Responses to Questions from Congressman Edward J. Markey
Letter of April 15, 2011

1. Who at the Commission made the decisions to a) initially direct its inspectors to limit the scope of the inspections to Design Basis Events and b) subsequently direct its inspectors not to record findings or observations of any beyond Design Basis Events in a manner that would result in the public disclosure of any identified vulnerabilities? Please provide me with a copy of all documents (including reports, emails, correspondence, memos, phone or meeting minutes or other materials) related to both the decisions regarding the scope of the inspections as well as the manner in which inspection findings and observations would be recorded and reported.

On March 23, 2011, the NRC issued inspection requirements and guidance to its inspectors in NRC Inspection Manual Temporary Instruction (TI) 2515/183, “Follow up to the Fukushima Daiichi Nuclear Station Fuel Damage Event” (Enclosure 2). The intent of the TI was to provide the NRC with a high-level look at the industry’s preparedness in the following areas including beyond design basis events. The scope of the inspection included, but was not limited to four primary areas of investigation for NRC inspectors at NRC-licensed operating nuclear power plants:

1) Assessing a licensee’s capability to mitigate conditions that result from beyond design basis events, typically bounded by security threats, committed to as part of NRC Security Order Section B.5.b and severe accident management guidelines;

2) Assessing a licensee’s capability to mitigate station blackout conditions;

3) Assessing a licensee’s capability to mitigate internal and external flooding events required by station design; and

4) Assessing the thoroughness of a licensee’s walk downs and inspections of important equipment needed to mitigate fire and flood events to identify the potential that the equipment’s function could be lost during seismic events possible for the site.

In addition to the TI, Manual Chapter 0612, “Power Reactor Inspection Reports” provides the NRC’s guidance to inspectors for documenting this inspection, as well as all routine inspections. The Manual Chapter is also used by the NRC staff to determine if issues warrant documentation in inspection reports and to ensure inspection reports clearly communicate significant inspection results in a consistent manner to licensees, NRC staff, and the public. To ensure consistency in the reports, we issued a draft template (Enclosure 3) to document all of the results of the inspections. These inspections have now been completed and the inspection reports were made publicly available on May 13, 2011. The Temporary Instruction, the Manual Chapter and the template comprise the documents regarding the scope of the inspections as well as the manner in which inspection findings and observations would be recorded and reported. Additionally, copies of emails relating to the Temporary Instruction are enclosed (Enclosure 4) and are marked “Not For Public Disclosure.”

The inspections performed as a result of this TI represent only a first step in our follow-up to events in Japan, and are intended to provide only a high-level “check” of licensee preparedness. As noted in the TI, if necessary, a more specific follow-up inspection will be performed at a later date. Beyond these initial inspections, the NRC task force established following events in Japan

will be examining the results of the TI inspections and other information in the near term to develop recommendations, as appropriate, for potential changes to NRC's regulatory requirements, programs, processes, and other actions as needed. The task force also will recommend a framework for a longer-term review.

On April 29, 2001, the NRC issued TI 2515/184, "Availability and Readiness Inspection of Severe Accident Management Guidelines (SAMGs)" (Enclosure 5). The TI assesses whether licensees have the SAMGs maintained and available for use, and plant staff are trained on their use. The TI was completed on May 27, 2011.

On May 11, 2011, the NRC issued Bulletin 2011-01, "Mitigating Strategies" (Enclosure 6). The Bulletin requires licensees to confirm, by June 10, 2011, that mitigating strategy equipment is in place and available, as well as that the strategies can be carried out with current plant staffing. The bulletin also requires licensees to provide by July 11, 2011, additional information regarding licensee compliance with requirements for mitigating strategies programs. The NRC will use this information to determine if 1) additional assessment of program implementation is needed, 2) the current inspection program should be enhanced, or 3) further regulatory action is warranted.

