the most significant hazards at the MTW site?*

A: The most significant radiological hazard at the MTW site is the release of UF₆ material. The most significant chemical hazard is the release of HF or NH₃.

Although significant changes are not expected, new results may be obtained as a result of the assessment underway.

Q: *Does the MTW site have a spent fuel storage area?*

A: No. The conversion stage of the fuel cycle, which converts U₃O₈ into UF₆, occurs prior to the enriched UF₆ being fabricated into UO₂ fuel for use in nuclear reactors. Therefore the radioactive material currently being generated or previously generated at a conversion plant has never been used in a nuclear reactor and there is no need to store spent fuel at such a site.

Q: *How many people live near the MTW site?*

A: The plant site is located in a predominantly agricultural area. Within a two mile radius of the plant, approximately 68% of the land is undeveloped (e.g., cropland, forest, or wetland) and the remainder is developed. Within a one-mile radius of the facility, the total population is 558 persons; most of these are concentrated in the E to ESE sectors near the city of Metropolis. The nearest residence sampling device is currently located between the two nearest residences, approximately 1850 feet NNE of the Feed Materials Building. MTW is located in Massac County, IL which has approximately 15,000 residents in the year 2000. The nearest cities are Metropolis, IL which has about 6,500 people and is located 1 mile from the site. Paducah, KY is located 10 miles from the site and has a population of approximately 25,000. Within a 50 mile radius there are approximately 500,000 people. There are no facilities that would present significant evacuation problems within the immediate vicinity of the site. In addition, the Protective Action Recommendations provided in the Emergency Response Plan are limited to shelter-in-place only; no provisions are required for evacuation of the near-site public.

Q: *What would be the impact of a tsunami or flood at the MTW site?*

A: The MTW site is located inland, approximately 550 miles from the coast, so a tsunami is not a plausible scenario. Additionally, the site is not located near any large body of water that could cause a flood at the site. The nearest large surface waters are the Ohio River forms the site perimeter to the south. Flood control on the Ohio River is provided by dams. The nearest dam is located 7 miles (11 km) upriver at Brookport, Illinois. The historic maximum elevation flood at Metropolis was 342 feet in 1937. The 100-yr recurrence flood level developed by FEMA in 1983 is 337 feet. The 375' MTW site is at a relatively high elevation point. The town of Metropolis, for example is at a nominal elevation of 350', and the Kentucky side of the river is at about 350' elevation for a wide area. The West Kentucky Airpark (15 miles SE) landing strips are at 338' elevation. The Flood Map for Massac County indicates that the site is in flood zone C, which is not in either the 100-yr or 500-yr flood plain, and has no flood exposure. Thus the relative elevations make it apparent that the MTW site is not susceptible to rising river water floods. Therefore, flooding by the Ohio River can be considered a non-credible hazard. Although not credible, flooding caused by rising water in the Ohio River could cause building or tank farm damage leading to containment breach and release of contents.

Flooding is also under reassessment and new responses and strategies may be developed as a result.

U.S. Fuel Cycle Facilities - NEW!

Q: *What is the design basis earthquake for the Honeywell Metropolis Works (MTW) facility?*

A: The site area is in the northern part of the Mississippi Embayment, which has had a long history of seismic activity. The only major earthquakes in historic times were the New Madrid earthquakes of 1811-1812, centered about 60 miles southwest of the site. This earthquake was one of the strongest on record in this country. Major faults, trending toward New Madrid, are found approximately twenty-five to thirty miles east and west of the site. These faults, which occurred millions of years ago, have not been active in geologically recent time. Seismologists are unable to accurately predict the recurrence rates for destructive earthquakes such as those of 1811-1812 because of their infrequent occurrences. Nevertheless, experience indicates that major earthquakes originating along the New Madrid fault zone are capable of causing extensive damage in the Metropolis area. One such estimate concluded that a recurrence of an earthquake of the New Madrid intensity had a maximum likelihood of occurring once in 100-300 years in the entire seismic region.

While MTW is clearly located in a high seismicity zone, the original plant construction, which began in the late 1950s, did not adequately address seismic concerns. A 1993 report identified structural modifications to the Feed Materials Building and the tank farm to assure adequate performance. The structural modifications are designed to withstand a 475-yr recurrence site-specific earthquake. This is judged to be reasonably consistent with NUREG 1520 requirements even though NUREG 1520 specifies that the 500-yr earthquake be used, since the uncertainty in earthquake prediction is large relative to the small difference between recurrence periods. It is also stressed that MTW is not currently required to meet NUREG 1520, and other reasonable safety standards can be used.

The USGS completed the National Seismic Hazard Mapping Project in 1996. This project resulted in the development of new earthquake ground motion maps. These maps are also the referenced basis for IBC 2006 standards. A comparison of the 1996 ground motion parameters to that used in the Metropolis 1993 study shows considerable difference with the 1996 being more intense. As a result, MTW is in the process of investigating this new information and assessing the potential impacts and possible need for plant and system modifications.

Q: *What preparations are currently in place to respond to an emergency at MTW?*

A: The site Emergency Response Plan contains responsibilities, procedures, instructions, protective actions, and exposure guidelines for the postulated emergencies. The facility has an onsite emergency response organization with some limited medical, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, and hospital for additional emergency response resources as needed. Recommendations are based on the release quantity/chemical/physical state and atmospheric conditions and are discussed with participating government agencies as appropriate. The state and county have overall responsibility and authority for conducting appropriate emergency response and local implementation of recommended protective actions.

New responses and strategies may be developed as a result of the new assessment underway.

Q: *What would happen if an earthquake occurred in the vicinity of the MTW facility?*

A: In the current seismic assessment, a seismic event has the potential to result in a release of HF or UF₆. The seismic event probability is low. However, considerable work has been done to identify potential seismic risk to the Metropolis Works. Therefore the structural modifications constitute robust passive engineering to mitigate the event with low probability of failure.

As stated, a review is underway based on new USGS information.

Q: *What are the potential impacts of an earthquake that exceeded the design of the MTW facility?*

A: Under the current design and analysis, we identify that the greatest impact of an earthquake most likely would be structural failure increasing the risk that occupants would be injured and unable to escape. Additionally, the likelihood of fires and releases would increase. Although offsite impacts may increase, the potential for radiation doses large enough to cause an acute fatality or early injury to a member of the public is not considered plausible.

As stated, a review is underway based on new USGS information.

U.S. Fuel Cycle Facilities - NEW!

Q: *Are the MTW emergency power diesels built to withstand the effects of an earthquake, if not what happens if the MTW facility were to lose offsite power as a result of an earthquake?*

A: The plant itself has many redundant safety systems, such as sensors and mitigation systems. Critical safety systems have back-up power sources in case of a power outage and it is expected those systems would perform as designed. The plant also has an uninterruptible power supply (UPS) system that immediately provides power in the event of a power failure to assist us to shut down the plant processes safely.

Standby utilities are maintained in order to facilitate a safe and orderly shutdown of the process units during a complete power failure. Standby electrical power is provided from an electrical generator located in the Powerhouse Building. The standby electrical generator is diesel powered and delivers 480 volts AC. In the event that electrical power is interrupted, the standby generator automatically starts and comes to a standby mode. The standby power is then distributed, as required. As described in (1) above, MTW is in the process of investigating new seismic information and assessing the potential impacts and possible need for plant and system modifications.

New responses and strategies may be developed as a result of the new assessment underway.

Louisiana Enrichment Services

Q: *How would the NRC and LES respond to an earthquake at the facility?*

A: NRC and LES would respond consistent with standard procedures for emergency events. LES procedures are described in its Radiological Contingency and Emergency Plan.

Q: *Are the LES emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: Standby Diesel Generators are provided to power equipment that can tolerate a short break (short break load) in the normal power supply. This capability is needed to allow for an orderly shutdown of the facility. Each of the Standby Diesel Generators is sized for 100% of the short break load requirement of the equipment to which it is connected. The Standby Diesel Generators are not required for safe operation of the facility and are installed to provide protection of investment only. The functional requirement of the Standby Diesel Generator System is to provide backup power within approximately 10 seconds after a normal power interruption

A security diesel generator is provided to power select security equipment. The security diesel generator is not required for safe operation of the facility. Uninterruptible Power Supply (UPS) systems are provided to power the facility process equipment that do not tolerate a break (no break load) in the normal power supply. Input power for this UPS system is normally provided by the short break power system with backup from the Standby Diesel Generators. Batteries power the UPS if all other input power is lost. Each of the UPS systems is sized for 100% of its connected load.

Additional UPS systems with battery backup are installed to provide no break power to support systems such as emergency lighting. These systems are sized and located as necessary to provide the requirements of the equipment served. Systems requiring no break power are listed in Section 3.5.2.4, Operating Characteristics.

Duplicate batteries supply operating power for the 13 kV switchgear. Batteries provide starting power for each Standby Diesel Generator and operating power for each UPS system. The Standby Diesel Generator System provides backup 480V power to the NEF during a loss of normal power. The Standby Diesel Generator System is not a requirement for safe operation of the plant and is installed to provide protection of investment only. The Standby Diesel Generator System is comprised of two, 100% rated generators that supply the total backup power required. The Standby Diesel Generator System is installed in the Central Utilities Building. In the event of normal power failure, the Standby Diesel Generator System maintains plant services that protect the capital investment.

Q: *Are special procedures employed at LES for other natural phenomena events?*

A: For hurricanes and other natural phenomena events where advance warning can be provided, LES will shutdown operations and place systems into a safe configuration in advance of the event.

U.S. Fuel Cycle Facilities - NEW!

Q: *What happens if the LES facility were to lose offsite power as a result of an earthquake?*

A: LES's overall electrical power system is designed with a high level of redundancy to maintain a reliable power supply to the process equipment for investment protection. Total loss of electrical power does not have any safety implications. Safety systems for the facility are not dependent on electrical power. Safety systems fail to a fail-safe configuration on loss of power without the need for operator actions.

Q: *What are the potential impacts of an earthquake that exceeded the design of the LES facility?*

A: All buildings and structures, including such items as equipment supports, are designed to withstand the earthquake loads defined in Sections 1615 through 1617 of the International Building Code. The applicant proposed the method outlined in DOE-STD-1020 (DOE, 2002) or ASCE/Structural Engineering Institute (SEI) 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities" (ASCE/SEI, 2005) to demonstrate compliance to a target performance goal of 1.0×10^{-5} annual probability by designing to a seismic hazard of 1.0×10^{-4} annual probability.

Q: *What is the design basis earthquake for the Louisiana Enrichment Services (LES) facility?*

A: A site-specific probabilistic seismic hazard analysis was performed for LES using the seismic source zone geometries and earthquake recurrence models. The modeling included attenuation models suited for the regional and local seismic wave transmission characteristics. Total seismic ground motion hazard to a site results from summation of ground motion effects from all distant and local seismically active areas. The 250-year and 475-year return period peak horizontal ground accelerations are estimated at 0.024 g and 0.036 g, respectively. The respective 10,000- and 100,000-year return period peak horizontal ground accelerations were estimated at 0.15 and 0.31g. This return period is equivalent to a mean annual probability of E^{-4} . The associated peak vertical ground motion is estimated at 0.10 g. The seismic hazard calculated for facility site is similar to that calculated for the nearby Waste Isolation Pilot Plant. The calculated 10,000-year return period peak ground acceleration at the Waste Isolation Pilot Plant is slightly less than 0.15 g. Based on all the information available, the staff concludes that the seismic hazard described in the ISA Summary is acceptable because it is based on a method that follows current industry practice and includes available data.

Q: *What would happen if an earthquake occurred in the vicinity of the LES facility?*

A: The majority of earthquakes in the United States are located in the tectonically active western portion of the country. However, areas within New Mexico and the southwestern United States also experiences earthquakes, although at a lower rate and at lower intensities. Earthquakes in the region around the NEF site are isolated or occur in small clusters of low to moderate size events toward the Rio Grande Valley of New Mexico and in Texas, southeast of the NEF site. The largest earthquake within 322 km (200 mi) of the NEF is the August 16, 1931 earthquake located near Valentine, Texas. This earthquake has an estimated magnitude of 6.0 to 6.4 and produced a maximum epicentral intensity of VIII on the Modified Mercalli Intensity (MMI) Scale. The intensity observed at the NEF site is IV on the MMI scale.

Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF)

Q: *What is the design basis earthquake for the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) facility?*

A: The MFFF is located on Department of Energy's (DOE) Savannah River Site (SRS) near Aiken, South Carolina. At the SRS, there are no known capable or active faults within the 320-km (200-mile) radius that influence the seismicity of the region, with the exception of faults associated with the Charleston seismic zone (MFFF License Application). Earthquakes that could affect safe operations in the MFFF are associated with two seismic sources, repeat of Charlestown 1886 earthquake and small shallow earthquake of the South Carolina Piedmont. The MFFF facility is design to nuclear power plant requirements and utilizes the spectrum that is found in Regulatory Guide (RG) 1.60 anchored at 0.20g PGA.

Q: *What are the potential impacts of an earthquake that exceeded the design of the MFFF?*

A: The MFFF facility was design using DOE's methodologies for natural phenomena hazards that establishes performance goals for nuclear facilities. In the unlikely event of an earthquake exceeding the design basis of the MFFF, it is expected that major structures such as buildings will suffer major damage, but the damage is limited in the extent such that the occupants can safely exit the building.

U.S. Fuel Cycle Facilities - NEW!

Q: *What would happen if an earthquake occurred in the vicinity of the MFFF?*

A: In the event that an earthquake occurs in the vicinity of the facility the seismic monitoring and trip system initiates a shutdown of process-related systems if a seismic event exceeds a specified set point. The seismic monitoring and trip system shuts down normal and standby power systems, ensuring that all movements of nuclear material are stopped in a safe manner.

Q: *Would overpressurized uranium hexafluoride cylinders stored at MFFF greatly increase the risk to members of the public during an earthquake?*

A: No. There are no large inventories of Special Nuclear Material (SNM) stored or used in a gaseous or highly dispersible form similar to the uranium hexafluoride at the MFFF facility. The primary form of SNM in the MOX facility would be powder. There are no significant additional hazards to members of the public due to the powder form during an earthquake.

Q: *How would the NRC and the MFFF respond to an earthquake at the MFFF?*

A: MOX Services will follow the DOE Savannah River Emergency plan during an event at the facility. MOX Services will contact the NRC and DOE and interactions with State and local officials are conducted through the SRS Emergency Duty Officer who oversees the SRS Operations Center. The MFFF emergency preparedness program incorporates plans for radiation monitoring, repair and recovery efforts, search and rescue, and initial medical response.

Q: *What happens if the MFFF were to lose offsite power as a result of an earthquake?*

A: The design of the MFFF electric power supply system consists of a normal power system, a standby power system, and an emergency power system. Two separate and independent incoming offsite power feeders supply MFFF facility. In the rare event of a total loss of all incoming power to the facility, a standby power system composed of two independent standby diesel generators will automatically start and continue the supply of electrical power to the facility.

Q: *Are the MFFF emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: In the unlikely event that a total loss of all incoming power occurs and both standby diesel generator systems fail to start as discussed before, an independent and redundant seismically qualified emergency power system will provide electrical power to the facility. The emergency power system consists of two redundant and independent emergency diesel generator systems each of which has been designed to carry all important loads during an extended period of time until either the normal or standby power system can be restored. The emergency power system is qualified to survive the MFFF design-basis earthquake.

Q: *Can an earthquake as large as happened in Japan also happen near the MFFF?*

A: Near the Savannah River Site (SRS), instrumented historical seismic records indicate that seismicity associated with the SRS and surrounding region is closely related to the earthquake activity within the South Carolina Piedmont. This activity is characterized by shallow, small-magnitude and infrequent earthquakes. At the SRS, there are no known capable or active faults within the 320-km (200-mile) radius that influence the seismicity of the region, with the exception of faults associated with the Charleston seismic zone (MFFF License Application).

Nuclear Fuels Services (NFS)

Q: *What happens if the NFS facility were to lose offsite power as a result of an earthquake?*

A: If electrical power is lost, the processing equipment is designed to shut down to a safe condition. Site emergency power is available from two independent, uninterruptible power supply (UPS) systems. The UPS systems would maintain power to safety related equipment such as criticality alarms, fire alarms, security systems, radiation detectors, and emergency lighting. Each UPS would transfer load to batteries, send a start signal to the diesel generator, and transfer the load to the generator when the appropriate voltage is reached.

U.S. Fuel Cycle Facilities - NEW!

Q: *What would be the impact of a tsunami, hurricane, or flood at the NFS site?*

A: NFS is located approximately 350 miles from the Atlantic Ocean and at an elevation of 1640 feet above sea level where any significant impacts from a hurricane or tsunami are very unlikely. The regulatory analysis of emergency preparedness in NUREG-1140 concluded that tornados might cause large releases, but the material would disperse so widely that significant doses would not result. The NFS site is on the edge of the 100 year floodplain for the Nolichucky River. It is possible that water could get inside buildings during a severe flood. However, there would be flood warnings, evacuation orders, and other notices that would allow the licensee to secure processing equipment, seal containers, and prepare for the flood. We believe such actions would prevent a release having a significant impact on the environment.

Q: *Are the NFS emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: There is no special earthquake protection for the diesel generators. Buildings housing diesel generators meet the same building code requirements as other buildings on the site. Processing equipment will stop operating when normal power is lost. Emergency systems (detectors, alarms, lighting, etc.) would stop operating if emergency power is lost. At that point, emergency response workers would rely on portable equipment such as survey meters, flashlights, and portable breathing air equipment.

Q: *How would the NRC and NFS respond to an earthquake at the facility?*

A: NFS would declare an emergency, activate its emergency response organization and begin implementing its emergency plan. NFS would promptly notify State and local authorities and recommend protective actions NFS believes should be implemented for the public offsite. Licensee resources would be focused on controlling the immediate safety hazards at the site (i.e., safe shutdown, fire fighting, etc.).

State and local authorities would activate their emergency response organizations and begin implementing response procedures. The procedures include tools such as a Reverse 911 call to advise members of the public of the need to shelter or evacuate.

NRC would be notified immediately after State and local authorities are notified. NRC would activate its emergency response organization and begin an independent assessment of site conditions and protective actions being taken. NRC would share its independent assessment with State and local authorities. If warranted by events, NRC will dispatch an emergency site team to monitor licensee activities and serve in an advisory role to state and local officials who may be considering protective actions to further ensure the protection on the public. In addition, NRC would coordinate assistance from other Federal agencies if needed.

Q: *What would happen if an earthquake occurred in the vicinity of the NFS facility?*

A: If an earthquake occurred near NFS, it is expected that the facility will experience structural damage, but not catastrophic collapse of the buildings. The vast majority of uranium is in solid form or liquid form. We would expect some material to be spilled, but most of it would remain inside the buildings. Fires from broken gas lines and other flammable materials are possible. Firefighting efforts may be hampered by blocked roads, broken water lines, and other earthquake related damage. A major fire may require protective actions to minimize radiation dose. If a dose between 1 rem and 5 rem is possible, the initial recommendation in the NFS Emergency Plan is to shelter the public within 1 mile of the site (approx. 2800 people). If a dose greater than 5 rem is possible, the plan recommends limited evacuations within 550 yards of the site. The plan is consistent with guidance in the EPA Manual of Protective Action Guides.

U.S. Fuel Cycle Facilities - NEW!

Q: *What are the potential impacts of an earthquake that exceeded the design of the NFS facility?*

A: Earthquakes can cause damage which results in fires and releases of hazardous materials. A criticality accident is possible, but earthquakes are considered an unlikely cause of a criticality accident because such accidents require enriched uranium to be accumulated in a critical mass. The contents of broken pipes and containers tend to be dispersed, not accumulated. It has been noted that one control used for criticality safety is keeping moderators, such as water, away from enriched uranium. Broken pipes and damaged roofs may allow water into normally dry areas. The loss of a criticality control would increase the risk of an accident, but nuclear facilities are required to be designed so that at least two unlikely, independent, and concurrent changes in process conditions would be needed before criticality is possible. The accidental accumulation of enriched uranium in conditions which favor a criticality accident are possible, but unlikely. A criticality accident is most hazardous to workers near the accident site, not members of the public offsite, because radiation levels decrease rapidly with distance.

Building codes incorporate occupant safety margins such as maintaining structural integrity long enough to allow occupants to leave the building. Maintaining structural integrity will reduce the damage that could lead to fires and releases of hazardous materials. If an earthquake exceeded the design of the facility, structural failure would be more likely increasing the risk that occupants would be injured and unable to escape. The likelihood of fires and releases would increase. The regulatory analysis of emergency preparedness in NUREG-1140 concluded that the potential for radiation doses large enough to cause an acute fatality or early injury to a member of the public is not considered plausible at a nonreactor facility. An earthquake exceeding the design of a facility similar to NFS may cause offsite impacts to increase, but we believe large radiation doses are still unlikely.

Q: *What is the design basis earthquake for the NFS facility?*

A: Buildings were constructed in accordance with the Standard Building Code in effect at the time of construction. Buildings constructed recently were designed to meet the seismic load resistance specified in American Society of Civil Engineers (ASCE) Standard 7, Minimum Design Loads for Buildings and Other Structures. The ASCE 7 design loads are based on an earthquake with a 2500 year return period and a peak ground acceleration of 0.31 g. Buildings codes provide requirements for structures to be design and constructed taking in consideration earthquakes and other natural phenomena events. The provisions of the codes are aimed to ensure that buildings don't fail catastrophically under an event providing for the safe egress of its occupants.

Q: *Can an earthquake as large as happened in Japan also happen in the area surrounding NFS?*

A: The largest known earthquakes East of the Rocky Mountains were the New Madrid earthquakes along the Mississippi River in the 1800's. These are estimated to have been a magnitude of approximately 8. The seismic hazard maps published by the U.S. Geological Survey don't indicate a potential for such large earthquakes in Eastern Tennessee where NFS is located.

Q: *Would overpressurized uranium hexafluoride cylinders stored at NFS greatly increase the risk to members of the public during an earthquake?*

A: No. Our analysis of the potential breakdown of uranium hexafluoride (UF₆) into uranium pentafluoride (UF₅) and elemental fluorine (F₂) concluded that potential chemical consequences are low. All of the cylinders are small. A few are about two gallons in size, but most are less than a quart each. They are stored in shipping containers inside buildings. If the containers were outside and all were to fail simultaneously, a person standing on the property line might notice a strong odor and experience discomfort and irritation, which would be of short duration and cease when the person moves away. However, it is unlikely all cylinders will fail simultaneously. The shipping containers would confine any releases. In addition, the buildings would help confine any material that managed to escape the shipping containers. In the unlikely event of a container breach, it is unlikely that a person standing on the property line would experience anything more than an unpleasant odor.

Paducah Gaseous Diffusion Plant (PGDP)

Q: *What preparations are currently in place to respond to an emergency at the PGDP?*

A: The site Emergency Response Plan contains responsibilities, procedures, instructions, protective actions, and exposure guidelines for the postulated emergencies. The facility has an onsite emergency response organization with some limited medical, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, and hospital for additional emergency response resources as needed. Recommendations are based on the release quantity/chemical/physical state and atmospheric conditions and are discussed with participating government agencies as appropriate. In support of emergency response operations at the plant, the USEC Emergency Operations Facility (EOF), located in Bethesda, Maryland, provides oversight, makes appropriate notifications, coordinates interactions with the public and media, and may request assistance from Federal agencies. The state and county have overall responsibility and authority for conducting appropriate emergency response and local implementation of recommended protective actions.

Q: *Can an earthquake as large as Japan also happen at PGDP?*

A: The most recent earthquake in Japan was caused by a "subduction zone" event, which is the type of mechanism that produces the largest magnitude earthquakes. A subduction zone is a tectonic plate boundary where one tectonic plate is pushed under another plate. In the continental US, the only subduction zone is the Cascadia subduction zone which lies off the coast of northern California, Oregon and Washington. So, an earthquake and tsunami this large could only happen in that region. The Japan earthquakes experienced a ground motion corresponding to 2.7 g and the maximum credible ground motion at MTW is between 0.4 g and 1 g.

Q: *What are the most significant hazards at the PGDP site?*

A: A criticality accident represents the potential for a lethal radiation dose to a worker within 10 to 50 feet of the criticality, and lesser but significant doses out to 100 feet or more. The total radiation to an individual at the site boundary would exceed regulatory limits, but would not result in the potential for radiation doses large enough to cause injury. There are no likely criticality scenarios that would be initiated by an earthquake or loss of offsite power event. Another significant offsite hazard is the potential release of UF₆ due to an earthquake via damage to process structures, systems and components.

Q: *Does the PGDP site have a spent fuel storage area?*

A: No. The enrichment stage of the fuel cycle occurs prior to the enriched UF₆ being fabricated into UO₂ fuel for use in nuclear reactors. Therefore the radioactive material currently being generated or previously generated at an enrichment plant has never been used in a nuclear reactor and there is no need to store spent fuel at such a site.

Q: *How many people live near the PGDP site?*

A: Paducah, Kentucky, approximately 10 miles east, with a 2010 population of 25,000 is the largest city in the immediate region. The city of Metropolis, Illinois, with a 2010 population of about 6,000, is situated approximately 5 miles east of the plant. Two unincorporated communities, Grahamville and Heath, are located approximately 2 miles east of the plant. Part of 28 counties in 4 states fall within a 50-mile radius of the plant.

U.S. Fuel Cycle Facilities - NEW!

Q: *Are the emergency power diesels built to withstand the effects of an earthquake, if not what happens if the PGDP facility were to lose offsite power as a result of an earthquake?*

A: Loss of electrical power ("station blackout") is not a design basis event at PGDP. Loss of electrical power results in: UF6 compressors stop; autoclaves' containment valves shut (fail-safe); withdrawal compression sources (Normetex pumps) stop; UF6 cylinder handling cranes fail "as-is". No residual heat removal is required at PGDP. PGDP is supplied by 18 individual 161-KV power lines from three separate suppliers connected to four interconnected site switchyards. Diesel-driven electric generators are installed in all process buildings. These are not safety systems. The process buildings have large electrical storage battery banks to provide backup DC power. Loss of cascade compressors reduces cascade pressure to below atmospheric pressure (most of the cascade was below atmospheric pressure initially) and UF6 cools by ambient heat loss to solid state. Diesel generators provide power to close cascade valves for economic reasons, but are not safety significant. Liquid UF6 in cylinders and piping cools by ambient heat loss to solid state. Ambient heat loss is the normal cooling method for liquid UF6 in cylinders. Liquid UF6 cylinder handling cranes are seismically qualified: load remains suspended. HPFW is supplied by the C-611-R elevated storage tank and the diesel-driven fire pump. In summary, there is no safety impact from a loss of all offsite power. The equipment shuts down and cools to ambient temperatures by natural means. There are no residual heat issues.

Q: *How would the NRC and USEC respond to an earthquake at the PGDP facility?*

A: The NRC currently has two permanent resident inspectors on site at Paducah. These inspectors are available and on call 24/7 in the case of any plant emergency including an earthquake. If applicable the licensee would notify the NRC Operations Center and additional resources would be mobilized from the R-II office or headquarters depending on the severity of the hazard and the particular area of concerns within the plant. The onsite NRC inspectors would relay pertinent information through the Operations Center to the appropriate headquarters staff to continually reassess the hazards and the agency response. The NRC Operations Center would also assist in coordinating additional government agency responses. It is important to note that the licensee would have the lead in the response and the NRC would monitor the licensee's response providing oversight, assistance and coordination as necessary.

Q: *What would happen if an earthquake occurred in the vicinity of the PGDP facility?*

A: Only failures of cascade UF6 piping operating above atmospheric pressure contribute to exposures from a seismic event. It is possible, although highly unlikely, that a Design Basis seismic event could result in a criticality accident. The radiological consequence of a seismic event is 37 mrem at the site boundary. The chemical consequence of a seismic event is 4 mg U and 2 ppm HF at the site boundary. The consequence of a criticality accident is 7.74 rem at the site boundary.

Q: *What are the potential impacts of an earthquake that exceeded the design of the PGDP facility?*

A: Earthquakes can cause damage which results in fires and releases of hazardous materials. Earthquakes are considered an unlikely cause of a criticality accident because such accidents require enriched uranium to accumulate in an unsafe, critical mass. The contents of broken pipes and containers tend to be dispersed, not accumulated. Liquid and gaseous UF6 operations as well as waste storage are potentially at risk in the seismic event. A severe earthquake could result in toppling of equipment and collapse of structural walls and members, causing a release of uranium to the environment, and possibly a nuclear criticality accident, potentially resulting in a high consequence event to workers.

Q: *What is the design basis earthquake for the Paducah Gaseous Diffusion Plant (PGDP)?*

A: Design Basis seismic event causes (0.165g peak ground acceleration) rupture of cascade process piping (expansion joints) operating above atmospheric pressure. The failures predicted at withdrawal facilities either do not involve UF6 or else occur in piping and equipment that contains gaseous UF6 operating at subatmospheric pressure, therefore, no UF6 release is postulated for the withdrawal facilities. Process building oil systems can withstand the design basis earthquake. All facilities that store significant quantities of UF6 were analyzed for seismic effects up to the evaluation bases earthquake. All facilities have structural capacities at least equal to this peak ground acceleration, i.e. no building collapse.

U.S. Fuel Cycle Facilities - NEW!

Q: *What would be the impact of a tsunami or flood at the PGDP site?*

A: The PGDP site is located inland, approximately 550 miles from the coast, so a tsunami is not a plausible scenario. Additionally, the site is not located near any large body of water that could cause a flood at the site. The nearest large surface waters are the Ohio, Mississippi, Tennessee, and Cumberland Rivers which are about 20 miles from the PGDP site. PGDP is at least 12 feet above any conceivable flooding event, and, therefore, there is no safety impact from a flood at PGDP.

Westinghouse Columbia Fuel Fabrication Facility (CFFF)

Q: *What happens if the CFFF were to lose offsite power as a result of an earthquake?*

A: Emergency generators and Uninterruptable Power Supply (UPS) provide backup power for critical loads, including crucial process equipment:

- emergency lighting systems
- cooling system pumps
- fire alarm, hazard alarm, and other designated safety alarm systems
- conversion control room alarms
- health physics sampling systems
- emergency ventilation systems (including scrubbers)

Q: *What would be the worse-case scenarios from an accident at the CFFF?*

A: The worse-case scenario from a seismic event would be a catastrophic failure of the tank containing anhydrous ammonia concurrent with a complete loss of power to emergency equipment.

Q: *What would be the impact of a tsunami or flood at the CFFF?*

A: The site is not susceptible to a tsunami. Westinghouse is 100 miles from the nearest point of the Atlantic coast.

Q: *How many people live near the CFFF?*

A: Within 1 mile: 6
Within 3 miles: 963
Within 5 miles: 7,870

Q: *Are the CFFF emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: In a station blackout scenario, the most significant hazard would be the gradual buildup of chemicals from the vented vessels in the process. Worker Safety is assured through use of appropriate PPE (Primarily Respiratory protection). Offsite chemical releases are not postulated to exceed thresholds for public safety from a loss of power event.

Q: *Does the CFFF have a spent fuel storage area?*

A: The site does not have spent fuel. Occasionally, fuel assemblies are returned to the site because of minor damage or defects. Such assemblies have never been irradiated in a reactor; they pose no danger.

U.S. Fuel Cycle Facilities - NEW!

Q: *What would happen if an earthquake occurred in the vicinity of the CFFF?*

A: An analysis was performed to determine the consequences of the materials that would likely be released in a severe earthquake. The analysis indicated that an Intermediate Consequence event would occur for facility workers. The public is not at risk to high or intermediate consequence events.

In the event of a chemical hazard release an Emergency Response Planning Guidelines (ERPG)-2 hazard would be created for about one mile. The nearby hydrogen tank is a local fire and explosion hazard. UF6 cylinders in the storage area are not of concern; filled cylinders are at ground level; empty cylinders are racks above the filled cylinders. UF6 cylinders in the autoclaves are not of concern. The autoclaves are heavy-walled vessels that are partially recessed in the concrete floor of the plant.

A natural gas transmission pipeline is 2,800 ft north of the main manufacturing building. If the pipe were to rupture, a fire and explosion would occur.

Two diesel-powered fire pumps provide high water pressure to the above fire suppression equipment when these systems are activated. A total of 450,000 gal of water can be stored on site in two water storage tanks to provide water to fight a fire.

Q: *What is the design basis earthquake for the CFFF?*

A: The original manufacturing building, designed in 1968 was designed to comply with the Standard Building Code, 1965 Edition, with only minor exceptions. The building was designed to meet Seismic Zone I criteria. A building addition, designed in 1978 by Lockwood Greene, was designed to comply with the Standard Building Code, 1976 Edition.

The CFFF site is far from any center of significant earthquake activity. Several major earthquakes have occurred at distant points, and some minor to moderate shocks have occurred nearer to the site. No significant earthquake has been located nearer than about 20 mi from the site.

The large largest seismic event was the Charleston earthquake which was a magnitude 7.0 on August 31, 1886, which is the strongest earthquake documented in the southeastern United States in historic time. The earthquake was located about 90 mi southeast of Columbia and was felt as far away as Boston, Milwaukee, New York City, Cuba, and Bermuda. Damage from the earthquake was reported in Columbia, SC where Modified Mercalli (MM) intensities of VII-VIII were observed.

The nearest earthquake to the CFFF site occurred on April 20, 1964. The event had a magnitude of 3.5 and was located about 15 mi southwest of Columbia, South Carolina.

For the area near Columbia, a peak ground acceleration (PGA) of about 0.3 g would be expected for a 2,475-yr return period.

Q: *What are the most significant hazards at the CFFF?*

A: The most significant hazards at the site are the chemical hazards in the tank farm. The anhydrous ammonia tank has a capacity of 30,000 gallons; usually, only 10,000 gallons is present. Another fire and explosion hazard is from a tank of liquid hydrogen.

U.S. Fuel Cycle Facilities - NEW!

Q: *What are the potential impacts of an earthquake that exceeded the design of the CFFF?*

A: Earthquakes can cause damage which results in fires and releases of hazardous materials. Earthquakes are considered an unlikely cause of a criticality accident because such accidents require enriched uranium to accumulate in an unsafe, critical mass. The contents of broken pipes and containers tend to be dispersed, not accumulated.

Structure, systems and components were evaluated using a combination of the NRC-approved experience data methodology utilized by the Seismic Qualification Utilities Group of utilities to resolve natural phenomena hazards vulnerability and traditional structural risk assessment techniques. Preliminary findings indicate that a severe earthquake could result in toppling of equipment and collapse of structural walls and members, causing a release of uranium to the environment, and possibly a nuclear criticality accident, potentially resulting in a high consequence event to workers.

The seismic review was only a screening review. Detail fragility analyses have not been performed, it is not possible to reliably determine what size earthquake would result in severe damage to the building structure. The analyses indicate that an earthquake with a 0.2 g PGA most likely would result in some damage. Seismic engineers estimate that buildings should withstand an earthquake of up to 0.05 g PGA. For the Columbia region, the United States Geologic Survey (USGS) estimates a PGA of 0.07 g at a 500-year return period.

Q: *How would the NRC and the licensee respond to an earthquake at the CFFF?*

A: NRC Headquarters and Region II Emergency Response staff from the Fuel Cycle Safety Team and Protective Measures Team would monitor the situation.

The site Emergency Plan contains instructions and Protective Action Recommendations (PARs) for the postulated emergencies. The facility has an onsite Emergency Team with medical, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, and hospital. For these types of events, Westinghouse notifies the Department of South Carolina Department of Health and Environmental Control. Responses are based on the characteristics (e.g., chemical, quantity, physical state, atmospheric conditions). County officials would be notified.

U.S. Power Plants (General)

BWR Mark I Design

Q: *How are US BWRs similar and/or different from the plants experience problems in Japan?*

A: Thirty-five of the 104 operating nuclear power plants in the U.S. are boiling water reactors (BWRs), as are the reactors at Fukushima. Twenty-three of the U.S. BWRs have the same Mark I containment as the Fukushima reactors.

Two of the U.S. BWRs with a Mark I containment have an early nuclear steam supply system (NSSS) design designated as BWR-2. Six of the U.S. BWRs with Mark I containments have another early design, designated BWR-3, which are similar to Fukushima Unit 1. The remaining fifteen of the Mark I BWRs have the BWR-4 NSSS, similar to Fukushima Units 2, 3, and 4. The following table lists the operating BWRs in the United States.

The NRC is not aware of all differences that may exist between the Fukushima reactors and those of similar design and vintage operated in the U.S., neither do we have specific knowledge of implementation at Fukushima of the following improvements made to U.S. reactors:

Station Blackout (SBO) Rule - required the ability to cope with SBO for specified time and recover the plant

Anticipated Transient Without Scram (ATWS) Rule - required vendor specific improvements to enhance scram reliability

Hydrogen Control Rule - required modifications to reduce impact of hydrogen generated from beyond design basis events (DBEs)

Equipment Qualification Rule - required environmental qualification of electrical system equipment used for design basis accidents (DBAs)

Mark I Containment Improvement Program - (i) added hardened vent system for containment cooling and fission product scrubbing for beyond DBEs, and (ii) enhanced reliability of automatic depressurization system (ADS) and added an additional water injection capability independent of normal AC and emergency diesel power

Symptom-based Emergency Procedure Guides (EPGs) - provides emergency procedures that direct operator actions on the basis of critical safety parameter status rather than knowledge of the event initiator – applicable to any initiating event (DBA or beyond DBA).

Severe Accident Management Guidelines (SAMGs) - guidelines for minimizing radiological consequences of a damaged core event. Focuses on maintaining containment integrity, controlling releases, and emergency planning interface

Aircraft Impact Requirements - requires procedures to use all available equipment for core cooling, containment protection, and spent fuel pool cooling assuming a significant damage to the facility from an airplane crash

Mark I Containment Hydrodynamic Load Issue Resolution - resulted in structural strengthening of Mark I containments to better handle reactor system depressurization forces

Emergency Core Cooling System (ECCS) Pump Suction Strainer Improvements - larger surface area strainers installed with higher debris loading tolerance to ensure ECCS pump operation

Hydrogen explosions have been a major aspect of the Fukushima accident. In the U.S., NRC Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," conveyed the importance of having a robust pathway for venting primary containment, which contains the suppression pool, in certain severe accident scenarios. In response, all BWRs with Mark I containments that didn't have an existing strengthened or "hardened" pathway for venting directly from primary containment to the outside, made modifications to the plant consistent with the intent of the Generic Letter. This design feature permits a controlled depressurization of primary containment as well as a controlled release of radioactive materials and combustible hydrogen generated by damaged fuel, as may occur during severe accidents.

U.S. Power Plants (General)

Q: How many U.S. plants have designs similar to the affected Japanese reactors (and which ones)?

A: Thirty-five of the 104 operating nuclear power plants in the U.S. are boiling water reactors (BWRs), as are the reactors at Fukushima. Twenty-three of the U.S. BWRs have the same Mark I containment as the Fukushima reactors.

Two of the U.S. BWRs with a Mark I containment have an early nuclear steam supply system (NSSS) design designated as BWR-2. Six of the U.S. BWRs with Mark I containments have another early design, designated BWR-3, which are similar to Fukushima Unit 1. The remaining fifteen of the Mark I BWRs have the BWR-4 NSSS, similar to Fukushima Units 2, 3, and 4. The following table lists the operating BWRs in the United States.

Plant Name	NSSS Type	Containment Design	Location
Browns Ferry 1	BWR-4	Mark I	AL
Browns Ferry 2	BWR-4	Mark I	AL
Browns Ferry 3	BWR-4	Mark I	AL
Brunswick 1	BWR-4	Mark I	NC
Brunswick 2	BWR-4	Mark I	NC
Clinton	BWR-6	Mark III	IL
Columbia Generating Station	BWR-5	Mark II	WA
Cooper	BWR-4	Mark I	NE
Dresden 2	BWR-3	Mark I	IL
Dresden 3	BWR-3	Mark I	IL
Duane Arnold	BWR-4	Mark I	IA
Fermi 2	BWR-4	Mark I	OH
FitzPatrick	BWR-4	Mark I	NY
Grand Gulf 1	BWR-6	Mark III	MS
Hatch 1	BWR-4	Mark I	GA
Hatch 2	BWR-4	Mark I	GA
Hope Creek 1*	BWR-4	Mark I	NJ
La Salle 1	BWR-5	Mark II	IL
La Salle 2	BWR-5	Mark II	IL
Limerick 1	BWR-4	Mark II	PA
Limerick 2	BWR-4	Mark II	PA
Monticello	BWR-3	Mark I	MN
Nine Mile Point 1	BWR-2	Mark I	NY
Nine Mile Point 2	BWR-5	Mark II	NY
Oyster Creek	BWR-2	Mark I	NJ
Peach Bottom 2	BWR-4	Mark I	PA
Peach Bottom 3	BWR-4	Mark I	PA
Perry 1	BWR-6	Mark III	OH
Pilgrim 1	BWR-3	Mark I	MA
Quad Cities 1	BWR-3	Mark I	IL
Quad Cities 2	BWR-3	Mark I	IL
River Bend 1	BWR-6	Mark III	LA
Susquehanna 1	BWR-4	Mark II	PA
Susquehanna 2	BWR-4	Mark II	PA
Vermont Yankee	BWR-4	Mark I	VT

*has concrete secondary containment unlike other BWRs of this type

U.S. Power Plants (General)

Q: *Some in the media and in Hill briefings are suggesting the BWR Mark I containment is flawed. What are the concerns about this type of containment? Are the US plants with this safe?*

A: BWR Mark I containments have relatively small volumes in comparison with pressurized water reactor (PWR) containments. This makes the BWR Mark I containment relatively more susceptible to containment failure given a core meltdown severe enough to (1) fail the reactor vessel and also (2) severe enough so that the core melt reaches the containment boundary. On the positive side, BWRs have more ways of adding water to the core than PWRs to prevent core meltdown. The following improvements have been made to U.S. Mark I containment reactors:

Station Blackout (SBO) Rule: Required the ability to cope with SBO for specified time and recover the plant

Anticipated Transient Without Scram (ATWS) Rule: Required vendor specific improvements to enhance scram reliability

Hydrogen Control Rule: Required modifications to reduce impact of hydrogen generated from beyond design basis events (DBEs)

Equipment Qualification Rule: Required environmental qualification of electrical system equipment used for design basis accidents (DBAs)

Mark I Containment Improvement Program: (i) Added hardened vent system for containment cooling and fission product scrubbing for beyond DBAs, and (ii) Enhanced reliability of automatic depressurization system (ADS) and added an additional water injection capability independent of normal AC and emergency diesel power

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Aircraft Impact Requirements: Requires procedures to use all available equipment for core cooling, containment protection, and spent fuel pool cooling assuming a significant damage to the facility from an airplane crash

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Hydrogen explosions have been a major aspect of the Fukushima accident. In the U.S., NRC Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," conveyed the importance of having a robust pathway for venting primary containment, which contains the suppression pool, in certain severe accident scenarios. In response, all BWRs with Mark I containments that didn't have an existing strengthened or "hardened" pathway for venting directly from primary containment to the outside, made modifications to the plant consistent with the intent of the Generic Letter. This design feature permits a controlled depressurization of primary containment as well as a controlled release of radioactive materials and combustible hydrogen generated by damaged fuel, as may occur during severe accidents.