Please be assured that, should information at any time indicate that there is a basis to question the continued safe operation of NRC-licensed facilities, the NRC will take appropriate action as part of our ongoing safety oversight.

2. Will you immediately reverse the current direction to NRC inspectors to keep all findings and observations of vulnerabilities of U.S. reactors to beyond Design Basis events secret and excluded from all public reports on the Commission's Fukushima review? If not, why not?

All findings of the TI inspections have been documented in inspection reports following the template provided. Those inspection reports are publicly available on the NRC's web site.

3. The NRC review is supposed to evaluate the currently available information from the events that occurred in Japan to identify changes that might be needed at U.S. nuclear power plants of all designs. For each of the following events that are known to have occurred in Japan, please indicate a) whether the event in question is considered to be a "design basis event" by the NRC, b) whether NRC inspectors will be required to evaluate whether the U.S. nuclear power plants they are inspecting are capable of preventing or mitigating such an event, c) if not, why not, since the Commission clearly stated that all such events were supposed to be analyzed, d) if not, how regulatory or other recommendations will be developed that ensure that U.S. nuclear power plants are capable of preventing or mitigating such an event, e) whether the findings and observations associated with the inspections designed to evaluate U.S. ability to prevent or mitigate such an event will be made public as part of the NRC's 30, 60 and 90 day reports (and if not, why not), and f) whether the NRC intends to address U.S. vulnerability to the event at all through regulatory or other requirements.

- **i) An earthquake that is more severe than the one the nuclear power plant was designed to withstand.**
- **ii) For coastally-located nuclear power plants, a tsunami that is more severe than the one the nuclear power plant was designed to withstand.**

iii) A loss of operating power that is longer than current regulations are required to address.
iv) A total station blackout (i.e., loss of operating power and failure of emergency diesel generators) that is longer than current regulations are required to address.
v) A hydrogen explosion that occurs due to the buildup of hydrogen in the core or other areas of a nuclear reactor due to the failure of mitigation technologies such as hardened vents or hydrogen recombiners, and the causes of such failures.
vi) A hydrogen explosion that occurs due to the buildup of hydrogen in the spent fuel storage area of a nuclear reactor due to the absence of mitigation technologies such as hardened vents or hydrogen re-combiners.
vii) A breach in the containment vessel of a nuclear reactor core caused by a hydrogen explosion.
viii) A breach in the structure of a spent nuclear fuel storage area due to an earthquake or hydrogen explosion.
ix) The failure of the recirculation pump seals within the reactor pressure vessel which may prevent cooling water from fully filling the pressure vessel and thus covering and cooling the nuclear fuel rods contained therein.
x) The failure of one or more safety relief valves within the primary containment area that could enable the transfer of radioactive core material between the drywell and the torus.
xi) The potential melting of core material through the pressure vessel and into the drywell or torus of the nuclear reactor.
xii) The failure of the isolation condenser and/or reactor core isolation cooling systems and subsequent inability to provide cooling function to the nuclear reactor cores.
xiii) The failure of the primary containment vessel spray cooling and core spray systems.
xiv) The failure of systems used to cool spent nuclear fuel storage areas, including areas that contain varying amounts of spent nuclear fuel of varying ages.
xv) The failure of diagnostic equipment to accurately monitor temperature, water levels, hydrogen/oxygen concentrations, pressures and radiation onsite, both during a total station blackout and after basic electricity function is restored (such as if the devices have been damaged by water, radiation or other events).
xvi) The absence of a source of fresh cooling water with which to cool the reactor core and spent nuclear fuel storage areas.
xvii) The absence of a means by which to store large quantities of highly radioactive water that has leaked or spilled after being used to cool the core and spent nuclear fuel storage areas.
xviii) Repeated earthquake aftershocks that further threaten the integrity of the already-compromised reactor core, spent nuclear fuel storage areas, and emergency operations.
xix) The ability to manually repair or restore function associated with any of the above failures or events when faced with extremely high levels of radiation that may threaten the health and safety of those both on and offsite.