Continued Plant Safety

Q: *Is our battery backup power less effective than the Japanese?*

A: US regulations do not specify the length of time that a facility needs to have the batteries operate following a loss of offsite power. Instead, the amount of time is dependent on the site recovery strategy and is based on providing sufficient capacity to assure that the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

With respect to a comparison of battery backup power effectiveness, we currently do not have sufficient information to compare the differences in design requirements and performance characteristics of nuclear-grade batteries in the U.S. and Japanese nuclear power plants. However, in the U.S., nuclear power plants utilize redundant nuclear-grade (i.e., Class 1E, safety-related) batteries that are designed and constructed using rigorous standards and are routinely tested in accordance with plant technical specifications to ensure adequate capacity and capability exists to perform their intended safety functions. These batteries are located in structures that can withstand external environmental events such as earthquakes, tornadoes, tsunamis, and floods in accordance with NRC regulations. For U.S. nuclear power plants, the typical design duty cycles for safety grade batteries range from 1 - 8 hours (i.e., 1-2 hours for accident; 4 hours for station blackout; and 1-8 hours for a fire).

U.S. Power Plants (General)

Q: *Some in the media and in Hill briefings are suggesting the BWR Mark I containment is flawed. What are the concerns about this type of containment? Are the US plants with this safe?*

A: BWR Mark I containments have relatively small volumes in comparison with pressurized water reactor (PWR) containments. This makes the BWR Mark I containment relatively more susceptible to containment failure given a core meltdown severe enough to (1) fail the reactor vessel and also (2) severe enough so that the core melt reaches the containment boundary. On the positive side, BWRs have more ways of adding water to the core than PWRs to prevent core meltdown. The following improvements have been made to U.S. Mark I containment reactors:

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U.S. Power Plants (General)

Q: *What could you say about the dangers to the American public from our nuclear plants?*

A: The NRC remains convinced that U.S. nuclear power plants are designed and operated in a manner that protects public health and safety. The NRC established a senior level task force to conduct both short- and long-term analysis of the lessons that can be learned from the situation in Japan. The task force has completed its near-term analysis and concluded that continued operation and continued licensing activities do not pose an imminent risk to public health and safety. The current regulatory approach and the resultant plant capabilities allow the task force to conclude that a sequence of events like the Fukushima accident is unlikely to occur in the United States and some appropriate mitigation measures have been implemented, reducing the likelihood of core damage and radiological releases. The task force recommended rulemaking activities, orders, certain staff actions, and actions for long-term evaluation. The Commission will review the report and will provide the staff with direction. A long-term evaluation is planned and will assess whether any additional licensing actions are necessary. These actions may include Orders, information requests in accordance with Section 50.54(f) of Title 10 (10 CFR) of the Code of Federal Regulations, license amendments, rulemaking, etc.

Q: *Has this incident changed the NRC perception about earthquake risk?*

A: There has been no change in the NRC's perception of earthquake hazard (i.e. ground shaking levels) for US nuclear plants. The NRC continues to determine that US nuclear plants are safe. Even before the events in Japan, the NRC began reviewing the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting a generic review referred to as GI-199, "Implications of Updated Probabilistic Seismic Estimates in Central and Eastern United States on Existing Plants," to again assess the resistance of US nuclear plants to earthquakes. In addition, the NRC has been reviewing new seismic information regarding the plants in California for many years.

The NRC senior level task force has completed its near-term review. In the report, the task force recommends that the NRC require licensees to update their analysis of natural events, such as flooding or earthquakes. The task force did not find an immediate safety concern; rather, the task force is recommending the update since the original analyses were done forty or more years ago for some plants.

Q: *I live near a nuclear power plant similar to the ones having trouble in Japan. How can we now be confident that this plant won't experience a similar problem?*

A: U.S. nuclear power plants are built to withstand environmental hazards, including earthquakes and tsunamis. Even those plants that are located outside of areas with extensive seismic activity are designed for safety in the event of such a natural disaster. The NRC requires that safety-significant structures, systems, and components be designed to take into account the most severe natural phenomena historically reported for the site and surrounding area. The NRC is confident that the robust design of these plants makes it highly unlikely that a similar event could occur in the United States.

U.S. Power Plants (General)

Q: *Could an accident sequence like the one at Japan's Fukushima Daiichi nuclear plants happen in the US?*

A: The NRC task force has issued its near-term review of the event and its impact on U.S. plants ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission). The current regulatory approach, and more importantly, the resultant plant capabilities allow the task force to conclude that a sequence of events like the Fukushima accident is unlikely to occur in the United States and some appropriate mitigation measures have been implemented, reducing the likelihood of core damage and radiological release. Therefore, continued operation and continued licensing activities do not pose an imminent risk to public health and safety. The NRC is planning a longer-term review and will review any new specific information regarding the disaster at the Fukushima plant and its applicability to U.S. reactors, identify lessons learned, and determine if any changes to its regulatory requirements are necessary to continue to ensure the health and safety of the public and the environment.

The NRC relies primarily on information made available to it by the Japanese government and several organizations involved in responding, assessing, and mitigating the events at the Japanese nuclear plants. Those sources have described how Fukushima Daiichi Units 1-3 lost all offsite power and emergency diesel generators. This situation is called "station blackout." US nuclear power plants are designed to cope with a station blackout event that involves a loss of offsite power and onsite emergency power. The NRC's detailed regulations address this scenario. US nuclear plants conducted a "coping" assessment and developed a strategy to demonstrate to the NRC that they could maintain the plant in a safe condition during a station blackout scenario. These assessments, proposed modifications to the plant, and operating procedures were reviewed and approved by the NRC. Several plants added additional AC power sources to comply with this regulation.

In addition, US nuclear plant designs and operating practices since the terrorist events of September 11, 2001, are designed to mitigate severe accident scenarios such as aircraft impact, which include the complete loss of offsite power and all on-site emergency power sources.

US nuclear plant designs include consideration of seismic events and tsunamis. It is important not to extrapolate earthquake and tsunami data from one location of the world to another when evaluating these natural hazards. These catastrophic natural events are very region- and location-specific, based on tectonic and geological fault line locations.

Q: *Can significant damage to a nuclear plant like we see in Japan happen in the US due to an earthquake? Are the Japanese nuclear plants similar to US nuclear plants?*

A: All US nuclear plants are built to withstand environmental hazards, including earthquakes and tsunamis. Even those nuclear plants that are located within areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. The NRC requires that safety-significant structures, systems, and components be designed to take into account even rare and extreme seismic and tsunami events. In addition to the design of the plants, significant effort goes into emergency response planning and accident management. This approach is called defense-in-depth.

The Japanese facilities are similar in design to some US facilities. However, the NRC has required modifications to the plants since they were built, including design changes to control hydrogen and pressure in the containment. The NRC has also required plants to have additional equipment and measures to mitigate damage stemming from large fires and explosions from a beyond-design-basis event. The measures include providing core and spent fuel pool cooling and an additional means to power other equipment on site.

U.S. Power Plants (General)

Q: *The German government ordered some of its nuclear power plants to shut down in response to the events in Japan. Why is it safe to continue to operate the nuclear power reactors in the U.S. that are similar to the Japanese reactors at Fukushima Dai-ichi?*

A: Every regulatory body around the world that deals with nuclear reactors has considered many factors in determining their specific response to events in Japan. The NRC is not privy to all the factors influencing the decision by the German government. The Chairman of the NRC and the Executive Director for Operations at the NRC have briefed the White House and members of Congress on the situation in Japan and the impacts on the U.S.

The NRC continues to closely monitor the activities in Japan and is reviewing all available information; the agency continues to conclude that U.S. plants are operating safely. The NRC continues its licensing and oversight functions for all NRC licensees, including nuclear power plants. Information in a number of areas, including the principle of defense in depth, leads to the conclusion that the current fleet of reactors and materials licensees continue to protect the public health and safety.

Every reactor in the country is designed for severe natural events at its site. Every reactor has a wide range of diverse and redundant safety features as well as multiple physical barriers to contain radioactive material, in order to provide that public health and safety assurance. The NRC has a long regulatory history of conservative decision making. The NRC has been intelligently using risk insights to help inform the regulatory process and has required improvements to the plant designs as we learn from operating experience. Some of these include severe accident management guidelines, revisions to the emergency operating procedures, procedures and processes for dealing with large fires and explosions regardless of the cause, and requirements for coping with station blackout.

Q: *Why are US plants safe to operate considering the events in Japan?*

A: The NRC has been very closely monitoring the activities in Japan and reviewing all available information to allow us to conclude that the U.S. plants continue to operate safely. There has been no reduction in the licensing or oversight function of the NRC as it relates to any of the NRC licensees. Contributors to the conclusion that the current fleet of reactors and materials licensees continue to protect the public health and safety are based on a number of principles, including defense in depth.

Every U.S. reactor is designed for natural events, based on the specific site where the reactor is located. Every U.S. reactor has multiple fission product barriers, as well as a wide range of diverse and redundant safety features. All these factors support the NRC's conclusion that public health and safety can be assured. The NRC has a long regulatory history of conservative decisionmaking. The NRC has been intelligently using risk insights to help inform the regulatory process and has required improvements to the plant designs as we learn from operating experience. Some of these include severe accident management guidelines, revisions to the emergency operating procedures, procedures and processes for dealing with large fires and explosions regardless of the cause, and requirements for coping with station blackout.

The NRC's task force examining the accident at Fukushima Daiichi and its impact on U.S. plants ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission) has concluded that continued operation and continued licensing activities do not pose an imminent risk to public health and safety.

Q: *Has this crisis changed your opinion about the safety of U.S. nuclear power plants?*

A: No. The NRC remains confident that the design of U.S. nuclear power plants ensures the continued protection of public health and safety and the environment.

U.S. Power Plants (General)

Coordination Efforts

Q: *How is EPA monitoring, collecting and posting information related to the impacts in the U.S. of the accident in Japan?*

A: The EPA monitors, collects, and posts information related to the impacts of the Japanese events on the U.S. using their RadNet system. They have 100 fixed radiation monitoring sites in 48 states plus 40 additional deployable monitors that may be sent where needed. The fixed monitors provide information on beta and gamma radiation levels. The deployable monitors measure the external exposure rate and provide weather information. The data from these monitors is sent to a computer, where it is continually reviewed and is usually posted on the EPA's Central Data Exchange website (<http://epa.gov/cdx>) within 2 hours. However, if the computer picks up an abnormality in the radiation level, then the EPA laboratory staff is alerted and reviews the information prior to it being posted. In response to the events in Japan, EPA has sent additional monitors to Guam, Hawaii, and Alaska.

The EPA also monitors contamination in rainwater and drinking water as well as the level of iodine in milk. The EPA provides updates on these testing efforts and a summary of the air radiation monitoring results on its webpage, <http://www.epa.gov/japan2011/>. This webpage contains a link to Frequently Asked Questions, which was the source of information for this response. Additional information may be found there.

Q: *Where would I get IOSAT Potassium Iodide if my city should experience fallout from the Japanese nuclear disaster? Is this the right precaution or is there anything else that can be done to protect myself?*

A: We do not expect any U.S. states or territories to experience harmful levels of radioactivity. As such, we do not believe that there is any need for residents of the United States to take potassium iodide. U.S. residents should listen to the protective action decisions by their states and counties. As necessary, protective action decisions could include actions such as sheltering, evacuating, or taking potassium iodide.

Additional information regarding the use of potassium iodide can be found on NRC's webpage at the following link:

<http://www.nrc.gov/about-nrc/emerg-preparedness/about-emerg-preparedness/potassium-iodide-use.html>.

Since Potassium Iodide is classified as a drug. Additional information is on the Food and Drug Administration's web site: www.fda.gov.

Q: *My loved one is overseas, how do I find out if they are ok?*

A: We are directing public inquiries with regard to concern for loved ones overseas to the State Department, Consular Services at 202-647-7004.

Q: *What is the NRC doing about the emergencies at the nuclear power plants in Japan? Are you sending staff over there?*

A: We are closely following events in Japan, working with other agencies of the federal government, and have been in direct contact with our counterparts in that country. The NRC has sent several staff to Tokyo to assist with this emergency by working through the U.S. Ambassador in response to the Japanese government's request for assistance. The NRC continues to support the Japanese efforts with various staff members both in Japan and at the NRC headquarters in Rockville, MD.

Q: *What resources are the Japanese asking for?*

A: The Japanese have formally requested equipment needed to cool the reactor fuel. This includes such things as pumps, fire hoses, portable generators, and diesel fuel. The NRC is coordinating with General Electric, which has plant design specifications, to ensure any equipment provided will be capable of meeting the needs of the Japanese.

U.S. Power Plants (General)

Q: *What should the American public know about the incident in Japan?*

A: The events unfolding in Japan are the result of a catastrophic series of natural disasters. These include the fifth largest earthquake in recorded history and the resulting devastating tsunami. Despite these unique circumstances, the Japanese appear to have taken reasonable actions to mitigate the event and protect the surrounding population. Since the beginning of the event, the NRC has provided support to the Japanese government through the U.S. Ambassador to Japan, sent senior experience staff to Japan to provide technical assistance, and manned its Operations Center in Rockville, MD in order to gather and examine all available information as part of the effort to analyze the event and understand its implications both for Japan and the United States.

TEPCO (owner/operator of the Fukushima Daiichi facility) announced in late April that it established a plan toward restoring control of the Fukushima facility. The major aims of the plan involve two steps: (1) achieving a steady decline in radiation dose at the plant, and (2) bringing radioactive materials under control and significantly holding the radiation dose down. TEPCO categorized specific efforts under three major headings of “cooling,” “mitigation” and, “monitoring and decontamination.” These were further divided into the following five areas: (1) cooling the reactors, (2) cooling the spent fuel pools, (3) containing, storing, processing and reusing the water contaminated by radioactive materials (accumulated water), (4) mitigating radioactive materials in the atmosphere and soil, and (5) measuring, reducing and announcing the radiation doses in areas where evacuation has already taken place and where it is being planned, as well as areas where preparations are being made for emergency evacuation. The websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>) provide additional information on a daily basis.

Q: *Are any Americans in danger – armed forces, citizens in Tokyo?*

A: The NRC, in consultation with the White House and U.S. Embassy, has advised United States citizens in Japan to follow the protective measures recommended by the Japanese government. These measures appear to be consistent with steps the United States would take. The Department of Defense has personnel trained in radiation protective measures and is responsible for providing guidance to U.S. armed forces. Inquiries regarding U.S. citizens in Japan should be directed to the State Department, Consular Services at 202-647-7004.

Q: *Are we providing additional KI to the Japanese?*

A: We have not been asked to provide KI.

Q: *Does the NRC participate in inspection of the Japanese facilities?*

A: Unless the inspection is sponsored by the International Atomic Energy Agency (IAEA), the NRC does not normally participate in inspections of Japanese facilities.

Q: *Did the NRC consult the Department of Energy (DOE) or the Nuclear Energy Institute (NEI) for assistance in developing the protective action recommendation?*

A: Although the DOE assisted in providing radiation dose rate information to support the analysis performed by the NRC, the protective action recommendation was made by the NRC.

U.S. Power Plants (General)

Q: *How are the research activities conducted and coordinated at the NRC?*

A: NRC's Office of Research (RES) coordinates research activities with the other NRC program offices, as appropriate, and leads the agency's initiative for cooperative research with the U.S. Department of Energy (DOE) and other Federal agencies, the domestic nuclear industry, U.S. universities, and international partners. RES coordinates the development of consensus and voluntary standards for agency use, including appointment of agency staff to various standards committees. Based on research results and experience gained, RES works with the regulatory offices to develop appropriate regulatory actions to resolve potential safety issues for nuclear power plants and other facilities regulated by the NRC, including those issues designated as Generic Issues (GIs). GIs are technical or security issues that could impact two or more facilities or licensees. RES also develops the technical basis for those areas regulated by the NRC that have risk-informed, performance-based regulations.

RES supplies technical tools, analytical models, and experimental data needed to support the agency's regulatory decisions. RES does not conduct research for the primary purpose of developing improved technologies. That is more appropriately done by the Department of Energy or the nuclear industry. Rather, the NRC conducts research to confirm that the methods and data generated by the industry ensure that adequate safety margin is maintained.

RES activities support regulation of the commercial use of radioactive materials to protect public health and safety and to protect the environment. RES is also responsible for providing the technical basis for regulations to ensure the protection and safeguarding of nuclear materials and nuclear power plants in the interest of national security. Thus, while its primary focus is on supporting the licensing and regulatory process, the research conducted by and for the NRC plays an important role in supporting broad government-wide initiatives associated with national security.

Q: *What is the NRC doing to ensure that nuclear power plants update their preparedness and evacuation plans to include protections for the millions of people living within 50 miles of those facilities? NEW!*

A: The NRC has conducted numerous studies on evacuations and their associated phenomena, including assessments of several large-scale, mostly "ad-hoc" evacuations that have occurred within the U.S. over the past 15 years. From this research, the NRC gained valuable insights into the evacuation process (as well as affirmed that evacuations are an effective tool to protect public health and safety). As a result, the forthcoming revisions to the NRC's emergency preparedness regulations (approved by the Commission in August 2011) update the NRC requirements for the evacuation time estimates (ETEs) that licensees must prepare. ETEs are used as a tool to develop and improve evacuation plans in advance of an accident and to decide whether sheltering or evacuation is the appropriate protective action following an accident. The NRC issued Draft NUREG/CR-7002, "Criteria for Development of Evacuation Time Estimate Studies," (available electronically in the NRC's Agencywide Documents Access and Management System (ADAMS) Accession No. ML102790350) in May 2010 to provide the latest guidance for licensees on how to develop a comprehensive set of ETEs.

NRC continues to work actively with its Federal, State, and local partners to continue to enhance the state of emergency preparedness around domestic nuclear power plants. These efforts include the distribution of potassium iodide (to date 26 million potassium iodide tablets have been distributed to States), revisions to the NRC and Federal Emergency Management Agency EP regulations and requirements, and enhancements and updates to the Environmental Protection Agency's Protective Action Guidelines Manual.

Q: *What is the official agency to report radiation numbers and what is the public contact?*

A: NRC regulations require nuclear power plants to report any radiation doses detected at the plant that could be harmful to the public. This would include doses that are generated by the plant or by an external source. During an event in the U.S., it is the state's responsibility to provide protective action decisions for public health and safety. For this incident, the Japanese are responsible for reporting the public dose; nevertheless, should radiation doses be detected within the U.S., it would still be the state's responsibility to provide protective action decisions for public health and safety.

U.S. Power Plants (General)

Q: *How did the NRC develop its computer-based projections that supported the evacuation decision?*

A: The NRC uses the RASCAL computer code to perform offsite radiation dose projections. The RASCAL computer program contains information about U.S. nuclear reactor design types, radiation release pathways from the nuclear power plant to the environment, radionuclide source terms and meteorology. However, RASCAL is not capable of evaluating concurrent and multiple nuclear plant failures. So, to approximate the events unfolding at the Fukushima Daiichi facility, the NRC developed a model that aggregated information from the three operating reactors and the spent fuel pool. This aggregate model was then evaluated using the RASCAL computer code. The radiation doses calculated by the RASCAL code were predicted to exceed the protective action guidelines (PAGs) established by the U.S. Environmental Protection Agency (EPA) well beyond the 10-mile exposure pathway EPZ and beyond the 30 kilometer sheltering zone recommended by the Japanese authorities. Subsequent aerial monitoring by the U.S. Department of Energy (DOE) fixed-wing aircraft monitoring showed elevated radiation dose rates that were in excess of the EPA relocation PAGs to a distance beyond 25 miles from the facility.

Q: *I am traveling to Asia (not Japan). Should I adjust my travel plans to avoid flying through plume or being contaminated once on the ground?*

A: The NRC is not the responsible federal agency to advise U.S. citizens on foreign travel restrictions. That responsibility belongs to the Department of State.

Q: *Are air and sea shipments from Japan being checked for radiation contamination?*

A: U.S. Customs and Border Protection (CBP), a part of the Department of Homeland Security, is responsible for monitoring food and cargo at U.S. ports of entry. In accordance with established protocols, CBP uses radiation detection equipment at both air and sea ports, and uses this equipment, along with specific operational protocols, to resolve any security or safety risks that are identified with inbound travelers and cargo. CBP has issued field guidance reiterating its operational protocols and directing field personnel to specifically monitor maritime and air traffic from Japan. CBP will continue to evaluate the potential risks posed by radiation contamination on inbound travelers and cargo and will adjust its detection and response protocols, in coordination with its interagency partners, as developments warrant. The NRC works closely with CBP and the U.S. Environmental Protection Agency when CBP identifies radioactive materials that may involve licensed materials or radioactive materials shipped from other countries inadvertently.

Q: *Who are the Federal Contacts (for the state) to get information on what DOE & EPA are doing?*

A: States have an ongoing dialogue with the NRC and routinely ask questions through the NRC Regional State Liaison Officer. States also can ask questions through the NRC Headquarters Operations Center at 301-816-5100. Information regarding the following Federal departments and agencies can be obtained through their internet websites and through the NRC's public website:

- Department of State: <http://www.state.gov/>

- Federal Emergency Management Agency: <http://www.fema.gov/>

- Environmental Protection Agency: <http://www.epa.gov/japan2011/rert/radnet-data.html>

- Department of Energy: <http://blog.energy.gov/content/situation-japan>

Q: *When the states receive questions from the public / media that the NRC would be better to answer, where should they direct these calls?*

A: Members of state governments should first consult the NRC public website link for information. Some answers may already be provided. Press releases, information about boiling water reactor technology, frequently asked questions and an expanded set of frequently asked questions are already provided on the website at the following link: <http://www.nrc.gov/japan/japan-info.html>. If sufficient information is not available to address your inquiry, please call the NRC Headquarters Operations Center at (301) 816-5100.

U.S. Power Plants (General)

Q: *What is the NRC doing to ensure this (Japan event) doesn't happen at US plants?*

A: The NRC continues to conclude that US nuclear plants are safe. Since the beginning of the event, the NRC has provided support to the Japanese government through the U.S. Ambassador to Japan, sent senior experience staff to Japan to provide technical assistance, and manned its Operations Center in Rockville, MD in order to gather and examine all available information as part of the effort to analyze the event and understand its implications both for Japan and the United States.

The NRC has established a senior level task force to conduct both short- and long-term analysis of the lessons that can be learned from the situation in Japan. The task force is examining all the available information from Japan to understand the event's implications for the United States. They are performing a systematic and methodical review to see if there are changes that should be made to NRC programs and regulations to ensure protection of public health and safety. This will undoubtedly lead to the identification of issues that warrant further study in the longer term. The task force is scheduled to provide a report to the Commission in July 2011 identifying the results of its review and providing recommendations for short-term action, if necessary, and longer-term study.

The NRC has also issued the following documents related to the events in Japan:

- Information Notice 2011-05 provided information to licensees on the effects of the earthquake and resultant tsunami on the Fukushima Daiichi nuclear power station in Japan.

- Temporary Instruction 2515/183 provided instructions for NRC inspectors to perform independent assessments of the adequacy of industry-initiated efforts to respond to the fuel damage events at the Fukushima Daiichi nuclear station. This involves a high-level look at industry's preparedness for events that may exceed the design for a plant.

- Temporary Instruction 2515/184 provided instructions for NRC inspectors to determine: (i) that the severe accident management guidelines (SAMGs) are available and how they are being maintained, and (ii) the nature and extent of licensee implementation of SAMG training and exercises.

- Bulletin 2011-01 required all holders of operating licenses for nuclear power reactors to provide a comprehensive verification of their compliance with the regulatory requirements in 10 CFR 50.54(hh) associated with mitigating strategies for beyond design basis events.

Q: *How does the NRC ensure people can escape if an accident occurs from a natural disaster when the infrastructure is also affected or destroyed in an area around a plant?*

A: Each US nuclear power plant has an Emergency Plan for ensuring the health and safety of people who live within the emergency planning zone. Emergency plans contain contingencies for alternate evacuation routes, alternate means of notification, and other backup plans in the event of a natural disaster that damages the surrounding infrastructure. Licensees exercise these plans on a regular basis. The NRC performs oversight to verify the acceptable performance of the licensee's response during exercises, drills, and actual incidents and events. The Federal Emergency Management Agency (FEMA) provides oversight for offsite response.

For Incidents of National Significance where the critical infrastructure is severely damaged, the Department of Homeland Security (DHS) has a lead role as a coordinating agency to orchestrate Federal, State, and local assets. The Nuclear/Radiological Incident Annex to the National Response Framework provides for the NRC to be a coordinating agency for incidents involving NRC licensed materials.

Q: *Did the NRC share the post 9/11 enhancements to the U.S. facilities with the Japanese?*

A: Following the events of September 11, 2001, the NRC issued Orders requiring licensees to develop specific guidance and strategies to maintain or restore cooling of the core, containment, and spent fuel using existing or readily available resources (equipment and personnel). These strategies have to be implemented effectively even if large areas of the plant were lost due to explosions or fire, including those that an aircraft impact might create. Although it was recognized prior to September 11, 2001, that nuclear reactors already had significant capabilities to withstand a broad range of attacks, implementing these types of mitigation strategies would significantly enhance the plants' capabilities to withstand a broad range of threats. NRC's Japanese counterpart, the Japan Nuclear and Industrial Safety Agency (NISA), visited NRC in 2008. During that visit, NRC staff shared information contained in the NRC-issued Orders as referenced above. This cooperative exchange occurred under the authority of an international agreement between NRC and NISA for technical exchange.

U.S. Power Plants (General)

Q: *What is the NRC doing in response to the situation in Japan?*

A: The NRC has taken a number of actions:

(1) From March 11 until May 16, 2011, the NRC manned its Operations Center in Rockville, MD, in order to gather and examine all available information as part of the effort to analyze the event and understand its implications both for Japan and the United States.

(2) A team of NRC staff with expertise in boiling water nuclear reactors have deployed to Japan as part of a U.S. International Agency for International Development (USAID) team and continues to provide support.

(3) The NRC maintains communications with its counterpart agency in Japan, offering assistance and technical expertise as requested.

(4) The NRC continues to coordinate its actions with other Federal agencies as part of the U.S. government response. In addition, the NRC's senior-level task force conducted a near-term analysis of the lessons that can be learned from the situation in Japan.

The Near Term Task Force has examined all the available information from Japan to understand the event's implications for the United States and performed a systematic and methodical review to see if there should be changes to be made to NRC programs and regulations to ensure protection of public health and safety. Its report was issued on July 12 and is available to the public (ADAMS Accession No. ML111861807). On July 19, 2011, the Task Force presented its findings to the Commission in a meeting open to the public and proposed improvements in areas ranging from loss of power to earthquakes, flooding, spent fuel pools, venting and emergency preparedness. The Task Force discussed its report and recommendations with the public on July 28, 2011, in NRC headquarters in Rockville, Maryland.

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Q: *What other U.S. agencies are involved, and what are they doing?*

A: The entire federal family is responding to this event. The NRC is closely coordinating its efforts with the White House, DOE, DOD, USAID, and others. The U.S. government is providing whatever support requested by the Japanese government.

Q: *The United States has troops in Japan and has sent ships to help the relief effort – are they in danger from the radiation?*

A: The NRC is not the appropriate federal agency to answer this question. DOD is better suited to provide information regarding its personnel.

U.S. Power Plants (General)

Q: *Is there a danger of radiation making it to the United States?*

A: In response to nuclear emergencies, the NRC works with other U.S. agencies to monitor radioactive releases and predict their path. The NRC continues to monitor information regarding wind patterns near the Japanese nuclear power plants. Nevertheless, given the thousands of miles between the two countries, Hawaii, Alaska, the U.S. Territories and the U.S. West Coast are not expected to experience any harmful levels of radioactivity.

Q: *Is the U.S. government tracking the radiation released from the Japanese plants?*

A: Yes, a number of U.S. agencies are involved in monitoring and assessing radiation including EPA, DOE, and NRC. The best source of additional information is the Environmental Protection Agency.

Q: *Has the government set up radiation monitoring stations to track the release?*

A: The NRC understands that EPA is utilizing its existing nationwide radiation monitoring system, RadNet, to continuously monitor the nation's air and regularly monitors drinking water, milk and precipitation for environmental radiation. EPA has publicly stated its agreement with the NRC's assessment that we do not expect to see radiation at harmful levels reaching the U.S. from damaged Japanese nuclear power plants. Nevertheless, EPA has stated that it plans to work with its federal partners to deploy additional monitoring capabilities to parts of the western U.S. and U.S. territories.

Q: *Why are US plants safe to operate considering the events in Japan?*

A: The NRC has been very closely monitoring the activities in Japan and reviewing all available information to allow us to conclude that the U.S. plants continue to operate safely. There has been no reduction in the licensing or oversight function of the NRC as it relates to any of the NRC licensees. Contributors to the conclusion that the current fleet of reactors and materials licensees continue to protect the public health and safety are based on a number of principles, including defense in depth.

Every U.S. reactor is designed for natural events, based on the specific site where the reactor is located. Every U.S. reactor has multiple fission product barriers, as well as a wide range of diverse and redundant safety features. All these factors support the NRC's conclusion that public health and safety can be assured. The NRC has a long regulatory history of conservative decisionmaking. The NRC has been intelligently using risk insights to help inform the regulatory process and has required improvements to the plant designs as we learn from operating experience. Some of these include severe accident management guidelines, revisions to the emergency operating procedures, procedures and processes for dealing with large fires and explosions regardless of the cause, and requirements for coping with station blackout.

The NRC's task force examining the accident at Fukushima Daiichi and its impact on U.S. plants ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission) has concluded that continued operation and continued licensing activities do not pose an imminent risk to public health and safety.

Design: Risk-informed

Q: *Does the Seismic Core Damage represent a measurement of the risk of radiation release or only the risk of core damage (not accounting for additional containment)?*

A: Seismic core damage frequency is the probability of damage to the core resulting from a seismic initiating event. It does not imply either a meltdown or the loss of containment, which would be required for radiological release to occur. The likelihood of radiation release is far lower.

U.S. Power Plants (General)

Q: *Could there be core damage and radiation release at a U.S. plant if a natural disaster exceeding the plant design were to occur?*

A: U.S. nuclear power plants are built to withstand external hazards, including earthquakes, tsunamis, and flooding, as appropriate. The NRC has made substantial effort over time to ensure that vulnerabilities to both internal and external hazards were considered and mitigated in the plant current design and licensing basis of its regulated facilities. In 1988, the NRC's Generic Letter (GL) No. 88-20, "Individual Plant Examination [IPE] for Severe Accident Vulnerabilities," requested plant owners to perform a systematic evaluation of plant-specific vulnerabilities and report the results to the Commission. For many plants, the IPEs became the basis for the plant's initial Probabilistic Risk Assessment (PRA). Later the NRC issued Supplement 4 to GL 88-20, that requested licensees to evaluate vulnerabilities to external events (IPEEE). Most licensees made improvements to their facilities to reduce vulnerabilities identified in their IPEs and IPEEEs.

The ground motions that are used as seismic design bases at US nuclear plants are called the Safe Shutdown Earthquake (SSE) ground motions. In the 1990s, the NRC staff reviewed the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting a generic review (i.e., GI-199) to again assess the resistance of US nuclear plants to earthquakes. Based on NRC's preliminary analyses to date, the average probability of ground motions exceeding the SSE over the life of the plant for the plants in the Central and Eastern United States is less than about 1%. It is important to remember that structures, systems and components are required to have "adequate margin," meaning that they must continue be able withstand shaking levels that are above the plant's design basis.

Q: *Given that low probability events do occur, how does the U.S. ensure that U.S. plant designs are not significantly degraded by risk-informed changes?*

A: The NRC has established a policy for using risk information in its regulatory decision making. The NRC's policy statement on probabilistic risk assessment (PRA) encourages greater use of this analysis technique to improve safety decisionmaking and improve regulatory efficiency. The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy. In implementing risk-informed decisionmaking, licensing basis changes are expected to meet a set of key principles. Some of these principles are written in terms typically used in traditional engineering decisions (e.g., defense in depth). While written in these terms, it should be understood that risk analysis techniques can be, and are encouraged to be, used to help ensure and show that these principles are met. These principles are:

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change, i.e., a "specific exemption" under 10 CFR 50.12 or a "petition for rulemaking" under 10 CFR 2.802.
2. The proposed change is consistent with the defense-in-depth philosophy.
3. The proposed change maintains sufficient safety margins.
4. When proposed changes result in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.
5. The impact of the proposed change should be monitored using performance measurement strategies.

U.S. Power Plants (General)

Q: *What is the likelihood of the design basis or “SSE” ground motions being exceeded over the life of a nuclear plant?*

A: The ground motions that are used as seismic design bases at US nuclear plants are called the Safe Shutdown Earthquake ground motion (SSE). In the mid to late 1990s, the NRC staff reviewed the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting GI-199 to again assess the resistance of US nuclear plants to earthquakes. Based on NRC’s preliminary analyses to date, the mean probability of ground motions exceeding the SSE over the life of the plant for the plants in the Central and Eastern United States is less than about 1%.

It is important to remember that structures, systems and components are required to have “adequate margin,” meaning that they must continue be able withstand shaking levels that are above the plant’s design basis.

Q: *Does GI-199 provide rankings of US nuclear plants in terms of safety?*

A: The NRC does not rank nuclear plants by seismic risk. The objective of the GI-199 Safety/Risk Assessment was to perform a conservative, screening-level assessment to evaluate if further investigations of seismic safety for operating reactors in the central and eastern US (CEUS) are warranted, consistent with NRC directives. The results of the GI-199 safety risk assessment should not be interpreted as definitive estimates of plant-specific seismic risk because some analyses were conservative making the calculated risk higher than in reality. The nature of the information used (both seismic hazard data and plant-level fragility information) make these estimates useful only as a screening tool.

Q: *What do you mean by “increased estimates of seismic hazards” at nuclear plant sites?*

A: Seismic hazard (earthquake hazard) represents the chance (or probability) that a specific level of ground motion could be observed or exceeded at a given location. Our estimates of seismic hazard at some Central and Eastern United States locations have changed based on results from recent research, indicating that earthquakes occurred more often in some locations than previously estimated. Our estimates of seismic hazard have also changed because the models used to predict the level of ground motion, as caused by a specific magnitude earthquake at a certain distance from a site, changed. The increased estimates of seismic hazard at some locations in the Central and Eastern United States were discussed in a memorandum to the Commission, dated July 26, 2006. (The memorandum is available in the NRC Agencywide Documents Access and Management System [ADAMS] under Accession No. ML052360044).

Design: Defense-in-Depth

Q: *How would the U.S. have responded to the events in Japan of March 11, 2011?*

A: The NRC requires plant designs to include multiple and diverse safety systems, and plants must test their emergency response capabilities on a regular basis. Plant operators are very capable of responding to significant events. U.S. nuclear power plants have emergency operating procedures as well as severe accident management guidelines that ensure that the containment structure integrity takes priority in an accident situation. Therefore, in an event that goes beyond those analyzed in the original plant design (i.e., beyond design basis event), such as the one at Fukushima Daiichi, U.S. BWR operators are trained to preserve primary and secondary containment by venting to provide the greatest assurance of public protection during a severe accident. Each U.S. plant has an emergency plan that is coordinated with local, State and Federal departments and agencies to ensure the safety of the public within the Emergency Planning Zone. In addition, NRC regulations require plants to have plans in place that would allow them to mitigate even worst-case scenarios. Since 9/11, we have implemented requirements for licensees to have additional response capabilities for extreme situations.

U.S. Power Plants (General)

Q: *Why are US plants safe to operate considering the events in Japan?*

A: The NRC has been very closely monitoring the activities in Japan and reviewing all available information to allow us to conclude that the U.S. plants continue to operate safely. There has been no reduction in the licensing or oversight function of the NRC as it relates to any of the NRC licensees. Contributors to the conclusion that the current fleet of reactors and materials licensees continue to protect the public health and safety are based on a number of principles, including defense in depth.

Every U.S. reactor is designed for natural events, based on the specific site where the reactor is located. Every U.S. reactor has multiple fission product barriers, as well as a wide range of diverse and redundant safety features. All these factors support the NRC's conclusion that public health and safety can be assured. The NRC has a long regulatory history of conservative decisionmaking. The NRC has been intelligently using risk insights to help inform the regulatory process and has required improvements to the plant designs as we learn from operating experience. Some of these include severe accident management guidelines, revisions to the emergency operating procedures, procedures and processes for dealing with large fires and explosions regardless of the cause, and requirements for coping with station blackout.

The NRC's task force examining the accident at Fukushima Daiichi and its impact on U.S. plants ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission) has concluded that continued operation and continued licensing activities do not pose an imminent risk to public health and safety.

Design: External Events: Others (e.g., hurricanes, tornadoes, snow, ice, etc.)

Q: *Is there any information about how the Southeast Reactors performed during Katrina? What damage did the flood water do? Any power loss?*

A: The reactors performed as designed. Waterford was the most affected while River Bend also experienced some effects.

Waterford 3 (near New Orleans, LA) did not have damage to any safety equipment during, or shortly after Katrina. They shut down on August 28, 2005, in advance of the hurricane strike. The flooding did affect local infrastructure, including communications and power distribution. However, the plant successfully used their emergency diesel generators to furnish plant power. Access was maintained to the plant throughout the event. On September 9, 2005, after a comprehensive review by FEMA and the NRC, the plant was authorized to restart.

River Bend Station (30 miles north of Baton Rouge, LA) did not experience damage to any safety related equipment and only minimal damage to emergency planning equipment (one siren) during and after Hurricane Katrina. The station reduced power to 70 percent core thermal power on August 28, 2005, due to reduced electrical grid loads. Access was maintained to the plant throughout the event. On September 2, 2005, the plant returned to 100% power.

Also, in 1992 the eye of Hurricane Andrew, a category 5 hurricane, passed directly over the Turkey Point nuclear plant. The plant was shut down prior to the hurricane making landfall and an assessment of the plant following the hurricane demonstrated that the plant sustained very little damage and all of the safety equipment was intact.

U.S. Power Plants (General)

Q: *How has the NRC evaluated nuclear waste storage containers, dry cask or otherwise, to ensure their adequate protection under the loading stresses of a severe seismic event, or other catastrophic impact incident? NEW!*

A: U.S. Nuclear Regulatory Commission (NRC) regulations in Title 10 Code of Federal Regulations Part 72 and performance standards require licensees to analyze the environmental conditions and natural phenomena surrounding each Independent Spent Fuel Storage Installation (ISFSI) to determine severe design-basis events for each site. To be certified by the NRC, the storage casks must be evaluated and shown to withstand the forces and stresses from the most severe loading conditions for each type of event.

In evaluating the adequacy of each licensee's design basis events, the NRC uses historical seismic events, nearby seismic faults and site-specific ground characteristics for each ISFSI to determine the ground motions that could affect an ISFSI. The casks are designed to maintain stability, withstand the ground motions, and safely confine and shield the spent nuclear fuel under such events. Licensees have conservatively shown, and NRC has confirmed, freestanding dry storage cask components, such as the canister and overpack, will neither tip over nor fail during such events. The dry storage casks are also analyzed for other severe natural phenomena and accidents such as cask drops and tipover, explosions, fires, floods, and tornado winds, and tornado missiles. The radiological consequences of a cask tipover accident are addressed in NRC safety evaluation reports and have been shown to be negligible.

Design: External Events: Seismic

Q: *Do the Japan events of March 2011 mean that there should be more concerns about seismic risks at San Onofre Generating Station (SONGS)? NEW!*

A: U.S. nuclear plant designs consider seismic events and tsunamis. It is important not to extrapolate earthquake and tsunami data from one location of the world to another when evaluating these natural hazards. These catastrophic natural events are location specific, based on the locations of tectonic and geological fault lines. The March 2011 Japan earthquake occurred on a subduction zone, which is a very different type of tectonic environment than the region around SONGS, which is predominantly strike slip. A magnitude 9 earthquake can only occur on a subduction zone and cannot occur in the region around SONGS.

Q: *What do you mean by "increased estimates of seismic hazards" at nuclear plant sites?*

A: Seismic hazard (earthquake hazard) represents the chance (or probability) that a specific level of ground motion could be observed or exceeded at a given location. Our estimates of seismic hazard at some Central and Eastern United States locations have changed based on results from recent research, indicating that earthquakes occurred more often in some locations than previously estimated. Our estimates of seismic hazard have also changed because the models used to predict the level of ground motion, as caused by a specific magnitude earthquake at a certain distance from a site, changed. The increased estimates of seismic hazard at some locations in the Central and Eastern United States were discussed in a memorandum to the Commission, dated July 26, 2006. (The memorandum is available in the NRC Agencywide Documents Access and Management System [ADAMS] under Accession No. ML052360044).

Q: *Does the Seismic Core Damage represent a measurement of the risk of radiation release or only the risk of core damage (not accounting for additional containment)?*

A: Seismic core damage frequency is the probability of damage to the core resulting from a seismic initiating event. It does not imply either a meltdown or the loss of containment, which would be required for radiological release to occur. The likelihood of radiation release is far lower.

U.S. Power Plants (General)

Q: *Could an accident sequence like the one at Japan's Fukushima Daiichi nuclear plants happen in the US?*

A: The NRC task force has issued its near-term review of the event and its impact on U.S. plants ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission). The current regulatory approach, and more importantly, the resultant plant capabilities allow the task force to conclude that a sequence of events like the Fukushima accident is unlikely to occur in the United States and some appropriate mitigation measures have been implemented, reducing the likelihood of core damage and radiological release. Therefore, continued operation and continued licensing activities do not pose an imminent risk to public health and safety. The NRC is planning a longer-term review and will review any new specific information regarding the disaster at the Fukushima plant and its applicability to U.S. reactors, identify lessons learned, and determine if any changes to its regulatory requirements are necessary to continue to ensure the health and safety of the public and the environment.

The NRC relies primarily on information made available to it by the Japanese government and several organizations involved in responding, assessing, and mitigating the events at the Japanese nuclear plants. Those sources have described how Fukushima Daiichi Units 1-3 lost all offsite power and emergency diesel generators. This situation is called "station blackout." US nuclear power plants are designed to cope with a station blackout event that involves a loss of offsite power and onsite emergency power. The NRC's detailed regulations address this scenario. US nuclear plants conducted a "coping" assessment and developed a strategy to demonstrate to the NRC that they could maintain the plant in a safe condition during a station blackout scenario. These assessments, proposed modifications to the plant, and operating procedures were reviewed and approved by the NRC. Several plants added additional AC power sources to comply with this regulation.

In addition, US nuclear plant designs and operating practices since the terrorist events of September 11, 2001, are designed to mitigate severe accident scenarios such as aircraft impact, which include the complete loss of offsite power and all on-site emergency power sources.

US nuclear plant designs include consideration of seismic events and tsunamis. It is important not to extrapolate earthquake and tsunami data from one location of the world to another when evaluating these natural hazards. These catastrophic natural events are very region- and location-specific, based on tectonic and geological fault line locations.

Q: *How many plants are located in seismic areas?*

A: Although we often think of the US as having "active" and "non-active" earthquake zones, earthquakes can actually happen almost anywhere. Seismologists typically separate the US into low, moderate, and high seismicity zones. The NRC requires that every plant be designed for site-specific ground motions that are appropriate for their location. In addition, the NRC has specified a minimum ground shaking level to which the plants must be designed.

Q: *Has this incident changed the NRC perception about earthquake risk?*

A: There has been no change in the NRC's perception of earthquake hazard (i.e. ground shaking levels) for US nuclear plants. The NRC continues to determine that US nuclear plants are safe. Even before the events in Japan, the NRC began reviewing the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting a generic review referred to as GI-199, "Implications of Updated Probabilistic Seismic Estimates in Central and Eastern United States on Existing Plants," to again assess the resistance of US nuclear plants to earthquakes. In addition, the NRC has been reviewing new seismic information regarding the plants in California for many years.

The NRC senior level task force has completed its near-term review. In the report, the task force recommends that the NRC require licensees to update their analysis of natural events, such as flooding or earthquakes. The task force did not find an immediate safety concern; rather, the task force is recommending the update since the original analyses were done forty or more years ago for some plants.

U.S. Power Plants (General)

Q: *How many US reactors are located in active earthquake zones (and which reactors)?*

A: Although we often think of the US as having “active” and “non-active” earthquake zones, earthquakes can actually happen almost anywhere. Seismologists typically separate the US into low, moderate, and high seismicity zones. The NRC requires that every plant is designed for site-specific ground motions that are appropriate for their location. In addition, the NRC has specified a minimum ground shaking level to which the plants must be designed.

Q: *What could you say about the dangers to the American public from our nuclear plants?*

A: The NRC remains convinced that U.S. nuclear power plants are designed and operated in a manner that protects public health and safety. The NRC established a senior level task force to conduct both short- and long-term analysis of the lessons that can be learned from the situation in Japan. The task force has completed its near-term analysis and concluded that continued operation and continued licensing activities do not pose an imminent risk to public health and safety. The current regulatory approach and the resultant plant capabilities allow the task force to conclude that a sequence of events like the Fukushima accident is unlikely to occur in the United States and some appropriate mitigation measures have been implemented, reducing the likelihood of core damage and radiological releases. The task force recommended rulemaking activities, orders, certain staff actions, and actions for long-term evaluation. The Commission will review the report and will provide the staff with direction. A long-term evaluation is planned and will assess whether any additional licensing actions are necessary. These actions may include Orders, information requests in accordance with Section 50.54(f) of Title 10 (10 CFR) of the Code of Federal Regulations, license amendments, rulemaking, etc.

Q: *Why should the NRC not require the more sophisticated (3D) seismic studies being voluntarily conducted by licensees in California?*

A: Current NRC and American Nuclear Society (ANS) documentation provides guidance related to site investigations undertaken for the purpose of characterizing seismic sources and dynamic site properties. A variety of geophysical and geotechnical tools are available that can be used to investigate the earth from both a site-specific and a regional level. Each of these methods provides specific information by probing the earth in a different way. While some tools are universally useful, others are better suited to certain types of sub-surface materials and tectonic situations. While 3D seismic studies, such as those being performed in California, are sophisticated, they are not useful for all situations and the very large expense of the study could preclude broader application of techniques better suited to a specific site. The NRC would suggest the use of 3D seismic studies only in cases where it could be useful. The NRC attempts to provide regulations that call for techniques that would be the most suitable given the specific conditions of a plant and requested licensing actions.

Q: *What are the current findings of GI-199?*

A: Currently operating nuclear plants in the US remain safe, with no need for immediate action. This determination is based on NRC staff reviews of updated seismic hazard information and the conclusions of the first stage of GI-199. Existing nuclear plants were designed with considerable margin to be able to withstand the ground motions from the “deterministic” or “scenario earthquake” that accounted for the largest earthquakes expected in the area around the plant. The results of the GI-199 assessment demonstrate that the probability of exceeding the design basis ground motion may have increased at some sites, but only by a relatively small amount. In addition, the probabilities of seismic core damage are lower than the guidelines for taking immediate action. Although there is not an immediate safety concern, the NRC is focused on assuring safety during even very rare and extreme events. Therefore, the NRC has determined that assessment of updated seismic hazards and plant performance should continue.

Q: *The NRC Near-Term Task Force Report states that a sequence of events like the Fukushima accident is unlikely to occur in the U.S. and some appropriate mitigation measures have been implemented. What are those appropriate mitigation measures? NEW!*

A: The mitigation measures are what are commonly referred to as the B.5.b actions. These are the actions that were taken following the events of 9/11 in the United States. These measures would deal with the loss of large areas of the plant, including the use of portable equipment to provide some level of core cooling, spent fuel pool cooling and/or maintenance of containment integrity. They provide an additional level of mitigation capability that may be of assistance in the event of a significant accident similar to Fukushima.

U.S. Power Plants (General)

Q: *What is the current status of GI-199 regarding updated seismic analysis for U.S. nuclear reactors? NEW!*

A: As of September 1, 2011, the NRC staff is seeking public comments on a draft Generic Letter that would require U.S. nuclear power plants to re-examine their sites' seismic risk and provide that information to the NRC. Comments on the draft letter, published in the Federal Register and available on regulations.gov, will be accepted until Oct. 31. The letter represents the next step in the staff's ongoing multi-year examination of updated seismic hazard information for the eastern and central United States, through the NRC's Generic Issues program.

This effort, labeled GI-199, began long before the events at the Fukushima Dai-ichi nuclear plant in Japan and the recent Virginia earthquake. GI-199 was prompted by the seismic analyses included in applications from 2003 related to new reactor activity. The NRC issued an Information Notice in September 2010 regarding GI-199, including the agency's conclusion that existing plant designs safely account for possible earthquakes. More information on GI-199 is available on the NRC website. The NRC staff will consider the comments before finalizing the Generic Letter, which the staff expects to issue near the end of the year. The draft letter's approach would have U.S. nuclear power plants perform their analyses within either one or two years, depending on the analysis method used, and deliver their results to the NRC. The agency will then determine whether additional actions are necessary.

Q: *How has the NRC evaluated nuclear waste storage containers, dry cask or otherwise, to ensure their adequate protection under the loading stresses of a severe seismic event, or other catastrophic impact incident? NEW!*

A: U.S. Nuclear Regulatory Commission (NRC) regulations in Title 10 Code of Federal Regulations Part 72 and performance standards require licensees to analyze the environmental conditions and natural phenomena surrounding each Independent Spent Fuel Storage Installation (ISFSI) to determine severe design-basis events for each site. To be certified by the NRC, the storage casks must be evaluated and shown to withstand the forces and stresses from the most severe loading conditions for each type of event.

In evaluating the adequacy of each licensee's design basis events, the NRC uses historical seismic events, nearby seismic faults and site-specific ground characteristics for each ISFSI to determine the ground motions that could affect an ISFSI. The casks are designed to maintain stability, withstand the ground motions, and safely confine and shield the spent nuclear fuel under such events. Licensees have conservatively shown, and NRC has confirmed, freestanding dry storage cask components, such as the canister and overpack, will neither tip over nor fail during such events. The dry storage casks are also analyzed for other severe natural phenomena and accidents such as cask drops and tipover, explosions, fires, floods, and tornado winds, and tornado missiles. The radiological consequences of a cask tipover accident are addressed in NRC safety evaluation reports and have been shown to be negligible.

Q: *Have events in Japan changed our perception of earthquake risk to the nuclear plants in the US?*

A: The NRC continues to determine that US nuclear plants are safe. The Japanese quake does not change the NRC's perception of earthquake hazard (i.e., ground motion levels) at US nuclear plants. Even before the events in Japan, the NRC began reviewing the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting a generic review referred to as GI-199, "Implications of Updated Probabilistic Seismic Estimates in Central and Eastern United States on Existing Plants," to again assess the resistance of US nuclear plants to earthquakes. In addition, the NRC has been reviewing updated seismic information regarding the plants in California for many years. The NRC has established a senior level task force to identify areas of further evaluation as a result of the Japanese events. The task force has issued its near-term report, which recommends that the NRC require licensees to reevaluate and upgrade as necessary the design-basis seismic and flooding protection for each operating reactor. The Commission will review the report and will provide direction to the staff.

U.S. Power Plants (General)

Q: *With NRC moving to design certification, at what point is seismic capability tested – during design or modified to be site-specific? If in design, what strength seismic event must these be built to withstand?*

A: The regulations related to seismic requirements are contained in General Design Criterion 2 in Appendix A to Title 10 of the *Code of Federal Regulations*, Part 50.

During design certification, vendors propose a seismic design in terms of a ground motion spectrum for their nuclear facility. This spectrum is called a standard design response spectrum and is developed so that the proposed nuclear facility can be sited at most locations in the central and eastern United States. The vendors show that this design ground motion is suitable for a variety of different subsurface conditions such as hard rock, deep soil, or shallow soil over rock. Combined License and Early Site Permits applicants are required to develop a site specific ground motion response spectrum that takes into account all of the earthquakes in the region surrounding their site as well as the local site geologic conditions. Applicants estimate the ground motion from these postulated earthquakes to develop seismic hazard curves. These seismic hazard curves are then used to determine a site specific ground motion response spectrum that has a maximum annual likelihood of 1×10^{-4} of being exceeded. This can be thought of as a ground motion with a 10,000 year return period. This site specific ground motion response spectrum is then compared to the standard design response spectrum for the proposed design. If the standard design ground motion spectrum envelopes the site specific ground motion spectrum then the site is considered to be suitable for the proposed design. If the standard design spectrum does not completely envelope the site specific ground motion spectrum, then the COL applicant must do further detailed structural analysis to show that the design capacity is adequate. Margin beyond the standard design and site specific ground motions must also be demonstrated before fuel loading can begin.

Q: *How many US reactors are located in active earthquake zones?*

A: Although we often think of the US as having “active” and “non-active” earthquake zones, earthquakes can actually happen almost anywhere. Seismologists typically separate the US into low, moderate, and high seismicity zones. The NRC requires that every nuclear plant be designed for site-specific ground motions that are appropriate for their locations. In addition, the NRC has specified a minimum ground motion level to which nuclear plants must be designed.

Q: *What is the likelihood of the design basis or “SSE” ground motions being exceeded over the life of a nuclear plant?*

A: The ground motions that are used as seismic design bases at US nuclear plants are called the Safe Shutdown Earthquake ground motion (SSE). In the mid to late 1990s, the NRC staff reviewed the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting GI-199 to again assess the resistance of US nuclear plants to earthquakes. Based on NRC’s preliminary analyses to date, the mean probability of ground motions exceeding the SSE over the life of the plant for the plants in the Central and Eastern United States is less than about 1%.

It is important to remember that structures, systems and components are required to have “adequate margin,” meaning that they must continue be able withstand shaking levels that are above the plant’s design basis.

Q: *Does GI-199 provide rankings of US nuclear plants in terms of safety?*

A: The NRC does not rank nuclear plants by seismic risk. The objective of the GI-199 Safety/Risk Assessment was to perform a conservative, screening-level assessment to evaluate if further investigations of seismic safety for operating reactors in the central and eastern US (CEUS) are warranted, consistent with NRC directives. The results of the GI-199 safety risk assessment should not be interpreted as definitive estimates of plant-specific seismic risk because some analyses were conservative making the calculated risk higher than in reality. The nature of the information used (both seismic hazard data and plant-level fragility information) make these estimates useful only as a screening tool.

U.S. Power Plants (General)

Q: *Are U.S. nuclear power plants designed to withstand earthquakes? What would the effect be on [plant X] if a 9.0 earthquake hit?*

A: All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate. Even those plants that are located in areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. Each plant is designed to a ground-shaking level that is appropriate for its location, given the possible earthquake sources that may affect the site and its tectonic environment. Ground shaking is a function of both the magnitude of the earthquake and the distance from the fault plane to the specific site. The seismic responses of the structures, systems, and components associated with these facilities are site specific. The plants are analyzed for certain identified faults and tectonic capabilities in the area while others are analyzed for seismic zones.

Q: *Do U.S. nuclear plants have better capabilities to respond to natural disasters than the plants in Japan?*

A: The NRC is not yet aware of all of the differences that may exist between the reactors that are of similar design and vintage as those operated in the U.S. Many improvements have been made to U.S boiling water reactors (BWRs). For example, NRC Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," conveyed the importance of having a robust pathway for venting primary containment, which contains the suppression pool, in certain severe accident scenarios. In response, all BWRs with Mark I containments that didn't have an existing strengthened or "hardened" pathway for venting directly from primary containment to the outside, made modifications to the plant consistent with the intent of the Generic Letter. This design feature permits a controlled depressurization of primary containment as well as a controlled release of radioactive materials and combustible hydrogen that could be generated by damaged fuel, as may occur during severe accidents. U.S. nuclear power plants are built to withstand external hazards, including earthquakes tsunamis, and flooding, as appropriate. In addition to the design of the plants, significant effort goes into emergency response planning, preparation, and training. The NRC has also completed substantial research and analysis that resulted in the development and use of severe accident management guidelines. These insights have informed our decision making and review of licensed activities.

Q: *Could there be core damage and radiation release at a U.S. plant if a natural disaster exceeding the plant design were to occur?*

A: U.S. nuclear power plants are built to withstand external hazards, including earthquakes, tsunamis, and flooding, as appropriate. The NRC has made substantial effort over time to ensure that vulnerabilities to both internal and external hazards were considered and mitigated in the plant current design and licensing basis of its regulated facilities. In 1988, the NRC's Generic Letter (GL) No. 88-20, "Individual Plant Examination [IPE] for Severe Accident Vulnerabilities," requested plant owners to perform a systematic evaluation of plant-specific vulnerabilities and report the results to the Commission. For many plants, the IPEs became the basis for the plant's initial Probabilistic Risk Assessment (PRA). Later the NRC issued Supplement 4 to GL 88-20, that requested licensees to evaluate vulnerabilities to external events (IPEEE). Most licensees made improvements to their facilities to reduce vulnerabilities identified in their IPEs and IPEEEs.

The ground motions that are used as seismic design bases at US nuclear plants are called the Safe Shutdown Earthquake (SSE) ground motions. In the 1990s, the NRC staff reviewed the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting a generic review (i.e., GI-199) to again assess the resistance of US nuclear plants to earthquakes. Based on NRC's preliminary analyses to date, the average probability of ground motions exceeding the SSE over the life of the plant for the plants in the Central and Eastern United States is less than about 1%. It is important to remember that structures, systems and components are required to have "adequate margin," meaning that they must continue be able withstand shaking levels that are above the plant's design basis.

Q: *Is the NRC relooking at seismic analysis for US plants?*

A: The ground motions that are used as seismic design bases at US nuclear plants are called the Safe Shutdown Earthquake ground motion (SSE). In the mid to late 1990s, the NRC staff reviewed the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting a generic review referred to as GI-199, "Implications of Updated Probabilistic Seismic Estimates in Central and Eastern United States on Existing Plants," to again assess the resistance of US nuclear plants to earthquakes. In addition, the NRC has been reviewing new seismic information regarding the plants in California for many years.

U.S. Power Plants (General)

Q: *What is the seismic limit that Pilgrim Station, Seabrook Station and Vermont Yankee have been built to withstand?*

A: Each plant is designed to a ground-shaking level that is appropriate for its location, given the possible earthquake sources that may affect the site and its tectonic environment. Ground shaking is a function of both the magnitude of the earthquake and the distance from the fault plane to the site. The seismic responses of the structures, systems, and components associated with these facilities are dependent on several factors, as mentioned above; therefore, the responses may be different for the same magnitude earthquake. As a result, the NRC regulatory requirements focus on seismic limits based on ground shaking rather than limits defined by earthquake magnitude.

The ground motions associated with seismic events are determined for two categories of earthquakes: the Safe Shutdown Earthquake (SSE) which is generally defined as the maximum ground motion seismic response that the plant must be able to withstand and safely shut down and be maintained in a safely shut down condition, and; the Operating Basis Earthquake (OBE) which is defined as the ground motion seismic response that the plant must be able to withstand and to continue operating normally following such an event. The SSE and OBE reflect the horizontal acceleration of the ground in units of the earth's gravity, 'g'. The ground motions to which the Pilgrim, Seabrook, and Vermont Yankee plants are designed are: Pilgrim SSE of 0.150g and OBE of 0.080g; Seabrook SSE of 0.250g and OBE of 0.125g, and Vermont Yankee SSE of 0.140g and OBE of 0.070g.

Q: *What level of earthquake hazard are the US reactors designed for?*

A: Each reactor is designed for a different ground motion that is determined on a site-specific basis. The existing nuclear plants were designed on a "deterministic" or "scenario earthquake" basis that accounted for the largest earthquakes expected in the area around the plant, without consideration of the likelihood of the earthquakes considered. New reactors are designed using probabilistic techniques that characterize both the ground motion levels and uncertainty at the proposed site. These probabilistic techniques account for the ground motions that may result from all potential seismic sources in the region around the site. Technically speaking, this is the ground motion with an annual frequency of occurrence of 1×10^{-4} /year, but this can be thought of as the ground motion that occurs every 10,000 years on average. One important aspect is that probabilistic hazard and risk-assessment techniques account for beyond-design basis events. NRC's Generic Issue 199 (GI-199) project is using the latest probabilistic techniques used for new nuclear plants to review the safety of the existing plants.

Q: *What magnitude earthquake are currently operating US nuclear plants designed to?*

A: Ground motion is a function of both the magnitude of an earthquake and the distance from the fault to the site. Nuclear plants, and in fact all engineered structures, are actually designed based on ground motion levels, not earthquake magnitudes. The existing nuclear plants were designed based on a "deterministic" or "scenario earthquake" basis that accounted for the largest earthquakes expected in the area around the plant. A margin is further added to the predicted ground motions to provide added robustness.

Q: *If the same tragedy hit Pilgrim Station, Seabrook Station and Vermont Yankee would we be having the same major issues that the Japanese plants have? Please explain yes or no.*

A: The circumstances related to the events in Japan are highly unlikely in that the plant-specific external hazards profile is substantially different. All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate. Even those plants that are located in areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. The NRC requires that safety-significant structures, systems, and components be designed to take into account even very rare and extreme seismic and tsunami events. Pilgrim, Seabrook, and Vermont Yankee stations are designed to withstand the maximum credible natural events predicted for their specific sites. In addition to the design of the plants, significant effort goes into emergency response planning, preparation, and training. The NRC has also completed substantial research and analysis that resulted in the development and use of severe accident management guidelines. These insights have informed our decision making and review of licensed activities.

U.S. Power Plants (General)

Q: *Can significant damage to a nuclear plant like we see in Japan happen in the US due to an earthquake? Are the Japanese nuclear plants similar to US nuclear plants?*

A: All US nuclear plants are built to withstand environmental hazards, including earthquakes and tsunamis. Even those nuclear plants that are located within areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. The NRC requires that safety-significant structures, systems, and components be designed to take into account even rare and extreme seismic and tsunami events. In addition to the design of the plants, significant effort goes into emergency response planning and accident management. This approach is called defense-in-depth.

The Japanese facilities are similar in design to some US facilities. However, the NRC has required modifications to the plants since they were built, including design changes to control hydrogen and pressure in the containment. The NRC has also required plants to have additional equipment and measures to mitigate damage stemming from large fires and explosions from a beyond-design-basis event. The measures include providing core and spent fuel pool cooling and an additional means to power other equipment on site.

Q: *Could this happen at [any U.S. plant]?*

A: The events that have occurred in Japan are the result of a combination of highly unlikely natural disasters. These include the fifth largest earthquake in recorded history and the resulting devastating tsunami. This earthquake occurred on a "subduction zone", which is the type of tectonic region that produces earthquakes of the largest magnitude. A subduction zone is a tectonic plate boundary where one tectonic plate is pushed under another plate. Subduction zone earthquakes are also required to produce the kind of massive tsunami seen in Japan. In the continental US, the only subduction zone is the Cascadia subduction zone which lies off the coast of northern California, Oregon and Washington. So, a continental earthquake and tsunami as large as in Japan could only happen there. The only nuclear plant near the Cascadia subduction zone is the Columbia Generating Station. This plant is located a large distance from the coast (approximately 225 miles) and the subduction zone (approximately 300 miles), so the ground motions estimated at the plant are far lower than those seen at the Fukushima plants. This distance also precludes the possibility of a tsunami affecting the plant. Outside of the Cascadia subduction zone, earthquakes are not expected to exceed a magnitude of approximately 8. Magnitude is measured on a log scale and so a magnitude 9 earthquake is 32 times larger than a magnitude 8 earthquake.

The NRC believes that it is highly unlikely that a similar combination of events could occur in the United States. NRC and industry practices of defense in depth, conservative decision making, use of risk insights, and industry actions and coordination through the Institute of Nuclear Power Operations provides for further assurance that the facilities are safe.

U.S. Power Plants (General)

Q: *What is magnitude anyway? What is the Richter Scale? What is intensity?*

A: An earthquake's magnitude is a measure of the strength of the earthquake as determined from seismographic observations. Magnitude is essentially an objective, quantitative measure of the size of an earthquake. The magnitude can be expressed in various ways based on seismographic records (e.g., Richter Local Magnitude, Surface Wave Magnitude, Body Wave Magnitude, and Moment Magnitude). Currently, the most commonly used magnitude measurement is the Moment Magnitude, M_w , which is based on the strength of the rock that ruptured, the area of the fault that ruptured, and the average amount of slip. Moment magnitude is, therefore, a direct measure of the energy released during an earthquake. Because of the logarithmic basis of the scale, each whole number increase in magnitude represents a tenfold increase in measured amplitude; as an estimate of energy, each whole number step in the magnitude scale corresponds to the release of about 31 times more energy than the amount associated with the preceding whole number value.

The Richter magnitude scale was developed in 1935 by Charles F. Richter of the California Institute of Technology and was based on the behavior of a specific seismograph that was manufactured at that time. The instruments are no longer in use and the magnitude scale is, therefore, no longer used in the technical community. However, the Richter Scale is a term that is so commonly used by the public that scientists generally just answer questions about "Richter" magnitude by substituting moment magnitude without correcting the misunderstanding.

The intensity of an earthquake is a qualitative assessment of effects of the earthquake at a particular location. The intensity assigned is based on observed effects on humans, on human-built structures, and on the earth's surface at a particular location. The most commonly used scale in the US is the Modified Mercalli Intensity (MMI) scale, which has values ranging from I to XII in the order of severity. MMI of I indicates an earthquake that was not felt except by a very few, whereas MMI of XII indicates total damage of all works of construction, either partially or completely. While an earthquake has only one magnitude, intensity depends on the effects at each particular location.

Q: *How do magnitude and ground motion relate to each other?*

A: The ground motion experienced at a particular location is a function of the magnitude of the earthquake, the distance from the fault to the location of interest, and other elements such as the geologic materials through which the waves pass.

Q: *How big an earthquake is plant X designed to handle (for each plant)?*

A: All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate. Even those plants that are located in areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. Each plant is designed to a ground-shaking level that is appropriate for its location, given the possible earthquake sources that may affect the site and its tectonic environment. Ground shaking is a function of both the magnitude of the earthquake and the distance from the fault plane to the specific site. The seismic responses of the structures, systems, and components associated with these facilities are site specific. The plants are analyzed for certain identified faults and tectonic capabilities in the area while others are analyzed for seismic zones.

U.S. Power Plants (General)

Design: External Events: Tsunami

Q: *Could an accident sequence like the one at Japan's Fukushima Daiichi nuclear plants happen in the US?*

A: The NRC task force has issued its near-term review of the event and its impact on U.S. plants ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission). The current regulatory approach, and more importantly, the resultant plant capabilities allow the task force to conclude that a sequence of events like the Fukushima accident is unlikely to occur in the United States and some appropriate mitigation measures have been implemented, reducing the likelihood of core damage and radiological release. Therefore, continued operation and continued licensing activities do not pose an imminent risk to public health and safety. The NRC is planning a longer-term review and will review any new specific information regarding the disaster at the Fukushima plant and its applicability to U.S. reactors, identify lessons learned, and determine if any changes to its regulatory requirements are necessary to continue to ensure the health and safety of the public and the environment.

The NRC relies primarily on information made available to it by the Japanese government and several organizations involved in responding, assessing, and mitigating the events at the Japanese nuclear plants. Those sources have described how Fukushima Daiichi Units 1-3 lost all offsite power and emergency diesel generators. This situation is called "station blackout." US nuclear power plants are designed to cope with a station blackout event that involves a loss of offsite power and onsite emergency power. The NRC's detailed regulations address this scenario. US nuclear plants conducted a "coping" assessment and developed a strategy to demonstrate to the NRC that they could maintain the plant in a safe condition during a station blackout scenario. These assessments, proposed modifications to the plant, and operating procedures were reviewed and approved by the NRC. Several plants added additional AC power sources to comply with this regulation.

In addition, US nuclear plant designs and operating practices since the terrorist events of September 11, 2001, are designed to mitigate severe accident scenarios such as aircraft impact, which include the complete loss of offsite power and all on-site emergency power sources.

US nuclear plant designs include consideration of seismic events and tsunamis. It is important not to extrapolate earthquake and tsunami data from one location of the world to another when evaluating these natural hazards. These catastrophic natural events are very region- and location-specific, based on tectonic and geological fault line locations.

Q: *Are U.S. nuclear plants on the East Coast designed to withstand mega-tsunami waves 60–90 feet high that might be caused by a massive landslide into the Atlantic Ocean from eruption of a large volcano in the Canary Islands off northwest Africa?*

A: The NRC is aware of a study performed ten years ago which theorized that such a mega-tsunami could be generated by a massive landslide of one side of the Cumbre Vieja volcano on the Canary Island of La Palma. While this type of failure does occur to volcanoes in the Canary and Hawaiian Islands, significant problems with the original study have been identified by a number of subsequent studies performed in response to this study's theory. For example, studies performed since that time have refuted the highly conservative assumption that the landslide would hit the water in one coherent mass, but rather would do so in multiple landslides. In addition, improper modeling techniques and assumptions were used to assess how the resulting tsunami wave would propagate through the Atlantic Ocean. A recent report on tsunamis developed for the NRC by the US Geological Survey asserted that the tsunami generated by an eruption of this volcano would be, in fact, a small fraction of that size (i.e., no more than 3 feet). Nuclear stations on the East Coast of the United States are principally designed to deal with hurricane induced storm surges far higher than 3 feet. Thus based on the best, most reliable scientific studies currently available, the NRC has concluded that the existing flood protection measures of East Coast nuclear plants provide adequate margin against a tsunami induced by a flank landslide of this volcano.

U.S. Power Plants (General)

Q: *What could you say about the dangers to the American public from our nuclear plants?*

A: The NRC remains convinced that U.S. nuclear power plants are designed and operated in a manner that protects public health and safety. The NRC established a senior level task force to conduct both short- and long-term analysis of the lessons that can be learned from the situation in Japan. The task force has completed its near-term analysis and concluded that continued operation and continued licensing activities do not pose an imminent risk to public health and safety. The current regulatory approach and the resultant plant capabilities allow the task force to conclude that a sequence of events like the Fukushima accident is unlikely to occur in the United States and some appropriate mitigation measures have been implemented, reducing the likelihood of core damage and radiological releases. The task force recommended rulemaking activities, orders, certain staff actions, and actions for long-term evaluation. The Commission will review the report and will provide the staff with direction. A long-term evaluation is planned and will assess whether any additional licensing actions are necessary. These actions may include Orders, information requests in accordance with Section 50.54(f) of Title 10 (10 CFR) of the Code of Federal Regulations, license amendments, rulemaking, etc.

Q: *How many reactors are along coastal areas that could be affected by a tsunami? Is plant X designed to withstand a tsunami (for each coastal plant)?*

A: All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate. Many nuclear plants are located in coastal areas that could potentially be affected by a tsunami resulting from an earthquake. Two nuclear plants, Diablo Canyon and San Onofre, are on the Pacific Coast, which is known to have a tsunami hazard. There are many nuclear plants on the Atlantic Coast or on rivers that may be affected by a tidal bore resulting from a tsunami. These include St. Lucie, Turkey Point, Brunswick, Oyster Creek, Millstone, Pilgrim, Seabrook, Calvert Cliffs, Salem/Hope Creek, and Surry. In addition, there are two nuclear plants on the Gulf Coast, South Texas and Crystal River, that could potentially be affected by tsunami. Although tsunami on the Gulf and Atlantic Coasts may occur, it is very rare. Generally the flooding anticipated from hurricane storm surge exceeds the flooding expected from a tsunami for nuclear plants on these coasts.

Recent studies have looked at the potential of tsunami hitting the Gulf and Atlantic coasts, and have found that for many parts of the coast, tsunamigenic landslide (i.e., tsunami resulting from an underwater landslide) have the potential to exceed the seismically-induced tsunami. This research shows that the tsunamis produced by underwater landslides are localized, but can be extremely destructive in the nearby areas. The licensing basis for the coastal plants (i.e., FSARs) mentioned above did not specifically consider or assess this possibility, as the phenomenon was not well understood at the time. However, research supported by the NRC has been studying the issue since 2006. Although studies of tsunamigenic landslide continue, the current results indicated that flooding anticipated from hurricane storm surge, evaluated as part of the licensing basis for these plants, generally exceeds the flooding expected from a tsunami for nuclear plants on these coasts.

Q: *Do U.S. nuclear plants have better capabilities to respond to natural disasters than the plants in Japan?*

A: The NRC is not yet aware of all of the differences that may exist between the reactors that are of similar design and vintage as those operated in the U.S. Many improvements have been made to U.S. boiling water reactors (BWRs). For example, NRC Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," conveyed the importance of having a robust pathway for venting primary containment, which contains the suppression pool, in certain severe accident scenarios. In response, all BWRs with Mark I containments that didn't have an existing strengthened or "hardened" pathway for venting directly from primary containment to the outside, made modifications to the plant consistent with the intent of the Generic Letter. This design feature permits a controlled depressurization of primary containment as well as a controlled release of radioactive materials and combustible hydrogen that could be generated by damaged fuel, as may occur during severe accidents. U.S. nuclear power plants are built to withstand external hazards, including earthquakes tsunamis, and flooding, as appropriate. In addition to the design of the plants, significant effort goes into emergency response planning, preparation, and training. The NRC has also completed substantial research and analysis that resulted in the development and use of severe accident management guidelines. These insights have informed our decision making and review of licensed activities.

U.S. Power Plants (General)

Q: *If the same tragedy hit Pilgrim Station, Seabrook Station and Vermont Yankee would we be having the same major issues that the Japanese plants have? Please explain yes or no.*

A: The circumstances related to the events in Japan are highly unlikely in that the plant-specific external hazards profile is substantially different. All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate. Even those plants that are located in areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. The NRC requires that safety-significant structures, systems, and components be designed to take into account even very rare and extreme seismic and tsunami events. Pilgrim, Seabrook, and Vermont Yankee stations are designed to withstand the maximum credible natural events predicted for their specific sites. In addition to the design of the plants, significant effort goes into emergency response planning, preparation, and training. The NRC has also completed substantial research and analysis that resulted in the development and use of severe accident management guidelines. These insights have informed our decision making and review of licensed activities.

Q: *Are U.S. nuclear power plants designed to withstand tsunamis? What would the effect be on [plant X] if a subsequent tsunami hit?*

A: All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate. Many nuclear plants are located in coastal areas that could potentially be affected by a tsunami resulting from an earthquake. Two nuclear plants, Diablo Canyon and San Onofre, are on the Pacific Coast, which is known to have a tsunami hazard. There are many nuclear plants on the Atlantic Coast or on rivers that may be affected by a tidal bore resulting from a tsunami. These include St. Lucie, Turkey Point, Brunswick, Oyster Creek, Millstone, Pilgrim, Seabrook, Calvert Cliffs, Salem/Hope Creek, and Surry. In addition, there are two nuclear plants on the Gulf Coast, South Texas and Crystal River, that could potentially be affected by tsunami. Although tsunami on the Gulf and Atlantic Coasts may occur, it is very rare. Generally the flooding anticipated from hurricane storm surge exceeds the flooding expected from a tsunami for nuclear plants on these coasts.

Recent studies have looked at the potential of tsunami hitting the Gulf and Atlantic coasts, and have found that for many parts of the coast, tsunamigenic landslide (i.e., tsunami resulting from an underwater landslide) have the potential to exceed the seismically-induced tsunami. This research shows that the tsunamis produced by underwater landslides are localized, but can be extremely destructive in the nearby areas. The licensing basis for the coastal plants (i.e., FSARs) mentioned above did not specifically consider or assess this possibility, as the phenomenon was not well understood at the time. However, research supported by the NRC has been studying the issue since 2006. Although studies of tsunamigenic landslide continue, the current results indicated that flooding anticipated from hurricane storm surge, evaluated as part of the licensing basis for these plants, generally exceeds the flooding expected from a tsunami for nuclear plants on these coasts.

Emergency Preparedness

Q: *Is there a 50-mile emergency planning zone (EPZ) around U.S. reactors?*

A: There are two emergency planning zones (EPZ) established around a nuclear power plant. The first zone, the 10-mile EPZ, is where exposure from a radiological release event would likely be from the radioactive plume and it is in this EPZ where protective actions such as sheltering and/or evacuation would be appropriate. Beyond the 10-mile EPZ and out to the 50-mile EPZ is the ingestion exposure pathway where exposure to radionuclides would likely be from ingestion of contaminated food/milk and surface water. These zones are not limits but rather provide for a comprehensive emergency planning framework that would allow expansion of the response efforts beyond the zones should radiological conditions warrant such expansion.

U.S. Power Plants (General)

Q: *How is EPA monitoring, collecting and posting information related to the impacts in the U.S. of the accident in Japan?*

A: The EPA monitors, collects, and posts information related to the impacts of the Japanese events on the U.S. using their RadNet system. They have 100 fixed radiation monitoring sites in 48 states plus 40 additional deployable monitors that may be sent where needed. The fixed monitors provide information on beta and gamma radiation levels. The deployable monitors measure the external exposure rate and provide weather information. The data from these monitors is sent to a computer, where it is continually reviewed and is usually posted on the EPA's Central Data Exchange website (<http://epa.gov/cdx>) within 2 hours. However, if the computer picks up an abnormality in the radiation level, then the EPA laboratory staff is alerted and reviews the information prior to it being posted. In response to the events in Japan, EPA has sent additional monitors to Guam, Hawaii, and Alaska.

The EPA also monitors contamination in rainwater and drinking water as well as the level of iodine in milk. The EPA provides updates on these testing efforts and a summary of the air radiation monitoring results on its webpage, <http://www.epa.gov/japan2011/>. This webpage contains a link to Frequently Asked Questions, which was the source of information for this response. Additional information may be found there.

Q: *What is the NRC doing to ensure that nuclear power plants update their preparedness and evacuation plans to include protections for the millions of people living within 50 miles of those facilities? NEW!*

A: The NRC has conducted numerous studies on evacuations and their associated phenomena, including assessments of several large-scale, mostly "ad-hoc" evacuations that have occurred within the U.S. over the past 15 years. From this research, the NRC gained valuable insights into the evacuation process (as well as affirmed that evacuations are an effective tool to protect public health and safety). As a result, the forthcoming revisions to the NRC's emergency preparedness regulations (approved by the Commission in August 2011) update the NRC requirements for the evacuation time estimates (ETEs) that licensees must prepare. ETEs are used as a tool to develop and improve evacuation plans in advance of an accident and to decide whether sheltering or evacuation is the appropriate protective action following an accident. The NRC issued Draft NUREG/CR-7002, "Criteria for Development of Evacuation Time Estimate Studies," (available electronically in the NRC's Agencywide Documents Access and Management System (ADAMS) Accession No. ML102790350) in May 2010 to provide the latest guidance for licensees on how to develop a comprehensive set of ETEs.

NRC continues to work actively with its Federal, State, and local partners to continue to enhance the state of emergency preparedness around domestic nuclear power plants. These efforts include the distribution of potassium iodide (to date 26 million potassium iodide tablets have been distributed to States), revisions to the NRC and Federal Emergency Management Agency EP regulations and requirements, and enhancements and updates to the Environmental Protection Agency's Protective Action Guidelines Manual.

U.S. Power Plants (General)

Q: *What are the emergency planning zones established around U.S. nuclear power plants? NEW!*

A: One of the objectives of emergency response planning is to minimize the potential for public radiation exposure from a spectrum of accidents or incidents that could produce offsite doses in excess of protective action guidelines (PAGs). Two well-defined emergency planning zones have been established around domestic nuclear power plants: the 10-mile “plume exposure” emergency planning zone (EPZ) and the 50-mile “ingestion pathway” EPZ.

The size of the 10-mile EPZ is based on two principal factors: (1) projected doses from most accidents would not exceed the PAGs beyond 10 miles, and (2) projected doses from very low probability “worst-case” accidents would not exceed doses harmful to human health outside the 10-mile zone. In addition, the NRC and FEMA have concluded that detailed planning within 10 miles provides a substantial basis for expansion of response efforts in the event that expansion proved necessary. During the emergency at Fukushima, conditions degraded to a point that Japanese officials required additional protective actions beyond the established 10-km (6-mile) area around the facility. While the U.S. emergency preparedness (EP) framework has always considered the potential for expansion of the EPZ should it be necessary, the events in Japan provided a “real-world” look at the implementation of such an expansion. The NRC, as part of its longer-term review of its EP regulations, plans to examine the insights and lessons learned from the phased evacuations conducted beyond the established plume exposure EPZ around Fukushima.

A second and larger EPZ covers an area of about 50 miles around domestic nuclear power plants. The predominant concern for this area is exposure to radionuclides through ingestion. Scientific studies of the Chernobyl accident have shown that ingestion was the predominant exposure pathway to populations living at distances beyond the evacuation area. This ingestion exposure (e.g., drinking contaminated milk) resulted in elevated thyroid doses and the later development, in some children, of thyroid cancer. The predetermined protection action plans in place for the domestic 50-mile EPZ include prompt interdiction of contaminated food, dairy, and water products, as well as directives to provide stored feed to livestock and to remove them from pasture. In the days after the releases from the Fukushima site, Japanese officials worked quickly to monitor, identify, and interdict contaminated food products to prevent them from being consumed.

U.S. Power Plants (General)

Q: *The Severe Accident Management Guidelines (SAMGs) that licensees are supposed to have are voluntary and not part of the NRC's baseline inspections. Why are these considered voluntary? How does the NRC know that the licensees have SAMGs in place?*

A: The NRC carries out its mission to protect public health and safety by specifying licensing and operational requirements that nuclear power plants must meet, and by inspecting and enforcing compliance with these requirements. When a licensee complies with the regulations, "adequate protection" is presumed.

The NRC can only impose requirements beyond those necessary for adequate protection by satisfying the Backfit Rule (10CFR 50.109), which requires evidence of "a substantial increase in the overall protection of the public health and safety or the common defense and security", and that the costs of implementation are justified in view of the increased protection.

Protection against severe accidents is provided by regulatory requirements in two basic ways: 1) Prevention of core damage events such that the likelihood of events that lead to core damage is very low; and 2) mitigation of consequences in the event of a severe accident. The combination of these two aspects must result in an acceptably low risk to public health and safety. The NRC has determined that the combination of these two aspects does result in an acceptably low risk.

Severe Accident Management Guidelines (SAMGs) address mitigation of consequences in the event of a severe accident. A variety of regulations were already in place prior to the development of SAMGs to provide for the mitigation of accidents that were either postulated to occur (this is the deterministic approach) or were the most probable to occur (this is the probabilistic or risk-informed approach). The licensing basis for a plant typically contains a combination of these approaches to accident analysis. These include, for example, those regulations related to reactor containments (10 CFR 50, Appendix, A Section V) and fuel and radioactivity controls (10 CFR 50, Appendix A, Section VI), reactor siting criteria (10 CFR Part 100), and Emergency Planning requirements (10 CFR 50 Appendix E). The pre-SAMG "mitigation" requirements in conjunction with existing "prevention" requirements were judged to provide adequate protection. Therefore, while SAMGs further enhance mitigation capability, their contribution to risk reduction did not rise to the level of justifying a new requirement. Accordingly, the staff worked with industry to encourage voluntary implementation of SAMGs at all plants.

The Reactor Oversight Program is a risk-informed approach to inspection that focuses on assuring compliance with those requirements that are most risk significant. Since SAMGs are not a requirement (for the reasons noted above) they are not included in the NRC baseline inspection program. SAMGs provide an improvement/enhancement to the safety margins already inherent in meeting the regulatory requirements.

As part of the NRC response to the events in Japan, the NRC staff issued a temporary instruction to address the SAMGs. Temporary Instruction 2515/184 provided instructions for NRC inspectors to determine: (i) that the severe accident management guidelines (SAMGs) are available and how they are being maintained, and (ii) the nature and extent of licensee implementation of SAMG training and exercises.

Q: *The NRC has discussed the results of the inspections from the temporary instructions concerning station blackout and severe accident management guidelines in a public meeting. Where can I find more information on individual plants?*

A: The detailed inspection reports for these inspections are available at the NRC's public webpage at the following link: <http://www.nrc.gov/japan/japan-activities.html>.