a) The answer to items (ix), (x), and (xviii) is "yes." The answer to all others is "no," except for the following clarifications:

iii) On-site power systems are required to have supplies of consumable material that support a period of operation typically four to seven days, allowing resupply following extreme natural phenomena.

xii) Failure of the isolation condenser and/or reactor core isolation cooling systems is analyzed. Subsequent failure of a single train of backup systems also is analyzed. However, subsequent inability to provide cooling to the core (failure of multiple backup systems) is not a design basis accident.

xiii) Failure of a single train of these systems is analyzed. However, subsequent failure of backup trains is not a design basis accident.

xiv) Failure of a single pump in these systems is evaluated. Subsequent failure of the remaining pump is considered in establishing the necessary rate of make-up water supply.

b) The TI inspections were intended to provide a high-level look at industry preparedness. They are a first step in a multi-step assessment the agency is pursuing as a result of events in Japan. Aspects of some of the events listed in Question Three were addressed during the TI inspections, but the entire list of events was not addressed during those inspections in a comprehensive way. The NRC task force analyzed what we knew about events in Japan and making appropriate recommendations. Our longer-term review will analyze more complete technical information from the events in Japan, including specific information on the sequence of events and the status of equipment during the duration of the event. During that review, the agency will evaluate all relevant technical and policy issues related to the event and identify specific actions and further analysis, as appropriate, to ensure the U.S. reactor fleet continues to operate safely.

c) See response to (b) above.

d) See response to (b) above.

e) The results of the TI inspections have been made publicly available and, if relevant to the agenda, will be discussed at the scheduled meetings during which the staff will report to the Commission on its activities.

(f) The NRC's response to the events in Japan will consist of several components – initial inspections to assess licensee preparedness, a near-term look at what we now know about events in Japan, and a longer-term look once we have more complete technical information about those events. Decisions about regulatory changes or other actions will be made as each phase of this process is completed.

4. The Commission directed its staff to obtain external stakeholder input as part of both its near-term and longer-term work. Please fully describe all plans to solicit such input. Specifically, will any licensee or other nuclear industry personnel be accompanying inspectors during these inspections at nuclear power plants? If so, will NRC also ensure that appropriate non-industry individuals that possess the appropriate expertise and security clearances are also provided such an opportunity?

The near-term review had limited stakeholder involvement because of the accelerated nature of what the NRC was trying to accomplish. During that time, however, when information was needed to support the near-term review, the task force obtained that input from various sources. Specific agency actions that may result from the efforts of the task force will follow our normal processes for stakeholder involvement (e.g., public comment periods on rulemakings).

Regarding the conduct of the recent TI inspections, the NRC performs independent inspections. In some cases, we have ongoing arrangements for state representatives to participate in NRC

inspections. Licensees occasionally are requested to accompany the NRC on an inspection, but only if we believe we may need them to answer questions, not to participate in the inspection itself.

5. Why have inspectors only been provided with 40 hours (or 50-60, in the case of a multi-unit nuclear power plant) with which to complete their work? Why does the Commission have confidence that the necessary knowledge with which to inform our own safety efforts can be obtained in such a short period of time?

The TI indicates that the estimated average time to complete the TI inspection requirements is 40 hours per site. This estimate was based on how much time likely would be necessary to accomplish the required activities and recognition that these inspections needed to be conducted and results documented in a fairly short timeframe, approximately five weeks. There was no official direction that actual inspection hours could not be lower or higher than this estimate. The actual inspection effort to complete TI-183 accounted for over 2600 inspection hours, or about 40 hours per site. The actual inspection effort to complete TI-184 accounted for approximately 900 inspection hours, or about 14 per site. TI-184 focused on the availability and readiness of a plant’s severe accident management guidelines as requested by the Task Force, as compared to the broader high-level look at the industry’s preparedness per TI-183.