U.S. Power Plants (General)

- Q:** *How does the process for taking protective measures following an accident (evacuation, sheltering, KI) work in the U.S. including the roles and responsibilities of Federal Government Agencies, State and local governments?*
- A:** Every nuclear power plant operator in the U.S. has an approved Emergency Plan that includes procedures for performing specific actions in response to an emergency, including the necessary interactions with State and Local authorities and responders. These Emergency Plans are exercised on a regular basis (i.e., every 2 years) and include participation of plant personnel, State and Local authorities and responders. The NRC also participates in these exercises in addition to providing oversight and evaluation of the exercise. In addition, the Federal Emergency Management Agency (FEMA) provides oversight of the offsite responses during these exercises. In the event of an emergency that would require activation of this plan, plant operators would work together with state and local authorities to direct and guide the actions of off-site responders and together would determine the need for evacuation and/or sheltering to minimize radiation exposure to the public. Decision-making regarding evacuation and/or sheltering would involve information regarding the actual emergency, conditions at the plant, mitigating actions being taken at the plant, meteorological conditions that could affect the direction of travel of any radioactive plume and potential dispersion of this plume. Although the NRC has been involved in providing funding for the purchase of potassium iodide (KI) for communities neighboring nuclear power plants, distribution of KI and directions for ingestion of KI are made at the State and Local levels. Federal government agencies involved in emergency response to nuclear power plant emergencies include the NRC and the FEMA. Other federal agencies that may become involved, depending on the severity of the situation, include the Department of Homeland Security (DHS) and other federal agencies. For Incidents of National Significance where the critical infrastructure is severely damaged, DHS has a lead role as a coordinating agency to orchestrate Federal, State, and local assets. The Nuclear/Radiological Incident Annex to the National Response Framework provides for the NRC to be a coordinating agency for incidents involving NRC licensed materials. Information regarding the National Response Framework is available at the following link:
<http://www.fema.gov/emergency/nrf/>.

U.S. Power Plants (General)

Q: *Why does the NRC not establish a 50-mile EPZ in the U.S. if this was the NRC's recommendation for the accident in Japan?*

A: The United States government cannot intervene in the management of events internal to another sovereign nation. The US government can only make recommendations to its citizens in that country on actions for their safety. The State Department routinely issues such recommendations (known as travelers warning and advisories) for many different types of events; civil unrest, terrorism, natural disasters and technological accidents. It is within this context that the Nuclear Regulatory Commission made a recommendation to the US Ambassador in Japan for protective actions for US citizens residing in the regions surrounding the damaged Fukushima Daiichi Nuclear Power Plant site.

The decision-making environment that existed at the time in which the NRC decision was made was one in which: there was limited and often conflicting information about the exact conditions of the reactors and spent fuel pools at the Fukushima nuclear facility immediately following the earthquake and tsunami; radiation monitors showed significantly elevated readings in some areas of the plant site which would challenge plant crews attempting to stabilize the plant; analysis results from offsite samples indicated that some fuel damage had occurred; there was a level of uncertainty about whether or not efforts to stabilize the plant in the very near term were going to be successful, and; changing meteorological conditions resulted in the winds shifting rapidly from blowing out to sea to blowing back onto land.

In its evaluation of the rapidly changing and unprecedented event, the NRC performed a series of dose calculations to assess a "worst case" scenario. This was a conservative calculation which considered the rapidly changing course of the events and the very real possibility that these events were going to continue to degrade. As a result of these calculations, the progression of events and the uncertainty regarding the plans to bring the situation under control, the decision was made to recommend the evacuation of US citizens out to 50 miles from the facility.

In the United States, the NRC has direct access to the plant site including the control room and any and all vital plant areas. The NRC maintains two resident inspectors at each plant who have unfettered access to the site. In addition, the NRC has required that direct communications links between the NRC Operations Center and the plant be installed, tested, and routinely exercised. These links provide NRC staff and the Executive team with up-to-date and reliable information about the ongoing events at the plant. In addition, the Chairman can order the plant to take actions to mitigate the event if the NRC does not believe that the appropriate actions are being taken by the plant operators.

In the U.S., there are two emergency planning zones (EPZ) established around a nuclear power plant. The first zone, the 10-mile EPZ, is where exposure from a radiological release event would likely be from the radioactive plume and it is in this EPZ where protective actions such as sheltering and/or evacuation would be appropriate. Beyond the 10-mile EPZ and out to the 50-mile EPZ is the ingestion exposure pathway where exposure to radionuclides would likely be from ingestion of contaminated food/milk and surface water. Comprehensive planning is performed for these zones and is routinely tested and evaluated by way of the full participation exercises. These zones are not limits but rather provide for a comprehensive emergency planning framework that would allow expansion of the response efforts beyond the zones should radiological conditions warrant such expansion. Nuclear power plant licensees are required to have an emergency plan for both the onsite and offsite response that has been evaluated and tested prior to obtaining an operating license and must conduct such exercises on a biennial cycle. The NRC remains confident that its current regulatory framework for emergency preparedness, including the establishment of an EPZ, and the flexibility to respond to emergent radiological conditions, as necessary, provides adequate protection for the health and safety of the public.

The NRC's Near-Term Task Force issued its report on July 12 and it is available to the public (ADAMS Accession No. ML111861807). On July 19, 2011, the Task Force presented its findings to the Commission and proposed improvements in multiple areas including emergency preparedness. The Task Force considered the existing planning structure, including the 10-mile plume exposure pathway and 50-mile ingestion pathway emergency planning zones, and found no basis to recommend a change. The development of protective action recommendations by the Japanese government, including expansion of evacuations out to 20 km (~12 miles) from the plant supported effective and timely evacuation to minimize the impact of the radiological releases on public health and safety. Subsequent decisions by the Government of Japan to evacuate selected areas based on potential long-term exposures are consistent with the U.S. strategy to expand protective actions during an event consistent with developments at the time and provided timely and effective actions to protect the public in those areas. Therefore, the Task Force found no basis to recommend changes to the emergency planning zones.

U.S. Power Plants (General)

Q: *How would the U.S. have responded to the events in Japan of March 11, 2011?*

A: The NRC requires plant designs to include multiple and diverse safety systems, and plants must test their emergency response capabilities on a regular basis. Plant operators are very capable of responding to significant events. U.S. nuclear power plants have emergency operating procedures as well as severe accident management guidelines that ensure that the containment structure integrity takes priority in an accident situation. Therefore, in an event that goes beyond those analyzed in the original plant design (i.e., beyond design basis event), such as the one at Fukushima Daiichi, U.S. BWR operators are trained to preserve primary and secondary containment by venting to provide the greatest assurance of public protection during a severe accident. Each U.S. plant has an emergency plan that is coordinated with local, State and Federal departments and agencies to ensure the safety of the public within the Emergency Planning Zone. In addition, NRC regulations require plants to have plans in place that would allow them to mitigate even worst-case scenarios. Since 9/11, we have implemented requirements for licensees to have additional response capabilities for extreme situations.

Q: *Do U.S. nuclear plants have better capabilities to respond to natural disasters than the plants in Japan?*

A: The NRC is not yet aware of all of the differences that may exist between the reactors that are of similar design and vintage as those operated in the U.S. Many improvements have been made to U.S. boiling water reactors (BWRs). For example, NRC Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," conveyed the importance of having a robust pathway for venting primary containment, which contains the suppression pool, in certain severe accident scenarios. In response, all BWRs with Mark I containments that didn't have an existing strengthened or "hardened" pathway for venting directly from primary containment to the outside, made modifications to the plant consistent with the intent of the Generic Letter. This design feature permits a controlled depressurization of primary containment as well as a controlled release of radioactive materials and combustible hydrogen that could be generated by damaged fuel, as may occur during severe accidents. U.S. nuclear power plants are built to withstand external hazards, including earthquakes tsunamis, and flooding, as appropriate. In addition to the design of the plants, significant effort goes into emergency response planning, preparation, and training. The NRC has also completed substantial research and analysis that resulted in the development and use of severe accident management guidelines. These insights have informed our decision making and review of licensed activities.

Q: *What is the basis for the dose analyses attached to the March 16, 2011, NRC press release?*

A: The basis for the dose assessment was the limited and unverifiable information on the plant conditions at the Fukushima facility. The facility was modeled in a computer-based dose assessment code as a hypothetical, four reactor site. The dose assessment results are conservative predictions only and may not be representative of any actual radiation releases. The computer-based dose assessment model also utilized predicted meteorological conditions following the events at the Fukushima facility and, therefore, may not be representative of the actual meteorological conditions that occurred for this area. The NRC press release of March 16, 2011, and the predicted dose estimates are available on the NRC's public website and may be accessed at the following link: <http://www.nrc.gov/reading-rm/doc-collections/news/2011/11-050.pdf>.

The assumptions on plant conditions used as the basis for the analyses were indicative of the uncertain and unstable nature of the conditions on Fukushima Daiichi site at the time the analyses were done, and accounted for uncertainty in the future progression of events. Since that time, actions to mitigate the events at facility and to stabilize the reactors and spent fuel at the plant have continued. The NRC continues to support the protective action recommendations provided in the March 16, 2011, press release because conditions at the plant continue to change. The NRC continues to monitor the situation at the Fukushima facility and may reassess its protective action recommendations as additional detailed and verifiable information about actual conditions becomes available.

U.S. Power Plants (General)

Q: *Why are US plants safe to operate considering the events in Japan?*

A: The NRC has been very closely monitoring the activities in Japan and reviewing all available information to allow us to conclude that the U.S. plants continue to operate safely. There has been no reduction in the licensing or oversight function of the NRC as it relates to any of the NRC licensees. Contributors to the conclusion that the current fleet of reactors and materials licensees continue to protect the public health and safety are based on a number of principles, including defense in depth.

Every U.S. reactor is designed for natural events, based on the specific site where the reactor is located. Every U.S. reactor has multiple fission product barriers, as well as a wide range of diverse and redundant safety features. All these factors support the NRC's conclusion that public health and safety can be assured. The NRC has a long regulatory history of conservative decisionmaking. The NRC has been intelligently using risk insights to help inform the regulatory process and has required improvements to the plant designs as we learn from operating experience. Some of these include severe accident management guidelines, revisions to the emergency operating procedures, procedures and processes for dealing with large fires and explosions regardless of the cause, and requirements for coping with station blackout.

The NRC's task force examining the accident at Fukushima Daiichi and its impact on U.S. plants ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission) has concluded that continued operation and continued licensing activities do not pose an imminent risk to public health and safety.

Q: *How does the NRC ensure people can escape if an accident occurs from a natural disaster when the infrastructure is also affected or destroyed in an area around a plant?*

A: Each US nuclear power plant has an Emergency Plan for ensuring the health and safety of people who live within the emergency planning zone. Emergency plans contain contingencies for alternate evacuation routes, alternate means of notification, and other backup plans in the event of a natural disaster that damages the surrounding infrastructure. Licensees exercise these plans on a regular basis. The NRC performs oversight to verify the acceptable performance of the licensee's response during exercises, drills, and actual incidents and events. The Federal Emergency Management Agency (FEMA) provides oversight for offsite response.

For Incidents of National Significance where the critical infrastructure is severely damaged, the Department of Homeland Security (DHS) has a lead role as a coordinating agency to orchestrate Federal, State, and local assets. The Nuclear/Radiological Incident Annex to the National Response Framework provides for the NRC to be a coordinating agency for incidents involving NRC licensed materials.

Q: *How did the NRC develop its computer-based projections that supported the evacuation decision?*

A: The NRC uses the RASCAL computer code to perform offsite radiation dose projections. The RASCAL computer program contains information about U.S. nuclear reactor design types, radiation release pathways from the nuclear power plant to the environment, radionuclide source terms and meteorology. However, RASCAL is not capable of evaluating concurrent and multiple nuclear plant failures. So, to approximate the events unfolding at the Fukushima Daiichi facility, the NRC developed a model that aggregated information from the three operating reactors and the spent fuel pool. This aggregate model was then evaluated using the RASCAL computer code. The radiation doses calculated by the RASCAL code were predicted to exceed the protective action guidelines (PAGs) established by the U.S. Environmental Protection Agency (EPA) well beyond the 10-mile exposure pathway EPZ and beyond the 30 kilometer sheltering zone recommended by the Japanese authorities. Subsequent aerial monitoring by the U.S. Department of Energy (DOE) fixed-wing aircraft monitoring showed elevated radiation dose rates that were in excess of the EPA relocation PAGs to a distance beyond 25 miles from the facility.

U.S. Power Plants (General)

Q: *Why did the NRC decide to recommend evacuation out to 50 miles from the Fukushima Daiichi facility for U.S. citizens in Japan?*

A: The decision to expand evacuation of U.S. citizens out to 50 miles from the Fukushima Daiichi facility was a conservative decision that was made out of consideration of several factors including an abundance of caution resulting from limited and unverifiable information concerning event progression at several units at the Fukushima Daiichi facility. The NRC based its assessment on information available at the time regarding the condition of the units conditions at Fukushima Daiichi that included significant damage to Units 1, 2, and 3 that appeared to have been a result of hydrogen explosions. Prior to the earthquake and tsunami, Unit 4 was in a refueling outage and its entire core had been transferred to the spent fuel pool only 3 months earlier so the fuel was quite fresh. Radiation monitors showed significantly elevated readings in some areas of the plant site which would challenge plant crews attempting to stabilize the plant. Based on analysis results, there were indications from some offsite contamination sampling smears that fuel damage had occurred. There was a level of uncertainty about whether or not efforts to stabilize the plant in the very near term were going to be successful. Changing meteorological conditions resulted in the winds shifting rapidly from blowing out to sea to blowing back onto land.

Q: *Did the NRC share the post 9/11 enhancements to the U.S. facilities with the Japanese?*

A: Following the events of September 11, 2001, the NRC issued Orders requiring licensees to develop specific guidance and strategies to maintain or restore cooling of the core, containment, and spent fuel using existing or readily available resources (equipment and personnel). These strategies have to be implemented effectively even if large areas of the plant were lost due to explosions or fire, including those that an aircraft impact might create. Although it was recognized prior to September 11, 2001, that nuclear reactors already had significant capabilities to withstand a broad range of attacks, implementing these types of mitigation strategies would significantly enhance the plants' capabilities to withstand a broad range of threats. NRC's Japanese counterpart, the Japan Nuclear and Industrial Safety Agency (NISA), visited NRC in 2008. During that visit, NRC staff shared information contained in the NRC-issued Orders as referenced above. This cooperative exchange occurred under the authority of an international agreement between NRC and NISA for technical exchange.

Fuel Cycle Facilities

Q: *How many people live near the MTW site?*

A: The plant site is located in a predominantly agricultural area. Within a two mile radius of the plant, approximately 68% of the land is undeveloped (e.g., cropland, forest, or wetland) and the remainder is developed. Within a one-mile radius of the facility, the total population is 558 persons; most of these are concentrated in the E to ESE sectors near the city of Metropolis. The nearest residence sampling device is currently located between the two nearest residences, approximately 1850 feet NNE of the Feed Materials Building. MTW is located in Massac County, IL which has approximately 15,000 residents in the year 2000. The nearest cities are Metropolis, IL which has about 6,500 people and is located 1 mile from the site. Paducah, KY is located 10 miles from the site and has a population of approximately 25,000. Within a 50 mile radius there are approximately 500,000 people. There are no facilities that would present significant evacuation problems within the immediate vicinity of the site. In addition, the Protective Action Recommendations provided in the Emergency Response Plan are limited to shelter-in-place only; no provisions are required for evacuation of the near-site public.

Q: *How would the NRC and USEC respond to an earthquake at the PGDP facility?*

A: The NRC currently has two permanent resident inspectors on site at Paducah. These inspectors are available and on call 24/7 in the case of any plant emergency including an earthquake. If applicable the licensee would notify the NRC Operations Center and additional resources would be mobilized from the R-II office or headquarters depending on the severity of the hazard and the particular area of concerns within the plant. The onsite NRC inspectors would relay pertinent information through the Operations Center to the appropriate headquarters staff to continually reassess the hazards and the agency response. The NRC Operations Center would also assist in coordinating additional government agency responses. It is important to note that the licensee would have the lead in the response and the NRC would monitor the licensee's response providing oversight, assistance and coordination as necessary.

U.S. Power Plants (General)

Q: *What would happen if an earthquake occurred in the vicinity of the PGDP facility?*

A: Only failures of cascade UF6 piping operating above atmospheric pressure contribute to exposures from a seismic event. It is possible, although highly unlikely, that a Design Basis seismic event could result in a criticality accident. The radiological consequence of a seismic event is 37 mrem at the site boundary. The chemical consequence of a seismic event is 4 mg U and 2 ppm HF at the site boundary. The consequence of a criticality accident is 7.74 rem at the site boundary.

Q: *What are the potential impacts of an earthquake that exceeded the design of the PGDP facility?*

A: Earthquakes can cause damage which results in fires and releases of hazardous materials. Earthquakes are considered an unlikely cause of a criticality accident because such accidents require enriched uranium to accumulate in an unsafe, critical mass. The contents of broken pipes and containers tend to be dispersed, not accumulated. Liquid and gaseous UF6 operations as well as waste storage are potentially at risk in the seismic event. A severe earthquake could result in toppling of equipment and collapse of structural walls and members, causing a release of uranium to the environment, and possibly a nuclear criticality accident, potentially resulting in a high consequence event to workers.

Q: *What is the design basis earthquake for the Paducah Gaseous Diffusion Plant (PGDP)?*

A: Design Basis seismic event causes (0.165g peak ground acceleration) rupture of cascade process piping (expansion joints) operating above atmospheric pressure. The failures predicted at withdrawal facilities either do not involve UF6 or else occur in piping and equipment that contains gaseous UF6 operating at subatmospheric pressure, therefore, no UF6 release is postulated for the withdrawal facilities. Process building oil systems can withstand the design basis earthquake. All facilities that store significant quantities of UF6 were analyzed for seismic effects up to the evaluation bases earthquake. All facilities have structural capacities at least equal to this peak ground acceleration, i.e. no building collapse.

Q: *Can an earthquake as large as Japan also happen at MTW?*

A: The Japan earthquakes experienced a ground motion corresponding to 2.7 g. Under current design bases for MTW the maximum credible ground motion at MTW is between than 0.4 g and 1 g. Thus, an earthquake as large as the Japan event was not plausible.

Under the new USGS information, a ground acceleration of 2.124g is associated with the Maximum Credible Earthquake. However, MTW is in the process of investigating this new information and assessing the potential impacts and possible need for plant and system modifications.

Q: *What would be the impact of a tsunami or flood at the MTW site?*

A: The MTW site is located inland, approximately 550 miles from the coast, so a tsunami is not a plausible scenario. Additionally, the site is not located near any large body of water that could cause a flood at the site. The nearest large surface waters are the Ohio River forms the site perimeter to the south. Flood control on the Ohio River is provided by dams. The nearest dam is located 7 miles (11 km) upriver at Brookport, Illinois. The historic maximum elevation flood at Metropolis was 342 feet in 1937. The 100-yr recurrence flood level developed by FEMA in 1983 is 337 feet. The 375' MTW site is at a relatively high elevation point. The town of Metropolis, for example is at a nominal elevation of 350', and the Kentucky side of the river is at about 350' elevation for a wide area. The West Kentucky Airpark (15 miles SE) landing strips are at 338' elevation. The Flood Map for Massac County indicates that the site is in flood zone C, which is not in either the 100-yr or 500-yr flood plain, and has no flood exposure. Thus the relative elevations make it apparent that the MTW site is not susceptible to rising river water floods. Therefore, flooding by the Ohio River can be considered a non-credible hazard. Although not credible, flooding caused by rising water in the Ohio River could cause building or tank farm damage leading to containment breach and release of contents.

Flooding is also under reassessment and new responses and strategies may be developed as a result.

Q: *Does the MTW site have a spent fuel storage area?*

A: No. The conversion stage of the fuel cycle, which converts U3O8 into UF6, occurs prior to the enriched UF6 being fabricated into UO2 fuel for use in nuclear reactors. Therefore the radioactive material currently being generated or previously generated at a conversion plant has never been used in a nuclear reactor and there is no need to store spent fuel at such a site.

U.S. Power Plants (General)

Q: *What preparations are currently in place to respond to an emergency at the PGDP?*

A: The site Emergency Response Plan contains responsibilities, procedures, instructions, protective actions, and exposure guidelines for the postulated emergencies. The facility has an onsite emergency response organization with some limited medical, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, and hospital for additional emergency response resources as needed. Recommendations are based on the release quantity/chemical/physical state and atmospheric conditions and are discussed with participating government agencies as appropriate. In support of emergency response operations at the plant, the USEC Emergency Operations Facility (EOF), located in Bethesda, Maryland, provides oversight, makes appropriate notifications, coordinates interactions with the public and media, and may request assistance from Federal agencies. The state and county have overall responsibility and authority for conducting appropriate emergency response and local implementation of recommended protective actions.

Q: *What preparations are currently in place to respond to an emergency at MTW?*

A: The site Emergency Response Plan contains responsibilities, procedures, instructions, protective actions, and exposure guidelines for the postulated emergencies. The facility has an onsite emergency response organization with some limited medical, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, and hospital for additional emergency response resources as needed. Recommendations are based on the release quantity/chemical/physical state and atmospheric conditions and are discussed with participating government agencies as appropriate. The state and county have overall responsibility and authority for conducting appropriate emergency response and local implementation of recommended protective actions.

New responses and strategies may be developed as a result of the new assessment underway.

Q: *Does the PGDP site have a spent fuel storage area?*

A: No. The enrichment stage of the fuel cycle occurs prior to the enriched UF₆ being fabricated into UO₂ fuel for use in nuclear reactors. Therefore the radioactive material currently being generated or previously generated at an enrichment plant has never been used in a nuclear reactor and there is no need to store spent fuel at such a site.

Q: *Are the MTW emergency power diesels built to withstand the effects of an earthquake, if not what happens if the MTW facility were to lose offsite power as a result of an earthquake?*

A: The plant itself has many redundant safety systems, such as sensors and mitigation systems. Critical safety systems have back-up power sources in case of a power outage and it is expected those systems would perform as designed. The plant also has an uninterruptable power supply (UPS) system that immediately provides power in the event of a power failure to assist us to shut down the plant processes safely.

Standby utilities are maintained in order to facilitate a safe and orderly shutdown of the process units during a complete power failure. Standby electrical power is provided from an electrical generator located in the Powerhouse Building. The standby electrical generator is diesel powered and delivers 480 volts AC. In the event that electrical power is interrupted, the standby generator automatically starts and comes to a standby mode. The standby power is then distributed, as required. As described in (1) above, MTW is in the process of investigating new seismic information and assessing the potential impacts and possible need for plant and system modifications.

New responses and strategies may be developed as a result of the new assessment underway.

Q: *What are the most significant hazards at the MTW site?*

A: The most significant radiological hazard at the MTW site is the release of UF₆ material. The most significant chemical hazard is the release of HF or NH₃.

Although significant changes are not expected, new results may be obtained as a result of the assessment underway.

U.S. Power Plants (General)

Q: *Are the emergency power diesels built to withstand the effects of an earthquake, if not what happens if the PGDP facility were to lose offsite power as a result of an earthquake?*

A: Loss of electrical power (“station blackout”) is not a design basis event at PGDP. Loss of electrical power results in: UF6 compressors stop; autoclaves’ containment valves shut (fail-safe); withdrawal compression sources (Normetex pumps) stop; UF6 cylinder handling cranes fail “as-is”. No residual heat removal is required at PGDP. PGDP is supplied by 18 individual 161-KV power lines from three separate suppliers connected to four interconnected site switchyards. Diesel-driven electric generators are installed in all process buildings. These are not safety systems. The process buildings have large electrical storage battery banks to provide backup DC power. Loss of cascade compressors reduces cascade pressure to below atmospheric pressure (most of the cascade was below atmospheric pressure initially) and UF6 cools by ambient heat loss to solid state. Diesel generators provide power to close cascade valves for economic reasons, but are not safety significant. Liquid UF6 in cylinders and piping cools by ambient heat loss to solid state. Ambient heat loss is the normal cooling method for liquid UF6 in cylinders. Liquid UF6 cylinder handling cranes are seismically qualified: load remains suspended. HPFW is supplied by the C-611-R elevated storage tank and the diesel-driven fire pump. In summary, there is no safety impact from a loss of all offsite power. The equipment shuts down and cools to ambient temperatures by natural means. There are no residual heat issues.

Q: *How would the NRC and the licensee respond to an earthquake at the MTW facility?*

A: The NRC currently has two permanent resident inspectors on site at the Paducah USEC facility which is approximately 15 miles from MTW. These inspectors are available and on call 24/7 in the case of any plant emergency including an earthquake. These inspectors could quickly respond to an emergency at MTW from the USEC site if necessary. If applicable the licensee would notify the NRC Operations Center and additional resources would be mobilized from the R-II office or headquarters depending on the severity of the hazard and the particular area of concerns within the plant. The onsite NRC inspectors dispatched from PGDP would relay pertinent information through the Operations Center to the appropriate headquarters staff to continually reassess the hazards and the agency response. The NRC Operations Center would also assist in coordinating additional government agency responses. It is important to note that the licensee would have the lead in the response and the NRC would monitor the licensee’s response providing oversight, assistance and coordination as necessary.

New responses and strategies may be developed as a result of the new assessment underway.

Q: *How many people live near the PGDP site?*

A: Paducah, Kentucky, approximately 10 miles east, with a 2010 population of 25,000 is the largest city in the immediate region. The city of Metropolis, Illinois, with a 2010 population of about 6,000, is situated approximately 5 miles east of the plant. Two unincorporated communities, Grahamville and Heath, are located approximately 2 miles east of the plant. Part of 28 counties in 4 states fall within a 50-mile radius of the plant.

Q: *Are the MFFF emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: In the unlikely event that a total loss of all incoming power occurs and both standby diesel generator systems fail to start as discussed before, an independent and redundant seismically qualified emergency power system will provide electrical power to the facility. The emergency power system consists of two redundant and independent emergency diesel generator systems each of which has been designed to carry all important loads during an extended period of time until either the normal or standby power system can be restored. The emergency power system is qualified to survive the MFFF design-basis earthquake.

Q: *What are the most significant hazards at the PGDP site?*

A: A criticality accident represents the potential for a lethal radiation dose to a worker within 10 to 50 feet of the criticality, and lesser but significant doses out to 100 feet or more. The total radiation to an individual at the site boundary would exceed regulatory limits, but would not result in the potential for radiation doses large enough to cause injury. There are no likely criticality scenarios that would be initiated by an earthquake or loss of offsite power event. Another significant offsite hazard is the potential release of UF6 due to an earthquake via damage to process structures, systems and components.

U.S. Power Plants (General)

Q: *Can an earthquake as large as Japan also happen at PGDP?*

A: The most recent earthquake in Japan was caused by a “subduction zone” event, which is the type of mechanism that produces the largest magnitude earthquakes. A subduction zone is a tectonic plate boundary where one tectonic plate is pushed under another plate. In the continental US, the only subduction zone is the Cascadia subduction zone which lies off the coast of northern California, Oregon and Washington. So, an earthquake and tsunami this large could only happen in that region. The Japan earthquakes experienced a ground motion corresponding to 2.7 g and the maximum credible ground motion at MTW is between than 0.4 g and 1 g.

Q: *Can a U.S. fuel cycle facility have a criticality accident as the result of an earthquake?*

A: In general, fuel cycle facilities are constructed to the local Uniform Building Code so as to withstand anticipated earthquakes. Facilities’ Integrated Safety Analyses (ISAs) also consider natural phenomena hazards, including earthquakes and severe weather, and must demonstrate that chemical and radiological hazards, including criticality, have an acceptable level of risk. For example, criticality accidents must be shown to be highly unlikely.

Criticality, in general, requires the accumulation of a sufficient mass of nuclear material into a compact geometry, such as a sphere. It also requires a certain quantity of moderator, materials that slow down neutrons and enhance their ability to cause fission, the most common of which is water. During an earthquake, nuclear material would tend to be dispersed over a wide area, the opposite of what is needed for criticality. Nuclear facilities are required to be designed so that at least two unlikely, independent, and concurrent changes in process conditions would be needed before criticality is possible. Accumulation, rather than dispersion, of nuclear material, in the right geometric shape and with the right quantity of moderator for criticality to occur, would require the occurrence of several unlikely events and is considered to be very unlikely.

Q: *What would be the consequences of a criticality accident at a fuel cycle facility to a member of the public?*

A: A criticality accident is considered a high consequence event to a worker in the immediate area. Outside approximately fifteen feet, a lethal dose is unlikely. Beyond approximately 200 feet, a significant dose is unlikely. Fuel cycle facilities are required to perform analyses to evaluate the chemical and radiological consequences to members of the public. Most fuel facilities in the U.S. are situated on sites where the distance to the site boundary, and the presence of shielding material, such as concrete walls and steel containers, precludes any significant exposure to members of the public. Criticality is therefore considered a localized industrial hazard with little or no off-site consequences.

Q: *What is the design basis earthquake for the GE-Hitachi Laser-Based Uranium Enrichment Facility (GLE)?*

A: The NRC staff is currently in the process of evaluating GLEs license application. The proposed site of the facility is located in Wilmington, North Carolina. This location is considered a low earthquake hazard area. The proposed facility is located inland 10 miles west and 26 miles north of the Atlantic Ocean. In the guidance is NUREG/CR-6966, “Tsunami Hazard Assessment at Nuclear Power Plant Sites in the United States of America,” this location is greater than 1 mile inland from the coast and is not considered susceptible to a tsunami.

Q: *What are the potential impacts of an earthquake that exceeded the design of the GLE facility?*

A: The NRC staff is currently in the process of evaluating GLE’s license application.

Q: *What would happen if an earthquake occurred in the vicinity of the GLE facility?*

A: The NRC staff is currently in the process of evaluating GLE’s license application.

Q: *How would the NRC and GLE respond to an earthquake at the GLE facility?*

A: NRC and GLE would respond consistent with standard procedures for emergency events. GLE procedures are described in its Radiological Contingency and Emergency Plan.

Q: *What happens if the GLE facility were to lose offsite power as a result of an earthquake?*

A: The NRC staff is currently in the process of evaluating GLE’s license application.

U.S. Power Plants (General)

Q: *Are the GLE emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: The NRC staff is currently in the process of evaluating GLE's license application.

Q: *What would be the impact of a tsunami or flood at the PGDP site?*

A: The PGDP site is located inland, approximately 550 miles from the coast, so a tsunami is not a plausible scenario. Additionally, the site is not located near any large body of water that could cause a flood at the site. The nearest large surface waters are the Ohio, Mississippi, Tennessee, and Cumberland Rivers which are about 20 miles from the PGDP site. PGDP is at least 12 feet above any conceivable flooding event, and, therefore, there is no safety impact from a flood at PGDP.

Q: *Are the NFS emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: There is no special earthquake protection for the diesel generators. Buildings housing diesel generators meet the same building code requirements as other buildings on the site. Processing equipment will stop operating when normal power is lost. Emergency systems (detectors, alarms, lighting, etc.) would stop operating if emergency power is lost. At that point, emergency response workers would rely on portable equipment such as survey meters, flashlights, and portable breathing air equipment.

Q: *Are special procedures employed at GLE for other natural phenomena events?*

A: The NRC staff is currently in the process of evaluating GLE's license application.

Q: *How many people live near the B&W NOG site?*

A: The closest resident to the facility is approximately 4,500 feet directly west. The nearest potential off-site worker would be an occupational worker at the AREVA facility, approximately 3,000 feet to the northeast.

Offsite population totals are as follows:

1 mile: 302
2 miles: 1100
3 miles: 2424
4 miles: 4557
5 miles: 9070

Q: *What preparations are currently in place to respond to an emergency at GNF-A?*

A: The site Emergency Plan contains instructions and Protective Action Recommendations (PARs) for the postulated emergencies. The facility has an onsite Emergency Team with medical, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, hospitals, New Hanover County and the State of North Carolina to provide support and assistance if there is an emergency.

Q: *What happens if the GNF-A facility were to lose offsite power as a result of an earthquake?*

A: Process equipment will fail safe if the electrical service is interrupted. Emergency power is provided for a supervised alarm system and essential equipment. There are four emergency diesel generators onsite. Additionally, the facility has emergency power capabilities for support services, the controlled areas, and the emergency control center. A diesel-operated generator will provide an automatic startup and a switch over to the emergency system in the event of a power failure.

Q: *Are the GNF-A emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: The emergency diesels are not seismically qualified. There are no safety significant scenarios which would result from a loss of offsite power. Risk-significant safety controls are passive or fail to a safe position without relying upon electric power.

U.S. Power Plants (General)

Q: *What would be the impacts of a tsunami or a flood at the GFN-A site? How would the NRC and the licensee respond to these events?*

A: The facility is more than 10 miles from the Atlantic coastline and at an elevation 25–30 feet above sea level. Tidal bores are also highly unlikely given the distance from the coastline, the quick dissipation as bores travel upstream, and the elevation of the facility site.

Q: *What are the most significant hazards at the GNF-A site?*

A: A criticality accident represents the potential for a lethal radiation dose to a worker within 10 to 50 feet of the criticality, and lesser but significant doses out to 100 feet or more. The total radiation to an individual at the site boundary would exceed regulatory limits, but would not result in the potential for radiation doses large enough to cause injury. There are no criticality scenarios that would be initiated by an earthquake or loss of offsite power event.

Q: *What is the design basis earthquake for the NFS facility?*

A: Buildings were constructed in accordance with the Standard Building Code in effect at the time of construction. Buildings constructed recently were designed to meet the seismic load resistance specified in American Society of Civil Engineers (ASCE) Standard 7, Minimum Design Loads for Buildings and Other Structures. The ASCE 7 design loads are based on an earthquake with a 2500 year return period and a peak ground acceleration of 0.31 g. Buildings codes provide requirements for structures to be design and constructed taking in consideration earthquakes and other natural phenomena events. The provisions of the codes are aimed to ensure that buildings don't fail catastrophically under an event providing for the safe egress of its occupants.

Q: *What are the potential impacts of an earthquake that exceeded the design of the NFS facility?*

A: Earthquakes can cause damage which results in fires and releases of hazardous materials. A criticality accident is possible, but earthquakes are considered an unlikely cause of a criticality accident because such accidents require enriched uranium to be accumulated in a critical mass. The contents of broken pipes and containers tend to be dispersed, not accumulated. It has been noted that one control used for criticality safety is keeping moderators, such as water, away from enriched uranium. Broken pipes and damaged roofs may allow water into normally dry areas. The loss of a criticality control would increase the risk of an accident, but nuclear facilities are required to be designed so that at least two unlikely, independent, and concurrent changes in process conditions would be needed before criticality is possible. The accidental accumulation of enriched uranium in conditions which favor a criticality accident are possible, but unlikely. A criticality accident is most hazardous to workers near the accident site, not members of the public offsite, because radiation levels decrease rapidly with distance.

Building codes incorporate occupant safety margins such as maintaining structural integrity long enough to allow occupants to leave the building. Maintaining structural integrity will reduce the damage that could lead to fires and releases of hazardous materials. If an earthquake exceeded the design of the facility, structural failure would be more likely increasing the risk that occupants would be injured and unable to escape. The likelihood of fires and releases would increase. The regulatory analysis of emergency preparedness in NUREG-1140 concluded that the potential for radiation doses large enough to cause an acute fatality or early injury to a member of the public is not considered plausible at a nonreactor facility. An earthquake exceeding the design of a facility similar to NFS may cause offsite impacts to increase, but we believe large radiation doses are still unlikely.

Q: *What would happen if an earthquake occurred in the vicinity of the NFS facility?*

A: If an earthquake occurred near NFS, it is expected that the facility will experience structural damage, but not catastrophic collapse of the buildings. The vast majority of uranium is in solid form or liquid form. We would expect some material to be spilled, but most of it would remain inside the buildings. Fires from broken gas lines and other flammable materials are possible. Firefighting efforts may be hampered by blocked roads, broken water lines, and other earthquake related damage. A major fire may require protective actions to minimize radiation dose. If a dose between 1 rem and 5 rem is possible, the initial recommendation in the NFS Emergency Plan is to shelter the public within 1 mile of the site (approx. 2800 people). If a dose greater than 5 rem is possible, the plan recommends limited evacuations within 550 yards of the site. The plan is consistent with guidance in the EPA Manual of Protective Action Guides.

U.S. Power Plants (General)

Q: *Would overpressurized uranium hexafluoride cylinders stored at NFS greatly increase the risk to members of the public during an earthquake?*

A: No. Our analysis of the potential breakdown of uranium hexafluoride (UF₆) into uranium pentafluoride (UF₅) and elemental fluorine (F₂) concluded that potential chemical consequences are low. All of the cylinders are small. A few are about two gallons in size, but most are less than a quart each. They are stored in shipping containers inside buildings. If the containers were outside and all were to fail simultaneously, a person standing on the property line might notice a strong odor and experience discomfort and irritation, which would be of short duration and cease when the person moves away. However, it is unlikely all cylinders will fail simultaneously. The shipping containers would confine any releases. In addition, the buildings would help confine any material that managed to escape the shipping containers. In the unlikely event of a container breach, it is unlikely that a person standing on the property line would experience anything more than an unpleasant odor.

Q: *What is the design basis earthquake for the Honeywell Metropolis Works (MTW) facility?*

A: The site area is in the northern part of the Mississippi Embayment, which has had a long history of seismic activity. The only major earthquakes in historic times were the New Madrid earthquakes of 1811-1812, centered about 60 miles southwest of the site. This earthquake was one of the strongest on record in this country. Major faults, trending toward New Madrid, are found approximately twenty-five to thirty miles east and west of the site. These faults, which occurred millions of years ago, have not been active in geologically recent time. Seismologists are unable to accurately predict the recurrence rates for destructive earthquakes such as those of 1811-1812 because of their infrequent occurrences. Nevertheless, experience indicates that major earthquakes originating along the New Madrid fault zone are capable of causing extensive damage in the Metropolis area. One such estimate concluded that a recurrence of an earthquake of the New Madrid intensity had a maximum likelihood of occurring once in 100-300 years in the entire seismic region.

While MTW is clearly located in a high seismicity zone, the original plant construction, which began in the late 1950s, did not adequately address seismic concerns. A 1993 report identified structural modifications to the Feed Materials Building and the tank farm to assure adequate performance. The structural modifications are designed to withstand a 475-yr recurrence site-specific earthquake. This is judged to be reasonably consistent with NUREG 1520 requirements even though NUREG 1520 specifies that the 500-yr earthquake be used, since the uncertainty in earthquake prediction is large relative to the small difference between recurrence periods. It is also stressed that MTW is not currently required to meet NUREG 1520, and other reasonable safety standards can be used.

The USGS completed the National Seismic Hazard Mapping Project in 1996. This project resulted in the development of new earthquake ground motion maps. These maps are also the referenced basis for IBC 2006 standards. A comparison of the 1996 ground motion parameters to that used in the Metropolis 1993 study shows considerable difference with the 1996 being more intense. As a result, MTW is in the process of investigating this new information and assessing the potential impacts and possible need for plant and system modifications.

Q: *What happens if the NFS facility were to lose offsite power as a result of an earthquake?*

A: If electrical power is lost, the processing equipment is designed to shut down to a safe condition. Site emergency power is available from two independent, uninterruptible power supply (UPS) systems. The UPS systems would maintain power to safety related equipment such as criticality alarms, fire alarms, security systems, radiation detectors, and emergency lighting. Each UPS would transfer load to batteries, send a start signal to the diesel generator, and transfer the load to the generator when the appropriate voltage is reached.

Q: *What would happen if an earthquake occurred in the vicinity of the MTW facility?*

A: In the current seismic assessment, a seismic event has the potential to result in a release of HF or UF₆. The seismic event probability is low. However, considerable work has been done to identify potential seismic risk to the Metropolis Works. Therefore the structural modifications constitute robust passive engineering to mitigate the event with low probability of failure.

As stated, a review is underway based on new USGS information.

U.S. Power Plants (General)

Q: *Can an earthquake as large as happened in Japan also happen in the area surrounding NFS?*

A: The largest known earthquakes East of the Rocky Mountains were the New Madrid earthquakes along the Mississippi River in the 1800's. These are estimated to have been a magnitude of approximately 8. The seismic hazard maps published by the U.S. Geological Survey don't indicate a potential for such large earthquakes in Eastern Tennessee where NFS is located.

Q: *What would be the impact of a tsunami, hurricane, or flood at the NFS site?*

A: NFS is located approximately 350 miles from the Atlantic Ocean and at an elevation of 1640 feet above sea level where any significant impacts from a hurricane or tsunami are very unlikely. The regulatory analysis of emergency preparedness in NUREG-1140 concluded that tornados might cause large releases, but the material would disperse so widely that significant doses would not result. The NFS site is on the edge of the 100 year floodplain for the Nolichucky River. It is possible that water could get inside buildings during a severe flood. However, there would be flood warnings, evacuation orders, and other notices that would allow the licensee to secure processing equipment, seal containers, and prepare for the flood. We believe such actions would prevent a release having a significant impact on the environment.

Q: *What is the design basis earthquake for the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) facility?*

A: The MFFF is located on Department of Energy's (DOE) Savannah River Site (SRS) near Aiken, South Carolina. At the SRS, there are no known capable or active faults within the 320-km (200-mile) radius that influence the seismicity of the region, with the exception of faults associated with the Charleston seismic zone (MFFF License Application). Earthquakes that could affect safe operations in the MFFF are associated with two seismic sources, repeat of Charlestown 1886 earthquake and small shallow earthquake of the South Carolina Piedmont. The MFFF facility is design to nuclear power plant requirements and utilizes the spectrum that is found in Regulatory Guide (RG) 1.60 anchored at 0.20g PGA.

Q: *How would the NRC and GNF-A respond to an earthquake at the facility?*

A: Local (New Hanover County) and the State of North Carolina emergency management agencies are both notified.

The licensee provides recommendations based on the release quantity/chemical / physical state and atmospheric conditions.

Q: *What are the potential impacts of an earthquake that exceeded the design of the MFFF?*

A: The MFFF facility was design using DOE's methodologies for natural phenomena hazards that establishes performance goals for nuclear facilities. In the unlikely event of an earthquake exceeding the design basis of the MFFF, it is expected that major structures such as buildings will suffer major damage, but the damage is limited in the extent such that the occupants can safely exit the building.

Q: *What would happen if an earthquake occurred in the vicinity of the GNF-A facility?*

A: Impacts to the facility will depend on the magnitude of the earthquake and the location of its epicenter. The licensee will implement its emergency plan and will coordinate with local and State emergency management agencies and organizations to respond to the event.

Q: *What would happen if an earthquake occurred in the vicinity of the MFFF?*

A: In the event that an earthquake occurs in the vicinity of the facility the seismic monitoring and trip system initiates a shutdown of process-related systems if a seismic event exceeds a specified set point. The seismic monitoring and trip system shuts down normal and standby power systems, ensuring that all movements of nuclear material are stopped in a safe manner.

Q: *Would overpressurized uranium hexafluoride cylinders stored at MFFF greatly increase the risk to members of the public during an earthquake?*

A: No. There are no large inventories of Special Nuclear Material (SNM) stored or used in a gaseous or highly dispersible form similar to the uranium hexafluoride at the MFFF facility. The primary form of SNM in the MOX facility would be powder. There are no significant additional hazards to members of the public due to the powder form during an earthquake.

U.S. Power Plants (General)

Q: *How would the NRC and the MFFF respond to an earthquake at the MFFF?*

A: MOX Services will follow the DOE Savannah River Emergency plan during an event at the facility. MOX Services will contact the NRC and DOE and interactions with State and local officials are conducted through the SRS Emergency Duty Officer who oversees the SRS Operations Center. The MFFF emergency preparedness program incorporates plans for radiation monitoring, repair and recovery efforts, search and rescue, and initial medical response.

Q: *What happens if the MFFF were to lose offsite power as a result of an earthquake?*

A: The design of the MFFF electric power supply system consists of a normal power system, a standby power system, and an emergency power system. Two separate and independent incoming offsite power feeders supply MFFF facility. In the rare event of a total loss of all incoming power to the facility, a standby power system composed of two independent standby diesel generators will automatically start and continue the supply of electrical power to the facility.

Q: *Can an earthquake as large as happened in Japan also happen near the MFFF?*

A: Near the Savannah River Site (SRS), instrumented historical seismic records indicate that seismicity associated with the SRS and surrounding region is closely related to the earthquake activity within the South Carolina Piedmont. This activity is characterized by shallow, small-magnitude and infrequent earthquakes. At the SRS, there are no known capable or active faults within the 320-km (200-mile) radius that influence the seismicity of the region, with the exception of faults associated with the Charleston seismic zone (MFFF License Application).

Q: *What are the potential impacts of an earthquake that exceeded the design of the MTW facility?*

A: Under the current design and analysis, we identify that the greatest impact of an earthquake most likely would be structural failure increasing the risk that occupants would be injured and unable to escape. Additionally, the likelihood of fires and releases would increase. Although offsite impacts may increase, the potential for radiation doses large enough to cause an acute fatality or early injury to a member of the public is not considered plausible.

As stated, a review is underway based on new USGS information.

Q: *How would the NRC and NFS respond to an earthquake at the facility?*

A: NFS would declare an emergency, activate its emergency response organization and begin implementing its emergency plan. NFS would promptly notify State and local authorities and recommend protective actions NFS believes should be implemented for the public offsite. Licensee resources would be focused on controlling the immediate safety hazards at the site (i.e., safe shutdown, fire fighting, etc.).

State and local authorities would activate their emergency response organizations and begin implementing response procedures. The procedures include tools such as a Reverse 911 call to advise members of the public of the need to shelter or evacuate.

NRC would be notified immediately after State and local authorities are notified. NRC would activate its emergency response organization and begin an independent assessment of site conditions and protective actions being taken. NRC would share its independent assessment with State and local authorities. If warranted by events, NRC will dispatch an emergency site team to monitor licensee activities and serve in an advisory role to state and local officials who may be considering protective actions to further ensure the protection on the public. In addition, NRC would coordinate assistance from other Federal agencies if needed.

U.S. Power Plants (General)

Q: *What are the potential impacts of an earthquake that exceeded the design of the CFFF?*

A: Earthquakes can cause damage which results in fires and releases of hazardous materials. Earthquakes are considered an unlikely cause of a criticality accident because such accidents require enriched uranium to accumulate in an unsafe, critical mass. The contents of broken pipes and containers tend to be dispersed, not accumulated.

Structure, systems and components were evaluated using a combination of the NRC-approved experience data methodology utilized by the Seismic Qualification Utilities Group of utilities to resolve natural phenomena hazards vulnerability and traditional structural risk assessment techniques. Preliminary findings indicate that a severe earthquake could result in toppling of equipment and collapse of structural walls and members, causing a release of uranium to the environment, and possibly a nuclear criticality accident, potentially resulting in a high consequence event to workers.

The seismic review was only a screening review. Detail fragility analyses have not been performed, it is not possible to reliably determine what size earthquake would result in severe damage to the building structure. The analyses indicate that an earthquake with a 0.2 g PGA most likely would result in some damage. Seismic engineers, estimate that buildings should withstand an earthquake of up to 0.05 g PGA. For the Columbia region, the United States Geologic Survey (USGS) estimates a PGA of 0.07 g at a 500-year return period.

Q: *Are the B&W NOG emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: The emergency diesels are not seismically qualified. There are no safety significant consequences that would result from a loss of offsite power. Risk-significant safety controls are passive or fail to a safe position without relying upon electric power.

Q: *Does the Areva Richland site have a spent fuel storage area?*

A: No.

Q: *What are the most significant hazards at the Areva Richland site?*

A: A criticality accident represents the potential for a lethal radiation dose to a worker within 10 to 50 feet of the criticality, and lesser but significant doses out to 100 feet or more. The total radiation to an individual at the site boundary would exceed regulatory limits, but would not result in the potential for radiation doses large enough to cause injury. There are no criticality scenarios that would be initiated by an earthquake or loss of offsite power event.

Q: *What is the design basis earthquake for the B&W NOG facility?*

A: The B&W NOG site is located near the western limit of the Piedmont physiographic province. Seismic activity in the Central Virginia region is classified as moderate. The site falls within the western part of the Central Virginia cluster region which is classified as Seismic Zone 2, indicating that moderate damage could occur as the result of earthquakes. Since 1774, there have been 18 earthquakes reported as having an intensity VI or higher on the Modified Mercalli Scale, defined as "felt by all, many frightened and run indoors, falling plaster and chimney bricks, damage small." It is comparable to 4.5 on the Richter scale.

The building structures are designed to meet the requirements of the IBC (International Building Code) National Building Code as adopted by the Commonwealth of Virginia. The IBC requires a design based on maximum earthquake spectral response acceleration for periods of 0.2 and 1 second. The peak ground acceleration is 0.26g and 0.09g for 0.2 and 1 second periods, respectively. The peak ground acceleration may be equated to an earthquake with a return period of 2500 years.

U.S. Power Plants (General)

Q: *What are the potential impacts of an earthquake that exceeded the design of the B&W NOG facility?*

A: Building codes incorporate occupant safety margins such as maintaining structural integrity long enough to allow occupants to leave the building. Maintaining structural integrity would reduce the damage that could lead to fires and releases of hazardous materials. If an earthquake exceeded the design of the facility, structural failure would be more likely increasing the risk that occupants would be injured and unable to escape.

The likelihood of a high consequence event would increase. Although offsite impacts may increase, the potential for radiation doses large enough to cause an acute fatality or early injury to a member of the public is not considered plausible.

Q: *What would happen if an earthquake occurred in the vicinity of the B&W NOG facility?*

A: A significant earthquake could cause damage resulting in fires or the release of hazardous materials. Earthquakes are considered an unlikely cause of a criticality accident because such accidents require enriched uranium must be accumulated in an unsafe, critical mass. The contents of broken pipes and containers tend to be dispersed, not accumulated.

Q: *How would the NRC and the licensee respond to an earthquake at the B&W NOG facility?*

A: The NRC has a resident inspector at the facility. NRC Headquarters and Region II Emergency Response staff from the Fuel Cycle Safety Team and Protective Measures Team would monitor the situation. If warranted by events, NRC will dispatch an emergency site team to monitor licensee activities and serve in an advisory role to state and local officials who may be considering protective actions to further ensure the protection on the public.

Q: *What preparations are currently in place to respond to such an emergency at the B&W NOG?*

A: The site Emergency Plan contains instructions and Protective Action Recommendations (PARs) for the postulated emergencies. The facility has an onsite Emergency Team with medical, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, and hospital. For these types of events B&W NOG notifies the Virginia Department of Emergency Services, and Campbell, Amherst, and Appomattox counties. Recommendations are based on the release quantity/chemical/physical state and atmospheric conditions and are discussed with the agencies listed above. Campbell County officials are the decision-makers for evacuations.

Q: *What happens if the B&W NOG facility were to lose offsite power as a result of an earthquake?*

A: There are no safety significant consequences that would result from a loss of offsite power. Back-up power supplies are installed to allow an orderly shutdown of plant operations. There are no plant processes that require continuous electrical power after shutdown.

Q: *What would be the impact of a tsunami, hurricane, or flood at the B&W NOG site?*

A: B&W NOG is located approximately 170 miles from the Atlantic Ocean and at an elevation of 820 feet above sea level where any impact from a hurricane or tsunami is very unlikely. The James River borders three sides of the site. Flooding of the James River occurs infrequently. There have been 11 significant floods since 1771, which averaged 28 feet above the normal river elevation. The main manufacturing facility is located approximately 110 feet above the river, and 75 feet above the 100-year flood level.

Q: *Does the B&W NOG site have a spent fuel storage area?*

A: The Lynchburg Technology Center has a cask unloading pool. Note, however, that unlike a power reactor site, the LTC does not possess large quantities of fuel elements immediately following refueling, and is limited to a maximum of 4 irradiated fuel assemblies.

It is possible that a shielded cask could fall into the pool, causing a fuel element to rupture. The worst case would be the sudden and complete release of fission gases in the transfer canal. Normally, these gases are filtered by HEPA filters and flow up the stack. A significant earthquake which caused damage to the filters, exhaust system, or stack would not result in the potential for radiation doses large enough to cause an acute fatality or early injury to a member of the public.

U.S. Power Plants (General)

Q: *What would be the impact of a tsunami or flood at the Areva Richland site?*

A: The site is located approximately 120 miles from the Pacific Ocean. The Columbia River is approximately 1 1/2 miles east of the site. The facility is located approximately 25 feet above the Columbia River and 370 feet above sea level.

The combination of the Probable Maximum Flood (PMF) and failure of any of the dams on the Columbia River upstream of the Hanford area has been calculated to be 1 E^{-6} . It is also noted that following an upstream event, some time would be available for site personnel to place the facility in a safe shutdown mode.

Q: *What is the design basis earthquake for the CFFF?*

A: The original manufacturing building, designed in 1968 was designed to comply with the Standard Building Code, 1965 Edition, with only minor exceptions. The building was designed to meet Seismic Zone I criteria. A building addition, designed in 1978 by Lockwood Greene, was designed to comply with the Standard Building Code, 1976 Edition.

The CFFF site is far from any center of significant earthquake activity. Several major earthquakes have occurred at distant points, and some minor to moderate shocks have occurred nearer to the site. No significant earthquake has been located nearer than about 20 mi from the site.

The large largest seismic event was the Charleston earthquake which was a magnitude 7.0 on August 31, 1886, which is the strongest earthquake documented in the southeastern United States in historic time. The earthquake was located about 90 mi southeast of Columbia and was felt as far away as Boston, Milwaukee, New York City, Cuba, and Bermuda. Damage from the earthquake was reported in Columbia, SC where Modified Mercalli (MM) intensities of VII-VIII were observed.

The nearest earthquake to the CFFF site occurred on April 20, 1964. The event had a magnitude of 3.5 and was located about 15 mi southwest of Columbia, South Carolina.

For the area near Columbia, a peak ground acceleration (PGA) of about 0.3 g would be expected for a 2,475-yr return period.

Q: *Are the Areva Richland emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: The emergency diesels are not seismically qualified. There are no safety significant scenarios which would result from a loss of offsite power. Risk-significant safety controls are passive or fail to a safe position without relying upon electric power.

Q: *What would happen if an earthquake occurred in the vicinity of the CFFF?*

A: An analysis was performed to determine the consequences of the materials that would likely be released in a severe earthquake. The analysis indicated that an Intermediate Consequence event would occur for facility workers. The public is not at risk to high or intermediate consequence events.

In the event of a chemical hazard release an Emergency Response Planning Guidelines (ERPG)-2 hazard would be created for about one mile. The nearby hydrogen tank is a local fire and explosion hazard. UF6 cylinders in the storage area are not of concern; filled cylinders are at ground level; empty cylinders are racks above the filled cylinders. UF6 cylinders in the autoclaves are not of concern. The autoclaves are heavy-walled vessels that are partially recessed in the concrete floor of the plant.

A natural gas transmission pipeline is 2,800 ft north of the main manufacturing building. If the pipe were to rupture, a fire and explosion would occur.

Two diesel-powered fire pumps provide high water pressure to the above fire suppression equipment when these systems are activated. A total of 450,000 gal of water can be stored on site in two water storage tanks to provide water to fight a fire.

U.S. Power Plants (General)

Q: *How would the NRC and the licensee respond to an earthquake at the CFFF?*

A: NRC Headquarters and Region II Emergency Response staff from the Fuel Cycle Safety Team and Protective Measures Team would monitor the situation.

The site Emergency Plan contains instructions and Protective Action Recommendations (PARs) for the postulated emergencies. The facility has an onsite Emergency Team with medical, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, and hospital. For these types of events, Westinghouse notifies the Department of South Carolina Department of Health and Environmental Control. Responses are based on the characteristics (e.g., chemical, quantity, physical state, atmospheric conditions). County officials would be notified.

Q: *What happens if the CFFF were to lose offsite power as a result of an earthquake?*

A: Emergency generators and Uninterruptable Power Supply (UPS) provide backup power for critical loads, including crucial process equipment:

- emergency lighting systems
- cooling system pumps
- fire alarm, hazard alarm, and other designated safety alarm systems
- conversion control room alarms
- health physics sampling systems
- emergency ventilation systems (including scrubbers)

Q: *Are the CFFF emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: In a station blackout scenario, the most significant hazard would be the gradual buildup of chemicals from the vented vessels in the process. Worker Safety is assured through use of appropriate PPE (Primarily Respiratory protection). Offsite chemical releases are not postulated to exceed thresholds for public safety from a loss of power event.

Q: *What would be the impact of a tsunami or flood at the CFFF?*

A: The site is not susceptible to a tsunami. Westinghouse is 100 miles from the nearest point of the Atlantic coast.

Q: *How many people live near the CFFF?*

A: Within 1 mile: 6
Within 3 miles: 963
Within 5 miles: 7,870

Q: *Does the CFFF have a spent fuel storage area?*

A: The site does not have spent fuel. Occasionally, fuel assemblies are returned to the site because of minor damage or defects. Such assemblies have never been irradiated in a reactor; they pose no danger.

Q: *What is the design basis earthquake for the GNF-A facility?*

A: The GNF-A facility was built to the Uniform Building Code (UBC) in effect at the time of construction. The UBC has identified the facility area as Seismic Zone 1 and considers seismic events of minor magnitude (5.5 to 6.0 in the Richter scale). Since there are no significant geologic features in the Wilmington region that would produce a "major" earthquake, it is highly unlikely that a "major" earthquake could affect the GNF-A facility.

Q: *What would be the worse-case scenarios from an accident at the CFFF?*

A: The worse-case scenario from a seismic event would be a catastrophic failure of the tank containing anhydrous ammonia concurrent with a complete loss of power to emergency equipment.

Q: *What are the most significant hazards at the CFFF?*

A: The most significant hazards at the site are the chemical hazards in the tank farm. The anhydrous ammonia tank has a capacity of 30,000 gallons; usually, only 10,000 gallons is present. Another fire and explosion hazard is from a tank of liquid hydrogen.

U.S. Power Plants (General)

Q: *What are the potential impacts of an earthquake that exceeded the design of the GNF-A facility?*

A: Impacts to the facility will depend on the magnitude of the earthquake and the location of its epicenter. However, based on the inventory of material that GNF-A is licensed to possess, it is unlikely that there will be a significant release of radioactive material.

Q: *What are the most significant hazards at the B&W NOG site?*

A: A criticality accident represents the potential for a lethal radiation dose to a worker within 10 to 50 feet of the criticality, and lesser but significant doses out to 100 feet or more. The total radiation to an individual at the site boundary would exceed regulatory limits, but would not result in the potential for radiation doses large enough to cause injury. There are no criticality scenarios that would be initiated by an earthquake or loss of offsite power event.

Q: *Are special procedures employed at the ACP for other natural phenomena?*

A: Yes. The ACP's Emergency Plan describes emergency measures to be taken in response to emergencies such as accidents or natural phenomena events (i.e., earthquake, flood, high winds/tornadoes, lightning strikes, and snow load hazards). It describes the protective actions to be implemented on-site and off-site to ensure exposures of personnel and members of the public are limited in case of an accidental release of licensed material to the environment.

On-site protective actions include evacuation, shelter in place, accountability, search and rescue, and monitoring and decontamination.

For off-site protective actions the ACP's Incident Commander (IC) is responsible for providing protective action recommendations to local officials. These recommendations are based on assessment actions and an understanding of the actual or potential conditions. Recommendations include sheltering in place, evacuation or advisories that no action is needed. Appropriate off-site authorities are responsible for alerting and notifying the public on the recommended off-site protective actions.

Q: *What are the potential impacts of an earthquake that exceeded the design of the LES facility?*

A: All buildings and structures, including such items as equipment supports, are designed to withstand the earthquake loads defined in Sections 1615 through 1617 of the International Building Code. The applicant proposed the method outlined in DOE-STD-1020 (DOE, 2002) or ASCE/Structural Engineering Institute (SEI) 43-05, "Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities" (ASCE/SEI, 2005) to demonstrate compliance to a target performance goal of 1.0×10^{-5} annual probability by designing to a seismic hazard of 1.0×10^{-4} annual probability.

Q: *What would happen if an earthquake occurred in the vicinity of the LES facility?*

A: The majority of earthquakes in the United States are located in the tectonically active western portion of the country. However, areas within New Mexico and the southwestern United States also experiences earthquakes, although at a lower rate and at lower intensities. Earthquakes in the region around the NEF site are isolated or occur in small clusters of low to moderate size events toward the Rio Grande Valley of New Mexico and in Texas, southeast of the NEF site. The largest earthquake within 322 km (200 mi) of the NEF is the August 16, 1931 earthquake located near Valentine, Texas. This earthquake has an estimated magnitude of 6.0 to 6.4 and produced a maximum epicentral intensity of VIII on the Modified Mercalli Intensity (MMI) Scale. The intensity observed at the NEF site is IV on the MMI scale.

Q: *How would the NRC and LES respond to an earthquake at the facility?*

A: NRC and LES would respond consistent with standard procedures for emergency events. LES procedures are described in its Radiological Contingency and Emergency Plan.

Q: *What happens if the LES facility were to lose offsite power as a result of an earthquake?*

A: LES's overall electrical power system is designed with a high level of redundancy to maintain a reliable power supply to the process equipment for investment protection. Total loss of electrical power does not have any safety implications. Safety systems for the facility are not dependent on electrical power. Safety systems fail to a fail-safe configuration on loss of power without the need for operator actions.

U.S. Power Plants (General)

Q: *Are the LES emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: Standby Diesel Generators are provided to power equipment that can tolerate a short break (short break load) in the normal power supply. This capability is needed to allow for an orderly shutdown of the facility. Each of the Standby Diesel Generators is sized for 100% of the short break load requirement of the equipment to which it is connected. The Standby Diesel Generators are not required for safe operation of the facility and are installed to provide protection of investment only. The functional requirement of the Standby Diesel Generator System is to provide backup power within approximately 10 seconds after a normal power interruption

A security diesel generator is provided to power select security equipment. The security diesel generator is not required for safe operation of the facility. Uninterruptible Power Supply (UPS) systems are provided to power the facility process equipment that do not tolerate a break (no break load) in the normal power supply. Input power for this UPS system is normally provided by the short break power system with backup from the Standby Diesel Generators. Batteries power the UPS if all other input power is lost. Each of the UPS systems is sized for 100% of its connected load.

Additional UPS systems with battery backup are installed to provide no break power to support systems such as emergency lighting. These systems are sized and located as necessary to provide the requirements of the equipment served. Systems requiring no break power are listed in Section 3.5.2.4, Operating Characteristics.

Duplicate batteries supply operating power for the 13 kV switchgear. Batteries provide starting power for each Standby Diesel Generator and operating power for each UPS system. The Standby Diesel Generator System provides backup 480V power to the NEF during a loss of normal power. The Standby Diesel Generator System is not a requirement for safe operation of the plant and is installed to provide protection of investment only. The Standby Diesel Generator System is comprised of two, 100% rated generators that supply the total backup power required. The Standby Diesel Generator System is installed in the Central Utilities Building. In the event of normal power failure, the Standby Diesel Generator System maintains plant services that protect the capital investment.

Q: *Are special procedures employed at LES for other natural phenomena events?*

A: For hurricanes and other natural phenomena events where advance warning can be provided, LES will shutdown operations and place systems into a safe configuration in advance of the event.

Q: *What is the design basis earthquake for the ACP?*

A: Seismic specifications for the ACP design are based on the risks and potential consequences from seismic events involving the primary facilities. This approach results in two criteria being applied depending upon whether or not the normal operations therein involve liquid UF₆. Facilities where liquid UF₆ operations occur are required to withstand the forces resulting from a 10,000-year return period seismic event (Peak Ground Acceleration - PGA 0.48 G). All other primary facilities are required to withstand the forces resulting from a 1,000-year return period seismic event (PGA 0.15 G) because UF₆ operations therein involve UF₆ in either gas or solid form.

The existing buildings utilized by the ACP are designed and constructed to withstand a 1,000-year seismic event. All newly constructed buildings, including extensions to existing buildings, are designed to withstand a 1,000-year seismic event. The Customer Services Building will be the only building within the complex that will handle liquid UF₆ and is designed to withstand a 10,000-year seismic event. Process-related equipment in the ACP buildings will also be seismically qualified to meet the 1000-year or 10,000-year return earthquake, as applicable. Therefore, while the unprevented frequency of a seismic event that would be expected to impact these buildings is estimated to be "Not Unlikely," no release of hazardous material is expected during a design basis seismic event.

U.S. Power Plants (General)

Q: *What are the potential impacts of an earthquake that exceeded the design of the ACP facility?*

A: While a seismic event of a magnitude beyond the design of the facility could result in damage to the buildings and equipment, a review of design calculations for these buildings show significant reserve with respect to the design basis seismic capacity. The models and assumptions used in these calculations indicate that the capacity of the structural components would exceed the basic requirements with significant margin. If a seismic event were to occur such that there was a breach of process systems and equipment containing UF6 in these facilities, it is unlikely that a release of licensed material would produce any significant impact to the off-site public because the systems and equipment operate below atmospheric conditions, contain low inventories of licensed material, and the structures and equipment are not expected to fail catastrophically. Previous experience with similar equipment (at the GDP) has shown that a breach in equipment operating below atmospheric pressure has resulted in minimal release to the environment (the ACP operates at much lower pressures than the GDP). Besides, some of the ACP equipment (i.e., autoclaves with roll or tilt capability) are designed to withstand a 100,000-year return period seismic event.

The Customer Service building would be expected to survive the beyond design basis earthquake without experiencing a release of hazardous material that would produce a significant impact to the off-site public. However, since UF6 in liquid state could be located within the building during a seismic event, the building is considered to be a high seismic risk location and, as such, the margin of safety for the design has been increased appropriately. Liquid UF6 would be in cylinders and piping, all of which in turn would be located in autoclaves. Given this configuration, it is unlikely that the seismic damage suffered would be sufficient to breach a cylinder or process piping. Therefore, no release of hazardous material would result.

Q: *What would happen if an earthquake occurred in the vicinity of the ACP?*

A: There are no major geologic fault structures in the vicinity of the ACP and there have been no historical earthquake epicenters within 25 miles from the site. However, there have been eight earthquake epicenters within 50 miles. The maximum event had an epicenter intensity of over IV on the Modified Mercalli (MM) scale. These events had intensities between I and IV. The maximum peak ground acceleration (PGA) of a MM level IV event roughly corresponds to 0.02 G. Historically, the maximum earthquake-induced PGA experienced at the reservation was in 1955 and had a value of only 0.005 G.

Independent calculations and a review of the seismic hazard analyses for the reservation were performed by the USGS and the results were documented. For a return period of 500 years, the PGA was determined to have a value of 0.10 G. For a return period of 250 years, the PGA was determined to have a value of 0.05 G. Earthquakes with large ruptures are highly unlikely to occur near the reservation because of low values of maximum magnitude.

Q: *How would the NRC and USEC respond to an earthquake at the ACP facility?*

A: NRC and the ACP would respond consistent with standard procedures described in its Emergency Plan.

Q: *How many people live near the Areva Richland site?*

A: The closest resident to the facility is approximately 1 ½ miles to the southwest in the City of Richland.

Offsite population totals are as follows:

1 mile: 0
2 miles: 376
3 miles: 5020
4 miles: 15,560

U.S. Power Plants (General)

Q: *Are the ACP emergency power diesel generators built to withstand the effects of an earthquake? If not what happens when power to the facility is lost?*

A: The ACP maintains emergency power diesel generators that have been designed for key operational areas/buildings. The ACP maintains, inspects, and tests the emergency power diesel generators to ensure they are capable of activation in the event power is lost to the facility for they are supporting. The emergency power diesel generators are designed to support the critical items within a facility until such time that primary power can be restored. The existing buildings utilized by the ACP are designed and constructed to withstand a 1,000-year seismic event. All newly constructed buildings, including extensions to existing buildings, are designed to withstand a 1,000-year seismic event. In some cases, uninterruptable power supplies and batteries are in place. Although the emergency power diesels are not specifically designed for a seismic event, the facilities for they are in, or supporting, are designed to withstand a 1,000-year seismic event.

Key facilities within ACP, or supporting ACP, with emergency power diesels include: X-104 Guard Headquarters, X-112 Computer Support Facility, X-300 Plant Control Facility, X-1007 Fire Station, X-1020 Emergency Operations Center, X-3001 Process Building, and X-3012 Process Support Building. The X-3002, X-3346, and X-7725 will have emergency power diesels as well.

The X-640-1 and X-6644 Fire Water Pump Houses are equipped with emergency diesel fire water pumps.

Q: *What is the design basis earthquake for the Louisiana Enrichment Services (LES) facility?*

A: A site-specific probabilistic seismic hazard analysis was performed for LES using the seismic source zone geometries and earthquake recurrence models. The modeling included attenuation models suited for the regional and local seismic wave transmission characteristics. Total seismic ground motion hazard to a site results from summation of ground motion effects from all distant and local seismically active areas. The 250-year and 475-year return period peak horizontal ground accelerations are estimated at 0.024 g and 0.036 g, respectively. The respective 10,000- and 100,000-year return period peak horizontal ground accelerations were estimated at 0.15 and 0.31g. This return period is equivalent to a mean annual probability of E-4. The associated peak vertical ground motion is estimated at 0.10 g. The seismic hazard calculated for facility site is similar to that calculated for the nearby Waste Isolation Pilot Plant. The calculated 10,000-year return period peak ground acceleration at the Waste Isolation Pilot Plant is slightly less than 0.15 g. Based on all the information available, the staff concludes that the seismic hazard described in the ISA Summary is acceptable because it is based on a method that follows current industry practice and includes available data.

Q: *What is the design basis earthquake for the Areva Richland fuel fabrication facility?*

A: There is no design basis earthquake per se. Buildings were constructed in accordance with the local commercial building code in effect at the time of construction. Buildings constructed recently were designed to meet the seismic load resistance specified in the code for UBC Zone IIB. The code design loads are based on an earthquake with a 2,500 year return period and a peak ground acceleration of 0.20 g. The peak ground acceleration for a ground motion with an annual exceedance probability of 10⁻⁴ is 0.398g.

The staff determined that it would not be likely that the structures at the facility would suffer severe damage from a seismic event with a peak ground acceleration of 0.398g, because of sufficient safety margin associated with the design.

Q: *What happens if the Areva Richland facility were to lose offsite power as a result of an earthquake?*

A: There are no safety significant scenarios which would result from a loss of offsite power. Back-up power batteries are installed to allow an orderly shutdown of plant operations.

Q: *What preparations are currently in place to respond to such an emergency at Areva Richland?*

A: The site Emergency Plan contains instructions and Protective Action Recommendations (PARs) for the postulated emergencies. The facility has an onsite Plant Emergency Response Team with medical, incipient fire fighting, and incipient hazardous material response capabilities. Agreements are in place with the City of Richland fire department, police, and hospital. For these types of events Notifications are made to the Richland Emergency Dispatch Center, which in turn notifies both Benton and Franklin Counties Emergency Management personnel and the City of Richland. Additional assistance is available from the DOE Hanford fire department, approximately ½ mile away. The DOE Richland Operations Office is notified. The Washington State Emergency Management Division is notified.

U.S. Power Plants (General)

Q: *How would the NRC and the licensee respond to an earthquake at the Areva Richland facility?*

A: The NRC has a resident inspector at the nearby Columbia generating Station facility. NRC Headquarters and Region IV Emergency Response staff from the Fuel Cycle Safety Team and Protective Measures Team would monitor the situation.

Q: *What would happen if an earthquake occurred in the vicinity of the Areva Richland facility?*

A: A significant earthquake could cause damage resulting in fires or the release of hazardous materials. Earthquakes are considered an unlikely cause of a criticality accident because such accidents require enriched uranium must be accumulated in an unsafe, critical mass. The contents of broken pipes and containers tend to be dispersed, not accumulated.

Q: *What happens if the ACP were to lose offsite power as a result of an earthquake?*

A: Should an earthquake occur that causes the loss of offsite power, emergency backup power is or will be available to support an orderly shutdown of the enrichment processes. The enrichment processes are based on fail-safe operation so there are no active safety systems that are required to support safe shutdown. However, critical business operations do indicate a need for cost effective shutdown of enrichment equipment.

The ACP essential electrical system designs shall provide for continued operation of essential utility services in accordance with 10 CFR 70.64(a)(7). The ACP's essential electrical systems (EES) will be provided reliable electrical power as recommended by Articles 700 and 702 of ANSI C1/NFPA 70-2005. Each EES will be connected to a power source derived from the X-5000 Substation via a local single or double-ended 480V substation and another power source consisting of a standby diesel-generator set. This configuration will ensure reliable power for the essential electrical systems that support various systems that are necessary to protect life safety, maintain critical communications systems, and protect valuable process equipment.

Installation of cables, cable trays, and conduit systems will comply with ANSI C1/NFPA 70-7005. The cables will be suitable for their environment (hot, cold, wet, dry, and/or corrosive) and comply with applicable flame retardancy requirements. Physical supports for conduits, trays, panels, and cabinets will equal or exceed ANSI C1/NFPA 70-2005 requirements.

Q: *What are the potential impacts of an earthquake that exceeded the design of the Areva Richland facility?*

A: Building codes incorporate occupant safety margins such as maintaining structural integrity long enough to allow occupants to leave the building. Maintaining structural integrity would reduce the damage that could lead to fires and releases of hazardous materials. If an earthquake exceeded the design of the facility, structural failure would be more likely increasing the risk that occupants would be injured and unable to escape.

The likelihood of fires and releases would increase. Although offsite impacts may increase, the potential for radiation doses large enough to cause an acute fatality or early injury to a member of the public is not considered plausible.

Q: *What is the design basis earthquake for the EREF?*

A: The license application was submitted in December 2008 and is currently under review. The applicant submitted and staff reviewed a site-specific probabilistic seismic hazard assessment (PSHA) for the proposed EREF. The design basis earthquake for the EREF has a peak ground acceleration of 0.16 g. The EREF design will allow it to withstand, without serious consequences to the public, the effects of earthquakes that have less than a one in 100,000 chance of occurring.

The proposed EREF site is situated in a seismically inactive region of the Eastern Snake River Plain. The largest historical earthquake to strike the Snake River Plain was the 1905 Shoshone earthquake, with an estimated magnitude of between 5.3 and 5.7.

Q: *Are special procedures employed at the EREF for other natural phenomena?*

A: As part of the license review, the resistance of the proposed EREF to natural phenomena was evaluated, including consideration of: seismic hazards, volcanic hazards, tornado hazard, high winds, extreme precipitation, flood, snow, and lightning.

U.S. Power Plants (General)

Q: *How many people live near the EREF?*

A: The distance to the nearest resident to the proposed EREF is approximately 5 miles. The population density around the site and region is generally low. The nearest population center is Idaho Falls which is about 20 miles southeast of the site. Its estimated population is 52,786 (based on 2006 data).

Q: *What would be the impact of a tsunami or flood at the EREF?*

A: The proposed EREF site is located inland, approximately 575 miles from the coast. Additionally, the site is not located near any large body of water that could cause a flood at the site. The nearest large surface waters are the Snake River which is about 20 miles east and Lake Wolcott which is approximately 75 miles southwest of the proposed site.

Q: *What happens if the EREF were to lose offsite power?*

A: Items relied on for safety for the proposed EREF will be designed to maintain their safety functions or to fail into a safe state in the event of a loss of off-site power. Standby diesel generators will be provided for investment protection purposes only.

Q: *How would the NRC and EREF respond to an emergency, such as an earthquake, at the facility?*

A: NRC regulations require that the proposed EREF have an emergency plan. The emergency plan contains onsite and offsite Protective Action Recommendations (PARs) for emergencies. The EREF will have an onsite Emergency Response Organization with first aid, fire fighting, and hazardous material response capabilities. Agreements are in place with the local fire department, police, and hospital. In the event of an alert or site area emergency, as part of the emergency plan, EREF would notify key offsite agencies, namely the Bonneville County Emergency Management Services, Idaho Bureau of Homeland Security, and the NRC Emergency Operations Center. Once notified, NRC staff would monitor the situation.

Q: *What are the potential impacts of an earthquake that exceeds the design of the EREF?*

A: Although the facility is designed to withstand an earthquake with a one in 10,000 chance of occurring, analysis of the design shows that the facility design has sufficient margin to be able to maintain radiological safety even if it is shaken by an earthquake with a one in 100,000 chance of occurrence.

In the case of an earthquake which leads to a breach in the piping for the uranium hexafluoride (UF₆) systems, it is assumed that there would be a release of UF₆. Although the uranium feed material is radioactive, the primary consideration with regard to human health and safety is chemical rather than radiological. Thus, in the event of an unmitigated release, the chemical effects are greater health and environmental concerns than the amount of radiation that might be released. The NRC has modeled the potential consequences from such a release. Based on its modeling, NRC has found that the consequences to workers are potentially high, while consequences to offsite public are low (below the appropriate Acute Exposure Guideline Level). Mitigating measures will further reduce the consequences.

Generic Issues Program

Q: *Where can I get current information about Generic Issue 199?*

A: The public NRC Generic Issues Program (GIP) website (<http://www.nrc.gov/about-nrc/regulatory/generic-issues.html>) contains program information and documents, background and historical information, generic issue status information, and links to related programs. The latest Generic Issue Management Control System quarterly report, which has regularly updated GI-199 information, is publicly available at <http://www.nrc.gov/reading-rm/doc-collections/generic-issues/quarterly/index.html>. Additionally, the US Geological Survey provides data and results that are publicly available at <http://earthquake.usgs.gov/hazards/products/conterminous/2008/>.

U.S. Power Plants (General)

Q: *Is the NRC relooking at seismic analysis for US plants?*

A: The ground motions that are used as seismic design bases at US nuclear plants are called the Safe Shutdown Earthquake ground motion (SSE). In the mid to late 1990s, the NRC staff reviewed the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting a generic review referred to as GI-199, "Implications of Updated Probabilistic Seismic Estimates in Central and Eastern United States on Existing Plants," to again assess the resistance of US nuclear plants to earthquakes. In addition, the NRC has been reviewing new seismic information regarding the plants in California for many years.

GI-199

Q: *What is Generic Issue 199 about?*

A: Generic Issue 199 investigates the safety and risk implications of updated earthquake-related data and models. These data and models suggest that the probability for earthquake ground shaking above the seismic design basis for some nuclear power plants in the Central and Eastern United States is still low, but larger than previous estimates.

Q: *Does GI-199 provide rankings of US nuclear plants in terms of safety?*

A: The NRC does not rank nuclear plants by seismic risk. The objective of the GI-199 Safety/Risk Assessment was to perform a conservative, screening-level assessment to evaluate if further investigations of seismic safety for operating reactors in the central and eastern US (CEUS) are warranted, consistent with NRC directives. The results of the GI-199 safety risk assessment should not be interpreted as definitive estimates of plant-specific seismic risk because some analyses were conservative making the calculated risk higher than in reality. The nature of the information used (both seismic hazard data and plant-level fragility information) make these estimates useful only as a screening tool.

Q: *What are the current findings of GI-199?*

A: Currently operating nuclear plants in the US remain safe, with no need for immediate action. This determination is based on NRC staff reviews of updated seismic hazard information and the conclusions of the first stage of GI-199. Existing nuclear plants were designed with considerable margin to be able to withstand the ground motions from the "deterministic" or "scenario earthquake" that accounted for the largest earthquakes expected in the area around the plant. The results of the GI-199 assessment demonstrate that the probability of exceeding the design basis ground motion may have increased at some sites, but only by a relatively small amount. In addition, the probabilities of seismic core damage are lower than the guidelines for taking immediate action. Although there is not an immediate safety concern, the NRC is focused on assuring safety during even very rare and extreme events. Therefore, the NRC has determined that assessment of updated seismic hazards and plant performance should continue.

Q: *Where can I get current information about Generic Issue 199?*

A: The public NRC Generic Issues Program (GIP) website (<http://www.nrc.gov/about-nrc/regulatory/generic-issues.html>) contains program information and documents, background and historical information, generic issue status information, and links to related programs. The latest Generic Issue Management Control System quarterly report, which has regularly updated GI-199 information, is publicly available at <http://www.nrc.gov/reading-rm/doc-collections/generic-issues/quarterly/index.html>. Additionally, the US Geological Survey provides data and results that are publicly available at <http://earthquake.usgs.gov/hazards/products/conterminous/2008/>.

U.S. Power Plants (General)

Q: *What is the current status of GI-199 regarding updated seismic analysis for U.S. nuclear reactors? NEW!*

A: As of September 1, 2011, the NRC staff is seeking public comments on a draft Generic Letter that would require U.S. nuclear power plants to re-examine their sites' seismic risk and provide that information to the NRC. Comments on the draft letter, published in the Federal Register and available on regulations.gov, will be accepted until Oct. 31. The letter represents the next step in the staff's ongoing multi-year examination of updated seismic hazard information for the eastern and central United States, through the NRC's Generic Issues program.

This effort, labeled GI-199, began long before the events at the Fukushima Dai-ichi nuclear plant in Japan and the recent Virginia earthquake. GI-199 was prompted by the seismic analyses included in applications from 2003 related to new reactor activity. The NRC issued an Information Notice in September 2010 regarding GI-199, including the agency's conclusion that existing plant designs safely account for possible earthquakes. More information on GI-199 is available on the NRC website. The NRC staff will consider the comments before finalizing the Generic Letter, which the staff expects to issue near the end of the year. The draft letter's approach would have U.S. nuclear power plants perform their analyses within either one or two years, depending on the analysis method used, and deliver their results to the NRC. The agency will then determine whether additional actions are necessary.

Q: *What is the likelihood of the design basis or "SSE" ground motions being exceeded over the life of a nuclear plant?*

A: The ground motions that are used as seismic design bases at US nuclear plants are called the Safe Shutdown Earthquake ground motion (SSE). In the mid to late 1990s, the NRC staff reviewed the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting GI-199 to again assess the resistance of US nuclear plants to earthquakes. Based on NRC's preliminary analyses to date, the mean probability of ground motions exceeding the SSE over the life of the plant for the plants in the Central and Eastern United States is less than about 1%.

It is important to remember that structures, systems and components are required to have "adequate margin," meaning that they must continue be able withstand shaking levels that are above the plant's design basis.

Q: *What level of earthquake hazard are the US reactors designed for?*

A: Each reactor is designed for a different ground motion that is determined on a site-specific basis. The existing nuclear plants were designed on a "deterministic" or "scenario earthquake" basis that accounted for the largest earthquakes expected in the area around the plant, without consideration of the likelihood of the earthquakes considered. New reactors are designed using probabilistic techniques that characterize both the ground motion levels and uncertainty at the proposed site. These probabilistic techniques account for the ground motions that may result from all potential seismic sources in the region around the site. Technically speaking, this is the ground motion with an annual frequency of occurrence of 1×10^{-4} /year, but this can be thought of as the ground motion that occurs every 10,000 years on average. One important aspect is that probabilistic hazard and risk-assessment techniques account for beyond-design basis events. NRC's Generic Issue 199 (GI-199) project is using the latest probabilistic techniques used for new nuclear plants to review the safety of the existing plants.

License Renewal

Q: *Do you expect that applications for reactor extensions or power uprates will be slowed because of this review? What about new reactor licenses?*

A: The NRC will continue to process existing applications for power uprates and license renewal applications in accordance with the schedules that have been established. The NRC continues to believe that its regulatory framework and requirements provide for a rigorous and comprehensive license review process that examines the full extent of siting, system design and operations of nuclear power plants. The recommendations of the NRC's task force that was established to examine lessons learned from the events in Japan will certainly be taken into account in the performance of the NRC's review of these applications, as appropriate. Further, the NRC has the necessary regulatory tools to require changes to existing licenses or applications for certification should the agency determine that changes are necessary.

U.S. Power Plants (General)

Q: *How will the NRC's regulatory process protect the ability of communities to challenge the relicensing decision? NEW!*

A: In keeping with NRC's open and transparent processes, the NRC review will continue to have dialogue with all stakeholders, including public interest groups, industry, Federal, State, Tribes and local agencies, and members of the public, as well as making associated documents available on the NRC website, to enhance understanding of the regulatory decision-making process. In addition, members of the public have the opportunity to petition to intervene in the license renewal process.

Q: *How will the NRC consider the seismic risks in license renewal decision? NEW!*

A: The NRC's regulations for license renewal (10 CFR Part 54) require licensees to manage the age-related degradation of passive systems, structures, and components (SSCs) to ensure they will fulfill their safety-related functions, as specified in the current licensing basis, that will continue into the period of extended operation. A plant's licensing basis, including its seismic design basis, is established during initial plant licensing. The licensing basis dynamically evolves during subsequent license amendments and licensing actions, as new information and plant modifications are incorporated into the plant design and license. The NRC has multiple processes to evaluate the adequacy of current plant operations and licensing bases (e.g., Reactor Oversight Process, Generic Issues Program). If new information or operating experience warrants, the NRC will direct additional measures to maintain established safety thresholds commensurate with risk and safety benefit (e.g., require plant improvements through the backfit process). Any age-related degradation of SSCs in the application's aging management plan affected by seismic events will be evaluated by the applicant and reviewed by the NRC staff as part of the license renewal process.

Q: *Does the NRC expect that applications for reviews such as reactor extensions and new reactor licenses be affected from the Near Term Task Force review? NEW!*

A: The NRC will continue to process existing applications for new licenses and license renewal applications in accordance with the schedules that have been established. The NRC continues to believe that its regulatory framework and requirements provide for a rigorous and comprehensive license review process that examines the full extent of siting, system design and operations of nuclear power plants. The recommendations of the NRC's task force examining lessons learned from the events in Japan will be taken into account in the performance of the NRC's review of these applications, as appropriate. Further, the NRC has the necessary regulatory tools to require changes to existing licenses or applications for certification should the agency determine that changes are necessary.

Q: *How will the task force's timeline of 90 days for its short-term analysis and approximately six months for its long-term recommendations impact existing license applications?*

A: The timelines for the task force analyses and recommendations will not have any immediate effect on the review of existing license applications. The NRC will continue to process existing applications for new licenses and license renewal applications in accordance with the schedules that have been established. The NRC continues to believe that its regulatory framework and requirements provide for a rigorous and comprehensive license review process that examines the full extent of siting, system design and operations of nuclear power plants. The recommendations of the NRC's task force that was established to examine lessons learned from the events in Japan will certainly be taken into account in the performance of the NRC's review of these applications, as appropriate. Further, the NRC has the necessary regulatory tools to require changes to existing licenses or applications for certification should the agency determine that changes are necessary. For example, any new requirements that may result from the task force's recommendations could be implemented in accordance with existing agency policies that may involve rulemaking or backfitting.

Q: *Will the NRC continue processing existing license applications while the task force conducts its analysis?*

A: The NRC will continue to process existing applications for new licenses (i.e., early site permits, design certifications, and combined licenses) and license renewal applications in accordance with the schedules that have been established. The NRC continues to believe that its regulatory framework and requirements provide for a rigorous and comprehensive license review process that examines the full extent of siting, system design and operations of nuclear power plants. The recommendations of the NRC's task force that was established to examine lessons learned from the events in Japan will certainly be taken into account in the performance of the NRC's review of these applications, as appropriate. Further, the NRC has the necessary regulatory tools to require changes to existing licenses or applications for certification should the agency determine that changes are necessary.

U.S. Power Plants (General)

Q: *Why do license renewal reviews not include a review of the plant's response to external events?*

A: The regulations stipulating the requirements associated with license renewal were issued via rulemaking in 1991 (54 FR 64943). As described in the Statement of Considerations (SOC) for this license renewal rule, the Commission determined that, with the exception of age-related degradation unique to license renewal, the NRC's existing regulatory process is adequate to ensure that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety for operation. The Commission considered whether or not to include plant responses to external events that may be outside the licensing basis but reasoned that the existing regulatory process was sufficient to address those instances while at the same time avoiding duplicative and, perhaps, less efficient assessments. With this understanding, the Commission maintained that the focus of license application renewals should be limited to the age-related degradation management for systems, structures and components (SSCs) that are included in the scope of license renewal (e.g., important to safety, or whose failure could impact safety equipment). As a consequence, license renewal reviews consider applicant activities to detect, manage, and correct the effects of age-related materials degradation on SSCs to ensure that the functionality of safety equipment is not adversely impacted during the renewed license operating period.

Recent proceedings associated with Oyster Creek license renewal have reiterated the Commission's position that the NRC's comprehensive and ongoing oversight of licensed facilities will assure that useful data, operating experience, lessons learned, etc. will be absorbed by changes in NRC rules, orders, and license amendments, as needed, accompanied by the public participation required by statute and regulation. Therefore, plant response to external events will be reviewed when the need is identified, irrespective of the plant's status regarding license renewal (e.g., post-Fukushima review is being done for all plants, and actions will be taken and applied based on plant designs). The NRC has completed its near-term review of lessons learned from the events at Fukushima. The Commission is currently reviewing the report and will provide the staff with direction. Any changes will be applied to plants irrespective of whether a plant has a renewed license or not.

Q: *How will the events in Japan affect license renewal for U.S. plants?*

A: The NRC's recently initiated review of U.S. plants will examine current practice at operating reactors to ensure proper actions will be taken if a severe event occurs – this covers plants regardless of where they are in their license lifetime. The events in Japan, based on what's known at this time, appear to be unrelated to issues examined in license renewal. The NRC's long-term review of its regulations will determine whether any revisions to license renewal reviews are necessary.

Q: *Has NRC's plan with new reactor reviews changed following tragedy in Japan? Which four reactors are expected to be under construction? NEW!*

A: The NRC's plans with regard to New Reactor Licensing Reviews have not changed. Summer, Units 2 and 3, and Vogtle, Units 3 and 4, are currently conducting "pre-construction" activities; full construction can only occur after the NRC issues a Combined License. NRC has not received any indication from industry or current applicants that there is any change regarding their plans to obtain new licenses.

MOX Fuel

Q: *Given the situation in Japan where a radioactive release is anticipated, is the radiation release associated with MOX fuel different than a release from conventional uranium fuel?*

A: Yes, fission product release from MOX fuel is different than conventional uranium fuel since the MOX fuel consists of slightly different uranium and plutonium content than conventional fuel. Irradiated MOX fuel will have different fission product species as well as different concentrations of typical fission products. Thus the releases would consist of different concentrations of fission products.

The NRC staff has reviewed radiological consequences of releases of this type for domestic MOX fuel and has determined that they would meet the NRC's regulatory requirements.

U.S. Power Plants (General)

Q: *Can an earthquake as large as happened in Japan also happen near the MFFF?*

A: Near the Savannah River Site (SRS), instrumented historical seismic records indicate that seismicity associated with the SRS and surrounding region is closely related to the earthquake activity within the South Carolina Piedmont. This activity is characterized by shallow, small-magnitude and infrequent earthquakes. At the SRS, there are no known capable or active faults within the 320-km (200-mile) radius that influence the seismicity of the region, with the exception of faults associated with the Charleston seismic zone (MFFF License Application).

Q: *What does the latest accident in Japan tell us about the use of MOX fuel in boiling water reactors?*

A: The type of fuel used in these reactors had no impact on the occurrence of events that resulted in the current situation at the Fukushima Daiichi facility in Japan. MOX fuel has been used in BWRs and PWRs both domestically and internationally for many years. There is considerable international experience using MOX fuel in nuclear power reactors in both BWRs and PWRs. There are several dozen nuclear reactors worldwide that use MOX fuel. Additional information on experience using MOX in nuclear power reactors is available on the NRC's public website and may be accessed at the following link: <http://www.nrc.gov/materials/fuel-cycle-fac/mox/reactors.html>.

Q: *Is the risk of fuel failure more likely because of the use of MOX fuel?*

A: No. The mechanisms associated with fuel failure are not dependent on fuel type. Fuel failure mechanisms are predominantly associated with loss of sufficient heat removal capacity to the fuel rods.

Q: *Are there any active license applications for MOX fuel use or production?*

A: There are currently no active license applications for use of MOX fuel in nuclear power reactors in the U.S. Shaw AREVA MOX Services (MOX Services), under contract to the U.S. Department of Energy (DOE) applied to the NRC for approval to construct a Mixed Oxide Fuel Fabrication Facility (MFFF) at the Savannah River site in Aiken, South Carolina. The NRC issued a construction authorization in March 2005 for this facility. In September 2006, MOX Services submitted a License Application (LA) to possess and use radioactive material. The NRC reviewed the LA and published its Final Safety Evaluation Report (SER) in December 2010.

Upon verification of construction of the principal structures, systems and components (PSSCs) of the MFFF, the NRC may issue a license to possess and use radioactive material at the facility. The NRC understands that the schedule for completion of construction of the PSSCs is expected to be in the 2014/2015 timeframe to allow operations to begin by 2016. The NRC also understands that the DOE has solicited the commercial nuclear power industry to assess interest in future use of MOX fuel that will be produced at this facility.

Q: *Where in the U.S. are commercial nuclear power reactors currently licensed to use MOX fuel. Where is MOX fuel currently in use?*

A: There are currently no nuclear power plants in the U.S. that are utilizing mixed-oxide (MOX) fuel. In response to a license amendment request from Duke Energy Carolinas, LLC, the NRC authorized the use of four MOX fuel lead test assemblies (LTAs) in one of the two units at the Catawba Nuclear Station. The MOX LTAs were loaded into the reactor in the spring of 2005. The LTAs were irradiated for two operating cycles and were removed in the spring of 2008. Testing and evaluation of the MOX fuel lead test assemblies at Catawba is no longer ongoing and there are no plans for its resumption.

The four LTAs are currently in the spent fuel pool at Catawba Nuclear Station. Five pins from these assemblies were sent to Oak Ridge National Laboratory in January 2009 and are currently undergoing analysis.

Additional information on MOX fuel and its use in power reactors are available at the following links:

<http://www.nrc.gov/materials/fuel-cycle-fac/mox/reactors.html> and <http://www.nrc.gov/reading-rm/doc-collections/fact-sheets/mox-bg.html>.

U.S. Power Plants (General)

Q: *What happens if the MFFF were to lose offsite power as a result of an earthquake?*

A: The design of the MFFF electric power supply system consists of a normal power system, a standby power system, and an emergency power system. Two separate and independent incoming offsite power feeders supply MFFF facility. In the rare event of a total loss of all incoming power to the facility, a standby power system composed of two independent standby diesel generators will automatically start and continue the supply of electrical power to the facility.

Q: *How would the NRC and the MFFF respond to an earthquake at the MFFF?*

A: MOX Services will follow the DOE Savannah River Emergency plan during an event at the facility. MOX Services will contact the NRC and DOE and interactions with State and local officials are conducted through the SRS Emergency Duty Officer who oversees the SRS Operations Center. The MFFF emergency preparedness program incorporates plans for radiation monitoring, repair and recovery efforts, search and rescue, and initial medical response.

Q: *What is the design basis earthquake for the Mixed Oxide (MOX) Fuel Fabrication Facility (MFFF) facility?*

A: The MFFF is located on Department of Energy's (DOE) Savannah River Site (SRS) near Aiken, South Carolina. At the SRS, there are no known capable or active faults within the 320-km (200-mile) radius that influence the seismicity of the region, with the exception of faults associated with the Charleston seismic zone (MFFF License Application). Earthquakes that could affect safe operations in the MFFF are associated with two seismic sources, repeat of Charlestown 1886 earthquake and small shallow earthquake of the South Carolina Piedmont. The MFFF facility is design to nuclear power plant requirements and utilizes the spectrum that is found in Regulatory Guide (RG) 1.60 anchored at 0.20g PGA.

Q: *Would overpressurized uranium hexafluoride cylinders stored at MFFF greatly increase the risk to members of the public during an earthquake?*

A: No. There are no large inventories of Special Nuclear Material (SNM) stored or used in a gaseous or highly dispersible form similar to the uranium hexafluoride at the MFFF facility. The primary form of SNM in the MOX facility would be powder. There are no significant additional hazards to members of the public due to the powder form during an earthquake.

Q: *What would happen if an earthquake occurred in the vicinity of the MFFF?*

A: In the event that an earthquake occurs in the vicinity of the facility the seismic monitoring and trip system initiates a shutdown of process-related systems if a seismic event exceeds a specified set point. The seismic monitoring and trip system shuts down normal and standby power systems, ensuring that all movements of nuclear material are stopped in a safe manner.

Q: *What are the potential impacts of an earthquake that exceeded the design of the MFFF?*

A: The MFFF facility was design using DOE's methodologies for natural phenomena hazards that establishes performance goals for nuclear facilities. In the unlikely event of an earthquake exceeding the design basis of the MFFF, it is expected that major structures such as buildings will suffer major damage, but the damage is limited in the extent such that the occupants can safely exit the building.

Q: *Is the heat released from MOX fuel greater than the heat released from traditional uranium fuel?*

A: Yes. Plutonium fissioning releases slightly more heat than uranium fissioning. Core designers consider this effect when designing the fuel loading for the core.

Q: *Are the MFFF emergency power diesels built to withstand the effects of an earthquake, if not what happens when power to the facility is lost?*

A: In the unlikely event that a total loss of all incoming power occurs and both standby diesel generator systems fail to start as discussed before, an independent and redundant seismically qualified emergency power system will provide electrical power to the facility. The emergency power system consists of two redundant and independent emergency diesel generator systems each of which has been designed to carry all important loads during an extended period of time until either the normal or standby power system can be restored. The emergency power system is qualified to survive the MFFF design-basis earthquake.

U.S. Power Plants (General)

New Nuclear Power Plants

- Q:** *Does the NRC expect that applications for reviews such as reactor extensions and new reactor licenses be affected from the Near Term Task Force review? NEW!*
- A:** The NRC will continue to process existing applications for new licenses and license renewal applications in accordance with the schedules that have been established. The NRC continues to believe that its regulatory framework and requirements provide for a rigorous and comprehensive license review process that examines the full extent of siting, system design and operations of nuclear power plants. The recommendations of the NRC's task force examining lessons learned from the events in Japan will be taken into account in the performance of the NRC's review of these applications, as appropriate. Further, the NRC has the necessary regulatory tools to require changes to existing licenses or applications for certification should the agency determine that changes are necessary.
- Q:** *Has NRC's plan with new reactor reviews changed following tragedy in Japan? Which four reactors are expected to be under construction? NEW!*
- A:** The NRC's plans with regard to New Reactor Licensing Reviews have not changed. Summer, Units 2 and 3, and Vogtle, Units 3 and 4, are currently conducting "pre-construction" activities; full construction can only occur after the NRC issues a Combined License. NRC has not received any indication from industry or current applicants that there is any change regarding their plans to obtain new licenses.
- Q:** *Are the next generation of reactor designs beyond Gen III+, like small modular and high temperature gas reactors, safer than the Fukushima Daiichi reactors? NEW!*
- A:** The Nuclear Regulatory Commission's Advanced Reactor Policy Statement (73 FR 60612) sets an overall Commission expectation "that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions." The Policy Statement describes attributes of such designs, including "designs that minimize the potential for severe accidents and their consequences by providing sufficient inherent safety, reliability, redundancy, diversity, and independence in safety systems, with an emphasis on minimizing the potential for accidents over minimizing the consequences of such accidents."

Recently-proposed reactor designs claim to fulfill these expectations by providing simpler systems with increased passive cooling capabilities. However, since formal applications with detailed design information are not expected to be submitted to the NRC until at least the end of calendar year 2012 for review, it is premature to make definitive statements about these designs. While promising, the designers' claims must be verified through detailed review by the NRC staff that will include full consideration of lessons learned and insights from the Japanese earthquake and tsunami.

U.S. Power Plants (General)

Q: *Are those "Gen III+" reactors safer than the Fukushima Daiichi reactors? What are the safety advantages? NEW!*

A: The NRC has not completed any study that would compare the safety of the "GEN III+" reactors to the safety of the Fukushima Daiichi reactors. However, the "GEN III+" reactors do contain additional or innovative features that are different from the design of the Daiichi reactors and are intended to provide added reliability to system availability or to enhance the safety of the proposed design. "GEN III+" reactors can be divided into two categories – passive designs or evolutionary designs. A couple of examples of the design enhancements for each category are given below.

Passive designs

The most significant innovation of Gen III+ systems over second-generation designs is the incorporation of passive safety features that do not require active controls or operator intervention. Instead they rely on gravity or natural convection to mitigate the impact of abnormal events. For example, the AP1000 is equipped with reactor and containment cooling systems that employ water tanks, which can be emptied into the reactor vessel and into containment to flood in and around the reactor vessel. These systems provide cooling such that the reactor will remain in a safe condition for 72 hours without any external power or significant operator actions.

Another system advantage in some GEN III + designs is the use of a passive isolation condenser system. The ESBWR has a passive isolation condenser system that will remove heat from the reactor after it is shut down with no electrical power. Combined with its passive containment cooling system that removes heat from the containment, the system permits 72 hours of no operator actions and no external power for the power plant to remain in a safe mode. Additionally, part of the passive system is a gravity-driven cooling system which provides return water to the reactor core.

Evolutionary Designs

Evolutionary designs contain innovative enhancements or increased redundancy to existing reactor designs. For example, the EPR and USAPWR designs provide additional trains of both emergency AC power and emergency core cooling to enhance the availability and reliability of essential safety systems. Specifically, the major safety systems consist of four trains (instead of the current designs that provide only 2 full capacity trains), each capable of performing the entire safety function on its own. Each safety system is physically separated from the others and they are located in separate parts of the plant and have their own protection features. This reduces the likelihood of simultaneous failure of all the safety systems due to internal or external events, such as fire, flooding or airplane crash.

Another example is the use of containment refueling water storage. In today's reactors, the refueling water storage tank, which is the water source for the emergency core coolant system, is located outside the containment vessel. Its location makes it vulnerable to loss or damage from natural phenomena such as earthquakes, tornadoes, floods, etc. In reactors with evolutionary designs, the refueling water is placed inside containment, which effectively minimizes the potential for loss of this essential water source for core cooling. It also provides another heat and pressure removing feature for the containment to mitigate the potential for overpressurization of the containment building.

Finally, all the "GEN III+" designs contain some design features to mitigate a postulated severe accident such as core melt either by flooding the reactor cavity space immediately surrounding the reactor vessel with water to submerge the reactor vessel or by providing some form of core spreading device, commonly referred to as a core catcher, so that substantial releases of radioactivity are mitigated.

Q: *Will the NRC continue processing existing license applications while the task force conducts its analysis?*

A: The NRC will continue to process existing applications for new licenses (i.e., early site permits, design certifications, and combined licenses) and license renewal applications in accordance with the schedules that have been established. The NRC continues to believe that its regulatory framework and requirements provide for a rigorous and comprehensive license review process that examines the full extent of siting, system design and operations of nuclear power plants. The recommendations of the NRC's task force that was established to examine lessons learned from the events in Japan will certainly be taken into account in the performance of the NRC's review of these applications, as appropriate. Further, the NRC has the necessary regulatory tools to require changes to existing licenses or applications for certification should the agency determine that changes are necessary.

U.S. Power Plants (General)

Q: *With NRC moving to design certification, at what point is seismic capability tested – during design or modified to be site-specific? If in design, what strength seismic event must these be built to withstand?*

A: The regulations related to seismic requirements are contained in General Design Criterion 2 in Appendix A to Title 10 of the *Code of Federal Regulations*, Part 50.

During design certification, vendors propose a seismic design in terms of a ground motion spectrum for their nuclear facility. This spectrum is called a standard design response spectrum and is developed so that the proposed nuclear facility can be sited at most locations in the central and eastern United States. The vendors show that this design ground motion is suitable for a variety of different subsurface conditions such as hard rock, deep soil, or shallow soil over rock. Combined License and Early Site Permits applicants are required to develop a site specific ground motion response spectrum that takes into account all of the earthquakes in the region surrounding their site as well as the local site geologic conditions. Applicants estimate the ground motion from these postulated earthquakes to develop seismic hazard curves. These seismic hazard curves are then used to determine a site specific ground motion response spectrum that has a maximum annual likelihood of 1×10^{-4} of being exceeded. This can be thought of as a ground motion with a 10,000 year return period. This site specific ground motion response spectrum is then compared to the standard design response spectrum for the proposed design. If the standard design ground motion spectrum envelopes the site specific ground motion spectrum then the site is considered to be suitable for the proposed design. If the standard design spectrum does not completely envelope the site specific ground motion spectrum, then the COL applicant must do further detailed structural analysis to show that the design capacity is adequate. Margin beyond the standard design and site specific ground motions must also be demonstrated before fuel loading can begin.

Q: *Will this incident affect new reactor licensing?*

A: The NRC will continue to process existing applications for new reactor licenses (i.e., early site permits, design certifications, and combined licenses) in accordance with the schedules that have been established. The NRC continues to believe that its regulatory framework and requirements provide for a rigorous and comprehensive license review process that examines the full extent of siting, system design and operations of nuclear power plants. The recommendations of the NRC's task force that was established to examine lessons learned from the events in Japan will certainly be taken into account in the performance of the NRC's review of these applications, as appropriate. Further, the NRC has the necessary regulatory tools to require changes to existing licenses or applications for certification should the agency determine that changes are necessary.

Q: *With all this happening, how can the NRC continue to approve new nuclear power plants?*

A: The NRC continues to believe that its regulatory framework and requirements provide for a rigorous and comprehensive license review process that examines the full extent of siting, system design and operations of nuclear power plants.

The NRC's task force recommended that a regulatory framework that balances defense-in-depth and probabilistic risk analysis should be adopted for operating plants. It found that both of these elements are currently required for new plants. New applicants are required to submit a level whatever PRA while defense-in-depth severe accident requirements exist for new reactors.

The recommendations of the NRC's task force that was established to examine lessons learned from the events in Japan will certainly be taken into account in the performance of the NRC's review of these applications, as appropriate. Further, the NRC has the necessary regulatory tools to require changes to existing licenses or applications for certification should the agency determine that changes are necessary.

U.S. Power Plants (General)

Q: *How will the task force's timeline of 90 days for its short-term analysis and approximately six months for its long-term recommendations impact existing license applications?*

A: The timelines for the task force analyses and recommendations will not have any immediate effect on the review of existing license applications. The NRC will continue to process existing applications for new licenses and license renewal applications in accordance with the schedules that have been established. The NRC continues to believe that its regulatory framework and requirements provide for a rigorous and comprehensive license review process that examines the full extent of siting, system design and operations of nuclear power plants. The recommendations of the NRC's task force that was established to examine lessons learned from the events in Japan will certainly be taken into account in the performance of the NRC's review of these applications, as appropriate. Further, the NRC has the necessary regulatory tools to require changes to existing licenses or applications for certification should the agency determine that changes are necessary. For example, any new requirements that may result from the task force's recommendations could be implemented in accordance with existing agency policies that may involve rulemaking or backfitting.

Potential Consequences

Q: *Has the government set up radiation monitoring stations to track the release?*

A: The NRC understands that EPA is utilizing its existing nationwide radiation monitoring system, RadNet, to continuously monitor the nation's air and regularly monitors drinking water, milk and precipitation for environmental radiation. EPA has publicly stated its agreement with the NRC's assessment that we do not expect to see radiation at harmful levels reaching the U.S. from damaged Japanese nuclear power plants. Nevertheless, EPA has stated that it plans to work with its federal partners to deploy additional monitoring capabilities to parts of the western U.S. and U.S. territories.

Q: *Why is KI administered during nuclear emergencies?*

A: KI – potassium iodide – is one of the protective measures that might be taken in a radiological emergency in this country. A KI tablet will saturate the thyroid with non radioactive iodine and prevent the absorption of radioactive iodine that could be part of the radioactive material mix of radionuclides in a release. KI does not prevent exposure from these other radionuclides.

Q: *What should be done to protect people in Alaska, Hawaii and the West Coast from radioactive fallout?*

A: The NRC believes that the actions of the Japanese to control, stabilize and mitigate radioactive releases from the reactors at Fukushima have prevented harmful levels of radiation from reaching U.S. territory. The NRC continues to believe that protective measures are unnecessary in the United States. No U.S. states or territories have detected harmful levels of radioactivity. In the unlikely event that circumstances change, U.S. residents should listen to the protective action decisions of their states and counties. These protective action decisions could include actions such as sheltering, evacuation, or taking potassium iodide. The NRC will provide technical assistance to the states should they request it.

Q: *What are the short-term and long-term effects of exposure to radiation?*

A: The NRC does not expect that residents of the United States or its territories are at any risk of exposure to harmful levels of radiation resulting from the events in Japan.

On a daily basis, people are exposed to naturally occurring sources of radiation, such as from the sun or medical X-rays. The resulting effects are dependent on the strength and type of radiation as well as the duration of exposure.

Q: *My family has planned a vacation to Hawaii/Alaska/Seattle next week – is it safe to go, or should we cancel our plans?*

A: The NRC does not expect that residents of the United States or its territories are at any risk of exposure to harmful levels of radiation resulting from the events in Japan. Any changes to travel are a personal decision. The NRC is unaware of any travel restrictions within the United States or its territories.

U.S. Power Plants (General)

Q: *What are the risks to my children?*

A: The NRC continues to believe that protective measures are unnecessary in the United States. No U.S. states or territories have detected harmful levels of radioactivity. In the unlikely event that circumstances change, U.S. residents should listen to the protective action decisions of their states and counties. These protective action decisions could include actions such as sheltering, evacuation, or taking potassium iodide. The NRC will provide technical assistance to the states should they request it. United States citizens in Japan are encouraged to follow the protective measures recommended by the Japanese government. These measures appear to be consistent with steps the United States would take.

Q: *Are there other protective measures I should be taking?*

A: The NRC continues to believe that protective measures are unnecessary in the United States. No U.S. states or territories have detected harmful levels of radioactivity. In the unlikely event that circumstances change, U.S. residents should listen to the protective action decisions of their states and counties. These protective action decisions could include actions such as sheltering, evacuation, or taking potassium iodide. The NRC will provide technical assistance to the states should they request it. United States citizens in Japan are encouraged to follow the protective measures recommended by the Japanese government. These measures appear to be consistent with steps the United States would take.

Q: *Are air and sea shipments from Japan being checked for radiation contamination?*

A: U.S. Customs and Border Protection (CBP), a part of the Department of Homeland Security, is responsible for monitoring food and cargo at U.S. ports of entry. In accordance with established protocols, CBP uses radiation detection equipment at both air and sea ports, and uses this equipment, along with specific operational protocols, to resolve any security or safety risks that are identified with inbound travelers and cargo. CBP has issued field guidance reiterating its operational protocols and directing field personnel to specifically monitor maritime and air traffic from Japan. CBP will continue to evaluate the potential risks posed by radiation contamination on inbound travelers and cargo and will adjust its detection and response protocols, in coordination with its interagency partners, as developments warrant. The NRC works closely with CBP and the U.S. Environmental Protection Agency when CBP identifies radioactive materials that may involve licensed materials or radioactive materials shipped from other countries inadvertently.

Q: *The radiation "plume" seems to be going out to sea -- what is the danger of it reaching Alaska? Hawaii? The west coast?*

A: In response to nuclear emergencies, the NRC works with other U.S. agencies to monitor radioactive releases and predict their path. The NRC continues to monitor information regarding wind patterns near the Japanese nuclear power plants. Nevertheless, given the thousands of miles between the two countries, Hawaii, Alaska, the U.S. Territories and the U.S. West Coast are not expected to experience any harmful levels of radioactivity.

Q: *Is the U.S. government tracking the radiation released from the Japanese plants?*

A: Yes, a number of U.S. agencies are involved in monitoring and assessing radiation including EPA, DOE, and NRC. The best source of additional information is the Environmental Protection Agency.

Q: *Is there a danger of radiation making it to the United States?*

A: In response to nuclear emergencies, the NRC works with other U.S. agencies to monitor radioactive releases and predict their path. The NRC continues to monitor information regarding wind patterns near the Japanese nuclear power plants. Nevertheless, given the thousands of miles between the two countries, Hawaii, Alaska, the U.S. Territories and the U.S. West Coast are not expected to experience any harmful levels of radioactivity.

Q: *Was there any damage to US reactors from either the earthquake or the resulting tsunami?*

A: No.

U.S. Power Plants (General)

Q: *I live in the Western United States – should I be taking potassium iodide (KI)?*

A: The NRC continues to believe that protective measures are unnecessary in the United States. No U.S. states or territories have detected harmful levels of radioactivity. In the unlikely event that circumstances change, U.S. residents should listen to the protective action decisions of their states and counties. These protective action decisions could include actions such as sheltering, evacuation, or taking potassium iodide. The NRC will provide technical assistance to the states should they request it.

Power Supplies

Q: *What are US plants required to have for backup power?*

A: U.S. plants are required to meet General Design Criterion 17 in Appendix A to Title 10 of the *Code of Federal Regulations*, Part 50. Reactor units must have 2 independent power supplies. All U.S. plants, except one (i.e., Oconee), have diesel generators and battery backup systems. The remaining plant has a hydroelectric power facility for backup. Most of the U.S. plants with diesels have two diesels per unit and those that have only one dedicated diesel have a swing diesel available. The regulations do not specify the length of time that the diesels and batteries must operate following a loss of offsite power (most sites plan to run the diesels for multiple days and have battery backup capability for 8 hours). Instead the amount of time is dependent on the site recovery strategy and is based on providing sufficient capacity to assure that the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

Q: *Is our battery backup power less effective than the Japanese?*

A: US regulations do not specify the length of time that a facility needs to have the batteries operate following a loss of offsite power. Instead, the amount of time is dependent on the site recovery strategy and is based on providing sufficient capacity to assure that the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

With respect to a comparison of battery backup power effectiveness, we currently do not have sufficient information to compare the differences in design requirements and performance characteristics of nuclear-grade batteries in the U.S. and Japanese nuclear power plants. However, in the U.S., nuclear power plants utilize redundant nuclear-grade (i.e., Class 1E, safety-related) batteries that are designed and constructed using rigorous standards and are routinely tested in accordance with plant technical specifications to ensure adequate capacity and capability exists to perform their intended safety functions. These batteries are located in structures that can withstand external environmental events such as earthquakes, tornadoes, tsunamis, and floods in accordance with NRC regulations. For U.S. nuclear power plants, the typical design duty cycles for safety grade batteries range from 1 - 8 hours (i.e., 1-2 hours for accident; 4 hours for station blackout; and 1-8 hours for a fire).

U.S. Power Plants (General)

Q: *Are U.S. nuclear power plants designed for scenarios similar to what happened in Japan where all power to the reactors (i.e., both the power grid and emergency onsite power) was lost as a result of the earthquake and resultant tsunami?*

A: The NRC requires that all nuclear power plants are able to withstand a station blackout (SBO) - a complete loss of AC electric power to the station. These requirements are specified in 10 CFR 50.63, Loss of all alternating current power, and a more detailed definition is provided in 10 CFR 50.2, Definitions. The definition of coping is the time it takes until off site power is restored (i.e., the grid) or an emergency diesel generator, located either onsite or offsite, is restored to service. To meet this requirement, all nuclear power plants performed an SBO coping analysis that determined how long the plant could cope without AC power. The NRC has provided guidance for determining a plant specific SBO duration in Regulatory Guide 1.155, "Station Blackout," (August 1988). In general, SBO durations range from 2 to 16 hours, though licensees may propose alternate durations based on specific factors relating to the offsite and onsite power characteristics. There are two methods of coping with an SBO event. They are either: (i) AC independent (i.e., relying on battery power), or; (ii) alternate AC (AAC).

AC independent plants had to satisfy all the requirements for maintaining a plant in a safe condition for maximum duration of 4 hours.

If the configuration of offsite power (i.e., the grid system), onsite power (i.e., emergency diesel generators) and reliability of these sources could be affected by weather related events, and if restoration of these sources was not possible within 4 hours, then plants had to use an alternate AC source (i.e. AAC). Some plants decided to comply with the SBO rule by using the AAC as they already had that capability on their sites. Plants using an AAC source had a variable coping duration between 2 hours and 16 hours. This duration was subject to factors affecting the restoration of onsite or offsite power sources. The capability for coping with an SBO of specified duration must be determined by a coping analysis for plants with an AC independent method (i.e., batteries) and for plants with an AAC if that source is not available within 10 minutes of the initiating event.

Q: *A recent newswire article claimed that 93 of the U.S. nuclear power plants only had a 4-hour coping capacity for SBO. The rest of the plants could cope for 8 hours. Is that information correct?*

A: That information is not correct. First to clarify SBO coping capacity, the definition of coping is the time required to restore off site (i.e., the grid) or onsite power (i.e., emergency diesel generator). There are two different methods for coping with an SBO event:

- Relying only on battery power (AC-independent)

- Relying on an Alternate AC power source (i.e., an emergency diesel generator, hydro-powered generator, or a gas turbine)

The NRC only allows up to a 4-hour SBO coping analysis with batteries, anything longer requires an alternate AC source. The SBO coping time for an alternate AC source ranges from 2 hours to a maximum of 16 hours. For the 104 operating plants in the U.S., the basic breakdown with respect to power source is that 44 plants are "battery coping plants" and 60 plants are "alternate AC source" plants.

With respect to SBO coping times, a further breakdown includes 44 plants that have adopted the AC-independent method and have battery power for 4 hours. Another 43 plants use the AAC methodology and can restore AC power (i.e., offsite power or emergency diesel generator) within 4 hours. Hence a total of 87 plants have 4-hour SBO coping duration.

For the remainder of U.S. plants, 14 plants use the AAC methodology and can restore AC power (i.e., offsite power or emergency diesel generator) within 8 hours; hence these 14 plants have an 8-hour SBO coping duration. There are 3 plants that use the AAC methodology and have 16-hour duration for restoration of AC power; hence the remaining 3 plants have a 16-hour SBO coping duration.

U.S. Power Plants (General)

Price-Anderson Act

Q: *Why does the NRC let a private insurance company determine the amount of insurance coverage? Why does this private company control public protection?*

A: The intent of the Price-Anderson Act was to allow the government to regulate the safety of nuclear power while allowing the private insurance industry to provide financial protection. The NRC is the government agency that is responsible for ensuring that nuclear power plants are designed and operated in a way that protects public health and safety. The NRC is confident that the amount of insurance coverage determined by the private insurance company is adequate to provide financial compensation in the event of a nuclear accident.

Q: *When does the Price-Anderson Act expire?*

A: In 2005, the Price-Anderson Act was extended through December 31, 2025.

Q: *Has Price-Anderson ever been used?*

A: Only once. During the 1979 accident at the Three Mile Island Nuclear Power Plant, the Price-Anderson Act provided liability insurance to the public. The day after the accident, insurance company representatives established a local claims office in Pennsylvania. Advertisements were placed in local newspapers to inform residents of claims procedures. The insurance paid for the living expenses of families who decided to evacuate, although evacuation was not immediately ordered. When Pennsylvania's governor recommended the evacuation of pregnant women and families with young children who lived near the plant, the insurance paid for those evacuation expenses, too. In 1979, more than 3,000 people received nearly \$1.2 million in evacuation claims. More than 600 people were also reimbursed for lost wages as a result of the accident. In the months after the accident, numerous lawsuits were filed alleging various injuries and property damages. To date, the Price-Anderson insurance has paid about \$71 million in claims and litigation costs associated with the Three Mile Island accident. All payments were made from the primary insurance coverage. Money from the secondary layer of insurance was not needed.

Q: *The accidents in Japan affected the reactors and the spent fuel pools. Does the Price-Anderson Act cover all nuclear plant accidents or just some of them?*

A: The Price-Anderson Act covers all property and liability claims resulting from nuclear accidents at commercial nuclear power plants. This includes any incident related to the reactor or the spent fuel pool. Price-Anderson also covers claims related to transporting nuclear fuel and nuclear waste in and out of the plant.

Q: *More than a million people live within 50 miles of Plant X. How is a \$375 million insurance policy supposed to cover all of us?*

A: The Price-Anderson Act is a federal law that requires owners of nuclear power plants to purchase \$375 million of offsite liability insurance for each reactor at the plant. If a nuclear accident causes damages of more than \$375 million, the insurance is supplemented by additional coverage that is shared by every nuclear power plant in the country. There are currently 104 reactors licensed to operate in the United States, so this secondary pool of money contains about \$12.6 billion ... all of this secondary money is used, Congress would determine whether to provide additional disaster relief.

Q: *My insurance company is a nationally known, reputable business that I trust. What insurance company does the nuclear plant use – a good one or the cheapest one they can find?*

A: All U.S. nuclear power plant owners purchase their Price-Anderson insurance from American Nuclear Insurers (ANI), which is made of several large and reputable insurance companies. About half of the ANI companies are foreign insurance businesses. On average, a nuclear power plant owner pays about \$400,000 per year for Price-Anderson insurance at a single-unit reactor site. For power plants with more than one reactor, the total annual insurance cost is typically discounted, similar to automobile insurance for households with more than one car.

Q: *The Price-Anderson Act is a federal law? Why does the government spend my tax dollars on providing nuclear insurance to big energy companies?*

A: The Price-Anderson Act is a federal law, but your tax dollars do not pay for the insurance it requires owners of nuclear power plants to purchase. The extra insurance protection required for large commercial nuclear power companies is purchased at no cost to the public or the federal government.

U.S. Power Plants (General)

Q: *My insurance agent said that my homeowner's insurance does not cover nuclear accidents. Does Price-Anderson protect me?*

A: Your homeowner's insurance policy does not cover nuclear accidents because Price-Anderson covers claims related to nuclear accidents. By law, owners of nuclear power plants are required to purchase \$375 million of offsite liability insurance for each reactor at the plant. If a nuclear accident causes damages of more than \$375 million, the insurance is supplemented by additional coverage that is shared by every nuclear power plant in the country. There are currently 104 reactors licensed to operate in the United States, so this secondary pool of money contains about \$12.6 billion. If all of this secondary money is used, Congress would determine whether to provide additional disaster relief.

Q: *What is the Price-Anderson Act?*

A: In 1957, a federal law called the Price-Anderson Act was established to ensure that adequate money would be available to pay insurance claims following an accident at a commercial nuclear power plant. That law is still in place to protect those that live around nuclear power plants.

Q: *I'll have to find another place to stay if I have to evacuate my home during a nuclear accident. I can't afford to pay for a hotel or apartment for several months while the government tries to clean things up. How am I supposed to pay for that?*

A: Insurance under the Price-Anderson Act covers bodily injury, sickness, disease or resulting death, property damage and loss, and reasonable living expenses for people who are evacuated from a nuclear accident. The Stafford Act is another federal law that provides disaster relief to state and local governments. If a nuclear accident is declared an emergency or major disaster by the President, the Stafford Act will also be available to provide assistance to accident victims. The Stafford Act allows the federal and state governments to share costs of temporary housing for up to 18 months. It also provides additional money for home repair and temporary mortgage or rental payments. Distribution of Stafford Act funding is done through the Federal Emergency Management Agency. Together, the Price-Anderson and Stafford Acts provide money for a variety of expenses following a nuclear accident.

Radiation Protection

Q: *Where would I get IOSAT Potassium Iodide if my city should experience fallout from the Japanese nuclear disaster? Is this the right precaution or is there anything else that can be done to protect myself?*

A: We do not expect any U.S. states or territories to experience harmful levels of radioactivity. As such, we do not believe that there is any need for residents of the United States to take potassium iodide. U.S. residents should listen to the protective action decisions by their states and counties. As necessary, protective action decisions could include actions such as sheltering, evacuating, or taking potassium iodide.

Additional information regarding the use of potassium iodide can be found on NRC's webpage at the following link:

<http://www.nrc.gov/about-nrc/emerg-preparedness/about-emerg-preparedness/potassium-iodide-use.html>.

Since Potassium Iodide is classified as a drug. Additional information is on the Food and Drug Administration's web site: www.fda.gov.

Q: *How is EPA monitoring, collecting and posting information related to the impacts in the U.S. of the accident in Japan?*

A: The EPA monitors, collects, and posts information related to the impacts of the Japanese events on the U.S. using their RadNet system. They have 100 fixed radiation monitoring sites in 48 states plus 40 additional deployable monitors that may be sent where needed. The fixed monitors provide information on beta and gamma radiation levels. The deployable monitors measure the external exposure rate and provide weather information. The data from these monitors is sent to a computer, where it is continually reviewed and is usually posted on the EPA's Central Data Exchange website (<http://epa.gov/cdx>) within 2 hours. However, if the computer picks up an abnormality in the radiation level, then the EPA laboratory staff is alerted and reviews the information prior to it being posted. In response to the events in Japan, EPA has sent additional monitors to Guam, Hawaii, and Alaska.

The EPA also monitors contamination in rainwater and drinking water as well as the level of iodine in milk. The EPA provides updates on these testing efforts and a summary of the air radiation monitoring results on its webpage, <http://www.epa.gov/japan2011/>. This webpage contains a link to Frequently Asked Questions, which was the source of information for this response. Additional information may be found there.

U.S. Power Plants (General)

Q: *Why does the NRC not establish a 50-mile EPZ in the U.S. if this was the NRC's recommendation for the accident in Japan?*

A: The United States government cannot intervene in the management of events internal to another sovereign nation. The US government can only make recommendations to its citizens in that country on actions for their safety. The State Department routinely issues such recommendations (known as travelers warning and advisories) for many different types of events; civil unrest, terrorism, natural disasters and technological accidents. It is within this context that the Nuclear Regulatory Commission made a recommendation to the US Ambassador in Japan for protective actions for US citizens residing in the regions surrounding the damaged Fukushima Daiichi Nuclear Power Plant site.

The decision-making environment that existed at the time in which the NRC decision was made was one in which: there was limited and often conflicting information about the exact conditions of the reactors and spent fuel pools at the Fukushima nuclear facility immediately following the earthquake and tsunami; radiation monitors showed significantly elevated readings in some areas of the plant site which would challenge plant crews attempting to stabilize the plant; analysis results from offsite samples indicated that some fuel damage had occurred; there was a level of uncertainty about whether or not efforts to stabilize the plant in the very near term were going to be successful, and; changing meteorological conditions resulted in the winds shifting rapidly from blowing out to sea to blowing back onto land.

In its evaluation of the rapidly changing and unprecedented event, the NRC performed a series of dose calculations to assess a "worst case" scenario. This was a conservative calculation which considered the rapidly changing course of the events and the very real possibility that these events were going to continue to degrade. As a result of these calculations, the progression of events and the uncertainty regarding the plans to bring the situation under control, the decision was made to recommend the evacuation of US citizens out to 50 miles from the facility.

In the United States, the NRC has direct access to the plant site including the control room and any and all vital plant areas. The NRC maintains two resident inspectors at each plant who have unfettered access to the site. In addition, the NRC has required that direct communications links between the NRC Operations Center and the plant be installed, tested, and routinely exercised. These links provide NRC staff and the Executive team with up-to-date and reliable information about the ongoing events at the plant. In addition, the Chairman can order the plant to take actions to mitigate the event if the NRC does not believe that the appropriate actions are being taken by the plant operators.

In the U.S., there are two emergency planning zones (EPZ) established around a nuclear power plant. The first zone, the 10-mile EPZ, is where exposure from a radiological release event would likely be from the radioactive plume and it is in this EPZ where protective actions such as sheltering and/or evacuation would be appropriate. Beyond the 10-mile EPZ and out to the 50-mile EPZ is the ingestion exposure pathway where exposure to radionuclides would likely be from ingestion of contaminated food/milk and surface water. Comprehensive planning is performed for these zones and is routinely tested and evaluated by way of the full participation exercises. These zones are not limits but rather provide for a comprehensive emergency planning framework that would allow expansion of the response efforts beyond the zones should radiological conditions warrant such expansion. Nuclear power plant licensees are required to have an emergency plan for both the onsite and offsite response that has been evaluated and tested prior to obtaining an operating license and must conduct such exercises on a biennial cycle. The NRC remains confident that its current regulatory framework for emergency preparedness, including the establishment of an EPZ, and the flexibility to respond to emergent radiological conditions, as necessary, provides adequate protection for the health and safety of the public.

The NRC's Near-Term Task Force issued its report on July 12 and it is available to the public (ADAMS Accession No. ML111861807). On July 19, 2011, the Task Force presented its findings to the Commission and proposed improvements in multiple areas including emergency preparedness. The Task Force considered the existing planning structure, including the 10-mile plume exposure pathway and 50-mile ingestion pathway emergency planning zones, and found no basis to recommend a change. The development of protective action recommendations by the Japanese government, including expansion of evacuations out to 20 km (~12 miles) from the plant supported effective and timely evacuation to minimize the impact of the radiological releases on public health and safety. Subsequent decisions by the Government of Japan to evacuate selected areas based on potential long-term exposures are consistent with the U.S. strategy to expand protective actions during an event consistent with developments at the time and provided timely and effective actions to protect the public in those areas. Therefore, the Task Force found no basis to recommend changes to the emergency planning zones.

U.S. Power Plants (General)

Q: *What should be done to protect people in Alaska, Hawaii and the West Coast from radioactive fallout?*

A: The NRC believes that the actions of the Japanese to control, stabilize and mitigate radioactive releases from the reactors at Fukushima have prevented harmful levels of radiation from reaching U.S. territory. The NRC continues to believe that protective measures are unnecessary in the United States. No U.S. states or territories have detected harmful levels of radioactivity. In the unlikely event that circumstances change, U.S. residents should listen to the protective action decisions of their states and counties. These protective action decisions could include actions such as sheltering, evacuation, or taking potassium iodide. The NRC will provide technical assistance to the states should they request it.

Q: *What is the official agency to report radiation numbers and what is the public contact?*

A: NRC regulations require nuclear power plants to report any radiation doses detected at the plant that could be harmful to the public. This would include doses that are generated by the plant or by an external source. During an event in the U.S., it is the state's responsibility to provide protective action decisions for public health and safety. For this incident, the Japanese are responsible for reporting the public dose; nevertheless, should radiation doses be detected within the U.S., it would still be the state's responsibility to provide protective action decisions for public health and safety.

Q: *I live in the Western United States – should I be taking potassium iodide (KI)?*

A: The NRC continues to believe that protective measures are unnecessary in the United States. No U.S. states or territories have detected harmful levels of radioactivity. In the unlikely event that circumstances change, U.S. residents should listen to the protective action decisions of their states and counties. These protective action decisions could include actions such as sheltering, evacuation, or taking potassium iodide. The NRC will provide technical assistance to the states should they request it.

Q: *Why is KI administered during nuclear emergencies?*

A: KI – potassium iodide – is one of the protective measures that might be taken in a radiological emergency in this country. A KI tablet will saturate the thyroid with non radioactive iodine and prevent the absorption of radioactive iodine that could be part of the radioactive material mix of radionuclides in a release. KI does not prevent exposure from these other radionuclides.

Reactor Oversight

Q: *Why do license renewal reviews not include a review of the plant's response to external events?*

A: The regulations stipulating the requirements associated with license renewal were issued via rulemaking in 1991 (54 FR 64943). As described in the Statement of Considerations (SOC) for this license renewal rule, the Commission determined that, with the exception of age-related degradation unique to license renewal, the NRC's existing regulatory process is adequate to ensure that the licensing bases of all currently operating plants provide and maintain an acceptable level of safety for operation. The Commission considered whether or not to include plant responses to external events that may be outside the licensing basis but reasoned that the existing regulatory process was sufficient to address those instances while at the same time avoiding duplicative and, perhaps, less efficient assessments. With this understanding, the Commission maintained that the focus of license application renewals should be limited to the age-related degradation management for systems, structures and components (SSCs) that are included in the scope of license renewal (e.g., important to safety, or whose failure could impact safety equipment). As a consequence, license renewal reviews consider applicant activities to detect, manage, and correct the effects of age-related materials degradation on SSCs to ensure that the functionality of safety equipment is not adversely impacted during the renewed license operating period.

Recent proceedings associated with Oyster Creek license renewal have reiterated the Commission's position that the NRC's comprehensive and ongoing oversight of licensed facilities will assure that useful data, operating experience, lessons learned, etc. will be absorbed by changes in NRC rules, orders, and license amendments, as needed, accompanied by the public participation required by statute and regulation. Therefore, plant response to external events will be reviewed when the need is identified, irrespective of the plant's status regarding license renewal (e.g., post-Fukushima review is being done for all plants, and actions will be taken and applied based on plant designs). The NRC has completed its near-term review of lessons learned from the events at Fukushima. The Commission is currently reviewing the report and will provide the staff with direction. Any changes will be applied to plants irrespective of whether a plant has a renewed license or not.

U.S. Power Plants (General)

Regulatory Requirements (US)

Q: *How do we know that the spent fuel in pools at reactor sites is safe, in light of the knowledge of seismic risks?*

A: The agency continues to believe that spent fuel pools provide adequate protection of public health and safety. Over the course of many years, the NRC has taken advantage of the lessons learned from previous operating experience to implement a program of continuous improvement in the regulation of U.S. commercial nuclear reactors. This has included regular examination of topics related to spent fuel storage, as well as implementation of changes that have improved the safety of spent fuel pools. In addition, following the terrorist attacks of September 11, 2001, the NRC undertook an extensive re-examination of spent fuel pool safety and security. As a result of this reexamination, the Commission issued orders requiring licensees to implement additional strategies to keep spent fuel pools cool in the aftermath of a large explosion or fire at the plant. These requirements have since been incorporated into NRC regulations. The NRC's Japan Task Force has recommended the Commission consider additional enhancements to spent fuel pool makeup capability and instrumentation. As directed by the Commission, the NRC staff will implement any changes found to be appropriate to maintain the safety of spent fuel storage systems.

Q: *The NRC has proposed recommendations regarding station blackout. Would NRC consider putting a hold on any changes to all plants for changes dealing with on-site power generation?*

A: The NRC requires that all nuclear power plants are able to withstand a station blackout (SBO) - a complete loss of AC electric power to the station. These requirements are specified in 10 CFR 50.63, Loss of all alternating current power, and in 10 CFR 50.2, Definitions. The NRC's Japan Task Force has recommended the agency use certain lessons learned from Fukushima to improve the NRC's regulatory framework. The Commission has noted that some Task Force's recommendations raise very complex technical and regulatory questions that will require significant analysis. Since the events in Japan continue to evolve, the NRC has used and will continue to use the analytical resources and stakeholder engagement capabilities of the agency to ensure the consideration of many issues. There has been no reduction in the licensing or oversight function of the NRC as it relates to any of the NRC licensees. Contributors to the conclusion that the current fleet of reactors and materials licensees continue to protect the public health and safety are based on a number of principles, including defense in depth. Every U.S. reactor is designed for natural events based on the specific site where the reactor is located.

Research and Development

Q: *Is the NRC involved in research and development of new electric generation technologies? Is the NRC taking action to ensure a robust fuel mix portfolio including nuclear power?*

A: The NRC plays no role in the promotion and research of energy technologies, including nuclear power. The Energy Reorganization Act of 1974 divided the former Atomic Energy Commission into the Energy Research and Development Administration, which later became the U.S. Department of Energy (DOE) and the NRC. DOE's mission is to perform research and development activities in support of a national goal of energy independence, whereas NRC's mission is to regulate the nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, to promote the common defense and security, and to protect the environment.

U.S. Power Plants (General)

Q: *Does NRC have any research and development functions, especially in light of the events at Fukushima Daiichi Plant?*

A: The NRC's Office of Nuclear Regulatory Research (or RES), a major NRC program office, was mandated by Congress and created as part of the NRC in 1975. RES plans, recommends, and implements programs of nuclear regulatory research, standards development, and resolution of generic safety issues for nuclear power plants and other facilities regulated by the NRC. The Office coordinates research activities within and outside the agency, including NRC participation in national and international volunteer standards efforts. RES is responsible for developing methods, technical expertise and computer codes that are used by the NRC to assess safety and regulatory issues for materials licensees, fuel cycle facilities and operating reactors, as well as new and advanced reactor designs. RES develops the data needed to assess these codes by conducting experiments at national laboratories, universities, or in collaboration with international organizations.

The NRC regulatory research program addresses issues concerning nuclear reactors, nuclear materials, and radioactive waste. The research program is designed to improve the agency's knowledge where uncertainty exists, where safety margins are not well-characterized, and where regulatory decisions need to be confirmed in existing or new designs and technologies.

Q: *How are the research activities conducted and coordinated at the NRC?*

A: NRC's Office of Research (RES) coordinates research activities with the other NRC program offices, as appropriate, and leads the agency's initiative for cooperative research with the U.S. Department of Energy (DOE) and other Federal agencies, the domestic nuclear industry, U.S. universities, and international partners. RES coordinates the development of consensus and voluntary standards for agency use, including appointment of agency staff to various standards committees. Based on research results and experience gained, RES works with the regulatory offices to develop appropriate regulatory actions to resolve potential safety issues for nuclear power plants and other facilities regulated by the NRC, including those issues designated as Generic Issues (GIs). GIs are technical or security issues that could impact two or more facilities or licensees. RES also develops the technical basis for those areas regulated by the NRC that have risk-informed, performance-based regulations.

RES supplies technical tools, analytical models, and experimental data needed to support the agency's regulatory decisions. RES does not conduct research for the primary purpose of developing improved technologies. That is more appropriately done by the Department of Energy or the nuclear industry. Rather, the NRC conducts research to confirm that the methods and data generated by the industry ensure that adequate safety margin is maintained.

RES activities support regulation of the commercial use of radioactive materials to protect public health and safety and to protect the environment. RES is also responsible for providing the technical basis for regulations to ensure the protection and safeguarding of nuclear materials and nuclear power plants in the interest of national security. Thus, while its primary focus is on supporting the licensing and regulatory process, the research conducted by and for the NRC plays an important role in supporting broad government-wide initiatives associated with national security.

Severe Accidents

Q: *The NRC has proposed recommendations regarding station blackout. Would NRC consider putting a hold on any changes to all plants for changes dealing with on-site power generation?*

A: The NRC requires that all nuclear power plants are able to withstand a station blackout (SBO) - a complete loss of AC electric power to the station. These requirements are specified in 10 CFR 50.63, Loss of all alternating current power, and in 10 CFR 50.2, Definitions. The NRC's Japan Task Force has recommended the agency use certain lessons learned from Fukushima to improve the NRC's regulatory framework. The Commission has noted that some Task Force's recommendations raise very complex technical and regulatory questions that will require significant analysis. Since the events in Japan continue to evolve, the NRC has used and will continue to use the analytical resources and stakeholder engagement capabilities of the agency to ensure the consideration of many issues. There has been no reduction in the licensing or oversight function of the NRC as it relates to any of the NRC licensees. Contributors to the conclusion that the current fleet of reactors and materials licensees continue to protect the public health and safety are based on a number of principles, including defense in depth. Every U.S. reactor is designed for natural events based on the specific site where the reactor is located.

U.S. Power Plants (General)

Q: *Could there be core damage and radiation release at a U.S. plant if a natural disaster exceeding the plant design were to occur?*

A: U.S. nuclear power plants are built to withstand external hazards, including earthquakes, tsunamis, and flooding, as appropriate. The NRC has made substantial effort over time to ensure that vulnerabilities to both internal and external hazards were considered and mitigated in the plant current design and licensing basis of its regulated facilities. In 1988, the NRC's Generic Letter (GL) No. 88-20, "Individual Plant Examination [IPE] for Severe Accident Vulnerabilities," requested plant owners to perform a systematic evaluation of plant-specific vulnerabilities and report the results to the Commission. For many plants, the IPEs became the basis for the plant's initial Probabilistic Risk Assessment (PRA). Later the NRC issued Supplement 4 to GL 88-20, that requested licensees to evaluate vulnerabilities to external events (IPEEE). Most licensees made improvements to their facilities to reduce vulnerabilities identified in their IPEs and IPEEEs.

The ground motions that are used as seismic design bases at US nuclear plants are called the Safe Shutdown Earthquake (SSE) ground motions. In the 1990s, the NRC staff reviewed the potential for ground motions beyond the design basis as part of the Individual Plant Examination of External Events (IPEEE). From this review, the staff determined that seismic designs of operating nuclear plants in the US have adequate safety margins for withstanding earthquakes. Currently, the NRC is in the process of conducting a generic review (i.e., GI-199) to again assess the resistance of US nuclear plants to earthquakes. Based on NRC's preliminary analyses to date, the average probability of ground motions exceeding the SSE over the life of the plant for the plants in the Central and Eastern United States is less than about 1%. It is important to remember that structures, systems and components are required to have "adequate margin," meaning that they must continue be able withstand shaking levels that are above the plant's design basis.

Q: *How has the NRC evaluated nuclear waste storage containers, dry cask or otherwise, to ensure their adequate protection under the loading stresses of a severe seismic event, or other catastrophic impact incident? NEW!*

A: U.S. Nuclear Regulatory Commission (NRC) regulations in Title 10 Code of Federal Regulations Part 72 and performance standards require licensees to analyze the environmental conditions and natural phenomena surrounding each Independent Spent Fuel Storage Installation (ISFSI) to determine severe design-basis events for each site. To be certified by the NRC, the storage casks must be evaluated and shown to withstand the forces and stresses from the most severe loading conditions for each type of event.

In evaluating the adequacy of each licensee's design basis events, the NRC uses historical seismic events, nearby seismic faults and site-specific ground characteristics for each ISFSI to determine the ground motions that could affect an ISFSI. The casks are designed to maintain stability, withstand the ground motions, and safely confine and shield the spent nuclear fuel under such events. Licensees have conservatively shown, and NRC has confirmed, freestanding dry storage cask components, such as the canister and overpack, will neither tip over nor fail during such events. The dry storage casks are also analyzed for other severe natural phenomena and accidents such as cask drops and tipover, explosions, fires, floods, and tornado winds, and tornado missiles. The radiological consequences of a cask tipover accident are addressed in NRC safety evaluation reports and have been shown to be negligible.

Q: *The NRC Near-Term Task Force Report states that a sequence of events like the Fukushima accident is unlikely to occur in the U.S. and some appropriate mitigation measures have been implemented. What are those appropriate mitigation measures? NEW!*

A: The mitigation measures are what are commonly referred to as the B.5.b actions. These are the actions that were taken following the events of 9/11 in the United States. These measures would deal with the loss of large areas of the plant, including the use of portable equipment to provide some level of core cooling, spent fuel pool cooling and/or maintenance of containment integrity. They provide an additional level of mitigation capability that may be of assistance in the event of a significant accident similar to Fukushima.

Q: *Do the Japan events of March 2011 mean that there should be more concerns about seismic risks at San Onofre Generating Station (SONGS)? NEW!*

A: U.S. nuclear plant designs consider seismic events and tsunamis. It is important not to extrapolate earthquake and tsunami data from one location of the world to another when evaluating these natural hazards. These catastrophic natural events are location specific, based on the locations of tectonic and geological fault lines. The March 2011 Japan earthquake occurred on a subduction zone, which is a very different type of tectonic environment than the region around SONGS, which is predominantly strike slip. A magnitude 9 earthquake can only occur on a subduction zone and cannot occur in the region around SONGS.

U.S. Power Plants (General)

Q: *The Severe Accident Management Guidelines (SAMGs) that licensees are supposed to have are voluntary and not part of the NRC's baseline inspections. Why are these considered voluntary? How does the NRC know that the licensees have SAMGs in place?*

A: The NRC carries out its mission to protect public health and safety by specifying licensing and operational requirements that nuclear power plants must meet, and by inspecting and enforcing compliance with these requirements. When a licensee complies with the regulations, "adequate protection" is presumed.

The NRC can only impose requirements beyond those necessary for adequate protection by satisfying the Backfit Rule (10CFR 50.109), which requires evidence of "a substantial increase in the overall protection of the public health and safety or the common defense and security", and that the costs of implementation are justified in view of the increased protection.

Protection against severe accidents is provided by regulatory requirements in two basic ways: 1) Prevention of core damage events such that the likelihood of events that lead to core damage is very low; and 2) mitigation of consequences in the event of a severe accident. The combination of these two aspects must result in an acceptably low risk to public health and safety. The NRC has determined that the combination of these two aspects does result in an acceptably low risk.

Severe Accident Management Guidelines (SAMGs) address mitigation of consequences in the event of a severe accident. A variety of regulations were already in place prior to the development of SAMGs to provide for the mitigation of accidents that were either postulated to occur (this is the deterministic approach) or were the most probable to occur (this is the probabilistic or risk-informed approach). The licensing basis for a plant typically contains a combination of these approaches to accident analysis. These include, for example, those regulations related to reactor containments (10 CFR 50, Appendix, A Section V) and fuel and radioactivity controls (10 CFR 50, Appendix A, Section VI), reactor siting criteria (10 CFR Part 100), and Emergency Planning requirements (10 CFR 50 Appendix E). The pre-SAMG "mitigation" requirements in conjunction with existing "prevention" requirements were judged to provide adequate protection. Therefore, while SAMGs further enhance mitigation capability, their contribution to risk reduction did not rise to the level of justifying a new requirement. Accordingly, the staff worked with industry to encourage voluntary implementation of SAMGs at all plants.

The Reactor Oversight Program is a risk-informed approach to inspection that focuses on assuring compliance with those requirements that are most risk significant. Since SAMGs are not a requirement (for the reasons noted above) they are not included in the NRC baseline inspection program. SAMGs provide an improvement/enhancement to the safety margins already inherent in meeting the regulatory requirements.

As part of the NRC response to the events in Japan, the NRC staff issued a temporary instruction to address the SAMGs. Temporary Instruction 2515/184 provided instructions for NRC inspectors to determine: (i) that the severe accident management guidelines (SAMGs) are available and how they are being maintained, and (ii) the nature and extent of licensee implementation of SAMG training and exercises.

Q: *What happens when/if a plant "melts down"?*

A: In short, nuclear power plants in the United States are designed to be safe. To prevent the release of radioactive material, there are multiple barriers between the radioactive material and the environment, including the fuel cladding, the heavy steel reactor vessel itself and the containment building, usually a heavily reinforced structure of concrete and steel several feet thick.

Q: *How would the U.S. have responded to the events in Japan of March 11, 2011?*

A: The NRC requires plant designs to include multiple and diverse safety systems, and plants must test their emergency response capabilities on a regular basis. Plant operators are very capable of responding to significant events. U.S. nuclear power plants have emergency operating procedures as well as severe accident management guidelines that ensure that the containment structure integrity takes priority in an accident situation. Therefore, in an event that goes beyond those analyzed in the original plant design (i.e., beyond design basis event), such as the one at Fukushima Daiichi, U.S. BWR operators are trained to preserve primary and secondary containment by venting to provide the greatest assurance of public protection during a severe accident. Each U.S. plant has an emergency plan that is coordinated with local, State and Federal departments and agencies to ensure the safety of the public within the Emergency Planning Zone. In addition, NRC regulations require plants to have plans in place that would allow them to mitigate even worst-case scenarios. Since 9/11, we have implemented requirements for licensees to have additional response capabilities for extreme situations.

U.S. Power Plants (General)

Q: *Do U.S. nuclear plants have better capabilities to respond to natural disasters than the plants in Japan?*

A: The NRC is not yet aware of all of the differences that may exist between the reactors that are of similar design and vintage as those operated in the U.S. Many improvements have been made to U.S. boiling water reactors (BWRs). For example, NRC Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," conveyed the importance of having a robust pathway for venting primary containment, which contains the suppression pool, in certain severe accident scenarios. In response, all BWRs with Mark I containments that didn't have an existing strengthened or "hardened" pathway for venting directly from primary containment to the outside, made modifications to the plant consistent with the intent of the Generic Letter. This design feature permits a controlled depressurization of primary containment as well as a controlled release of radioactive materials and combustible hydrogen that could be generated by damaged fuel, as may occur during severe accidents. U.S. nuclear power plants are built to withstand external hazards, including earthquakes tsunamis, and flooding, as appropriate. In addition to the design of the plants, significant effort goes into emergency response planning, preparation, and training. The NRC has also completed substantial research and analysis that resulted in the development and use of severe accident management guidelines. These insights have informed our decision making and review of licensed activities.

Q: *How do we know that the spent fuel in pools at reactor sites is safe, in light of the knowledge of seismic risks?*

A: The agency continues to believe that spent fuel pools provide adequate protection of public health and safety. Over the course of many years, the NRC has taken advantage of the lessons learned from previous operating experience to implement a program of continuous improvement in the regulation of U.S. commercial nuclear reactors. This has included regular examination of topics related to spent fuel storage, as well as implementation of changes that have improved the safety of spent fuel pools. In addition, following the terrorist attacks of September 11, 2001, the NRC undertook an extensive re-examination of spent fuel pool safety and security. As a result of this reexamination, the Commission issued orders requiring licensees to implement additional strategies to keep spent fuel pools cool in the aftermath of a large explosion or fire at the plant. These requirements have since been incorporated into NRC regulations. The NRC's Japan Task Force has recommended the Commission consider additional enhancements to spent fuel pool makeup capability and instrumentation. As directed by the Commission, the NRC staff will implement any changes found to be appropriate to maintain the safety of spent fuel storage systems.

Q: *What is the worst-case scenario?*

A: In a nuclear emergency, the most important action is to ensure the core is covered with water to provide cooling to remove any heat from the fuel rods. Without adequate cooling, the fuel rods will melt. Recent reports from Japan have indicated that considerable amounts of fuel melted inside the reactor vessels of Units 1, 2, and 3, although the fuel remains inside the reactor vessel and adequate cooling water is being provided to the fuel; however damage and leakage of the reactor pressure vessels is suspected. Should the final containment structure fail, radiation from these melting fuel rods would be released to the atmosphere and additional protective measures may be necessary depending on such meteorological factors such as prevailing wind patterns and rainfall.

Spent Fuel

Q: *The waste-confidence revision seems like a long-term effort. What is the NRC doing to improve safety of spent fuel storage now?*

A: The NRC staff is currently reviewing its processes to identify near-term ways to improve efficiency and effectiveness in licensing, inspection, and enforcement. We expect to identify enhancements to the certification and licensing of storage casks, to the integration of inspection and licensing, and to our internal procedures and guidance. More information on the staff's plans can be found in COMSECY-10-0007.

Q: *How do you know the fuel pools are safe? Does the NRC inspect these facilities, or just the reactor itself?*

A: NRC inspectors are responsible for verifying that spent fuel pools and related operations are consistent with a plant's license. For example, our staff inspects spent fuel pool operations during each refueling outage. We also performed specialized inspections to verify that new spent fuel cooling capabilities and operating practices were being implemented properly.

U.S. Power Plants (General)

Q: *What would happen to a spent fuel pool during an earthquake? How can I be sure the pool wouldn't be damaged?*

A: All spent fuel pools are designed to seismic standards consistent with other important safety-related structures on the site. The pool and its supporting systems are located within structures that protect against natural phenomena and flying debris. The pools' thick walls and floors provide structural integrity and further protection of the fuel from natural phenomena and debris. In addition, the deep water above the stored fuel (typically more than 20 feet above the top of the spent fuel rods) would absorb the energy of debris that could fall into the pool. Finally, the racks that support the fuel are designed to keep the fuel in its designed configuration after a seismic event.

Q: *Can spent fuel pools leak?*

A: Spent fuel pools lined with stainless steel are designed to protect against a substantial loss of the water that cools the fuel. Pipes typically enter the pool above the level of the stored fuel, so that the fuel would stay covered even if there were a problem with one of the pipes. The only exceptions are small leakage-detection lines and, at two pressurized water reactor (PWR) sites, robust fuel transfer tubes that enter the spent fuel pool directly. The liner normally prevents water from being lost through the leak detection lines, and isolation valves or plugs are available if the liner experiences a large leak or tear.

Q: *How would you know about a leak in such a large pool of water?*

A: The spent fuel pools associated with all but one operating reactor have liner leakage collection to allow detection of very small leaks. In addition, the spent fuel pool and fuel storage area have diverse instruments to alert operators to possible large losses of water, which could be indicated by low water level, high water temperature, or high radiation levels.

Q: *How can operators get water back in the pool if there is a leak or a failure?*

A: All plants have systems available to replace water that could evaporate or leak from a spent fuel pool. Most plants have at least one system designed to be available following a design basis earthquake. In addition, the industry's experience indicates that systems not specifically designed to meet seismic criteria are likely to survive a design basis earthquake and be available to replenish water to the spent fuel pools. Furthermore, plant operators can use emergency and accident procedures that identify temporary systems to provide water to the spent fuel pool if normal systems are unavailable. In some cases, operators would need to connect hoses or install short pipes between systems. The fuel is unlikely to become uncovered rapidly because of the large water volume in the pool, the robust design of the pool structure, and the limited paths for loss of water from the pool.

Q: *Do U.S. nuclear power plants store their fuel above grade? Why is this considered safe?*

A: For boiling water reactor (BWR) Mark I and II designs, the spent fuel pool structures are located in the reactor building at an elevation several stories above the ground (about 50 to 60 feet above ground for the Mark I reactors). The spent fuel pools at other operating reactors in the U.S. are typically located with the bottom of the pool at or below plant grade level. Regardless of the location of the pool, its robust construction provides the potential for the structure to withstand events well beyond those considered in the original design. In addition, there are multiple means of restoring water to the spent fuel pools in the unlikely event that any is lost.

Q: *How are spent fuel pools kept cool? What happens if the cooling system fails?*

A: The spent fuel pool is cooled by an attached cooling system. The system keeps fuel temperatures low enough that, even if cooling were lost, operators would have substantial time to recover cooling before boiling could occur in the spent fuel pool. Licensees also have backup means to cool the spent fuel pool, using temporary equipment that would be available even after fires, explosions, or other unlikely events that could damage large portions of the facility and prevent operation of normal cooling systems. Operators have been trained to use this backup equipment, and it has been evaluated to provide adequate cooling even if the pool structure loses its water-tight integrity.

U.S. Power Plants (General)

Q: *What keeps spent fuel from re-starting a nuclear chain reaction in the pool?*

A: Spent fuel pools are designed with appropriate space between fuel assemblies and neutron-absorbing plates attached to the storage rack between each fuel assembly. Under normal conditions, these design features mean that there is substantial margin to prevent criticality (i.e., a condition where nuclear fission would become self-sustaining). Calculations demonstrate that some margin to criticality is maintained for a variety of abnormal conditions, including fuel handling accidents involving a dropped fuel assembly.

Q: *How long is spent fuel allowed to be stored in a pool or cask?*

A: NRC regulations do not specify a maximum time for storing spent fuel in pool or cask. The agency's "waste confidence decision" expresses the Commission's confidence that the fuel can be stored safely in either pool or cask for at least 60 years beyond the licensed life of any reactor without significant environmental effects. At current licensing terms (40 years of initial reactor operation plus 20 of extended operation), that would amount to at least 120 years of safe storage. However, it is important to note that this does not mean NRC "allows" or "permits" storage for that period. Dry casks are licensed or certified for 20 years, with possible renewals of up to 40 years. This shorter licensing term means the casks are reviewed and inspected, and the NRC ensures the licensee has an adequate aging management program to maintain the facility.

Q: *Does the waste confidence decision mean that a particular cask is safe?*

A: Not specifically. When the NRC issues certificates and licenses for specific dry cask storage systems, the staff makes a determination that the designs provide reasonable assurance that the waste will be stored safely for the term of the license or certificate. The Commission's Waste Confidence Decision is a generic action where the Commission found reasonable assurance that the waste from the nation's nuclear facilities can be stored safely and with minimal environmental impacts until a repository becomes available.

Q: *The NRC is reviewing applications for new nuclear power plants. What is the environmental impact of all that extra fuel?*

A: Continued use and potential growth of nuclear power is expected to increase the amount of waste in storage. This increased amount of spent fuel affects the environmental impacts to be assessed by the NRC staff, such as the need for larger storage capacities. In the staff's plan to develop an environmental impact statement for longer-term spent fuel storage, a preliminary scoping assumption is that nuclear power grows at a "medium" rate (as defined by the Department of Energy), in which nuclear power continues to supply about 20 percent of U.S. electricity production.

Q: *What about security? How do you know terrorists won't use all of this waste against us?*

A: For spent fuel, as with reactors, the NRC sets security requirements and licensees are responsible for providing the protection. We constantly remain aware of the capabilities of potential adversaries and threats to facilities, material, and activities, and we focus on physically protecting and controlling spent fuel to prevent sabotage, theft, and diversion. Some key features of these protection programs include intrusion detection, assessment of alarms, response to intrusions, and offsite assistance when necessary. Over the last 20 years, there have been no radiation releases that have affected the public. There have also been no known or suspected attempts to sabotage spent fuel casks or storage facilities. The NRC responded to the terrorist attacks on September 11, 2001, by promptly requiring security enhancements for spent fuel storage, both in spent fuel pools and dry casks.

Q: *How do we know that the spent fuel in pools at reactor sites is safe, in light of the knowledge of seismic risks?*

A: The agency continues to believe that spent fuel pools provide adequate protection of public health and safety. Over the course of many years, the NRC has taken advantage of the lessons learned from previous operating experience to implement a program of continuous improvement in the regulation of U.S. commercial nuclear reactors. This has included regular examination of topics related to spent fuel storage, as well as implementation of changes that have improved the safety of spent fuel pools. In addition, following the terrorist attacks of September 11, 2001, the NRC undertook an extensive re-examination of spent fuel pool safety and security. As a result of this reexamination, the Commission issued orders requiring licensees to implement additional strategies to keep spent fuel pools cool in the aftermath of a large explosion or fire at the plant. These requirements have since been incorporated into NRC regulations. The NRC's Japan Task Force has recommended the Commission consider additional enhancements to spent fuel pool makeup capability and instrumentation. As directed by the Commission, the NRC staff will implement any changes found to be appropriate to maintain the safety of spent fuel storage systems.

U.S. Power Plants (General)

Q: *What do you look at when you license a fuel storage facility? How do I know it can withstand a natural disaster?*

A: The NRC's requirements for both wet and dry storage can be found in Title 10 of the Code of Federal Regulations (10 CFR), including the general design criteria in Appendix A to Part 50 and the spent-fuel storage requirements in Part 72. The staff uses these rules to determine that the fuel will remain safe under anticipated operating and accident conditions. There are requirements on topics such as radiation shielding, heat removal, and criticality. In addition, the staff reviews fuel storage designs for protection against external environmental such as seismic events, tornados, and flooding, dynamic effects such as flying debris or drops from fuel handling equipment and drops of fuel storage and handling equipment, and hazards to the storage site from nearby activities.

Q: *How has the NRC evaluated nuclear waste storage containers, dry cask or otherwise, to ensure their adequate protection under the loading stresses of a severe seismic event, or other catastrophic impact incident? NEW!*

A: U.S. Nuclear Regulatory Commission (NRC) regulations in Title 10 Code of Federal Regulations Part 72 and performance standards require licensees to analyze the environmental conditions and natural phenomena surrounding each Independent Spent Fuel Storage Installation (ISFSI) to determine severe design-basis events for each site. To be certified by the NRC, the storage casks must be evaluated and shown to withstand the forces and stresses from the most severe loading conditions for each type of event.

In evaluating the adequacy of each licensee's design basis events, the NRC uses historical seismic events, nearby seismic faults and site-specific ground characteristics for each ISFSI to determine the ground motions that could affect an ISFSI. The casks are designed to maintain stability, withstand the ground motions, and safely confine and shield the spent nuclear fuel under such events. Licensees have conservatively shown, and NRC has confirmed, freestanding dry storage cask components, such as the canister and overpack, will neither tip over nor fail during such events. The dry storage casks are also analyzed for other severe natural phenomena and accidents such as cask drops and tipover, explosions, fires, floods, and tornado winds, and tornado missiles. The radiological consequences of a cask tipover accident are addressed in NRC safety evaluation reports and have been shown to be negligible.

Q: *The recent waste confidence findings say that fuel can be stored safely for 60 years beyond the reactor's licensed life. Does this mean fuel will be unsafe after 60 years? What if the spent fuel pool runs out of room before the end of a license?*

A: The NRC staff is currently developing an extended storage and transportation (EST) regulatory program. One aspect of this program is a safety and environmental analysis to support long-term (up to 300 years) storage and handling of spent fuel, as well as associated updates to the "waste confidence" rulemaking. This analysis will include an Environmental Impact Statement (EIS) on the environmental impacts of extended storage of fuel. The 300-year timeframe is appropriate for characterizing and predicting aging effects and aging management issues for EST. The staff plans to consider a variety of cask technologies, storage scenarios, handling activities, site characteristics, and aging phenomena—a complex assessment that relies on multiple supporting technical analyses. Any revisions to the waste confidence rulemaking, however, would not be an "approval" for waste to be stored longer than before—we do that through the licensing and certification of ISFSIs and casks. More information on the staff's plan can be found in SECY-11-0029

Q: *If the SFPs at U.S. plants are not in hardened structures (i.e., concrete containments) as has shown to be the case at Fukushima, why is this acceptable given the risks?*

A: The Government of Japan has issued a report including the sequence of events during the accident. This report is still under review by the NRC. However, according to previous information available to the NRC, it appears that the design of the building housing the spent fuel pools at Fukushima did not create or initiate the problems that have been observed. The NRC focuses on ensuring that cooling capability is maintained in order to prevent fuel damage. This has been accomplished at U.S. plants by redundant and/or diverse capabilities to provide forced cooling and water addition. As our understanding of the sequence of events at Fukushima improves, we will continue to assess spent fuel storage at U.S. plants.

Q: *What keeps fuel cool in dry casks?*

A: Fuel is often moved to dry cask storage after several years in spent fuel pools, so the residual heat given off by the fuel has significantly decreased. These casks are typically thick, leak-tight steel containers inside a robust steel or concrete overpack. The fuel is cooled by natural airflow around the cask.

U.S. Power Plants (General)

Q: *What is the corresponding radiological risk to that amount of fuel, should there be a fuel pool event. Is that factored into the licensee's emergency planning?*

A: The Final Safety Analysis Report (FSAR) for each plant analyzes a spectrum of accidents, including those that could occur in the spent fuel pool. These analyses are reviewed by the NRC to ensure that they demonstrate that any postulated radiological release would be below regulatory limits. These limits are selected to protect the public health and safety.

A licensee's emergency plan is symptom based to deal with any radiological hazard that could occur onsite. Regardless of the source, it is designed to ensure that appropriate protective actions are taken onsite and appropriate protective actions are recommended offsite.

Q: *How long are ISFSIs at U.S. plants good for (or "designed for")? What kind of analysis does NRC do to support extending their licenses?*

A: Utilities can apply for a site specific license under 10 CFR 72.42 or a general license under 10 CFR 72.212. The general license limits storage of spent fuel in casks that have been pre-approved for use by the NRC. In both cases the NRC's regulations provide for an initial 20-year license term for ISFSI licenses. License renewals are submitted with information consistent with the original license and the NRC staff reviews this information for continued acceptability. Site specific renewals can be requested for a time period chosen and justified by the licensee. License renewals under the general license are limited to 20 years for each renewal application.

BACKGROUND:

[The NRC issued a renewed license in December 2004 for the Surry ISFSI for a 40-year renewal term, through an exemption (ML043430234). In March 2005, NRC also granted a 40-year renewal period for the H.B. Robinson ISFSI (ML050890357).]

Q: *Are the spent fuel pools at U.S. plants cooled by safety-related cooling systems at [Plant XYZ]?*

A: Whether the spent fuel pool cooling system is "safety-related" at a particular plant depends on the plants specific accident analysis. Each plant's spent fuel pool cooling system is designed to provide cooling for both normal and accident conditions.

Q: *What amount of fuel was originally intended for spent fuel pool storage when the plants in the U.S. were initially licensed (and for how long)?*

A: The amount of fuel that can be stored in a spent fuel pool is governed by each plants' Technical Specifications. The original limit, as well as any increases to the limit are reviewed by the NRC on a plant-specific basis. The spent fuel may be stored in the pool for the duration of the license, including the time taken to decommission the plant.

BACKGROUND:

[Most spent fuel pools at U.S. nuclear power plants were not originally designed to have a storage capacity for all the spent fuel generated by their reactors. Depending upon when a plant was licensed, long-term planning for the spent fuel considered either reprocessing or shipment to a geologic repository. Since reprocessing or storage in a geologic repository are not currently an available option, nuclear power plant licensees have had to employ other options such as increasing the capacity of the spent fuel pool or an independent spent fuel storage installation. Either of these options would receive NRC review and approval prior to use.]

U.S. Power Plants (General)

Q: *Is the NRC going to make changes to spent fuel storage/safety requirements in light of the Japanese events (including possibly requiring spent fuel to be transferred to dry cask storage after a certain period of time)?*

A: The NRC continues to believe that U.S. nuclear power plants, including their spent fuel storage facilities, can and do operate safely. Following the events in Japan, the Commission directed the staff to establish a senior level task to conduct a methodical and systematic review of NRC processes and regulations to determine whether the agency should make additional improvements to its regulatory system and make recommendations to the Commission for its policy direction. The task force has completed its near-term review and has recommended that the NRC require licensees to install instrumentation in the spent fuel pools to monitor key parameters, to provide safety-related AC electrical power for the spent fuel makeup system, and to have installed a seismically qualified means to spray water into the spent fuel pools. The task force was silent on whether to accelerate spent fuel transfers to dry cask storage.

BACKGROUND:

[In Staff Requirements Memorandum (SRM-SECY 09-0090) issued in September 2010, the Commission approved revisions to the draft final rule on nuclear waste confidence and directed the staff to initiate a long-term rulemaking to address impacts of storage of spent fuel at onsite storage facilities, offsite storage facilities or both for extended periods. The Commission affirmed its confidence that spent nuclear fuel can be stored safely and securely without significant environmental impacts for at least 60 years after operation at any nuclear power plant either in the SFP or either onsite or offsite ISFSIs. Prior to the events in Japan, the staff provided a proposed plan for the long-term update to the Waste Confidence Rule (10 CFR 51.23) to the Commission in SECY-11-0029 which may be accessed at the following link: <http://www.nrc.gov/reading-rm/doc-collections/commission/secys/2011/2011-0029scy.pdf>.

Following the events in Japan, the Commission directed the staff to establish a senior level task to conduct a methodical and systematic review of NRC processes and regulations to determine whether the agency should make additional improvements to its regulatory system and make recommendations to the Commission for its policy direction. This direction is provided in tasking memorandum (COMSECY-COMGBJ-11-0002) which may be accessed at the following link: <http://www.nrc.gov/reading-rm/doc-collections/commission/comm-secy/2011/2011-0002comgbj-srm.pdf>. The task force will provide briefings to the Commission on a 30-day quick look report, a 60-day status of the ongoing near term review, and a 90-day completion of near term review. The task force has completed its near term review. Its report is available on the NRC homepage, www.nrc.gov.]

Q: *A recent report from the California Coastal Commission, dated March 24, 2011, identified the Humboldt Bay Power Plant as being susceptible to the same type of “megathrust earthquake” and resultant tsunami that occurred in Japan. Is that plant safe?*

A: The nuclear power reactor at the Humboldt Bay Power Plant is located near Eureka, California, and ceased operation in 1976. The plant is currently being decommissioned. All fuel, which was stored in the spent fuel pool since operations ceased, has been moved from the spent fuel pool into an onsite independent spent fuel storage installation (ISFSI) that was reviewed and approved by the NRC. The ISFSI provides an adequate means for safely storing the plant’s spent fuel, which has been cooling for several decades, in dry cask/storage containers.

Q: *What is spent nuclear fuel?*

A: “Spent nuclear fuel” refers to fuel elements that have been used at commercial nuclear reactors, but that are no longer capable of economically sustaining a nuclear reaction. Periodically, about one-third of the nuclear fuel in an operating reactor needs to be unloaded and replaced with fresh fuel.

Q: *Why doesn’t the NRC have up-to-date figures on how much spent fuel is stored at U.S. nuclear plants? Doesn’t the regulator have a clue about how much of this stuff is out there?*

A: The NRC and Department of Energy (NNSA) operate the Nuclear Material Management and Safeguards System (NMMSS), a database that tracks Special Nuclear Material (enriched uranium and plutonium). This database does not distinguish between fresh and irradiated material, and the information is withheld from the public for security reasons. That’s why figures on spent fuel inventory come from the industry.

U.S. Power Plants (General)

Q: *How much fuel is currently in dry cask storage?*

A: As of November 2010, there were 63 “independent spent fuel storage installations” (or ISFSIs) licensed to operate at 57 sites in 33 states. These locations are shown on a map on the NRC website at: <http://www.nrc.gov/waste/spent-fuel-storage/locations.pdf>. Over 1400 casks are stored in these independent facilities.

Q: *How much fuel is stored at decommissioned reactors? Is it in pools or casks?*

A: There are currently 10 decommissioned nuclear power reactors at 9 sites with no other nuclear operations. According to a 2008 Department of Energy report to Congress, approximately 2800 metric tons of spent fuel is stored at these nine sites. As of the writing of that report, seven of the sites had independent spent fuel storage installations, or ISFSIs. Two additional sites had approximately 1000 metric tons of spent fuel remaining in pool storage.

Q: *What is dry cask storage?*

A: Dry cask storage allows spent fuel that has already been cooled in the spent fuel pool for several years to be surrounded by inert gas inside a container called a cask. The casks are typically steel cylinders that are either welded or bolted closed. The steel cylinder provides containment of the spent fuel. Each cylinder is surrounded by additional steel, concrete, or other material to provide radiation shielding to workers and members of the public.

Q: *How do you know the dry casks are safe? Does the NRC inspect these facilities, or just the reactor and spent fuel pool?*

A: Designs for dry casks are reviewed by the NRC to ensure compliance with regulatory requirements for protection of the public and the environment. If the NRC determines that a storage cask meets the necessary requirements, the NRC will certify that storage cask for use. The NRC is also responsible for inspection of dry cask storage. Before casks are loaded, inspectors with specific knowledge of ISFSI operations assess the adequacy of a “dry run” by the licensee; they then observe all initial cask loadings. The on-site resident inspectors or region-based inspectors may observe later cask loadings, and the regional offices also perform periodic inspections of routine ISFSI operations.

Q: *What is an “ISFSI”?*

A: An independent spent fuel storage installation, or ISFSI, is a facility that is designed and constructed for the interim storage of spent nuclear fuel. These facilities are licensed separately from a nuclear power plant and are considered independent even though they may be located on the site of another NRC-licensed facility.

Q: *What kind of license is required for an ISFSI?*

A: NRC authorizes storage of spent nuclear fuel at an ISFSI in two ways: site-specific or general license. For site-specific applications, the NRC reviews the safety, environmental, physical security and financial aspects of the licensee and proposed ISFSI and, if we conclude it can operate safely, we issue a valid license. This license contains requirements on topics such as leak testing and monitoring and specifies the quantity and type of material the licensee is authorized to store at the site. A general license authorizes storage of spent fuel in casks previously approved by the NRC at a site already licensed to possess fuel or operate a nuclear power plant. Licensees must show the NRC that it is safe to store spent fuel in dry casks at their site, including analysis of earthquake intensity and tornado missiles. Licensees also review their programs (such as security or emergency planning) and make any changes needed to incorporate an ISFSI at their site. Of the currently licensed ISFSIs, 48 are operating under general licenses and 15 have specific licenses.

Q: *How much fuel is in the spent fuel pool(s) at [Plant XYZ]?*

A: The NRC does not disclose the exact amount of fuel currently stored at a plant's spent fuel storage pool. The Technical Specifications for each plant specify the maximum amount of assemblies that may be stored in the spent fuel pool. The design of the spent fuel pool is specifically reviewed by the NRC to ensure that the spent fuel can be safely stored under normal and accident conditions. Changes to the number of spent fuel assemblies that can be stored in the spent fuel pool must receive prior NRC review and approval.

U.S. Power Plants (General)

Q: *Why is spent fuel hot?*

A: Spent fuel generates what is called “residual heat” because of radioactive decay of the elements inside the fuel. After the fission reaction is stopped and the reactor is shut down, the products left over from the fuel’s time in the reactor are still radioactive and emit heat as they decay into more stable elements. Although the heat production drops rapidly at first, heat is still generated many years after shutdown. Therefore, the NRC sets requirements on the handling and storage of this fuel to ensure protection of the public and the environment.

Station Blackout

Q: *A recent newswire article claimed that 93 of the U.S. nuclear power plants only had a 4-hour coping capacity for SBO. The rest of the plants could cope for 8 hours. Is that information correct?*

A: That information is not correct. First to clarify SBO coping capacity, the definition of coping is the time required to restore off site (i.e., the grid) or onsite power (i.e., emergency diesel generator). There are two different methods for coping with an SBO event:

- Relying only on battery power (AC-independent)

- Relying on an Alternate AC power source (i.e., an emergency diesel generator, hydro-powered generator, or a gas turbine)

The NRC only allows up to a 4-hour SBO coping analysis with batteries, anything longer requires an alternate AC source. The SBO coping time for an alternate AC source ranges from 2 hours to a maximum of 16 hours. For the 104 operating plants in the U.S., the basic breakdown with respect to power source is that 44 plants are “battery coping plants” and 60 plants are “alternate AC source” plants.

With respect to SBO coping times, a further breakdown includes 44 plants that have adopted the AC-independent method and have battery power for 4 hours. Another 43 plants use the AAC methodology and can restore AC power (i.e., offsite power or emergency diesel generator) within 4 hours. Hence a total of 87 plants have 4-hour SBO coping duration.

For the remainder of U.S. plants, 14 plants use the AAC methodology and can restore AC power (i.e., offsite power or emergency diesel generator) within 8 hours; hence these 14 plants have an 8-hour SBO coping duration. There are 3 plants that use the AAC methodology and have 16-hour duration for restoration of AC power; hence the remaining 3 plants have a 16-hour SBO coping duration.

Q: *Does the SBO coping capacity take into consideration the B5b mitigating measures that were issued with NRC security orders following the events of 9/11?*

A: No. The Station Blackout (SBO) rule (10 CFR 50.63, Loss of all alternating current power) was issued as a final rule on June 21, 1988. The SBO requirements were in place well before the B5b mitigating measures in response to the 9/11 event were established so the determination of SBO coping capacity did not take the B5b measures into consideration.

U.S. Power Plants (General)

Q: *Are U.S. nuclear power plants designed for scenarios similar to what happened in Japan where all power to the reactors (i.e., both the power grid and emergency onsite power) was lost as a result of the earthquake and resultant tsunami?*

A: The NRC requires that all nuclear power plants are able to withstand a station blackout (SBO) - a complete loss of AC electric power to the station. These requirements are specified in 10 CFR 50.63, Loss of all alternating current power, and a more detailed definition is provided in 10 CFR 50.2, Definitions. The definition of coping is the time it takes until off site power is restored (i.e., the grid) or an emergency diesel generator, located either onsite or offsite, is restored to service. To meet this requirement, all nuclear power plants performed an SBO coping analysis that determined how long the plant could cope without AC power. The NRC has provided guidance for determining a plant specific SBO duration in Regulatory Guide 1.155, "Station Blackout," (August 1988). In general, SBO durations range from 2 to 16 hours, though licensees may propose alternate durations based on specific factors relating to the offsite and onsite power characteristics. There are two methods of coping with an SBO event. They are either: (i) AC independent (i.e., relying on battery power), or; (ii) alternate AC (AAC).

AC independent plants had to satisfy all the requirements for maintaining a plant in a safe condition for maximum duration of 4 hours.

If the configuration of offsite power (i.e., the grid system), onsite power (i.e., emergency diesel generators) and reliability of these sources could be affected by weather related events, and if restoration of these sources was not possible within 4 hours, then plants had to use an alternate AC source (i.e. AAC). Some plants decided to comply with the SBO rule by using the AAC as they already had that capability on their sites. Plants using an AAC source had a variable coping duration between 2 hours and 16 hours. This duration was subject to factors affecting the restoration of onsite or offsite power sources. The capability for coping with an SBO of specified duration must be determined by a coping analysis for plants with an AC independent method (i.e., batteries) and for plants with an AAC if that source is not available within 10 minutes of the initiating event.

Q: *The NRC has proposed recommendations regarding station blackout. Would NRC consider putting a hold on any changes to all plants for changes dealing with on-site power generation?*

A: The NRC requires that all nuclear power plants are able to withstand a station blackout (SBO) - a complete loss of AC electric power to the station. These requirements are specified in 10 CFR 50.63, Loss of all alternating current power, and in 10 CFR 50.2, Definitions. The NRC's Japan Task Force has recommended the agency use certain lessons learned from Fukushima to improve the NRC's regulatory framework. The Commission has noted that some Task Force's recommendations raise very complex technical and regulatory questions that will require significant analysis. Since the events in Japan continue to evolve, the NRC has used and will continue to use the analytical resources and stakeholder engagement capabilities of the agency to ensure the consideration of many issues. There has been no reduction in the licensing or oversight function of the NRC as it relates to any of the NRC licensees. Contributors to the conclusion that the current fleet of reactors and materials licensees continue to protect the public health and safety are based on a number of principles, including defense in depth. Every U.S. reactor is designed for natural events based on the specific site where the reactor is located.

Statutory Responsibility

Q: *Is the NRC involved in research and development of new electric generation technologies? Is the NRC to taking action to ensure a robust fuel mix portfolio including nuclear power?*

A: The NRC plays no role in the promotion and research of energy technologies, including nuclear power. The Energy Reorganization Act of 1974 divided the former Atomic Energy Commission into the Energy Research and Development Administration, which later became the U.S. Department of Energy (DOE) and the NRC. DOE's mission is to perform research and development activities in support of a national goal of energy independence, whereas NRC's mission is to regulate the nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, to promote the common defense and security, and to protect the environment.

U.S. Power Plants (General)

Q: *Does NRC have any research and development functions, especially in light of the events at Fukushima Daiichi Plant?*

A: The NRC's Office of Nuclear Regulatory Research (or RES), a major NRC program office, was mandated by Congress and created as part of the NRC in 1975. RES plans, recommends, and implements programs of nuclear regulatory research, standards development, and resolution of generic safety issues for nuclear power plants and other facilities regulated by the NRC. The Office coordinates research activities within and outside the agency, including NRC participation in national and international volunteer standards efforts. RES is responsible for developing methods, technical expertise and computer codes that are used by the NRC to assess safety and regulatory issues for materials licensees, fuel cycle facilities and operating reactors, as well as new and advanced reactor designs. RES develops the data needed to assess these codes by conducting experiments at national laboratories, universities, or in collaboration with international organizations.

The NRC regulatory research program addresses issues concerning nuclear reactors, nuclear materials, and radioactive waste. The research program is designed to improve the agency's knowledge where uncertainty exists, where safety margins are not well-characterized, and where regulatory decisions need to be confirmed in existing or new designs and technologies.

U.S. Response (Immediate actions at U.S. reactors)

Q: *What is the schedule for the NRC task force to complete its review of lessons learned from the events at the Fukushima Daiichi nuclear power station in Japan and to provide their recommendations to the Commission?*

A: The senior-level task force established by the NRC to review lessons learned from the events at the Fukushima Daiichi nuclear power station in Japan was directed by the Commission to provide a 30-day, 60-day, and 90-day status briefing and a final report providing its recommendations to the Commission. The task force completed its report and provided the final briefing to the Commission on July 19, 2011. Information provided at this briefing and the previous briefings, including agenda, slides, transcript, the Staff Requirements Memorandum specifying staff actions requested by the Commission, and Webcast Archive may be accessed on the NRC's public webpage at the following link: <http://www.nrc.gov/japan/japan-meeting-briefing.html>. The task force's public meeting on July 28, 2011, discussed the results of its review.

Q: *Has NRC's plan with new reactor reviews changed following tragedy in Japan? Which four reactors are expected to be under construction? NEW!*

A: The NRC's plans with regard to New Reactor Licensing Reviews have not changed. Summer, Units 2 and 3, and Vogtle, Units 3 and 4, are currently conducting "pre-construction" activities; full construction can only occur after the NRC issues a Combined License. NRC has not received any indication from industry or current applicants that there is any change regarding their plans to obtain new licenses.

Q: *Does the NRC expect that applications for reviews such as reactor extensions and new reactor licenses be affected from the Near Term Task Force review? NEW!*

A: The NRC will continue to process existing applications for new licenses and license renewal applications in accordance with the schedules that have been established. The NRC continues to believe that its regulatory framework and requirements provide for a rigorous and comprehensive license review process that examines the full extent of siting, system design and operations of nuclear power plants. The recommendations of the NRC's task force examining lessons learned from the events in Japan will be taken into account in the performance of the NRC's review of these applications, as appropriate. Further, the NRC has the necessary regulatory tools to require changes to existing licenses or applications for certification should the agency determine that changes are necessary.

Q: *Based on the information available to date about the events at Fukushima, will the NRC task force develop an event progression to inform its review and for the benefit of the U.S. public?*

A: The task force did not develop a detailed event progression for the Fukushima facility. The Japanese have provided a sequence of events in a report to International Atomic Energy Agency that can be accessed through the IAEA website: <http://www.iaea.org/newscenter/focus/fukushima/japan-report/>. The NRC is currently reviewing the report. The NRC's initial assessment is that is consistent with the agency's understanding of the events that transpired at Fukushima 1 Daiichi following the March 11 earthquake and tsunami. At this early point in the review, no immediate actions for the NRC were identified beyond the Temporary Instructions and the Bulletin that were issued. The NRC is planning to thoroughly review the report and will identify any lessons learned that may be applicable in the United States.

U.S. Power Plants (General)

Q: *Based on the information that has been learned so far about the reactors and spent fuel at Fukushima, has the NRC task force identified any technical areas or issues that it is focusing on for lessons learned or additional NRC action?*

A: It is important to note that the task force's work reinforces the NRC's confidence in the continued safety and emergency planning for U.S. nuclear power plants. The task force found that the operating nuclear power plants are protected against low likelihood severe natural phenomena and have accident mitigation capabilities such that continued operation poses no imminent risk to public health and safety. The task force also found that, over the years, the NRC has address low likelihood events on a case by case basis, thereby creating a patchwork regulatory framework. The task force recommended that the Commission establish a policy for balanced layers of defense against severe accidents, including protection, mitigation and emergency preparedness, and enhance its requirements within the new framework. The task force recommended a combination of orders, rulemakings, and further evaluation to address these issues. A long-term review, expected to begin shortly, could identify additional areas for further NRC action.

Q: *How can the public get involved in the NRC task force review process and provide its recommendations for what they should look at?*

A: The task force has completed its near-term review; unfortunately, because of the 90-day time constraint on the near-term review, the NRC task force did not have the opportunity for the kind of public participation that is typical for an agency action. As the task force transitions to the longer-term review, the Commission has asked the staff to more fully engage stakeholders to ensure the public has a voice in the process. Any task force recommendations for action approved by the Commission will follow the NRC's normal process for public involvement (e.g., proposed rulemakings to change regulations and proposed regulatory guidance will continue to include the public comment periods).

Q: *All the world's nuclear powers are reviewing the events at Fukushima. Is the NRC task force coordinating with any other countries to ensure a more consistent approach to lessons learned and potential need for additional regulations?*

A: The task force has begun some international interactions. These include task force member attendance at a Nuclear Energy Agency (NEA) Steering Committee where the Japanese made a presentation on the events at Fukushima and follow-on discussions. In addition, another task force member has chaired the Multinational Design Evaluation Program (MDEP) Steering Committee, where the Japanese made another presentation. Various NRC staff members have interacted with various Japanese delegations and have shared insights from those interactions with the task force. The task force is focusing on where those insights lead with respect to what can be learned for U.S. plants rather than focusing on the specific day-to-day events that continue at Fukushima. NRC line organizations and their staffs are monitoring and reviewing the details of those specific day-to-day events.

Q: *The German government ordered some of its nuclear power plants to shut down in response to the events in Japan. Why is it safe to continue to operate the nuclear power reactors in the U.S. that are similar to the Japanese reactors at Fukushima Dai-ichi?*

A: Every regulatory body around the world that deals with nuclear reactors has considered many factors in determining their specific response to events in Japan. The NRC is not privy to all the factors influencing the decision by the German government. The Chairman of the NRC and the Executive Director for Operations at the NRC have briefed the White House and members of Congress on the situation in Japan and the impacts on the U.S.

The NRC continues to closely monitor the activities in Japan and is reviewing all available information; the agency continues to conclude that U.S. plants are operating safely. The NRC continues its licensing and oversight functions for all NRC licensees, including nuclear power plants. Information in a number of areas, including the principle of defense in depth, leads to the conclusion that the current fleet of reactors and materials licensees continue to protect the public health and safety.

Every reactor in the country is designed for severe natural events at its site. Every reactor has a wide range of diverse and redundant safety features as well as multiple physical barriers to contain radioactive material, in order to provide that public health and safety assurance. The NRC has a long regulatory history of conservative decision making. The NRC has been intelligently using risk insights to help inform the regulatory process and has required improvements to the plant designs as we learn from operating experience. Some of these include severe accident management guidelines, revisions to the emergency operating procedures, procedures and processes for dealing with large fires and explosions regardless of the cause, and requirements for coping with station blackout.

U.S. Power Plants (General)

Q: *What short-term and long-term actions to ensure the safety of the U.S. operating nuclear power plants is the NRC taking in response to the events at the Japanese nuclear power plants at Fukushima Daiichi?*

A: Shortly after the March 11, 2011, earthquake and tsunami and the resulting crisis at the Japanese Fukushima Daiichi nuclear power plant, the NRC launched a two-pronged review of U.S. nuclear power plant safety. The Near-Term Task Force was established in response to Commission direction to conduct a systematic and methodical review of NRC processes and regulations to determine whether the agency should make additional improvements to its regulatory system and to make recommendations to the Commission for its policy direction. The Near-Term Task Force, made up of senior managers and staff with relevant experience, conducted a short-term review of the lessons that can be learned from the situation in Japan, and issued its report on July 12, 2011. NRC inspectors who are posted at every U.S. nuclear power plant supported the task force's short-term effort, supplemented as necessary by experts from the agency's regional and headquarters offices. The task force's report is available to the public (ADAMS Accession No. ML111861807). On July 19, 2011, the Task Force presented its findings to the Commission and proposed improvements in areas ranging from loss of power to earthquakes, flooding, spent fuel pools, venting and emergency preparedness. The Task Force discussed its report and recommendations with the public on July 28, 2011, in NRC headquarters in Rockville, Maryland.

The NRC expects to shortly establish a long-term review under the oversight of a steering committee of senior agency executives. As new information about the Fukushima accident becomes available, it will be considered and may form the basis for additional actions arising from the long-term review. The longer-term review is expected to include more public meetings and more stakeholder involvement. The Task Force remained aware of concerns being raised by the public during the near-term review, but the relatively short duration of their assignment prevented a systematic engagement with the public. Also, the lessons learned will be evaluated for applicability to non-power reactor NRC licensed facilities.

Q: *What action is the NRC taking regarding licensee plans to walk down their plants to confirm systems, procedures, etc., are in place to deal with natural phenomena? Are the resident inspectors going to accompany the licensees during the walkdowns?*

A: The NRC issued Information Notice 2011-05, "Tohoku-Taiheiyou-Oki Earthquake Effects on Japanese Nuclear Power Plants", on March 18, 2011, to all holders of or applicants for operating licenses for nuclear power plants. The issuance of Information Notice (IN 2011-05) was also the subject of an NRC Press Release on March 18, 2011. Both documents are available on the NRC's public webpage at the following link: <http://www.nrc.gov/reading-rm/doc-collections/news/2011/11-052.pdf>. The IN provided a summary discussion of actions that the U.S. nuclear power industry has begun taking at each licensed reactor site to confirm systems, procedures, staged equipment, etc. are in place to deal with natural phenomena. On March 23, 2011, the NRC issued Temporary Instruction 2515/183, "Followup to the Fukushima Daiichi Nuclear Station Fuel Damage Event", which instructs NRC inspectors to independently assess the adequacy of the actions taken by licensees in response to the Fukushima Daiichi event and to coordinate their inspection efforts with the licensees schedule for verification of plant capabilities. TI 2515/183 is available on the NRC's public webpage at the following link: <http://pbadupws.nrc.gov/docs/ML1107/ML11077A007.pdf>.

Q: *How will the U.S. learn from the failures at the Japanese reactors?*

A: The NRC has established a senior level task force to analyze the events in Japan and develop lessons learned and recommendations to improve plant safety, as appropriate. The task force issued its near-term report ("Recommendations for the Enhancing Reactor Safety in the 21st Century: The Near-term Task Force Review of Insights from the Fukushima Daiichi Accident," July 12, 2011, Nuclear Regulatory Commission) and concluded that continued operation and continued licensing activities do not pose an imminent risk to public health and safety. The task force recommended rulemaking activities, orders, certain staff actions, and actions for long-term evaluation. The Commission will review the report and will provide the staff with direction. A long-term evaluation is planned and will assess whether any additional licensing actions are necessary. These actions may include Orders, information requests in accordance with Section 50.54(f) of Title 10 (10 CFR) of the Code of Federal Regulations, license amendments, rulemaking, etc.

The NRC issued an information notice to inform licensees about the effects of the earthquake on nuclear power plants in Japan. In addition, the NRC's staffs at every reactor site have performed targeted inspections to confirm facility responses to beyond design-basis events. The NRC has also issued Bulletin 2011-01 that requires all licensees to verify under oath and affirmation that their mitigation strategies and capabilities are in compliance with relevant NRC regulations.

U.S. Power Plants (General)

Q: *What is the NRC doing to correct misinformation in the public/media?*

A: The NRC Office of Public Affairs is working closely with stakeholders to disseminate information via press releases to keep the public informed. The NRC and Regional Offices are also working closely with State and Local officials to maintain an open dialogue on the events and safety of the NRC licensed facilities. The NRC has recently established a link on its public internet website <http://www.nrc.gov/japan/japan-info.html> that provides information and links to other sources of related information.

Q: *What has the NRC task force learned from the targeted inspections performed at U.S. nuclear power plants in response to the events at Fukushima?*

A: The task force has completed its near-term review and concluded that continued operation and continued licensing activities do not pose an imminent risk to public health and safety. The task force made several recommendations in its report, such as evaluating seismic and flooding protection at plants, ensuring that existing mitigation equipment is stored in areas that are protected from severe flooding, and strengthening emergency plan staffing and facilities to address multi-unit events.

Based on the targeted inspections performed in response to Temporary Instructions 2515/183, none of the observations made by the NRC inspectors posed a significant safety issue. Temporary Instruction 2515/183 provided instructions for NRC inspectors to perform independent assessments of the adequacy of industry-initiated efforts to respond to the fuel damage events at the Fukushima Daiichi nuclear station. This involves a high-level look at industry's preparedness for events that may exceed the design for a plant. In summary, observations were made that there were discrepancies in terms of procedures, equipment, and training. The detailed inspection reports for these inspections are available at the NRC's public webpage at the following link: <http://www.nrc.gov/japan/japan-activities.html>.

U.S. Response (Long-term actions at U.S. reactors)

Q: *What is the schedule for the NRC task force to complete its review of lessons learned from the events at the Fukushima Daiichi nuclear power station in Japan and to provide their recommendations to the Commission?*

A: The senior-level task force established by the NRC to review lessons learned from the events at the Fukushima Daiichi nuclear power station in Japan was directed by the Commission to provide a 30-day, 60-day, and 90-day status briefing and a final report providing its recommendations to the Commission. The task force completed its report and provided the final briefing to the Commission on July 19, 2011. Information provided at this briefing and the previous briefings, including agenda, slides, transcript, the Staff Requirements Memorandum specifying staff actions requested by the Commission, and Webcast Archive may be accessed on the NRC's public webpage at the following link: <http://www.nrc.gov/japan/japan-meeting-briefing.html>. The task force's public meeting on July 28, 2011, discussed the results of its review.

Q: *The NRC has proposed recommendations regarding station blackout. Would NRC consider putting a hold on any changes to all plants for changes dealing with on-site power generation?*

A: The NRC requires that all nuclear power plants are able to withstand a station blackout (SBO) - a complete loss of AC electric power to the station. These requirements are specified in 10 CFR 50.63, Loss of all alternating current power, and in 10 CFR 50.2, Definitions. The NRC's Japan Task Force has recommended the agency use certain lessons learned from Fukushima to improve the NRC's regulatory framework. The Commission has noted that some Task Force's recommendations raise very complex technical and regulatory questions that will require significant analysis. Since the events in Japan continue to evolve, the NRC has used and will continue to use the analytical resources and stakeholder engagement capabilities of the agency to ensure the consideration of many issues. There has been no reduction in the licensing or oversight function of the NRC as it relates to any of the NRC licensees. Contributors to the conclusion that the current fleet of reactors and materials licensees continue to protect the public health and safety are based on a number of principles, including defense in depth. Every U.S. reactor is designed for natural events based on the specific site where the reactor is located.

U.S. Power Plants (General)

Q: *Based on the information available to date about the events at Fukushima, will the NRC task force develop an event progression to inform its review and for the benefit of the U.S. public?*

A: The task force did not develop a detailed event progression for the Fukushima facility. The Japanese have provided a sequence of events in a report to International Atomic Energy Agency that can be accessed through the IAEA website: <http://www.iaea.org/newscenter/focus/fukushima/japan-report/>. The NRC is currently reviewing the report. The NRC's initial assessment is that is consistent with the agency's understanding of the events that transpired at Fukushima 1 Daiichi following the March 11 earthquake and tsunami. At this early point in the review, no immediate actions for the NRC were identified beyond the Temporary Instructions and the Bulletin that were issued. The NRC is planning to thoroughly review the report and will identify any lessons learned that may be applicable in the United States.

Q: *How will the U.S. learn from the failures at the Japanese reactors?*

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U.S. Power Plants (Plant-specific)

Chernobyl

Q: *Is the event in Japan worse than TMI and Chernobyl?*

A: Initially, the events in Japan were classified as Level 3, "Serious Incidents," and were reclassified as Level 5, "Accidents with Wider Consequences," on the International Nuclear Events Scale (INES). In comparison, TMI was a Level 5 event and Chernobyl was a Level 7 event.

On April 12, 2011, the Japanese Nuclear and Industrial Safety Agency (NISA) government raised the rating for the events at the Fukushima Daiichi site on the International Nuclear and Radiological Event Scale (INES) from 5, "Accident with Wider Consequences," to 7, "Major Accident," citing calculations by both NISA and the Nuclear Safety Commission of Japan (NSC) of radioactive materials released from the Fukushima Daiichi reactors. This new provisional rating considers the accidents that occurred at Units 1, 2, and 3 as a single event on INES. NISA notes that while an INES rating of 7 is the same as that of the Chernobyl accident, their current estimated amount of radioactive materials released is approximately 10% of the amount from the Chernobyl accident.

Q: *If Chernobyl was a 7 and Three Mile Island was a 5, when does this event move from the 4 level?*

A: The International Atomic Energy Agency (IAEA) rates nuclear events in accordance with its International Nuclear and Radiological Event Scale (INES). IAEA initially assigned the events in Japan an INES rating of 4, "Accident with Local Consequences." This rating is subject to change as events unfold and additional information becomes available. INES classifies nuclear accidents based on the radiological effects on people and the environment and the status of barriers to the release of radiation. IAEA determinations regarding the INES rating of events are made independently.

Three Mile Island was assigned an INES rating of 5, "Accident with Wider Consequences," due to the severe damage to the reactor core.

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Diablo Canyon

Q: *Why should the NRC not require the more sophisticated (3D) seismic studies being voluntarily conducted by licensees in California?*

A: Current NRC and American Nuclear Society (ANS) documentation provides guidance related to site investigations undertaken for the purpose of characterizing seismic sources and dynamic site properties. A variety of geophysical and geotechnical tools are available that can be used to investigate the earth from both a site-specific and a regional level. Each of these methods provides specific information by probing the earth in a different way. While some tools are universally useful, others are better suited to certain types of subsurface materials and tectonic situations. While 3D seismic studies, such as those being performed in California, are sophisticated, they are not useful for all situations and the very large expense of the study could preclude broader application of techniques better suited to a specific site. The NRC would suggest the use of 3D seismic studies only in cases where it could be useful. The NRC attempts to provide regulations that call for techniques that would be the most suitable given the specific conditions of a plant and requested licensing actions.

Q: *How will the NRC consider and utilize the new information on seismic risks in the licensing proceedings, such as those pertaining to nuclear plants in California including San Onofre and Diablo Canyon? NEW!*

A: The scope of the Generic Issue 199 (GI-199) Safety/Risk Assessment Report is limited to plants in the Central and Eastern United States, and while Western plants such as Diablo Canyon and San Onofre sites are not included in the GI-199 Safety/Risk Assessment Report, the Information Notice on GI-199 is addressed to all operating power plants in the U.S. (as well as all independent spent fuel storage installation licensees). The staff plans to consider inclusion of operating reactors in the Western U.S. in its future generic communication information requests.

U.S. Power Plants (Plant-specific)

Indian Point

Q: *Why is Indian Point safe if there is a fault line underneath it?*

A: The Ramapo fault system, located near the Indian Point Nuclear Power Plant, is an example of an old fault system that, based on geologic field evidence, has not been active in the last 65.5 million years. The Ramapo fault system extends primarily from southeastern New York to northern New Jersey and is made up of a series of northeast-oriented faults. Even though there is minor earthquake activity in the vicinity of the Ramapo faults, this earthquake activity cannot be directly correlated with any individual fault within the Ramapo fault system. U.S. nuclear power plants are designed and built to withstand the largest expected earthquake in the site region, based on observed historical seismicity and field evidence for prehistoric earthquakes, and are also designed to incorporate seismic safety margins. A potential earthquake in and around the vicinity of the Ramapo fault system was taken into account during the NRC licensing process for the Indian Point plants, and the plant design incorporated the largest expected earthquake in the site region. In summary, the Ramapo fault system exhibits no definitive evidence for recent fault displacement (i.e., no evidence for fault activity in the last 65.5 million years) and the Indian Point nuclear power plant was designed and built to safely shutdown in the event of an earthquake having the highest magnitude observed in the geologic record near the site.

Pilgrim

Q: *What is the seismic limit that Pilgrim Station, Seabrook Station and Vermont Yankee have been built to withstand?*

A: Each plant is designed to a ground-shaking level that is appropriate for its location, given the possible earthquake sources that may affect the site and its tectonic environment. Ground shaking is a function of both the magnitude of the earthquake and the distance from the fault plane to the site. The seismic responses of the structures, systems, and components associated with these facilities are dependent on several factors, as mentioned above; therefore, the responses may be different for the same magnitude earthquake. As a result, the NRC regulatory requirements focus on seismic limits based on ground shaking rather than limits defined by earthquake magnitude.

The ground motions associated with seismic events are determined for two categories of earthquakes: the Safe Shutdown Earthquake (SSE) which is generally defined as the maximum ground motion seismic response that the plant must be able to withstand and safely shut down and be maintained in a safely shut down condition, and; the Operating Basis Earthquake (OBE) which is defined as the ground motion seismic response that the plant must be able to withstand and to continue operating normally following such an event. The SSE and OBE reflect the horizontal acceleration of the ground in units of the earth's gravity, 'g'. The ground motions to which the Pilgrim, Seabrook, and Vermont Yankee plants are designed are: Pilgrim SSE of 0.150g and OBE of 0.080g; Seabrook SSE of 0.250g and OBE of 0.125g, and Vermont Yankee SSE of 0.140g and OBE of 0.070g.

Q: *For Pilgrim Station and Seabrook Station, what design and safety precautions have been installed at your plant to sustain a devastating tsunami that would hit as did the tragedy at the Japanese plants?*

A: All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate. Even those plants that are located in areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. The NRC requires that safety-significant structures, systems, and components be designed to take into account even very rare and extreme seismic and tsunami events. The Pilgrim and Seabrook Stations are designed to withstand the maximum credible natural events predicted for their specific sites. In addition to the design of the plants, significant effort goes into emergency response planning, preparation, and training. The NRC has also completed substantial research and analysis that resulted in the development and use of severe accident management guidelines. These insights have informed our decision making and review of licensed activities.

Q: *Please explain the outcome at each plant (Pilgrim Station, Seabrook Station and Vermont Yankee) if it was hit with a 8.9 earthquake (i.e., the same as what hit Japan)?*

A: Each plant is designed to a ground-shaking level that is appropriate for its location, given the possible earthquake sources that may affect the site and its tectonic environment. Ground shaking is a function of both the magnitude of the earthquake and the distance from the fault plane to the site. The seismic hazards associated with the earthquake in Japan cannot be duplicated by the geology of New England, where the Pilgrim, Seabrook, and Vermont Yankee facilities are located, so this makes a postulated comparison of facility responses to the same seismic event even less likely.

U.S. Power Plants (Plant-specific)

Q: *If the same tragedy hit Pilgrim Station, Seabrook Station and Vermont Yankee would we be having the same major issues that the Japanese plants have? Please explain yes or no.*

A: The circumstances related to the events in Japan are highly unlikely in that the plant-specific external hazards profile is substantially different. All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate. Even those plants that are located in areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. The NRC requires that safety-significant structures, systems, and components be designed to take into account even very rare and extreme seismic and tsunami events. Pilgrim, Seabrook, and Vermont Yankee stations are designed to withstand the maximum credible natural events predicted for their specific sites. In addition to the design of the plants, significant effort goes into emergency response planning, preparation, and training. The NRC has also completed substantial research and analysis that resulted in the development and use of severe accident management guidelines. These insights have informed our decision making and review of licensed activities.

San Onofre

Q: *Do the Japan events of March 2011 mean that there should be more concerns about seismic risks at San Onofre Generating Station (SONGS)? NEW!*

A: U.S. nuclear plant designs consider seismic events and tsunamis. It is important not to extrapolate earthquake and tsunami data from one location of the world to another when evaluating these natural hazards. These catastrophic natural events are location specific, based on the locations of tectonic and geological fault lines. The March 2011 Japan earthquake occurred on a subduction zone, which is a very different type of tectonic environment than the region around SONGS, which is predominantly strike slip. A magnitude 9 earthquake can only occur on a subduction zone and cannot occur in the region around SONGS.

Q: *How will the NRC consider and utilize the new information on seismic risks in the licensing proceedings, such as those pertaining to nuclear plants in California including San Onofre and Diablo Canyon? NEW!*

A: The scope of the Generic Issue 199 (GI-199) Safety/Risk Assessment Report is limited to plants in the Central and Eastern United States, and while Western plants such as Diablo Canyon and San Onofre sites are not included in the GI-199 Safety/Risk Assessment Report, the Information Notice on GI-199 is addressed to all operating power plants in the U.S. (as well as all independent spent fuel storage installation licensees). The staff plans to consider inclusion of operating reactors in the Western U.S. in its future generic communication information requests.

Seabrook

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U.S. Power Plants (Plant-specific)

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A: The circumstances related to the events in Japan are highly unlikely in that the plant-specific external hazards profile is substantially different. All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate. Even those plants that are located in areas with low and moderate seismic activity are designed for safety in the event of such a natural disaster. The NRC requires that safety-significant structures, systems, and components be designed to take into account even very rare and extreme seismic and tsunami events. Pilgrim, Seabrook, and Vermont Yankee stations are designed to withstand the maximum credible natural events predicted for their specific sites. In addition to the design of the plants, significant effort goes into emergency response planning, preparation, and training. The NRC has also completed substantial research and analysis that resulted in the development and use of severe accident management guidelines. These insights have informed our decision making and review of licensed activities.

Q: *Please explain the outcome at each plant (Pilgrim Station, Seabrook Station and Vermont Yankee) if it was hit with a 8.9 earthquake (i.e., the same as what hit Japan)?*

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The ground motions associated with seismic events are determined for two categories of earthquakes: the Safe Shutdown Earthquake (SSE) which is generally defined as the maximum ground motion seismic response that the plant must be able to withstand and safely shut down and be maintained in a safely shut down condition, and; the Operating Basis Earthquake (OBE) which is defined as the ground motion seismic response that the plant must be able to withstand and to continue operating normally following such an event. The SSE and OBE reflect the horizontal acceleration of the ground in units of the earth's gravity, 'g'. The ground motions to which the Pilgrim, Seabrook, and Vermont Yankee plants are designed are: Pilgrim SSE of 0.150g and OBE of 0.080g; Seabrook SSE of 0.250g and OBE of 0.125g, and Vermont Yankee SSE of 0.140g and OBE of 0.070g.

Three Mile Island

Q: *Is the event in Japan worse than TMI and Chernobyl?*

A: Initially, the events in Japan were classified as Level 3, "Serious Incidents," and were reclassified as Level 5, "Accidents with Wider Consequences," on the International Nuclear Events Scale (INES). In comparison, TMI was a Level 5 event and Chernobyl was a Level 7 event.

On April 12, 2011, the Japanese Nuclear and Industrial Safety Agency (NISA) government raised the rating for the events at the Fukushima Daiichi site on the International Nuclear and Radiological Event Scale (INES) from 5, "Accident with Wider Consequences," to 7, "Major Accident," citing calculations by both NISA and the Nuclear Safety Commission of Japan (NSC) of radioactive materials released from the Fukushima Daiichi reactors. This new provisional rating considers the accidents that occurred at Units 1, 2, and 3 as a single event on INES. NISA notes that while an INES rating of 7 is the same as that of the Chernobyl accident, their current estimated amount of radioactive materials released is approximately 10% of the amount from the Chernobyl accident.

U.S. Power Plants (Plant-specific)

Q: *Compare this incident to the Three Mile Island. What are the similarities?*

A: The events at Three Mile Island (TMI) in 1979 were the result of an equipment malfunction that resulted in the loss of cooling water to the reactor fuel. Subsequent operator actions compounded the malfunction ultimately resulting in the partial core meltdown. The events in Japan appear to be the result of an earthquake and subsequent tsunami that knocked out electrical power to emergency safety systems designed to cool the reactor fuel. TEPCO (owner/operator of the Fukushima Daiichi facility) estimates that considerable melting of the reactor cores occurred in three of the six units at this facility. However, the core material in these units is now being cooled with adequate amounts of water. In comparison, only one of the two units at TMI experienced partial core melting in 1979. In both events the final safety barrier, the containment building, remained largely intact and contained the majority of the radioactivity preventing its release to the environment. There appears to have been more radiation released from the Fukushima facility than from TMI and, as a result, the International Atomic Energy Agency on April 12, 2011, raised its rating of the Fukushima accident on the International Nuclear and Radiological Event Scale (INES) from 5, "Accident with Wider Consequences", which was the TMI rating, to 7, "Major Accident." The conditions at the Fukushima Daiichi nuclear facility in Japan continue to be monitored and assessed and actions to mitigate and prevent further releases of radiation to the environment are being actively employed by TEPCO. The websites for TEPCO (<http://www.tepco.co.jp/en/index-e.html>) and JAIF (<http://www.jaif.or.jp/english>) provide additional information on a daily basis.

Q: *If Chernobyl was a 7 and Three Mile Island was a 5, when does this event move from the 4 level?*

A: The International Atomic Energy Agency (IAEA) rates nuclear events in accordance with its International Nuclear and Radiological Event Scale (INES). IAEA initially assigned the events in Japan an INES rating of 4, "Accident with Local Consequences." This rating is subject to change as events unfold and additional information becomes available. INES classifies nuclear accidents based on the radiological effects on people and the environment and the status of barriers to the release of radiation. IAEA determinations regarding the INES rating of events are made independently.

Three Mile Island was assigned an INES rating of 5, "Accident with Wider Consequences," due to the severe damage to the reactor core.

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U.S. Power Plants (Plant-specific)

U.S. BWR Mark I Plants

Q: *Some in the media and in Hill briefings are suggesting the BWR Mark I containment is flawed. What are the concerns about this type of containment? Are the US plants with this safe?*

A: BWR Mark I containments have relatively small volumes in comparison with pressurized water reactor (PWR) containments. This makes the BWR Mark I containment relatively more susceptible to containment failure given a core meltdown severe enough to (1) fail the reactor vessel and also (2) severe enough so that the core melt reaches the containment boundary. On the positive side, BWRs have more ways of adding water to the core than PWRs to prevent core meltdown. The following improvements have been made to U.S. Mark I containment reactors:

Station Blackout (SBO) Rule: Required the ability to cope with SBO for specified time and recover the plant

Anticipated Transient Without Scram (ATWS) Rule: Required vendor specific improvements to enhance scram reliability

Hydrogen Control Rule: Required modifications to reduce impact of hydrogen generated from beyond design basis events (DBEs)

Equipment Qualification Rule: Required environmental qualification of electrical system equipment used for design basis accidents (DBAs)

Mark I Containment Improvement Program: (i) Added hardened vent system for containment cooling and fission product scrubbing for beyond DBAs, and (ii) Enhanced reliability of automatic depressurization system (ADS) and added an additional water injection capability independent of normal AC and emergency diesel power

Symptom-based Emergency Procedure Guides (EPGs): Provides emergency procedures that direct operator actions on the basis of critical safety parameter status rather than knowledge of the event initiator – applicable to any initiating event (DBA or beyond DBA)

Severe Accident Management Guidelines (SAMGs): Guidelines for minimizing radiological consequences of a damaged core event. Focuses on maintaining containment integrity, controlling releases, and emergency planning interface

Aircraft Impact Requirements: Requires procedures to use all available equipment for core cooling, containment protection, and spent fuel pool cooling assuming a significant damage to the facility from an airplane crash

Mark I Containment Hydrodynamic Load Issue Resolution: Resulted in structural strengthening of Mark I containments to better handle reactor system depressurization forces

Emergency Core Cooling System (ECCS) Pump Suction Strainer Improvements: Larger surface area strainers installed with higher debris loading tolerance to ensure ECCS pump operation

Hydrogen explosions have been a major aspect of the Fukushima accident. In the U.S., NRC Generic Letter 89-16, "Installation of a Hardened Wetwell Vent," conveyed the importance of having a robust pathway for venting primary containment, which contains the suppression pool, in certain severe accident scenarios. In response, all BWRs with Mark I containments that didn't have an existing strengthened or "hardened" pathway for venting directly from primary containment to the outside, made modifications to the plant consistent with the intent of the Generic Letter. This design feature permits a controlled depressurization of primary containment as well as a controlled release of radioactive materials and combustible hydrogen generated by damaged fuel, as may occur during severe accidents.

U.S. Power Plants (Plant-specific)

U.S. Coastal Plants

Q: *Are U.S. nuclear power plants designed to withstand tsunamis? What would the effect be on [plant X] if a subsequent tsunami hit?*

A: All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate. Many nuclear plants are located in coastal areas that could potentially be affected by a tsunami resulting from an earthquake. Two nuclear plants, Diablo Canyon and San Onofre, are on the Pacific Coast, which is known to have a tsunami hazard. There are many nuclear plants on the Atlantic Coast or on rivers that may be affected by a tidal bore resulting from a tsunami. These include St. Lucie, Turkey Point, Brunswick, Oyster Creek, Millstone, Pilgrim, Seabrook, Calvert Cliffs, Salem/Hope Creek, and Surry. In addition, there are two nuclear plants on the Gulf Coast, South Texas and Crystal River, that could potentially be affected by tsunami. Although tsunami on the Gulf and Atlantic Coasts may occur, it is very rare. Generally the flooding anticipated from hurricane storm surge exceeds the flooding expected from a tsunami for nuclear plants on these coasts.

Recent studies have looked at the potential of tsunami hitting the Gulf and Atlantic coasts, and have found that for many parts of the coast, tsunamigenic landslide (i.e., tsunami resulting from an underwater landslide) have the potential to exceed the seismically-induced tsunami. This research shows that the tsunamis produced by underwater landslides are localized, but can be extremely destructive in the nearby areas. The licensing basis for the coastal plants (i.e., FSARs) mentioned above did not specifically consider or assess this possibility, as the phenomenon was not well understood at the time. However, research supported by the NRC has been studying the issue since 2006. Although studies of tsunamigenic landslide continue, the current results indicated that flooding anticipated from hurricane storm surge, evaluated as part of the licensing basis for these plants, generally exceeds the flooding expected from a tsunami for nuclear plants on these coasts.

Q: *How many reactors are along coastal areas that could be affected by a tsunami? Is plant X designed to withstand a tsunami (for each coastal plant)?*

A: All U.S. nuclear power plants are built to withstand external hazards, including earthquakes, flooding, and tsunamis, as appropriate. Many nuclear plants are located in coastal areas that could potentially be affected by a tsunami resulting from an earthquake. Two nuclear plants, Diablo Canyon and San Onofre, are on the Pacific Coast, which is known to have a tsunami hazard. There are many nuclear plants on the Atlantic Coast or on rivers that may be affected by a tidal bore resulting from a tsunami. These include St. Lucie, Turkey Point, Brunswick, Oyster Creek, Millstone, Pilgrim, Seabrook, Calvert Cliffs, Salem/Hope Creek, and Surry. In addition, there are two nuclear plants on the Gulf Coast, South Texas and Crystal River, that could potentially be affected by tsunami. Although tsunami on the Gulf and Atlantic Coasts may occur, it is very rare. Generally the flooding anticipated from hurricane storm surge exceeds the flooding expected from a tsunami for nuclear plants on these coasts.

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U.S. Power Plants (Plant-specific)

Q: *Why should the NRC not require the more sophisticated (3D) seismic studies being voluntarily conducted by licensees in California?*

A: Current NRC and American Nuclear Society (ANS) documentation provides guidance related to site investigations undertaken for the purpose of characterizing seismic sources and dynamic site properties. A variety of geophysical and geotechnical tools are available that can be used to investigate the earth from both a site-specific and a regional level. Each of these methods provides specific information by probing the earth in a different way. While some tools are universally useful, others are better suited to certain types of sub-surface materials and tectonic situations. While 3D seismic studies, such as those being performed in California, are sophisticated, they are not useful for all situations and the very large expense of the study could preclude broader application of techniques better suited to a specific site. The NRC would suggest the use of 3D seismic studies only in cases where it could be useful. The NRC attempts to provide regulations that call for techniques that would be the most suitable given the specific conditions of a plant and requested licensing actions.

Vermont Yankee

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